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# Idaho Nuclear Technology and Engineering Center Sodium-Bearing Waste – Waste-Incidental-to-Reprocessing Determination Report

INEEL Idaho National Engineering and Environmental Laboratory
Home of Science and Engineering Solutions

DOE/ID-10780

# Idaho Nuclear Technology and Engineering Center Sodium-Bearing Waste — Waste-Incidental-to-Reprocessing Determination Report

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# ABSTRACT

U.S. Department of Energy Manual 435.1-1, *Radioactive Waste Management*, Section I.1.C, requires that all radioactive waste subject to Department of Energy Order 435.1 be managed as high-level radioactive waste, transuranic waste, or low-level radioactive waste. Determining the radiological classification of the sodium-bearing waste currently in the Idaho Nuclear Technology and Engineering Center Tank Farm Facility inventory is important to its proper treatment and disposition. This report presents the technical basis for making the determination that the sodium-bearing waste is waste incidental to spent fuel reprocessing and should be managed as mixed transuranic waste.

This report focuses on the radiological characteristics of the sodium-bearing waste. The report does not address characterization of the nonradiological, hazardous constituents of the waste in accordance with Resource Conservation and Recovery Act requirements.

# SUMMARY

U.S. Department of Energy Manual 435.1-1, *Radioactive Waste Management Manual*, Section I.1.C, requires that all radioactive waste subject to Department of Energy Order 435.1 be managed as either high-level radioactive waste, transuranic waste, or low-level radioactive waste. DOE M 435.1-1 also states that waste resulting from reprocessing spent nuclear fuel determined to be incidental to reprocessing is not high-level radioactive waste and shall be managed in accordance with the requirements for transuranic waste or low-level radioactive waste, as appropriate. The determination that spent nuclear fuel reprocessing wastes are wastes incidental to reprocessing and therefore not high-level radioactive waste is called a "waste-incidental-to-reprocessing determination."

Determining the radiological classification of sodium-bearing waste currently in the Idaho Nuclear Technology and Engineering Center Tank Farm Facility inventory is important to its proper management, treatment, and disposition. This report presents the technical basis for the determination that the sodium-bearing waste is waste incidental to spent fuel reprocessing and should be managed as mixed transuranic waste. The sodium-bearing waste is currently stored in the Tank Farm Facility 300,000-gallon below-grade tanks.

For this report, *sodium-bearing waste* is defined as liquids and solids from the following sources:

- Decontamination solutions from past spent fuel reprocessing maintenance activities
- Tank heel solids
- Liquid wastes from ongoing maintenance and closure activities at the Idaho Nuclear Technology and Engineering Center
- Remaining second- and third-cycle spent fuel reprocessing extraction wastes
- Trace contamination from first-cycle spent fuel reprocessing extraction waste.

The report presents the:

- Requirements for waste-incidental-to-reprocessing determinations applicable to sodium-bearing waste
- Documentation that sodium-bearing waste meets the waste-incidental-to-reprocessing criteria for transuranic waste identified in Department of Energy Manual 435.1-1.

Department of Energy Manual 435.1-1, *Radioactive Waste Management Manual*, Section II.B.2(b), lists three criteria that must be satisfied to demonstrate through a waste-incidental-to-reprocessing determination that spent nuclear fuel related wastes should be managed as transuranic waste. The Department of Energy has evaluated sodium-bearing waste against the criteria listed below, and,

for the reasons presented, has concluded that sodium-bearing waste meets these criteria and can be managed and disposed of as transuranic waste.

**Criterion 1.** The waste must have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical [DOE M 435.1-1, II(B)(2)(b)(1)].

- DOE M 435.1-1 provides flexibility for DOE to determine which radionuclides are important for meeting disposal-site performance objectives. Therefore, DOE uses a disposal site-specific risk-based approach for determining key radionuclides.

- The planned disposal location for INTEC sodium-bearing waste is the Waste Isolation Pilot Plant (WIPP) in New Mexico. The important radionuclides for WIPP performance objectives that account for most radionuclide release and therefore the most risk are Am-241, Pu-238, Pu-239, and Pu-240. These were evaluated as key radionuclides for meeting the SBW waste incidental to reprocessing (WIR) determination criterion 1 requirements. Total source from all radionuclides was evaluated and found to be within WIPP limits.

