U. S. NUCLEAR REGULATORY COMMISSION
REVIEW OF IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER
DRAFT WASTE-INCIDENTAL-TO-REPROCESSING DETERMINATION FOR
TANK FARM FACILITY RESIDUALS

Attachment

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1. INTRODUCTION

1.1 Background – History of Reprocessing Activities at the Idaho National Engineering and Environmental Laboratory

The Idaho National Engineering and Environmental Laboratory (INEEL), a 2300-square kilometer (890-square mile) site located in eastern Idaho, was initially established to develop civilian and defense nuclear reactor technologies (see Figure 1). The Idaho Nuclear Technology and Engineering Center (INTEC) at the INEEL was established in 1953 to recover fissile uranium by reprocessing spent nuclear fuel. The spent fuel was dissolved, producing an acidic aqueous solution, which 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 U.S. Department of Energy Idaho Operations Office (DOE-ID) ceased reprocessing activities at INTEC.

The liquid waste from spent fuel reprocessing was stored in the Tank Farm Facility (TFF), which consists of 11 1000-cubic meter (m^3) (300,000-gallon) and four 100-m^3 (30,000-gallon) underground storage tanks (see Figure 2). First-cycle solvent extraction waste was initially stored separately from other reprocessing wastes. Other reprocessing wastes include decontamination solutions from maintenance and closure activities and second- and third-cycle reprocessing extraction wastes.

Beginning in 1963, INEEL began to stabilize the first-cycle and most of the second- and third-cycle extraction wastes in a solid form through removal from the tanks and subsequent calcination. Calcination is a thermal process where liquids are converted to solid oxides. In January 1990, the U.S. Environmental Protection Agency (EPA) issued a Notice of Noncompliance [1], since the 1000-m^3 (300,000-gallon) tanks did not meet the secondary containment requirements of the Resource Conservation and Recovery Act. The Notice of Noncompliance resulted in a Consent Order from the Idaho Department of Health and Welfare [2] that required INEEL to upgrade the tank system or permanently cease use of all 1000-m^3 (300,000-gallon) tanks before the end of calendar year 2012. As of March 1998, INEEL completed calcination of the first-cycle extraction waste and most of the second- and third-cycle extraction waste.

Figure 1

Figure 2

1.2 DOE-ID Tank Closure Strategy

The 11 large underground storage tanks[1] are contained in octagonal or square concrete vaults. The 1000-m^3 (300,000-gallon) tanks are stainless steel vessels with an inside diameter of 15 meters (m) [49 feet (ft)] and a wall height of 6.4 m (21 ft).[2] 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 of the tanks. The tops of the concrete vaults are covered with approximately 3 m (10 ft) of soil to provide radiation shielding. Figure 3 is a diagram of a belowground storage tank showing the tank, sand pad, concrete vault, and auxiliary piping.

An essential aspect of closing the underground storage tanks, containing the residual reprocessing and decontamination wastes, is the estimation of the tank residual waste inventory. DOE-ID has used a combination of historical process knowledge, modeling, and analytical sampling to characterize the tank contents. The physical form of the residual materials in the tank are solids and liquids. The liquid and solid phases from three tanks (WM-182, WM-183, and WM-188) were sampled and the resultant analytical data was used as a basis for estimating the composition of tank wastes that have not been recently sampled. Recognizing that there are numerous sources of uncertainty in the estimation of the residual tank inventory, DOE-ID introduced measures into the tank closure strategy to ensure that public health and safety would be protected. First, four different estimates of the residual inventory were developed to represent uncertainty in the calculations of future human exposure. These four estimates are hereafter referred to as *worst, conservative, realistic,* and *best*, and are described in more detail in Section 2.1 of this report. All four estimates are quite conservative with respect to preliminary cleaning results for tank WM-182 [3, 4]. Second, DOE-ID developed a sampling and analysis plan that it will use after tank cleaning to characterize the residual materials remaining in the tanks [5]. Tank closure will only proceed if radiological concentrations are acceptable (i.e., closure performance objectives can be met).

The first step in the DOE-ID TFF closure strategy is to remove the liquid waste from the tanks and empty the tank to the heel using the existing jet pumps. Next, the tanks and piping will be flushed with water and the tank will again be emptied to the heel level. The tank will then be washed and waste removed. Video surveillance and sampling of heel residuals will supply information to compare against closure performance objectives. If the residual waste can meet the performance objectives, then the heel will be displaced with a reducing grout. Absorbent materials would be added to eliminate any free liquids. If the performance objectives cannot be met, additional tank cleaning will be performed. After addition of the grout to the tank, the piping and the tank vault would be filled with grout. Finally, the remaining void space in the tank would be filled with grout.

---

1 The four-100 m^3 (30,000-gallon) underground storage tanks have been emptied and cleaned such that the residual inventory is insignificant compared to the 11 1000-m^3 (300,000-gallon) tanks. Therefore, the 100-m^3 (30,000-gallon) tanks are not further discussed. In addition, one of the 1000-m^3 (300,000-gallon) tanks, tank WM-190, only contained a very small amount of high-level waste (HLW) [0.2 m^3 (<50 gallons)] and 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-gallon) tanks. Therefore, only 10 tanks were included in the evaluation.

2 Two tanks (WM-180 and WM-181) have 7.0-m (23-ft) high walls.

Figure 3

DOE-ID, during the summer of 2002, demonstrated the effectiveness of tank cleaning on tank WM-182 at INTEC [4]. Cleaning operations took place on 18 separate days over a 3- to 4-month period. It was estimated that tank WM-182 contained 520 TBq (14,000 Ci) in solids and 510 TBq (13,800 Ci) in liquids before cleaning operations. WM-182 was cleaned using a washball and directional spray nozzles to rinse the tank and slurry the solids. The existing steam jet was modified during cleaning to remove as much of the solids as practical. Preliminary results suggest that the inventory was reduced to approximately 67 TBq (1,800 Ci), or about 93 percent removal. The rate of removal of radioactivity from the tank showed an exponential decrease, as would be expected of a system that approximates a continuously stirred tank. The exponential decrease provides a means to determine when removal of radioactivity from the tank is no longer practical. To put the inventory of WM-182 after washing into perspective, the aforementioned *best* inventory used in the performance assessment (PA) assumed that a tank has 460 TBq (12,500 Ci) of activity remaining after cleaning. Therefore, the results of PA and evaluation of key radionuclide removal are likely to be conservative.

In addition to residual waste remaining in the tanks, underground process piping will contain residual waste. DOE-ID developed a sampling plan to characterize the waste remaining in the process piping [6]. Estimates of the waste remaining in the process piping were developed from characterization of piping sections removed from the system. To account for uncertainty in the method used to collect the data and the limited amount of piping sampled, a safety factor of 500 was applied to the piping inventory [3]. The inventory remaining in the process piping is very small compared to the residual waste remaining in the tanks. Piping will be closed by multiple rinses and subsequent sampling of radionuclide concentrations. The piping will be grouted to provide stability and limit future releases.

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 inadvertent pumping of waste from a tank to the vault outside of the tank). The waste entered the tank vault sumps and was pumped back into the tanks approximately 24 hours later [3]. While the waste was in the sumps, radionuclides could diffuse into the sand pads underneath the tanks. 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, providing flushing for the sand pads. The contaminated sand pads have been included in the source term for DOE-ID's incidental waste determination.

1.3 Incidental Waste/Waste Incidental to Reprocessing Criteria

Since 1969, the U.S. Nuclear Regulatory Commission (NRC) has recognized the concept of incidental waste or waste incidental to reprocessing (WIR). Certain material that otherwise would be classified as high-level radioactive waste (HLW) need not be disposed of as HLW and sent to a geologic repository because the residual radioactive contamination after decommissioning is sufficiently low to not represent a hazard to the public health and safety if disposed of in a near-surface low-level radioactive waste (LLW) disposal facility. Consequently, incidental waste is not considered HLW.

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 [7]: (1) The waste has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; (2) The waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR Part 61; and (3) 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 the May 30, 2000, SRM on SECY-99-0284, "Classification of Savannah River Residual Tank Waste as Incidental," the Commission stated that a more generic, performance-based approach should be taken in regard to reviewing WIR determinations [8]. In effect, cleanup to the maximum extent that is technically and economically practical and demonstration that performance objectives could be met (consistent with those which the Commission demands for the disposal of LLW) should serve to provide adequate protection of the public health and safety and the environment. In the "Final Policy Statement for the Decommissioning Criteria for the West Valley Demonstration Project at the West Valley Site," dated February 1, 2002, the Commission noted the criteria that should be applied to the incidental waste determinations at West Valley:

(1) The waste should be processed (or should be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; and

(2) The waste should be managed so that safety requirements comparable to the performance objectives in 10 CFR Part 61, Subpart C, are satisfied [9].

These criteria are risk-informed and performance-based in that the criteria allow flexibility to develop innovative approaches to meeting the performance objectives in Part 61. In demonstrating that the performance objectives have been met, focus should be placed on the potential health consequences of leaving waste on-site (i.e., doses that might occur), rather than being concerned with more indirect measures of health risk, such as meeting specific radionuclide concentration limits. For HLW tank closure, it is not necessary to meet LLW Class C concentration limits for the residual materials remaining in the tanks, since the waste classification requirements for Part 61 were primarily derived to protect inadvertent intruders at an acute dose of 5 millisievert (mSv) [500 millirem (mrem)] [10]. Therefore, demonstration that the waste has been or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical and that the waste is managed so that safety requirements comparable to the performance objectives in Part 61, Subpart C (which include protection of the inadvertent intruder), are satisfied, is sufficient to provide for protection of the public health and safety and the environment.

On July 9, 1999, DOE issued DOE Order 435.1, "Radioactive Waste Management" [11]. DOE Order 435.1 and its associated manual and guidance [12, 13] require that all DOE radioactive waste be managed as HLW, transuranic (TRU) waste, or LLW. The Order states that waste, determined to be incidental to reprocessing, is not HLW and shall be managed in accordance with the requirements for TRU waste or LLW, if it meets appropriate criteria. DOE Order 435.1 discusses the WIR evaluation process and the criteria for a WIR determination.

1.4 NRC Review Approach

In September 2001, DOE-ID and NRC established a Memorandum of Understanding (MOU) that provides a basic framework for NRC review of the WIR determination of the INTEC TFF [14]. Under the terms and conditions of the MOU, NRC is acting in an advisory capacity, and any advice given to DOE-ID under the MOU does not constitute regulatory approval, authorization, or license for DOE activities.

NRC's initial review was based on DOE-ID's "Idaho Nuclear Technology and Engineering Center Tank Farm Facility Residuals – Waste-Incidental-to-Reprocessing Determination Report, Draft A," submitted in February 2002 [15]. To adequately evaluate the assumptions, parameters, models, and uncertainties, other additional documents were also reviewed [5,16-20]. NRC sent a request for additional information (RAI) to DOE-ID in June 2002 [21]. NRC and DOE-ID held a meeting on October 2, 2002, to discuss DOE-ID's preliminary responses to the RAI. In December 2002, DOE-ID submitted a revised determination for NRC review consisting of: 1) "Final Responses to the Request for Additional Information on the Idaho National Engineering and Environmental Laboratory Draft Waste Incidental to Reprocessing Determination for Tank Farm Facility Residuals"; 2) "Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory," Volumes 1, 2, and 3; and 3) "Idaho Nuclear Technology and Engineering Center Tank Farm Facility Residuals – Waste-Incidental-to-Reprocessing Determination Report, Draft B" [22,3,23]. NRC conclusions regarding DOE-ID's WIR determination, as provided in DOE-ID's revised draft, are presented in this report.

In the "Final Policy Statement of the Decommissioning Criteria for the West Valley Demonstration Project at the West Valley Site," dated February 1, 2002 [9], the Commission noted the criteria that should be applied to the incidental waste determinations at West Valley. NRC reviewed DOE-ID's WIR determination for the INTEC TFF to the criteria that are to be applied to incidental waste determinations at West Valley. NRC reviewed the determination to assess whether it had sound technical assumptions, analysis, and conclusions with regard to meeting these incidental waste criteria,[3] and thus, that DOE-ID's proposed management of residual tank farm waste as LLW is protective of public health and safety and the environment.

[3] (1) The waste should be processed (or should be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; and (2) The waste should be managed so that safety requirements comparable to the performance objectives in 10 CFR Part 61, Subpart C, are satisfied [9].

2. CRITERION ONE

The waste should be processed (or should be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical.

2.1 Tank Inventory and Sampling

A significant source of uncertainty in HLW tank closure can be the concentration of radionuclides in the residual materials remaining in the tanks. The inventory remaining in the tanks must be developed to determine that the waste has been processed to remove key radionuclides to the maximum extent that is technically and economically practical. DOE Order 435.1 provides an initial list of key radionuclides, based partly on those listed in Part 61. DOE-ID used this initial list to determine which radionuclides were key for TFF closure, via analyses of public, worker, and intruder exposures. It is expected that key radionuclides may vary for different sites and different scenarios. DOE-ID determined that: americium (Am)-241; carbon (C)-14; cesium (Cs)-137; iodine (I)-129; plutonium (Pu)-238; Pu-239; Pu-240; strontium (Sr)-90; and technetium (Tc)-99 were the key radionuclides for TFF closure. For the public exposures, Tc-99, I-129, and Sr-90 were the dominant radionuclides. For the intruder exposures, Cs-137, Sr-90, and Pu-238 dominated the radiological consequences. Hereafter, the term "key radionuclides" refers to those isotopes important for the assessment of TFF closure risks and not the generic list developed in DOE Order 435.1.

The physical form of the residual materials in the tanks is solid and liquid. The solid material, with an estimated specific gravity of 1.4, is easily dispersed [24]. The liquid and solid phases from tanks WM-182, WM-183, and WM-188 were sampled and the resultant analytical data were used as a basis for estimating the composition of tanks that have not been recently sampled. The concentrations for radionuclides lacking current analytical data were estimated using the ORIGEN2 model and analytical data for Cs-137 [20]. Sampling data show variability from tank to tank and also from sample to sample within a tank. Two of the radionuclides most important to the assessment of the risk to the public receptor are I-129 and Tc-99. The activity in the waste is dominated by Cs-137 and Sr-90. Cs-137 is a high-energy gamma emitter whereas I-129 and Tc-99 are low-energy beta emitters. Because of the high activity and the type of radiation emitted by the Cs-137 and Sr-90, it is difficult to characterize analytically how much I-129 and Tc-99 are present in the waste. DOE-ID provided an analysis using fission yield ratios to evaluate the analytical data for Tc-99 and I-129 [22]. It concluded that the measured values were reasonably consistent but that the analytical procedures were introducing additional uncertainty to the values for Tc-99 and I-129 concentrations. Therefore, model-generated values were used in lieu of measured values for I-129 and Tc-99.

A conservative tank residual waste inventory was prepared for demonstration of protection of public health and safety. The expected conservatisms were:

- Tank WM-188 was used as the basis for assigning radionuclide concentrations to all tanks. This is likely to be a conservative assumption because tank WM-188 has the highest measured radionuclide concentrations for any of the tanks that have been sampled.

- It was assumed that each tank will initially (before grouting) contain 2,317 kilograms (5108 pounds) of tank solids and 4,989 liters (L) (1318 gallons) of liquids, which corresponds to approximately 3.2 centimeters (cm) [1.3 inches (in.)] of material remaining in the bottom of the tank. These residual solid and liquid quantities are much larger than observed in the actual cleaning results for tank WM-182 [4].

- It was assumed that radionuclide concentrations in the solid materials will be unaffected by tank cleaning and that tank cleaning will only result in bulk material removal. This is a conservative assumption because flushing of the tanks with uncontaminated water would be expected to result in partitioning of radionuclides from the solid phase to the liquid phase and subsequent removal from the tank by pumping (thereby reducing the solid concentrations).

To account for uncertainty, the sensitivity to four different inventories was assessed in the performance assessment calculations (*worst*, *conservative*, *realistic*, and *best*). They are generally described by:

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

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

- The *realistic*-case inventory predicted 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 above refer to further reductions in the tank waste inventory after the existing transfer equipment has already removed as much bulk liquid and solid waste as possible. In addition, the total volume of liquid waste for each of the four inventories is assumed to remain the same. DOE-ID provided a sampling and analysis plan that it will use after tank cleaning, to characterize the residual materials remaining in the tanks [5]. Tank closure (grouting) will only proceed if closure performance objectives can be met, in accordance with the sampling plan.

2.2 Technical Practicality of Waste Removal Options

In making the technological selections for the INTEC TFF, DOE-ID evaluated numerous chemical and mechanical waste removal technologies for cleaning of HLW tanks, primarily as part of its participation in the Tanks Focus Area (TFA) Technical Team. The TFA Technical Team is a national group 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-ID did not need to conduct significant additional basic research on tank cleaning.

Chemical processes the TFA Team considered included the following [23]:

- 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.

- 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.

- 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 liquid, which contains the nonradioactive elements, is decanted.

The INTEC TFF waste is acidic and contains few solids, whereas waste at other DOE sites is typically nonacidic and contains many solids. Typically, the larger amount of solids at other DOE sites resulted from precipitation processes when the waste was neutralized for storage. None of the chemical processes described above was determined to be technically practical for the INTEC waste. In particular, chemical cleaning was not practical for the TFF because the waste was already strongly acidic, and employing stronger acids created concerns with tank corrosion. Washing of solids with a basic solution, as in enhanced sludge washing, could cause solids to precipitate out of the solution. Therefore, chemical cleaning and enhanced sludge washing were methods that were determined not to be technically practical for removing additional key radionuclides from the waste at INTEC. In addition, caustic recycle and sludge washing are not applicable to the TFF wastes, as these chemical processes focus on waste reduction processes (as opposed to removal of key radionuclides and tank wastes to the maximum extent that is technically and economically practical). Saltcake dissolution would also not be applicable to the TFF wastes, as the waste has not been observed to adhere strongly to tank surfaces.

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 tank bottoms. Most pumping systems were not developed to remove small quantities of liquids with small amounts of suspended solids. The pumping systems were developed to remove large quantities of solids and thus were not technically practical for INTEC waste. Since 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 system consisting of a spray ball (washball) and directional spray nozzles, combined with a modified steam-jet pumping system, was determined to be the preferred technology for the INTEC tanks. This technology was demonstrated in a full-scale mockup test, as well as in the cleaning of tank WM-182 [4].

2.3 Economic Practicality of Waste Removal Options

Because only limited technologies were technically applicable to and practical for the INTEC TFF, DOE-ID's economic evaluation only considered three main options: the preferred system (as described above); a hypothetical new system that would provide 50 percent more waste removal; and complete tank removal. Table 1 provides a comparison of the performance objectives and costs for the options considered. Because INEEL is employing the technology that it considers to be the best available technology, detailed calculations for those technologies considered not to be as effective were not presented. As stated above, the preferred system is a washball, directional nozzles, mechanical arms, and steam jets. Total cost of development and deployment is estimated as $27 million. DOE-ID used the cost of development and deployment of the preferred technology to estimate how much it may cost to develop a new technology. Based on engineering judgment and the progress made to date in tank cleaning technology, it is reasonably expected that a new technology could be developed. A new technology that is assumed to remove 50 percent more waste could provide some benefit, at an additional cost of approximately $14.4 million for 10 tanks. The new technology and the economic impact associated with its development are estimates of unknown accuracy. DOE-ID believes it is not economically practical to pursue development of a new technology because: 1) the new technology is not yet developed; 2) the performance objectives can be achieved with conservative assumptions in key models or parameters; and 3) removal efficiencies for key radionuclides are high for the preferred system (shown in Section 2.4). Complete tank removal has a very large economic impact, as well as a relatively 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 (9100 person-rem) compared to implementation of the preferred technology (6.5 person-rem).

Table 1. Comparison of PA Results to Performance Objectives

Performance Objective	DOE Limit	Preferred System*	New Technology**	Complete Tank Removal***
All-pathways dose to public (mrem/yr)	25	0.04	0.02	0
Acute intruder for drilling (mrem)	500	144	72	0
Acute intruder for construction (mrem)	500	0.93	0.47	0
Chronic intruder for postdrilling (mrem/yr)	100	53	26	0
Chronic intruder for postconstruction (mrem/yr)	100	33	17	0
Protection of individuals during operations (rem/yr)	5	0.04	0.04	1.07
Additional cost for 10 tanks ($)		0	14.4 million	5.33 billion

* Preferred system is washball, directional nozzles, mechanical arms, and steam jets. Total cost of development and deployment estimated as $27 million. Results are shown for the *best* inventory, rather than the *conservative* inventory used in the PA. This allows for a more realistic analysis of costs versus actual expected inventory.

** New technology is unknown, but assumed to be 50 percent more effective, with costs comparable to developing current best technology.