- The Idaho Nuclear Technology and Engineering Center segregated, removed, and converted the first-cycle extraction waste and most of the second- and third-cycle extraction waste (representing 96% of the key radionuclide curie inventory from reprocessing) to a stable solid waste form (calcine). It is planned that this solidified extraction waste will be further treated and disposed of as high-level radioactive waste.

- Additional key radionuclide removal from the remaining sodium-bearing waste would incur an additional cost between $373 million and $2.21 billion, depending upon the treatment process selected, to remove about 3,000 curies. It was determined that the large expenditure for this relatively small reduction in radionuclide release (risk) was not economically practical.

**Criterion 2.** The waste will be incorporated in a solid physical form and meet alternative requirements for waste classification and characteristics, as the Department of Energy may authorize [DOE M 435.1-1, II(B)(2)(b)(2)].

- The Department of Energy plans to remove, solidify, and dispose of the sodium-bearing waste remaining in the 300,000-gallon storage tanks as mixed transuranic waste at the WIPP geologic repository. The solidified waste would meet WIPP waste acceptance criteria.

**Criterion 3.** The waste is managed pursuant to Department of Energy's authority under the *Atomic Energy Act of 1954*, as amended, in accordance with the provisions of Chapter III of Department of Energy Manual 435.1-1, as appropriate [DOE M 435.1-1, II(B)(2)(b)(3)].

- The solidified sodium-bearing waste would meet the waste acceptance criteria for the WIPP geologic repository as contact-handled and/or remote-handled mixed transuranic waste. Solidified sodium-bearing waste would be managed and disposed of as transuranic waste in accordance with DOE M 435.1-1. WIPP is a permitted disposal site for contact-handled mixed transuranic wastes and is expected to be permitted for remote-handled waste by 2003, long before sodium-bearing waste is shipped from the Idaho Nuclear Technology and Engineering Center.

- Disposal of the sodium-bearing waste as a mixed transuranic waste in the WIPP geologic repository would provide public health and safety protection and meet the applicable environmental protection standard of 40 CFR Part 191.

Since sodium-bearing waste meets the above stated requirements as waste incidental to reprocessing, it should be managed, treated, and disposed of as transuranic waste. This document presents the detailed documentation that supports the waste-incidental-to-reprocessing determination and provides the basis for DOE Field Office approval.

**NRC Review and Conclusions**

As recommended in DOE G 435.1-1, the Nuclear Regulatory Commission has provided a technical review of this waste-incidental-to-reprocessing (WIR) determination document. The review occurred between September 2001 and August 2002. Because the SBW will be treated to meet WIPP repository requirements and disposed under DOE & EPA jurisdiction, the NRC did not review this WIR determination document for compliance with WIR Criteria 2 and 3, rather they focused on Criterion 1—the assessment of whether the waste has been processed, or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical. Accordingly, the NRC only provided comments and observations on the methodology for meeting Criteria 2 and 3 that were identified during the review.

Based on NRC's review (Greeves 2002) of the information provided by DOE-ID, NRC agreed that it is not technically practical to remove additional key radionuclides from the SBW solids prior to disposal. NRC agreed that even though the technology exists to remove additional key radionuclides from SBW liquid, it is not economically practical to do so. Therefore, the NRC agreed that the SBW has been process to remove key radionuclides to the maximum extent practical. NRC, in its role of providing technical assistance to DOE-ID and acting in an advisory capacity and not providing regulatory approval in this action, concluded that Criterion 1 has been met. The NRC recommendations (See Appendix C) have been incorporated into this document.