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

2.4 Removal of Key Radionuclides

Key radionuclide removal from the tank farm residuals at INTEC will be, and has been historically, primarily the result of bulk waste removal. After bulk removal through pumping and cleaning operations, waste residuals will remain as contamination in the TFF. The residuals will contain Cs-137, Sr-90, C-14, Tc-99, I-129, and some TRU isotopes considered to be key radionuclides because they are important in meeting the TFF performance objectives [23]. Of the approximately 3.2 million TBq (87 million Ci) of waste generated by spent fuel reprocessing at INTEC, approximately 3400 TBq (92,600 Ci) are estimated to remain in the TFF after closure, representing about 0.1 percent of the initial inventory. For Cs-137, Sr-90, Am-241, Pu-238, C-14, I-129, neptunium (Np)-237, and Tc-99, the removal efficiencies (compared to initial inventories) are estimated to be 99.9 percent, 99.9 percent, 99.7 percent, 99.7 percent, 99.8 percent, 99.6 percent, 99.9 percent, and 98.2 percent, respectively [23]. The removal percentages are based on 2.5 cm (1.0 in.) of heels plus 1500 L (400 gallons) of free liquid remaining in each tank at closure. It should be noted that development of the 3400-TBq (92,600-Ci) value for waste remaining in the TFF at closure is based on the best available radionuclide concentration estimates and therefore is different from the total curie value assigned for the *best* inventory in the PA (which was based on scaling of radionuclide concentrations with WM-188). Before grouting operations begin in a tank, the tank will be sampled and visually inspected to ensure that closure performance objectives can be met.

2.5 NRC Review and Conclusions (Criterion One)

NRC had a number of requests for additional information with respect to the source term inventory, its composition, and the associated uncertainty [21]. In general, DOE-ID asserted the source term inventory used to show compliance was conservative but NRC's initial review did not reach the same conclusion. In particular, it was unclear that the approach of scaling the quantities of some key radionuclides in the tank residuals based on analytical values for Cs-137 appropriately captured the variability in radionuclide concentrations. In addition, only limited sampling has been performed to characterize the solids remaining in the tanks. In response to NRC concerns, DOE-ID developed and analyzed a range of possible source term estimates.

The development of the source term estimates (*worst*, *conservative*, *realistic*, and *best*) by the DOE-ID are conservative, with respect to the information that is known. The following conservative assumptions were applied to all four inventories. First, the Cs-137 concentration from tank WM-188 (the highest measured) was assigned to all tanks, even those that had analytical sampling and showed much lower values. Second, DOE-ID assumed that concentration of radionuclides in the solid inventory would be unaffected by tank washing. It is likely that some fraction of the activity contained within the solid inventory is associated with sorption onto solid phases. Therefore, it is expected that washing of the tank contents would result in the partitioning of radionuclides into the liquid phase and removal from the tank. Third, the main conservatism with respect to the known information is in the assignment of the amount of material remaining in the tanks at closure. It was assumed that the *worst*-case inventory had 2.5 cm (1.0 in.) of heels remaining in the tanks, with an additional 1500 L (400 gallons) of free liquid. The *best*-case inventory had a 50 percent reduction in solid volume and a 95 percent reduction in liquid concentration. The total activity remaining in each tank for the *worst*-case inventory was 1050 TBq (28,500 Ci) and for the *best*-case inventory 463 TBq (12,500 Ci). Preliminary results for cleaning of tank WM-182 suggest the inventory for that tank may have been reduced to as low as 67 TBq (1,800 Ci) [4]. The emplacement of grout may result in the further removal of activity by redistributing residual waste to the pumps, which would facilitate removal from the tanks. DOE-ID accounted for the uncertainty associated with limited sampling of tank residuals and the uncertainty in the methodology to develop tank inventories by introducing conservatism in other aspects of the approach to inventory development.

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 (~ pH 2), it would not likely be technically practical to pursue bulk chemical cleaning. Cleaning of tank WM-182 has demonstrated that the mechanical processes selected for bulk waste removal will be effective [4]. Complete tank removal would reduce potential annual doses to the public and inadvertent intruders further below the performance objectives. However, tank removal would be economically unreasonable and would result in relatively large worker exposures compared to the preferred technology. The tank cleaning technologies DOE-ID selected appear to be the most economically and technically practical.

Using best available information, DOE-ID estimated bulk waste removal would reduce the original tank inventory of 3.2 million TBq (87 million Ci) to approximately 3400 TBq (92,600 Ci), a reduction of 99.9 percent of the activity. Considering actual tank cleaning results from tank WM-182, waste removal at the TFF may reach 99.95 percent of the original TFF inventory.

Conservative estimates DOE-ID provided for key radionuclide removal were verified to be greater than 98 percent.

The following assumptions were made in assessing conformance with Criterion One:

- Economic values assigned to equipment, labor, and other relevant variables are reasonable.
- Current estimates of the radiological concentrations of the waste are reasonably accurate.

The following conclusions are made with respect to Criterion One:

- The DOE-ID methodology for solid mass estimation and liquid volume estimation is technically adequate.
- The conservative source term is likely to bound the residual materials concentrations and quantities actually remaining in the tanks.
- DOE-ID's argument that key radionuclides will be removed to the extent technically and economically practical is reasonable.

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

- Sampling of the radiological composition of residual materials remaining in tanks after cleaning should be completed before tank grouting and final closure, in accordance with DOE-ID's sampling plan.
- Because of the cooperative physical characteristics of the residual materials remaining in the tanks and the relatively small economic impact associated with tank flushing, DOE-ID should follow its current plan for cessation of tank flushing only after removal of residual activity from the tank becomes insignificant (as discussed in Section 1.2).
- DOE-ID should stay abreast of tank cleaning technology for potential use in future tank cleaning, if such technology is technically and economically practical.

3. CRITERION TWO

The waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW, as set out in 10 CFR Part 61.

As discussed in Section 1.3, for HLW tank closure it is not necessary to meet LLW Class C concentration limits for the residual materials remaining in the tanks (Criterion Two). In effect, cleanup to the maximum extent that is technically and economically practical (Criterion One) and demonstration that the waste is managed so that safety requirements comparable to the performance objectives in Part 61, Subpart C, are satisfied (Criterion Three), should serve to provide adequate protection of the public health and safety and the environment. Therefore, this review did not evaluate Criterion Two.

4. CRITERION THREE

The waste should be managed so that safety requirements comparable to the performance objectives in 10 CFR Part 61, Subpart C, are satisfied.

Criterion Three is designed to protect the general population from releases of radioactivity, to protect individuals from inadvertent intrusion into the waste, to protect individuals during operations, and to evaluate stability of the disposal site after closure. Protection of the general population (including intruders) is typically evaluated through a PA calculation that takes into account the relevant physical processes and the temporal evolution of the system.

§ 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." [25]

The 0.25-mSv/yr (25-mrem/yr) limit applies for the post-closure period of a disposal facility. The other radiological control limits of 10 CFR Part 20, "Standards for Protection Against Radiation," apply during facility operation [26].

§ 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."

Although a particular dose limit is not specified in this performance objective, compliance with the technical requirements of Part 61 and, in particular, with the classification system of 10 CFR 61.55, is considered to provide adequate protection to intruders at a near-surface land disposal facility. In the Draft Environmental Impact Statement for Part 61 [10], NRC used a 5-mSv/yr (500-mrem/yr) dose limit to an inadvertent intruder to establish the concentration limits and other aspects of the waste classification system. In addition, Part 61 does not specify a time for institutional controls in the performance objectives, but does require, in 10 CFR 61.59(b), that "... controls may not be relied upon for more than 100 years."

§ 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 LLW disposal facility workers.

§ 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. A disposal facility should not require long-term maintenance and care. Stability is particularly important considering the requirements in 10 CFR 61.59(b) that “... institutional controls must not be relied upon for more than 100 years following transfer of control of the disposal site to the owner.”

4.1 Performance Assessment to Demonstrate Performance Objectives

4.1.1 Performance Assessment Overview

DOE-ID has prepared a PA to demonstrate protection of the general population from releases of radioactivity as well as protection of individuals from inadvertent intrusion [3]. Most of the sections that follow describe the models, parameters, and analyses used to demonstrate the safety of the public resulting from nominal (expected) behavior of the system. Separate sections are provided for the inadvertent intruder (public) and protection of individuals during operations. Various tools were used to model engineered system degradation, release of radionuclides from the source term, transport of contaminants through the unsaturated zone, transport of contaminants through the saturated zone, and interception of contaminants by the public, resulting in radiological exposure. Source term modeling and release used the Disposal Unit Source Term-Multiple Species (DUST-MS) model [27]. Different conceptual model options are available in DUST-MS for source releases (e.g., rinse with partitioning, diffusion, uniform degradation, and solubility-limited release). A surface rinse model was selected for the PA. The surface rinse model accounts for partitioning between the infiltrating water and the radionuclides in the waste form. DUST-MS was also used to model partitioning and retardation for radionuclide transport occurring in the grouted tank, sand pad, and vault floor. Transport of contaminants through the unsaturated and saturated zones used PORFLOW [28]. PORFLOW is a mathematical model used for the simulation of multi-phase 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 [11]. The well is used to withdraw water for personal consumption and for watering a small garden, as well as other domestic purposes. DOE-ID developed a FORTRAN program to convert radionuclide fluxes into annual doses following dose methodology presented in various reports [29-31]. 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 downgradient location of the potential receptor was based on the site hydrology, perpendicular to a row of two tanks instead of five. Therefore a two-dimensional model of unit thickness, containing a cross-section of two tanks, was employed to evaluate public exposures via the groundwater pathway. The water pathway modeling, therefore, conservatively neglected lateral dispersion of contaminants during transport in one of two directions perpendicular to the flow direction.

Initially, limited uncertainty and sensitivity analyses were performed for the model assumptions and parameter choices. In response to the NRC RAI, further uncertainty and sensitivity analyses were performed and the PA was expanded to include additional data from the literature, selection criteria, sensitivity analysis of the selected inventory and transport parameters, and additional simulations for vertical hydraulic conductivity and flooding [21]. DOE-ID performed modeling on four separate scenarios: *best*-, *realistic*-, *conservative*-, and *worst*-case. For these scenarios, model simulations were conducted for variations in source inventories, release and transport parameters, and infiltration rates. DOE-ID used the *conservative* case to compare to the performance objectives and to provide a demonstration of compliance. The uncertainty and sensitivity analyses using the *best*-, *realistic*-, and *worst*-case scenarios were intended to serve as an evaluation of whether the *conservative* case is indeed conservative.

4.1.2 Source Term

Tank sampling and estimated inventories are discussed in Section 2.1. DOE-ID has used a combination of historical process knowledge, modeling, and analytical sampling to characterize the tank contents. DOE-ID employed a variety of conservative assumptions (e.g., assigning the radionuclide concentrations in all tanks based on the highest sampled value) to provide a bounding estimate for tank residual radionuclide inventory with respect to current knowledge. In response to the NRC RAI, DOE-ID developed four different inventories for the PA, identified as *worst*-, *conservative*-, *realistic*-, and *best*-case, to account for uncertainty [22]. The *conservative* inventory assumed a layer of heels 2.5-cm (1.0 in.) thick would remain in the tanks after cleaning in addition to 1500 L (400 gallons) of free liquid. The final inventory at tank closure is expected to be residual solid and liquid waste distributed within and covered by a reducing grout layer. Any liquid waste that is not bound in the grouting process will be bound through introduction of an adsorbent before tank closure. The solid residuals remaining in the tank will contain the bulk of the activity.