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# ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AK | acceptable knowledge |
| BBWI | Bechtel BWXT Idaho, LLC |
| CFR | Code of Federal Regulations |
| CH-TRU | contact-handled transuranic |
| Ci | curie |
| cm | centimeter |
| CNS | Chem-Nuclear Systems |
| CsIX | cesium ion exchange |
| CST | crystalline sillicotitanate |
| DD&D | decontamination, demolition & disposal |
| DMT | Decision Management Team |
| DOE | United States Department of Energy |
| DOE-CBFO | Department of Energy, Carlsbad Field Office |
| DOE-EM | Department of Energy, Office of Environmental Management |
| DOE-ID | Department of Energy, Idaho Operations Office |
| DOE-RL | Department of Energy, Richland Operations Office |
| DOT | Department of Transportation |
| dpm | disintegrations per minute |
| EIS | environmental impact statement |
| EPA | United States Environmental Protection Agency |
| FEIS | final environmental impact statement |
| FGE | Pu-239 fissile gram equivalent |
| G | Guidance |
| HAC | hypothetical accident conditions |
| HEPA | high-efficiency particulate air |
| HLW | high-level radioactive waste |
| hr | hour |
| IHLW&FD | Idaho High-Level Waste and Facilities Disposition |
| INEEL | Idaho National Engineering and Environmental Laboratory |
| INTEC | Idaho Nuclear Technology and Engineering Center |
| kg | kilogram |
| l | liter |
| lb | pound |
| LDUA | light-duty utility arm |
| LLW | low-level radioactive waste |
| M | Manual |
| M&O | management and operating |
| $m^3$ | cubic meter |
| MA&T | material accountability & tracking |
| MCP | management control procedure |
| ml | milliliter |
| mR | milli-Roentgen |
| mrem | millirem |
| nCi/g | nanocuries/gram |
| NGLW | newly generated liquid waste |
| NOPR | notice of proposed rulemaking |
| NRC | United States Nuclear Regulatory Commission |
| NWCF | New Waste Calcining Facility |
| O | Order |
| PA | performance assessment |
| PCB | polychlorinated biphenyl |
| PE-Ci | Pu-239 equivalent activity |
| PK | process knowledge |
| ppm | parts per million |
| QA | Quality Assurance |
| QAPRD | Quality Assurance Program Requirements Document |
| R/hr | roentgen per hour |

| | |
|---|---|
| RCRA | Resource Conservation and Recovery Act |
| rem | roentgen equivalent man |
| RH | remote-handled |
| ROD | record of decision |
| RRWAC | reusable property, recyclable materials waste acceptance criteria |
| SAR | safety analysis report |
| SBW | sodium-bearing waste |
| SNF | spent nuclear fuel |
| SSC | systems, structures, and components |
| TFF | Tank Farm Facility |
| TRU | transuranic |
| TRUCON | Transuranic Content Code |
| TRUEX | TRU extraction |
| TSR | technical safety requirements |
| UNEX | universal extraction |
| VE | visual examination |
| VOC | volatile organic compound |
| W | watt |
| WAC | waste acceptance criteria |
| WCF | Waste Calcining Facility |
| WCP | Waste Certification Program |
| WIPP | Waste Isolation Pilot Plant |
| WIR | waste incidental to reprocessing |
| WVDP | West Valley Demonstration Project |
| WWIS | WIPP Waste Information System |

# Idaho Nuclear Technology and Engineering Center Sodium-Bearing Waste — Waste-Incidental-to Reprocessing Determination Report

## 1. INTRODUCTION

U.S. Department of Energy Manual 435.1-1 (DOE 1999c), requires that all radioactive waste subject to DOE Order (O) 435.1 (DOE 1999b) be managed as either high-level radioactive waste (HLW), transuranic waste (TRU waste), or low-level radioactive waste (LLW).[a] DOE M 435.1-1 also states that waste resulting from reprocessing spent nuclear fuel that is determined to be incidental to reprocessing is not HLW, and shall be managed in accordance with the requirements for TRU waste or LLW, as appropriate. The determination that spent nuclear fuel reprocessing wastes are wastes incidental to reprocessing (WIR), and, therefore, not HLW, is called a waste-incidental-to-reprocessing or WIR determination.

Determining the radiological classification of waste currently in the Idaho Nuclear Technology and Engineering Center (INTEC) Tank Farm inventory is required for proper waste treatment and disposition. This WIR determination report presents the technical basis for determining that the sodium-bearing waste (SBW) is incidental to reprocessing and should be managed and disposed of as TRU waste.

The report demonstrates that (a) the majority of key radionuclides have been removed from waste stored in the Tank Farm, (b) it is not economically practical to remove additional key radionuclides from the SBW that remains, and (c) that the SBW can be put in final waste forms acceptable for TRU waste disposal. The SBW will be removed from the Tank Farm and treated to solidify and stabilize the waste for disposal at the Waste Isolation Pilot Plant (WIPP) geologic repository.

For this report, *sodium-bearing waste* is defined as the liquids and solids in the Tank Farm from the following sources:

- Decontamination solutions from past spent fuel reprocessing maintenance activities
- Tank heel solids
- Liquid wastes from ongoing maintenance and closure activities at INTEC
- Remaining second- and third-cycle spent fuel reprocessing extraction wastes
- Trace contamination from first-cycle spent fuel reprocessing extraction waste.

SBW is a RCRA mixed waste and has been assigned the following characteristic hazardous wastes codes D002, D004, D005, D006, D007, D008, D009, D010, and D011. In addition, past waste management practices have resulted in assigning the following RCRA listed waste codes: F001, F002, F005, and U134[b] (LIMTCO 1999). All SBW treatment products are considered mixed radioactive

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a. See DOE M 435.1-1 for definitions of HLW, TRU waste, and LLW.

b. The current application for a class 3 modification to the WIPP Hazardous Waste Permit contains the U-134 code. All other codes are currently acceptable at WIPP.

hazardous waste due to the mixture and derived form rules for hazardous waste. In this WIR determination, all HLW, TRU waste, and LLW are assumed to be mixed waste. Because a WIR determination addresses the radiological classification of a waste, characterization of the nonradiological, hazardous constituents of the SBW in accordance with Resource Conservation and Recovery Act (RCRA) requirements is deferred to other documentation.

## 1.1 Purpose

This report demonstrates that sodium-bearing waste (SBW) is waste incidental to reprocessing and should be managed and disposed of as TRU waste. The report summarizes current WIR determination requirements and guidance and its application to SBW currently stored in the Idaho Nuclear Technology and Engineering Center Tank Farm Facility. The requirements and guidance are contained in DOE O 435.1, *Radioactive Waste Management* and its accompanying manual and guidance document, present a basis for classifying SBW as HLW, TRU waste, or LLW. This report presents:

- The requirements for waste incidental to reprocessing determinations that are applicable to SBW
- The documentation that SBW meets the waste incidental to reprocessing criteria identified in DOE Manual 435.1-1
- The appropriate radiological classification of SBW.

## 1.2 Background

The Idaho National Engineering and Environmental Laboratory (INEEL) is an approximately 890-square mile reservation owned by the United States Government and located in Eastern Idaho (see Figure 1-1). First established nearly 50 years ago as the National Reactor Testing Station, the INEEL's initial mission was to develop civilian and defense nuclear reactor technologies. Over the years, the INEEL mission evolved beyond the original focus, and the INEEL is currently involved in various environmental, defense, energy supply, and industrial technology programs. In recognition of this evolution to a multiprogram installation, the site was designated the Idaho National Engineering Laboratory in 1974. In January 1997, the name was changed to the Idaho National Engineering and Environmental Laboratory to reflect greater emphasis on the laboratory's environmental missions.

In 1953, the Idaho Chemical Processing Plant, now called the Idaho Nuclear Technology and Engineering Center (INTEC), was chartered to recover fissile uranium by reprocessing spent nuclear fuel (SNF). In 1992, the DOE officially discontinued reprocessing SNF at INTEC. This decision changed the mission of INTEC to management and storage of SNF, and treatment and storage of reprocessing wastes generated from past and current operations and activities. The Tank Farm, located within INTEC (see Figure 1-3), consists of 11 nominal 300,000-gallon belowgrade stainless steel tanks in unlined concrete vaults of various construction, and other smaller tanks, interconnecting waste transfer lines, and associated support instrumentation and valves. The smaller tanks include four inactive 30,000-gallon stainless steel tanks. Waste from SNF reprocessing, including first-cycle extraction waste and SBW, were stored in the Tank Farm Facility. The first-cycle extraction waste was removed from the tanks to heel[c] level, and the tanks were then used to store additional SBW. The 30,000 gallon tanks have been cleaned and deactivated and are no longer used for storage.