It is assumed that the concentration of radionuclides in the solids is unchanged during tank washing. DOE-ID expects that this is a conservative assumption as radionuclides would partition from the solid phase to the liquid phase and be removed during tank flushing. At the completion of tank flushing, DOE-ID will perform sampling of the tank residuals to verify radiological composition of the actual inventory remaining in the tank. The plan for tank sampling was provided to NRC to incorporate in its review [5]. Sampling of actual tank inventories will be used to manage uncertainties associated with development of the amount and composition of tank residuals from limited information. Tanks will not be closed if sampling indicates that protection of public health and safety cannot be reasonably assured (i.e., if closure performance objectives could not be met). Closure of the TFF tanks will be performed in two phases: tanks WM-182 and WM-183 will be closed in the first phase and will serve as a proof-of-process demonstration of the waste removal, decontamination, and sampling techniques for the closure of the remaining TFF tanks [3]. The remaining TFF tanks will be closed in the second phase.

In addition to residual waste remaining in the tanks, process piping will contain residual waste. DOE-ID developed a sampling plan to characterize the waste remaining in the process piping

[6} Estimates of the waste remaining in the process piping were developed from characterization of piping sections removed from the system. To account for uncertainty in the method used to collect the data and the limited amount of piping sampled, a safety factor of 500 was applied to the piping inventory [3]. The inventory remaining in the process piping is very small compared to the residual waste remaining in the tanks. The simulated risks from the piping residuals were small compared to the tanks or contaminated sand pads.

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, or the inadvertent pumping of waste from a tank to the vault outside of the tank. The waste entered the tank vault sumps and was pumped back into the tanks approximately 24 hours later [3]. While the waste was in the sumps, radionuclides could have diffused into the sand pads underneath the tanks. 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, providing flushing for the sand pads. The residual inventory in the sand pads was developed based on an analytical model with 38 flushing events. The actual number of water transfers from the sand pad likely exceeds 130 for each vault to date [3]. The sand pad was modeled to fill with water, radionuclides partition from the contaminated solid sand particles to the water, and are subsequently removed when the sump pump operated. No direct analytical characterization of the concentration of radionuclides in the sand pad has been performed. Information was provided to demonstrate that flushing did occur, but it is inconclusive with respect to the level of contamination remaining in the sand pad at eventual facility closure [22]. Sand pad inventories have been difficult to evaluate with confidence. DOE-ID calculations rely on a number of assumptions regarding initial contamination and incremental removal over a 40-year period.

4.1.3 NRC Evaluation – Source Term

DOE-ID has developed a reasonable source term estimate through a combination of historical process knowledge, modeling, and analytical sampling. Current tank cleaning results from tank WM-182 suggest that even the *best* inventory used by DOE-ID in the PA is more conservative than actual tank cleaning results [4]. Visual examination of the tank shows a solid layer of approximately 0.16 cm (0.063 in.) or less, with the liquid waste effectively removed and replaced with demineralized water. NRC had numerous questions, associated with the source term, that DOE-ID addressed with additional information [21]. To manage uncertainty associated with limited characterization, DOE-ID elected to use a conservative inventory for PA calculations. Before closure of the tanks, DOE-ID will perform sampling to verify that the actual tank inventory is comparable to or less than the estimated inventory. If performance objectives could not be met, then additional tank cleaning would be pursued.

Triple rinsing of process waste lines with water and employing a safety factor of 500 to the piping inventory should ensure the process lines source term employed in the PA modeling is a conservative representation of the actual inventory. In addition, the risks associated with the process waste lines are small relative to the performance objectives. Therefore, DOE-ID's approach to the process piping source term is reasonable.

The sand pad source term was estimated from knowledge and characterization of the contaminating events, combined with analytical modeling, to estimate concentrations at the time of facility closure. Because only 38 flushing events were used in the modeling when approximately 130 events are expected to have occurred, the sand pad inventory is likely to be overestimated. The methodology for developing the sand pad inventory is reasonable and is also likely to be conservative. The PA results show that with the current inventory under the scenario of *worst*-case geologic sorption and moderate or high infiltration, there is the potential of exceeding the performance objectives as a result of Sr-90 releases from the sand pad (as currently modeled). Greater confidence in the sand pad source term could be gained through direct or indirect characterization of the Sr-90 concentration. For example, pumping the vault full of water and then retrieving a sample of the water before pumping to a tank could provide a measurement of the Sr-90 concentration in the liquid phase, even though direct sampling of the solid phase is not possible. The Sr-90 concentration in the liquid phase could be compared to the model estimates of the present-day concentration and could be used to calculate the expected concentrations on the sand pad solid phase.

Confirmation of the source term for the tanks should be accomplished and documented through the sampling and analysis plan described in the "Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM-182 and WM-183 Tank Residuals" [5]. Before grouting, actual tank sample results should be used to evaluate the post-cleaning status of the tanks, to compare the actual residual radionuclide inventory to the PA radionuclide inventory and ensure that public health and safety can be protected. In addition, because Sr-90 from the sand pad inventory constitutes the second-highest ground water model dose contributor, DOE-ID should explore methods for confirming its model of sand pad contaminant levels [3].

4.1.4 Release and Engineered System Degradation

DOE-ID developed failure times of the engineered components based on modeling of tank and grout degradation under the expected environmental conditions at the INTEC TFF [3]. The engineered components simulated were the concrete vaults, grout between the tanks and vaults, the waste tanks, and the grout inside the waste tanks. Initially, the engineered components provide a significant hydrologic barrier to the ingress of water to the waste. The engineered components gradually fail over time until they reach a state of failure where they no longer act as a hydrologic barrier to infiltrating moisture. Degradation of the grout can also be important for the chemical environment for release of radionuclides. Table 2 summarizes the degradation analysis results compared to the failure times assigned to the components in the PA. The failure times assigned in the PA were less than those developed in the degradation analyses. Degradation analyses were conducted to provide support for the PA analyses that assumed degradation step changes of 100 years for the vault and grout between the vault and tanks; 500 years for the tanks and grout inside the tanks; and 500 years for the piping. Uniform corrosion rates observed in stainless steel tanks WM-180 through WM-189 [5×10^{-10} to 1.3×10^{-6} m/yr (2×10^{-9} to 4.3×10^{-6} ft/yr)] were used as a basis for the corrosion rate used in the degradation calculation [1×10^{-5} m/yr (3×10^{-5} ft/yr)] [3]. Vault failure time was estimated from the expansion-corrosion reaction of the reinforcing steel. For the tanks and piping, corrosion was modeled as occurring inside and outside of the walls. The potential degradation mechanisms and factors included in the analyses that could affect component degradation included: initial cracks and voids; sulfate and magnesium attack; calcium hydroxide leaching; alkali-aggregate reaction; carbonation; acid attack; and corrosion of the tank, pipes, and reinforcement.

Table 2. Summary of Degradation Analysis and PA Failure Time Assumptions

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	>10000
Grout (between vault and tank)	100	3500	500	>10000
Piping	500	8000	1750	>10000
Tank and grout inside the tank	500	8000	1750	>10000

Because of the limited availability of site-specific data (e.g., properties of the grout, chemistry of soil moisture and water entering the vault), simple models were used along with a number of assumptions [3].

DOE-ID used the DUST-MS model for release modeling [27]. One-dimensional DUST-MS transport simulations were conducted for radionuclide sources in the grouted tank, piping, and the sand pad beneath the tank. Infiltration was assumed to contact and transport radionuclides after concrete degradation and tank corrosion. The two waste forms for the release simulations were the radionuclides in the grouted tank and piping and the radionuclides in the sand pad. DUST-MS has four different models to estimate release rates: rinse with partitioning, diffusion, uniform degradation, and solubility-limited release. The conceptual model selected for the simulations was a surface-rinse model. The surface-rinse model accounts for partitioning between the infiltrating water and the radionuclides in the waste form. Partitioning and retardation were modeled for radionuclide transport occurring in the grouted tank, sand pad, and vault floor.

The DUST-MS code has received extensive testing and verification and predictions have compared favorably to known analytical solutions as well as other code predictions [32]. In the RAI, NRC had a number of questions associated with the release rate calculation results, including the conceptual model and parameterization of the model [21]. DOE-ID stated it had confidence in the release rates because DUST-MS model results were verified by comparison to release rates generated with PORFLOW and by comparison to analytical model (hand) calculations. The distribution coefficients (K_d) for the grout were taken from two references (which compiled the results of numerous studies) [17,33]. Distribution coefficients for reducing conditions were used for the grout. DOE-ID stated that it would add fly ash (or other additives) to the grout to maintain reducing conditions. For some radionuclides, in particular Tc-99, reducing conditions can be favorable to radionuclide retention as a result of higher distribution coefficients compared to oxidizing conditions. DOE-ID anticipates that the reducing grout will alter the chemistry of the water that flows through the degraded grout. The reducing grout will increase the pH and decrease the oxidation potential of the infiltrating water. The concrete is expected to exhibit reducing conditions (Eh from –300 to –500 mV). Even after the grout no longer prevents water from contacting the waste, the chemical effects conducive to waste retention are expected to persist. The distribution coefficients for sand were taken from

Sheppard and Thibault (1990) [18]. Site-specific observations were unavailable for grout and sand pad sorption coefficients.

4.1.5 NRC Evaluation – Release and Engineered System Degradation

DOE-ID has completed modeling to develop estimates of engineered system failure (provided in Table 2). The modeling considered various corrosion and degradation mechanisms. Engineered system degradation is not extremely important with respect to long-lived radionuclides, but can be important for short-lived radionuclides such as Cs-137 and Sr-90. DOE-ID's failure times of engineered components used in the PA were shorter than the modeling suggested. DOE-ID's degradation models are based on reasonable conceptual models and physical processes expected to occur at the INEEL site. The approach of biasing engineered system failure to pessimistic (i.e., earlier) values is reasonable and not overly conservative, considering the limited amount of site-specific supporting information and the unvalidated degradation models.

Degradation in DOE-ID's analysis was defined with respect to the engineered component's ability to limit water contact with the waste. Potentially more important, particularly with respect to the grout, is the ability of the engineered component to chemically limit the release of radionuclides. It is likely that large grouted structures in environments with relatively low infiltration, such as INEEL, may provide for reducing conditions for very long periods of time. The grout release partition coefficients are equivalent to the *conservative* case employed in sensitivity analyses for transport through the concrete vault [3]; however, these values do not coincide with the conservative values from the Bradbury and Sarott (1995) source cited.