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c. *Tank Heel* means the liquid remaining in each tank after lowering to the greatest extent possible by use of existing transfer equipment, such as steam-jet ejectors.

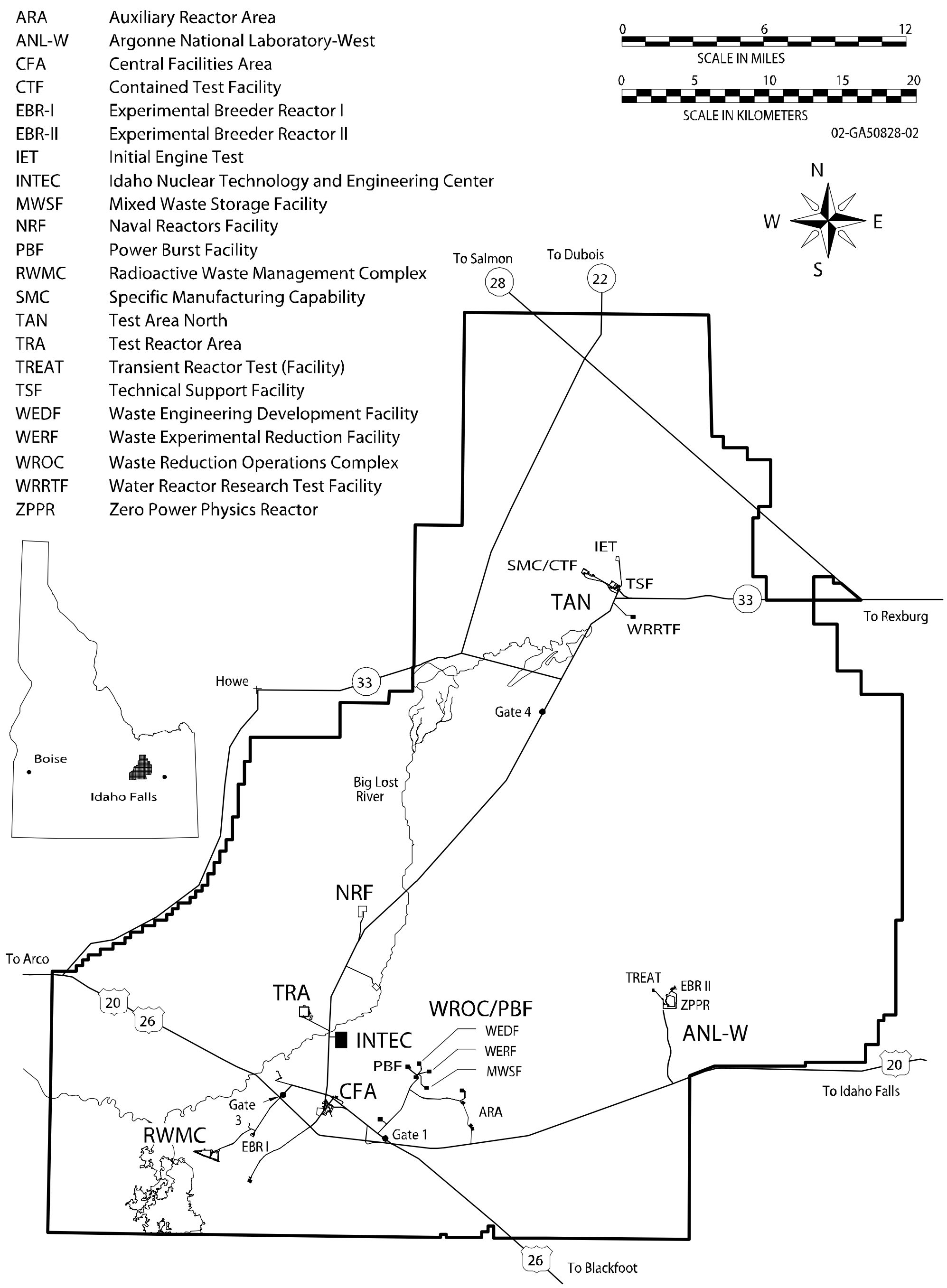


Figure 1-1. Idaho National Engineering and Environmental Laboratory site map.