DOE-ID will add fly ash, slag, or other substances to ensure reducing conditions in the grout. However, heterogeneity in the waste and engineered system, as well as fracturing and cracking of the grout, may provide for more oxidizing local conditions than otherwise expected (without a consideration of heterogeneity). Vadose zone systems with heterogeneity can be strongly oxidizing environments. Conclusions regarding the dissolution of grout, in particular the loss of calcium hydroxide, could be influenced by the presence of preferential pathways for flow. Leaching of calcium hydroxide on a local basis (e.g., associated with the preferential pathways) could be significant, whereas if the leaching were averaged over the total mass of grout (whether exposed to water or not), it may be insignificant. The need for additional technical bases for the use of reducing distribution coefficients for grout will depend on the risk significance. The sensitivity analyses DOE-ID provided used a range of reducing condition distribution coefficients. Future PA analyses should evaluate the sensitivity of the results to the use of oxidizing condition distribution coefficients for grout inside the tank and for the vaults. In particular, for the layer of degraded concrete just below the tanks, the selection of Bradbury and Sarott (1995) values for reducing conditions may not be appropriate for the redox-sensitive technetium [17]. The environment within the tanks is likely to be reducing because of the planned addition of reducing agents to the grout. However, it is not clear that those conditions will persist in the water as it flows into the degraded concrete in the unsaturated zone, where air exposure is likely. Therefore, it is possible that too much credit was taken for retardation of Tc-99 in the degraded concrete.

The sand pad partition coefficients are higher than the *worst*-case values chosen for interbedded sediments in the transport sensitivity/uncertainty analysis [3]. This contrast is notable because the sand pad was constructed with relatively clean sand that may be coarser, more free of clay minerals, and thus less sorptive than the interbedded sediments in the subsurface. In addition, sand pad partition coefficients were not included in the sensitivity/ uncertainty analysis. Sr-90 from the sand pads is the major dose contributor in some model cases [3]. DOE-ID should consider providing additional justification for sand pad partition coefficients in future PAs or demonstrate that the model results are not sensitive to this parameter.

Use of the surface-rinse model in DUST-MS is a reasonable representation of the physical processes expected to occur at the INEEL site. Confidence is gained in the approach by the comparison of the DUST-MS results to results from PORFLOW and analytical calculations. Confidence in the modeling results could be enhanced through the consideration of analog systems (e.g., release rates of radionuclides from grouted systems at other sites) or through experimentation on surrogates (e.g., laboratory or field experiments using non-radioactive elements).

4.1.6 Hydrology and Transport

A two-dimensional unsaturated/saturated model (PORFLOW) was used to simulate water and contaminant transport in the subsurface at the INTEC TFF [28]. Figure 4 is an illustration of the model. The two-dimensional model allowed a detailed approximation of the geology underlying the facility. With 103 vertical layers, DOE-ID modeled the complex geology underlying the TFF from the Big Lost River in the north, through the center of two tank vaults, and southward for a total distance of 2500 m (8200 ft) in the downgradient direction. The top boundary was located at the ground surface, the bottom at 200 m (700 ft) below ground surface, and the water table elevation varied between 134 and 139 m (440 and 456 ft) below the ground surface within the model domain. A uniform net infiltration flux formed the top boundary condition for the model, with the exception of simulations that incorporated seepage from the Big Lost River, and a no flow boundary condition was present at the model base. A wide range in infiltration rates at the site [0.41 to 12.0 cm/yr (0.16 to 4.7 in./yr)] is reported in the literature, some of which were determined with site-specific studies [3]. In the revised PA submitted to the NRC, infiltration rate was considered to be an uncertain parameter and was included as part of the sensitivity/uncertainty analyses. The north and south unsaturated zone boundaries were no flow, whereas the north and south saturated zone boundaries were set to constant hydraulic head values, based on the regional potentiometric surface.

As a result of hydrologic uncertainty, DOE-ID made certain assumptions in an attempt to provide a reasonable model of present and future hydrologic conditions. Existing hydrologic and geologic data were used if possible. When there were conflicting sources of data, an attempt was made to confirm the data by consulting additional sources. If there was no information available, then a conservative approach was adopted that resulted in using parameters that resulted in the highest transport rates. For example, if the only available information for a particular geologic unit was a range in hydraulic conductivity values, then the upper portion of the range was used in the model.

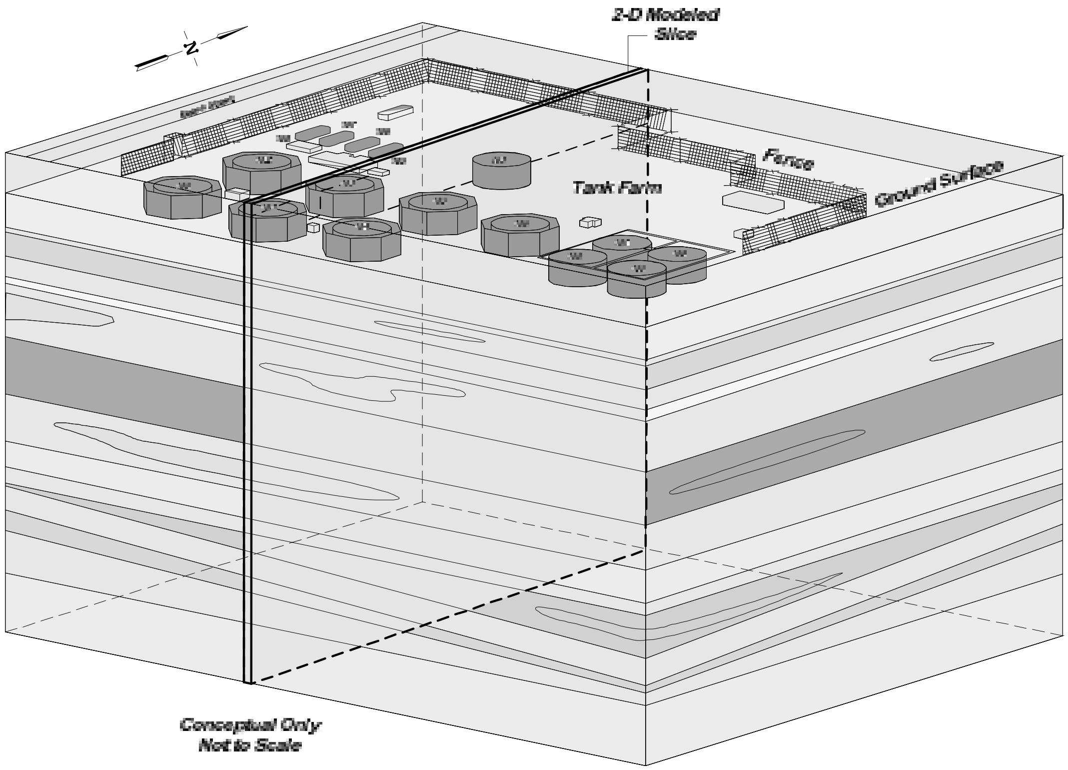

Figure 4. Illustration of the two-dimensional modeling slice used in PORFLOW

The geology underlying INTEC consists of a series of basalt flows and sedimentary interbeds with an alluvial veneer. There are over 30 individual geologic units that compose the unsaturated zone and upper regional aquifer at the site. The model used in the PA was based on a United States Geological Survey cross-section of the subsurface geology [34]. The cross-section contained alluvium, 18 basalt flows, and nine continuous and discontinuous sedimentary interbeds.

Perched ground water is present at INTEC. The mechanisms surrounding the factors that control the perched water zones underlying the TFF are not well understood and are controversial. It is postulated that the lithologic features contributing to contrasts in vertical hydraulic conductivity of the basalt layers and sedimentary interbeds provide the mechanisms for the development of perched water zones. The assumptions involving the perched water zones are perhaps the most important in developing and interpreting the numerical model because they control the hydraulic characteristics of the underlying formation. DOE-ID developed a calibrated model using existing observations, such as the distribution and extent of perched water bodies [3]. A complicating factor to the approach is that the major source of water for the perched water bodies is man-made. Upon eventual facility closure, the man-made sources for the perched water bodies will likely no longer exist. Additionally, there is uncertainty regarding the exact locations of all the perched zones because of the limited number of wells available for calibration. Therefore, an incremental modeling process was used to assess the impacts of key input parameters on the resulting numerical simulations. The calibrated model was then used for predictions of contaminant transport. Vertical hydraulic conductivity for the low-permeability sedimentary interbeds was subject to sensitivity analysis. Results indicate that the model is indeed sensitive to variations in vertical hydraulic conductivity for the key hydrostratigraphic units. The calibrated conductivity was within the measured range for lithologies of this type [3].

Dispersion of contaminants was modeled to occur only in the longitudinal direction (i.e., parallel to the direction of water flow). This is a conservative approach to the prediction of downgradient contaminant concentrations because it neglects transverse dispersion. Hydrodynamic dispersion is dependent on the pore-water velocity. Dispersivity values used in the TFF PA were developed from a literature review [3]. As the *best* and *realistic* values were identical for most parameters (including dispersivity, excluding inventory), DOE-ID generally used three scenarios (*best/realistic*-, *conservative*-, and *worst*-case) to propagate uncertainty in the PA analyses, thereby reducing the number of combinations evaluated in the overall uncertainty analyses.

Site-specific observations were typically unavailable for distribution coefficients associated with INEEL geologic materials (interbeds, basalts, alluvium) [3]. Knowledge of the sorptive properties of contaminants can be key to understanding contaminant movement. DOE-ID evaluated several compendia of soil and sediment distribution coefficient data. The most thorough of the compendia contained a breakout of distribution coefficients by major soil types [18]. The one significant exception to the use of generic distribution coefficients was the sediment value for Sr-90. The sediment was assigned distribution coefficients of 12 to 24 milliliters/gram (mL/g) in groundwater modeling studies at INTEC [35]. Because observations were made of Sr-90 transport (the Sr-90 source was not the TFF tanks), the investigators concluded that a sediment value of 24 mL/g resulted in a closer match of modeled concentrations and observed concentrations. Therefore, for the TFF PA, the *worst* scenario was assigned a value of 12 mL/g and the *best/realistic* scenario was assigned a value of 24 mL/g for Sr-90. In general, for most radionuclides, the sediment distribution coefficients for the *conservative* scenario (the scenario DOE-ID used to demonstrate that the performance objectives could be met) were assigned the minimum observed value for loam from Sheppard and Thibault (1990) [18]. As mentioned above, the main exception was Sr-90, which would have a range of 0.01 to 300 mL/g from Sheppard and Thibault (1990) but was assigned a range of 12 to 24 mL/g based on the site-specific modeling study and associated observations cited [35].

Transport modeling verification was possible because the disposal of tritium in percolation ponds and the associated monitoring data provided a tracer test to compare the transport of tritium predicted by the model with that observed under actual site conditions. The TFF PA model predicted tritium concentrations ranging from 0.4 to 3 becquerel/milliliter (Bq/mL) [10 to

80 picocuries/mL (pCi/mL)] while the observations were 0.20 to 1.36 Bq/mL (5.5 to 36.7 pCi/mL) [36].

DOE-ID analyzed the impact of a flood (resulting from failure of the Mackay Dam) on radionuclide transport. The flood was assumed to occur at tank failure (500-year postclosure). Infiltration was increased to 100 times the 12.4-cm/yr (4.9-in./yr) *worst*-case scenario infiltration rate. The PA results suggested that the transport simulations were very sensitive to infiltration rate, because it affected not only the transport rate through the unsaturated zone, but also the release rate from the waste form [3].

4.1.7 NRC Evaluation – Hydrology and Transport

The flood analysis that DOE-ID performed indicated that, while the peak concentration arrival time occurred slightly earlier than under non-flooding conditions, the peak concentration under flooding conditions was actually less than that estimated for non-flooding conditions. DOE-ID explained that the thickness and lateral extent of the perched water bodies beneath the TFF increase during the flooding scenario, 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.

The interpretation of the results of flow and transport models for a site-specific application can be difficult when generic information is heavily relied on. Sorption coefficients used in transport modeling through the aquifer appear to have been appropriately chosen, considering the large degree of uncertainty in the absence of site-specific data. In particular, the retardation behavior of fractured basalts at the site is not well understood. Distribution coefficients can vary widely from site to site as a result of varying mineral composition and differing geochemical environments. For example, in one compendium study, the distribution coefficient for strontium ranged from 0.05 to 190 mL/g in sand; 0.01 to 300 mL/g in loam; and 3.6 to 32000 mL/g in clay [18]. DOE-ID typically assigned the minimum observed values to the distribution coefficients for interbed porous material [3]. Because this is likely to be conservative compared to the actual amount of geologic sorption, it is a reasonable approach to manage distribution coefficient uncertainty. The main exception to this approach was strontium, which used site-specific values generated from observations of radionuclide migration [35]. The use of site-specific information is preferable, as long as uncertainty associated with the development of the information is recognized and accounted for. For example, as discussed earlier, the infiltration rate at the site is uncertain. The development of strontium distribution coefficients of 12 to 24 mL/g could have been conditional on the particular infiltration rate that was used in the study.

The overall modeling approach appears to be a reasonable attempt to manage uncertain hydrologic and geologic information. It appears, in general, that DOE-ID appropriately assessed the TFF transport system through use of conservative assumptions. These conservative assumptions help to simplify the transport model as well as add confidence in the output. Conceptual model uncertainty can, in some cases, significantly impact the risk when transport is through a fractured, unsaturated hydrologic system. Confidence is gained with the DOE-ID approach as a result of the comparisons of the model predictions for tritium movement with the observations [36].

4.1.8 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 INEEL site boundary. After active institutional controls cease at 100 years, the member of the public resides at the INTEC facility. An off-site 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 are predicted to occur. DOE Order 435.1 and its associated manual and guidance specify a receptor location of 100 m (300 ft) from the facility or as otherwise justified [11-13]. For the INTEC TFF, maximum contaminant concentrations are observed 600 m (2000 ft) downgradient from the facility in the PA, primarily as a result of the perched water bodies in the unsaturated zone [3]. 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 (30-ft) well-screen length.

The exposure pathways evaluated included the ingestion of contaminated water, ingestion of contaminated food, inhalation of contaminated airborne particulates, and external exposure to radionuclides in the air and on the ground (or soil) surface. The exposure pathways and mechanisms are more complex than the simplification provided here [3]. Release into the air pathway of volatile radionuclides was also considered. The analysis of the exposure pathways indicates that the ground-water 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) [29-31]. Parameters used in the dose model were primarily derived from values for the Yucca Mountain Project since the climate and geography are somewhat similar [37]. To account for uncertainty in the dose assessment modeling, most biosphere parameters were stochastic. DOE-ID 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) [38, 39].

The all-pathways total effective dose equivalent (TEDE) to a member of the public was 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.[4] Over 99 percent of the dose was from I-129 and Tc-99, with much smaller contributions from Sr-90 and C-14. DOE-ID applied a compliance period of 1,000 years as per the requirements of DOE Order 435.1 and its associated manual and guidance [11-13]. An evaluation was also performed for time periods out 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 dose [e.g., 0.014 mSv/yr (1.4 mrem/yr)] from I-129.

Table 3 provides a summary of the various scenarios analyzed and the performance objectives for protection of the public, protection of intruders, and protection of workers (individuals during

4 The dose methodology used in 10 CFR Part 61, Subpart C [based on International Commission on Radiological Protection Publication 2 (ICRP 2)], is different than that used in the newer ICRP 26. However, the resulting allowable doses are comparable, and DOE-ID used the newer methodology in ICRP 26.

operations). This table is pertinent to more than just this section ("Protection of the Public"), and should be consulted accordingly.

Table 3. Summary of Results Compared to Performance Objectives

Performance Objective (DOE Limit)	PA Result
All-pathways dose to public (not exceeding 25 mrem/yr)	1.4 mrem/yr
Acute drilling scenario (less than 500 mrem)	232 mrem
Acute construction scenario (less than 500 mrem)	0.93 mrem
Chronic postdrilling scenario (less than 100 mrem/yr)*	91.1 mrem/yr
Chronic postconstruction scenario (less than 100 mrem/yr)*	26.1 mrem/yr
Protection of individuals during operations (less than 5 rem/yr)	40.0 mrem/yr

* DOE Order 435.1 specifies 1 mSv/yr (100 mrem/yr) for a chronic intruder annual dose limit [12]. DOE Order 435.1 specifies separate performance objectives for airborne emissions and ground water protection. The air and water pathways were considered by the NRC in the review of the all-pathways dose assessment.

4.1.9 NRC Evaluation – Protection of the Public

DOE-ID has 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) 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"). DOE-ID's initial PA provided limited sensitivity analyses. NRC requested an expansion of the sensitivity analyses in a request for additional information [21]. Because the PA results were deterministic (with the exception of the dose model), DOE-ID provided a series of analyses to evaluate the impact of key uncertainties. The key uncertainties were residual radionuclide inventory, infiltration rate, transport parameters, and grout distribution coefficients. Table 4 contains a summary of the sensitivity analyses results. The shaded row is the result DOE-ID 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 4 represents 36 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 x 3 x 3 x 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 x 3 x 3 or 36. As an example, the eighth row down in Table 4 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 is provided to show 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. The

Table 4. Matrix of Sensitivity Analyses Results

Grout K_d	Transport	Infiltration (cm/yr)	Inventory	I-129 (mrem/yr)	Sr-90 (mrem/yr)	Total dose* (mrem/yr)
Worst case	Worst case	12.4	Worst case	40.4	85.8	85.8
			Conservative	15.2	85.8	85.8
			Realistic	11.7	85.8	85.8
			Best	7.76	85.8	85.8
Worst case	Worst case	4.1	Worst case	15.9	15	15
			Conservative	5.97	15	15
			Realistic	4.61	15	15
			Best	3.05	15	15
Worst case	Worst case	1.1	Worst case	4.65	0.18	4.65
			Conservative	1.75	0.18	1.75
			Realistic	1.35	0.18	1.35
			Best	0.89	0.18	0.89
Conservative	Conservative	12.4	Worst case	9.98	0.12	9.98
			Conservative	3.75	0.12	3.75
			Realistic	2.89	0.12	2.89
			Best	1.92	0.12	1.92
Conservative	Conservative	4.1	Worst case	3.59	0.006	3.59
			Conservative	***1.35***	***0.006***	***1.35***
			Realistic	1.04	0.006	1.04
			Best	0.69	0.006	0.69
Conservative	Conservative	1.1	Worst case	0.86	1.68×10^{-06}	0.86
			Conservative	0.32	1.68×10^{-06}	0.32
			Realistic	0.25	1.68×10^{-06}	0.25
			Best	0.17	1.68×10^{-06}	0.17
Best	Best	12.4	Worst case	2.61	2.36×10^{-04}	2.61
			Conservative	0.98	2.36×10^{-04}	0.98
			Realistic	0.76	2.36×10^{-04}	0.76
			Best	0.50	2.36×10^{-04}	0.50
Best	Best	4.1	Worst case	0.87	1.75×10^{-06}	0.87
			Conservative	0.33	1.75×10^{-06}	0.33
			Realistic	0.25	1.75×10^{-06}	0.25
			Best	0.17	1.75×10^{-06}	0.17
Best	Best	1.1	Worst case	0.24	5.65×10^{-12}	0.24
			Conservative	0.088	5.65×10^{-12}	0.09
			Realistic	0.068	5.65×10^{-12}	0.07
			Best	0.045	5.65×10^{-12}	0.04

* Total dose is not a summation of radionuclide-specific doses because of variability in the arrival times. Not all nuclides are shown in the table -- only two that elucidate system behavior and are risk-significant.

contribution of key radionuclides to peak annual dose and the timing of the peak annual dose can be significantly influenced by uncertainties.

Almost all the 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 1310 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. The arrival times for I-129 ranged from 538 years to 5670 years. Similar to Sr-90, the model results suggest that only under a very pessimistic scenario would the system not meet the performance objectives.

Cleaning results for tank WM-182 have achieved residual levels of waste significantly better than even the *best* inventory shown in Table 4. Therefore, actual risk to the public from tank residuals is likely to be significantly lower than the performance objectives even when uncertainty is considered. Sand pad risk to the public is also likely to be lower than the performance objectives because credit was taken for 38 flushing events, when approximately 130 are estimated to have occurred. Flushing of the sand pad with uncontaminated water results in a partitioning of radionuclides from the solid phase of the sand to the liquid phase, and the liquid phase is subsequently pumped back into the waste tanks.

DOE-ID applied a compliance period of 1,000 years as per the requirements of DOE Order 435.1 and its associated manual and guidance [11-13]. The performance objectives in Part 61, Subpart C do not specify a time period associated with protection of the general population. The recommendations of NRC's Performance Assessment Working Group, documented in NUREG-1573, include a time period of 10,000 years for analyzing performance with respect to 10 CFR 61.41 [40]. Disposal site performance is determined by many factors, including activity, half-life, and mobility of radionuclides in the waste inventory. Processes that control engineered barrier degradation, water infiltration and leaching of waste, and release and transport of radionuclides to the general environment can also significantly influence the disposal site performance. A detailed discussion and justification of a 10,000-year time period of analyses can be found in NUREG-1573 [40]. Because DOE-ID assessed longer-term risk in their PA model by analyzing out to 1 million years, DOE-ID's use of a 1000-year compliance period is not an issue for this WIR review. DOE should select a compliance period consistent with NUREG-1573 for future WIR determinations.

To better risk-inform the staff's review, NRC developed a PA model applicable to HLW tank closure, using the software platform GoldSim [41, 42]. The model was used to evaluate sensitivity of the PA results and to corroborate, in a general sense, the DOE-ID calculational results. Staff use of the model allowed a more focused review of those technical aspects of the problem more likely to influence the risks.

Staff concludes that there is reasonable assurance that safety requirements comparable to the 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 since the PA is the means to generate results to compare to performance objectives. Through demonstration of Criterion One (the waste has been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical), DOE-ID satisfied the intent of the provision to maintain releases of radioactivity to the environment ALARA. It should be noted that DOE-ID evaluated the commonly-used resident farmer scenario to assess the public exposures. This approach is acceptable to the NRC; however, NRC would consider other receptor scenarios proposed by DOE for analyzing public and intruder doses, as appropriate, in future WIR determinations.

4.1.10 Protection of Intruders

DOE-ID analyzed two 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 (30 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 bound 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-ID evaluated acute and chronic radiological impacts associated with both scenarios (intruder drilling and intruder construction). Approximately 1000 m (3300 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 with the waste residuals.

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 since the short-lived fission products (e.g., Sr-90 and Cs-137) are the main contributors to the intruder doses. Uncertainty exists in the state of concrete systems over time. However, DOE-ID asserts 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 are drilled directly through the waste. The acute intruder is exposed to drill cuttings spread on the land surface. Exposure time was set at 160 hours compared to the typical value of 6 hours, to account for the difficulty of developing an irrigation well at INEEL resulting from the presence of basalts in the subsurface [3]. The assumed diameter of an irrigation well was 0.56 m (1.84 ft) and the diameter of a residential drinking water well was 0.15 m (0.5 ft). Well diameters were 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. The major radionuclide contributors were Cs-137 and Pu-238 at 1.88 mSv (188 mrem) and 0.15mSv (15 mrem), respectively. Chronic exposure was considered as an extension of the acute drilling scenario. It was 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 included inhalation of resuspended drill cuttings, and ingestion of beef, milk, and vegetables contaminated via drill cuttings, but do not include the ground water 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 [0.52 mSv/yr and 0.37 mSv/yr (52 mrem/yr and 37 mrem/yr), respectively].

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- x 10-m [70- x 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 0.0093 mSv (0.93 mrem). 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), with Sr-90 and Cs-137 as the main contributors to dose [0.15 mSv/yr and 0.101 mSv/yr (15 mrem/yr and 10.1 mrem/yr), respectively].

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 [10]. DOE Order 435.1 and its associated manual and guidance specifies that the intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 1 mSv (100 mrem) in a year and 5 mSv (500 mrem) TEDE, excluding radon in the air [11-13]. All intruder scenario doses are less than 5 mSv (500 mrem) per year (all-pathways TEDE).

4.1.11 NRC Evaluation – Protection of Intruders

DOE-ID developed reasonable intruder scenarios to evaluate protection of inadvertent intruders and demonstrate that performance objectives comparable to 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 scenario. All intruder doses are calculated to be less than 5-mSv/yr (500-mrem/yr). DOE-ID invokes a likely conservatism in the analysis when it assumes that the drill cuttings are spread over an area that is one-half to one-sixth the size of typical lots in the region, thus concentrating the contamination. In addition and as discussed earlier, the tank residual inventory used in the PA is much larger than would be determined, based on recent cleaning results for tank WM-182 [4]. Therefore, it is expected that the maximum acute dose an intruder would receive is 2.32 mSv (232 mrem), which demonstrates with reasonable assurance protection of intruders.

4.1.12 Protection of Workers (Individuals During Operations)

The worker is protected by DOE regulations (10 CFR Part 835) which are analogous to the standards for radiation protection of individuals during operations set out in 10 CFR Part 20. DOE-ID developed estimates of worker doses based on experience from the cleaning of tank

WM-182. DOE-ID determined that most of the tank cleaning operations can be accomplished remotely and that worker dose is minimal. DOE-ID estimated the exposure per person to be about 0.40 mSv/yr (40 mrem/yr) for cleaning of a tank at the INTEC [43].

4.1.13 NRC Evaluation – Protection of Workers

The worker doses are significantly less than the 50-mSv/yr (5-rem/yr) regulatory limit, demonstrating that performance objectives comparable to 10 CFR 61.43 ("Protection of individuals during operations") could be achieved. Individual and collective doses would be significantly larger for complete tank removal [10.7 mSv/yr (1.07 rem/yr) per person] compared to the selected technology for tank cleaning. Therefore, it is expected that the worker protection performance objective (10 CFR 61.43) can be met.

4.1.14 Site Stability

DOE-ID plans to fill the tanks and vaults with grout, eliminating all voids to the extent practical, and thus providing structural stability of the vaults and tanks. As discussed previously, DOE-ID conducted degradation analysis for the grout, tanks, and vaults. Although cracking of the grout and corrosion of the stainless steel waste tanks are expected over long periods of time, significant structural collapse is not predicted.

The depth to the residual waste in the tanks is greater than 10 m (30 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.

4.1.15 NRC Evaluation – Site Stability

DOE-ID's plans to fill the tanks and vaults with 10 or more meters (30 or more feet) of grout and concrete appear sufficient to indicate that safety requirements comparable to 10 CFR 61.44 ("Stability of the disposal site after closure") can be met. Future actions to close the TFF will likely include an earthen cover. The cover design is not known at this time and therefore no credit was taken for an engineered cover that may limit infiltration of water to the waste. An engineered cover would likely 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 because of the depth of the waste. The arid environment, distance of the tanks from surface water bodies, and lack of significant grades is expected to limit fluvial erosion. 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 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.

4.2 NRC Review and Conclusions (Criterion Three)

<u>The following assumptions were used in assessing conformance with Criterion Three</u>:

- Active institutional controls will be maintained for 100 years.

- The dose calculations contained all the pathways that would provide a significant dose contribution.

- Sorption coefficients identified as conservative are sufficiently bounding in the release and transport models.

- The degradation analysis that produced the failure times for engineered components contained all significant degradation modes and mechanisms important for this application.

- The addition of further hydrologic model uncertainty in the sensitivity analyses with the PA would not significantly alter the conclusions.

- Current estimates of the radiological concentrations of the waste are reasonably accurate.

- Errors in documentation resulting from insufficient quality assurance controls will not significantly influence the calculational results.

<u>The following conclusions are made with respect to Criterion Three</u>:

- As indicated by the DOE-ID performance assessment, combined doses to the public from all pathways are projected to be well below the 0.25-mSv/yr (25-mrem/yr) limit; therefore, staff considers that there is reasonable assurance that safety requirements comparable to 10 CFR 61.41 can be satisfied, including ALARA requirements.

- Staff considers that there is reasonable assurance that safety requirements comparable to 10 CFR 61.42 for protection of individuals from inadvertent intrusion can be satisfied.

- The worker is protected by DOE regulations that are comparable to 10 CFR Part 20; therefore, the worker protection performance objective (10 CFR 61.43) can be considered to be met.

- DOE-ID's plans to fill the tanks, vaults, and ancillary piping with multiple layers of reducing grout appear sufficient to indicate that safety requirements comparable to 10 CFR 61.44 can be met.

<u>The following recommendations are made with respect to Criterion Three</u>:

- If sampling after tank cleaning indicates that the source term is significantly larger than that used in the current performance assessment, then the PA should be reevaluated.

- Although this assessment assumed that the *conservative* sorption coefficients for concrete, basalt, and interbedded sediments were sufficiently bounding, DOE-ID should consider expanding its literature review or conducting laboratory testing to provide additional confidence for the assertion of conservatism. Currently, the conservative values are simply calculated by interpolation between a lower bound and a realistic case. If retardation of Tc-99 in the degraded concrete layer at the base of the tanks provides a significant performance effect, a technical basis should be established for the assumption of reducing conditions in that location.

- Future PA analyses should evaluate the sensitivity of the results to the use of oxidizing condition distribution coefficients for grout.

- DOE-ID should investigate methods for measuring or better estimating the contaminated sand pad radionuclide inventories.

- DOE-ID should evaluate and, if needed, enhance quality assurance controls of documentation in future PAs as the TFF closure progresses. For example, Tables A-7 and A-8 of Revision 1 of the PA had some errors in the data for I-129 and Tc-99.

- As cleaning and closure of tanks progress, the closure strategy for each tank should be refined based on information obtained from prior tank and ancillary equipment closures at the TFF.

5. CONCLUSIONS AND RECOMMENDATIONS

The analysis performed regarding the proposed tank closure methodology was completed according to the terms and conditions of the established MOU [14]. It should be noted that NRC staff is providing technical assistance and advice to DOE-ID regarding the protection of public health and safety and the environment for the tank closure determination, which addresses the closure of tanks used to store HLW and sodium-bearing waste and evaluates whether the tanks and tank residuals may be managed as LLW. NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is incidental. NRC staff judgment as to the adequacy of the methodology is dependent on verification that the assumptions underlying the analysis are correct. This NRC assessment is a site-specific evaluation and is not a precedent for any future decisions regarding incidental waste activities at other sites.

NRC staff has concluded that the DOE-ID's WIR determination for residual tank farm waste demonstrates that the residual waste can meet the incidental waste criteria specified in the latest guidance for incidental waste determinations for tank closure (i.e., (1) the waste should be processed, or should be further processed, to remove key radionuclides to the maximum extent that is technically and economically practical; and (2) the waste should be managed so that safety requirements comparable to the performance objectives in 10 CFR Part 61, Subpart C, are satisfied) [9]. DOE-ID's determination that the residual waste from tank closure activities is incidental waste (to be managed as LLW), has sound technical assumptions, analyses, and conclusions with regard to protecting public health and safety and the environment.

5.1 NRC Recommendations for Future DOE-ID Tank Closure Activities

The following recommendations apply to future activities at the INEEL:

- The tank sampling protocol should be followed, with enough samples taken to adequately represent the residual tank contents after bulk waste removal and tank cleaning.

- If sampling after tank cleaning indicates that the source term is significantly larger than that used in the current PA, then the PA should be reevaluated.

- As the tank closure process will continue for approximately the next 13 years, technical feasibility of alternative waste removal and tank cleaning options, as well as tank grouting techniques, should continue to be evaluated.

- The closure strategy for the tanks or ancillary equipment should be refined, based on information obtained from ongoing tank and ancillary equipment closures throughout the DOE complex-wide Tank Focus Area Technical Team.

REFERENCES

[1] U.S. Environmental Protection Agency Notice of Noncompliance in the Matter of the United States Department of Energy Idaho Operations Office, January 29, 1990.

[2] Idaho Department of Health and Welfare Consent Order in the Matter of United States Department of Energy, Idaho National Engineering Laboratory, Idaho Code § 39-4413, dated April 3, 1992, modifications dated March 17, 1994, August 18, 1998, and April 20, 1999.

[3] U.S. Department of Energy, Idaho Operations Office, “Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory,” DOE/ID-10966, Revision 1 (Draft) (Volumes 1, 2, and 3), November 2002b.

[4] Kimmitt, R.R., Letter to V.L. Jacobson, “Tank WM-182 Cleaning Effectiveness,” October 30, 2002.

[5] Portage Environmental, “Sampling and Analysis Plan for the Post-Decontamination Characterization of the WM-182 and WM-183 Tank Residuals,” INEEL/EXT-01-00666, Revision 2, August 2002.

[6] Portage Environmental, “Sampling and Analysis Plan for the Post-Decontamination Characterization of the Process Waste Lines from INTEC Tank Farm Facility Tanks WM-182 and WM-183,” INEEL/EXT-01-01543, Revision 1, November 2001.

[7] U.S. Nuclear Regulatory Commission, Letter from R. Bernero/NRC, to J. Lytle/DOE, March 2, 1993.

[8] U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum - SECY-99-0284 - Classification of Savannah River Residual Tank Waste as Incidental," May 30, 2000.

[9] U.S. Nuclear Regulatory Commission, “Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement,” 67 FR 5003, February 1, 2002.

[10] U.S. Nuclear Regulatory Commission, *“Draft Environmental Impact Statement on 10 CFR Part 61, ‘Licensing Requirements for Land Disposal of Radioactive Waste,’”* NUREG-0782, Office of Nuclear Material Safety and Safeguards, September 1981.

[11] U.S. Department of Energy, “DOE Order 435.1: Radioactive Waste Management,” July 9, 1999.

[12] U.S. Department of Energy, “DOE Manual 435.1-1: Radioactive Waste Management Manual,” July 9, 1999.

[13] U.S. Department of Energy, “DOE Guide 435.1-1: Implementation Guide for Use with DOE M 435.1-1,” July 9, 1999.

[14] "Memorandum of Understanding Between the U.S. Department of Energy and the U.S. Nuclear Regulatory Commission Regarding the Management of (1) Existing Sodium-Bearing Waste and (2) Residual Waste from the Closure of the High-Level Waste Storage Tanks at the Idaho National Engineering and Environmental Laboratory," No. DE-GM07-01ID11435, September 7, 2001.

[15] U.S. Department of Energy, Idaho Operations Office, "Idaho Nuclear Technology and Engineering Center Tank Farm Facility Residuals--Waste-Incidental-to-Reprocessing Determination Report, Draft A," DOE/ID-10777, (and associated documents), February 2002a.

[16] U.S. Department of Energy, Idaho Operations Office, "Performance Assessment for the Tank Farm Facility at the Idaho National Engineering and Environmental Laboratory," DOE/ID-10966, December 2001 (Volumes 1, 2, and 3).

[17] Bradbury, M. H. and F. A. Sarott, "Sorption Databases for the Cementitious Near-Field of a L/ILW Repository for Performance Assessment." PSI Bericht 95-06. Würenlingen and Villigen, Switzerland: Paul Scherrer Institut. 117 pages, 1995.

[18] Sheppard, M. I. and D. H. Thibault, "Default soil solid/liquid partition coefficients, K_ds, for four major soil types: a compendium." Health Physics, Vol 59, pp 471-482, 1990.

[19] Idaho National Engineering and Environmental Laboratory, "INTEC Tank Farm Facility Closure Mockup Test Report Project File No. 015722," INEEL/EXT-2001-00299, March 2001.

[20] Wenzel, D.R., *Engineering Design File, Evaluation of Radionuclide Inventory for Sodium Bearing Waste*, EDR-FDO-006, Idaho National Engineering and Environmental Laboratory, November 26, 1997.

[21] U.S. Nuclear Regulatory Commission, Letter from T.H. Essig/NRC to J.T. Case/DOE-ID, "Request for Additional Information on the Idaho National Engineering and Environmental Laboratory Draft Waste Incidental to Reprocessing Determination for Tank Farm Facility Residuals (DOE/ID-10777)," June 5, 2002.

[22] Lockie, K.A., "Final Responses to the Request For Additional Information on the Idaho National Engineering and Environmental Laboratory Draft Waste Incidental to Reprocessing Determination for Tank Farm Facility Residuals," DOE-ID, December 4, 2002.

[23] U.S. Department of Energy, Idaho Operations Office, "Idaho Nuclear Technology and Engineering Center Tank Farm Facility Residuals--Waste-Incidental-to-Reprocessing Determination Report, Draft B," DOE/ID-10777, (and Associated Documents), December 2002c.

[24] Poloski, A., *Engineering Design File, Solids Characterization*, EDF-TST-001, Idaho National Engineering and Environmental Laboratory, September 19, 2000.

[25] U.S. Nuclear Regulatory Commission, 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," *Code of Federal Regulations*, Office of the Federal Register, January 1, 2001.

[26] U.S. Nuclear Regulatory Commission, 10 CFR Part 20, "Standards for Protection against Radiation," *Code of Federal Regulations*, Office of the Federal Register, January 1, 2001.

[27] Sullivan, T.M., *DUST-MS – Model Equations for Waste Form Leaching and Transport with Ingrowth Due to Radioactive Progeny*, Brookhaven National Laboratory, Environmental and Waste Technology Center, 1996.

[28] ACRi, *PORFLOW Users Manual*, Version 4.0, Rev. 4, Bel Air, California, 2000.

[29] NRC, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50," Regulatory Guide 1.109, Appendix I, Rev. 1, 1977.

[30] Maheras, S.J., et al., "Addendum to Radioactive Waste Management Complex Low-Level Waste Radiological Performance Assessment," INEEL/EXT-97-00462, April 1997.

[31] Peterson, H.T. Jr., "Terrestrial and Aquatic Food Chain Pathways," *Radiological Assessment - A Textbook on Environmental Dose Analysis*, J. E. Till and H. R. Meyer (eds.), NUREG/CR-3332, 1983.

[32] Sullivan, T.M., *DUST-MS - Disposal Unit Source Term - Multiple Species*, Informal Report BNL-EWMG, Brookhaven National Laboratory, 1998.

[33] Allard, B., G. Persson, and B. Torstenfelt, *Technical Reports 85-18, Actinide Solubilities and Speciation in a Repository Environment; 85-19, Organic Complexing Agents in Low- and Medium-Level Radioactive Waste; 85-20, Radionuclide Sorption on Carbonate-Clayish Rock; and 85-21, Radionuclide Sorption on Concrete*, NAGRA, Radiochemistry Consultant Group for Kematka Consultants Co., Stockholm, Sweden, 1985.

[34] Anderson, S.R., *Stratigraphy of the Unsaturated Zone and Uppermost Part of the Snake River Plain Aquifer at the Idaho Chemical Processing Plant and Test Reactor Area, Idaho National Engineering Laboratory, Idaho,* U.S. Geological Survey Water-Resources Investigations Report 91-4010, DOE/ID-22095, 1991.

[35] Rodriquez, R.R., et al., *Comprehensive RI/FS for the Idaho Chemical Processing Plant OU 3-13 at the INEEL - Part A, RI/BRA Report (Final)*, DOE/ID-10534, Binders 1-3, November 1997.

[36] Cecil, L.D., et al., *Formation of Perched Ground-water Zones and Concentrations of Selected Chemical Constituents in Water, Idaho National Engineering Laboratory, Idaho, 1986-1988*, U.S. Geological Survey Water-Resources Investigations Report 91-4166, DOE/ID-22100, 1991.

[37] LaPlante, P.A., and K. Poor, *Information and Analyses to Support Selection of Critical Groups and Reference Biospheres for Yucca Mountain Exposure Scenarios*, CNWRA 97-009, Center for Nuclear Waste Regulatory Analyses, Southwest Research Institute, San Antonio, Texas, 1997.

[38] U.S. Environmental Protection Agency, *External Exposure to Radionuclides in Air, Water, and Soil*, Federal Guidance Report No. 12, EPA-402-R-93-081, Office of Radiation and Indoor Air, Washington, DC, 1993.

[39] U.S. Environmental Protection Agency, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11, EPA-520/1-88-020, Office of Radiation Programs, Washington, DC, 1988.

[40] U.S. Nuclear Regulatory Commission, *A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities*, Recommendations of NRC's Performance Assessment Working Group, NUREG-1573, U.S. Nuclear Regulatory Commission, Washington, DC, October 2000.

[41] Kossik, R.F., Miller, I., and Knopf, S., "GoldSim Graphical Simulation Environment User's Guide," Version 7.4, Golder Associates Inc., Redmond, WA. 2001.

[42] Esh, D.W., A.H. Bradford, K.L. Banovac, and B.J. Davis, "Risks and Uncertainties Associated With High-Level Waste Tank Closure," *Proceedings Scientific Basis for Nuclear Waste Management XXVI*, Materials Research Society, Pittsburgh, PA, 2002.

[43] Jacobson, V. L., memo to K. Quigley, "Radiation Exposures from Tank WM-182 Cleaning," Idaho National Engineering and Environmental Laboratory, October 30, 2002.

LIST OF ABBREVIATIONS AND ACRONYMS

ALARA	As Low As Is Reasonably Achievable
Am	americium
Bq	becquerel
Bq/mL	becquerel/milliliter
C	carbon
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
Cs	cesium
DOE	U.S. Department of Energy
DOE-ID	U.S. Department of Energy, Idaho Operations Office
DUST-MS	Disposal Unit Source Term – Multiple Species
EPA	U.S. Environmental Protection Agency
HLW	high-level radioactive waste
I	iodine
ICRP	International Commission on Radiological Protection
in.	inches
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
K_d	distribution coefficient
L	liters
LLW	low-level radioactive waste
m	meters
m^3	cubic meters
mL/g	milliliters/gram
MOU	Memorandum of Understanding
mrem	millirem
mSv	millisievert
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
PA	performance assessment

pCi/mL	picocuries/milliliter
Pu	plutonium
RAI	request for additional information
Sr	strontium
SRM	Staff Requirements Memorandum
Sv	sievert
Tbq	terabecquerel
Tc	technetium
TEDE	Total Effective Dose Equivalent
TFA	Tanks Focus Area
TFF	Tank Farm Facility
TRU	transuranic
V	volts
WIR	waste-incidental-to-reprocessing
yr	year