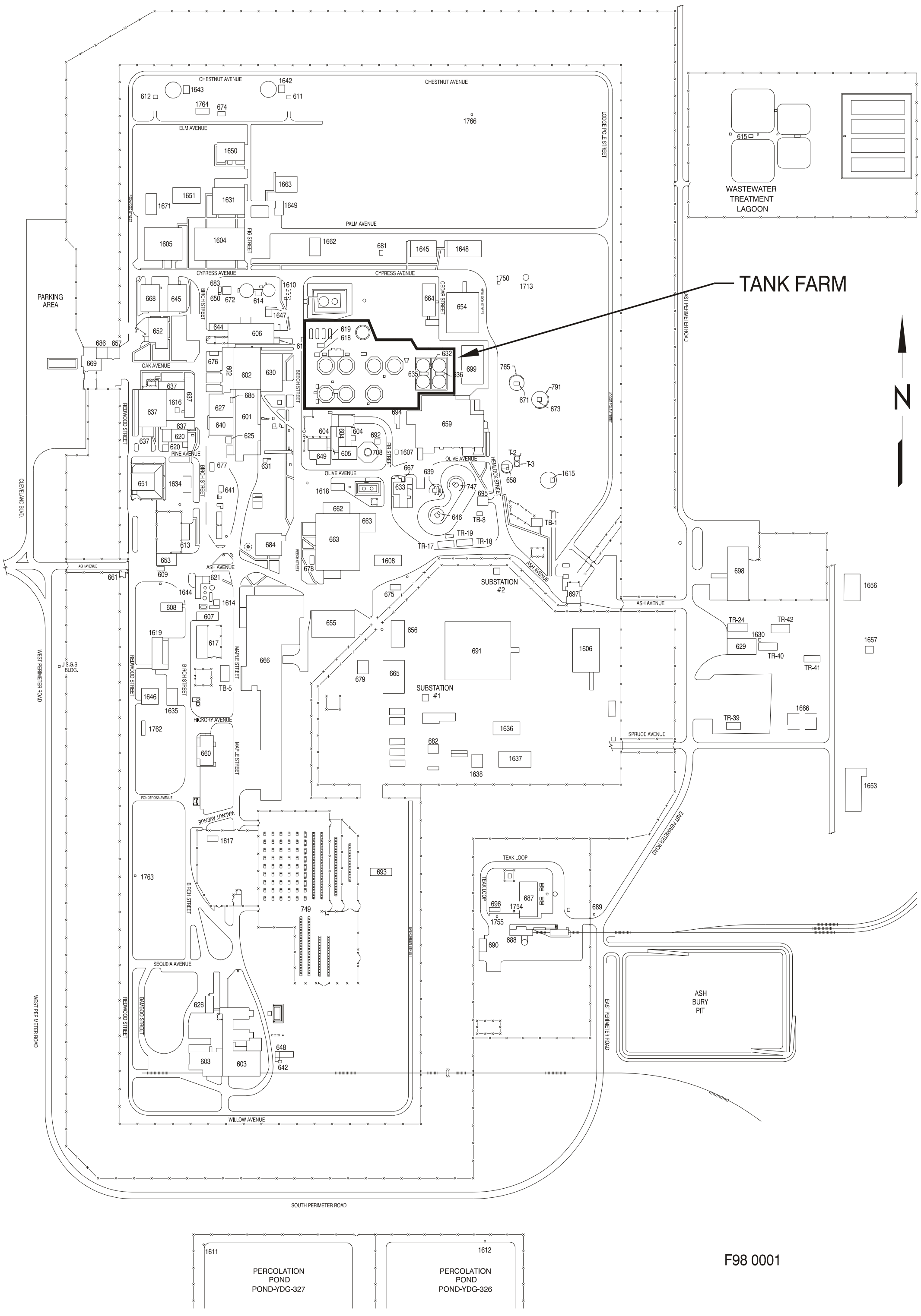


Figure 1-2. INTEC area plot plan (not to scale).

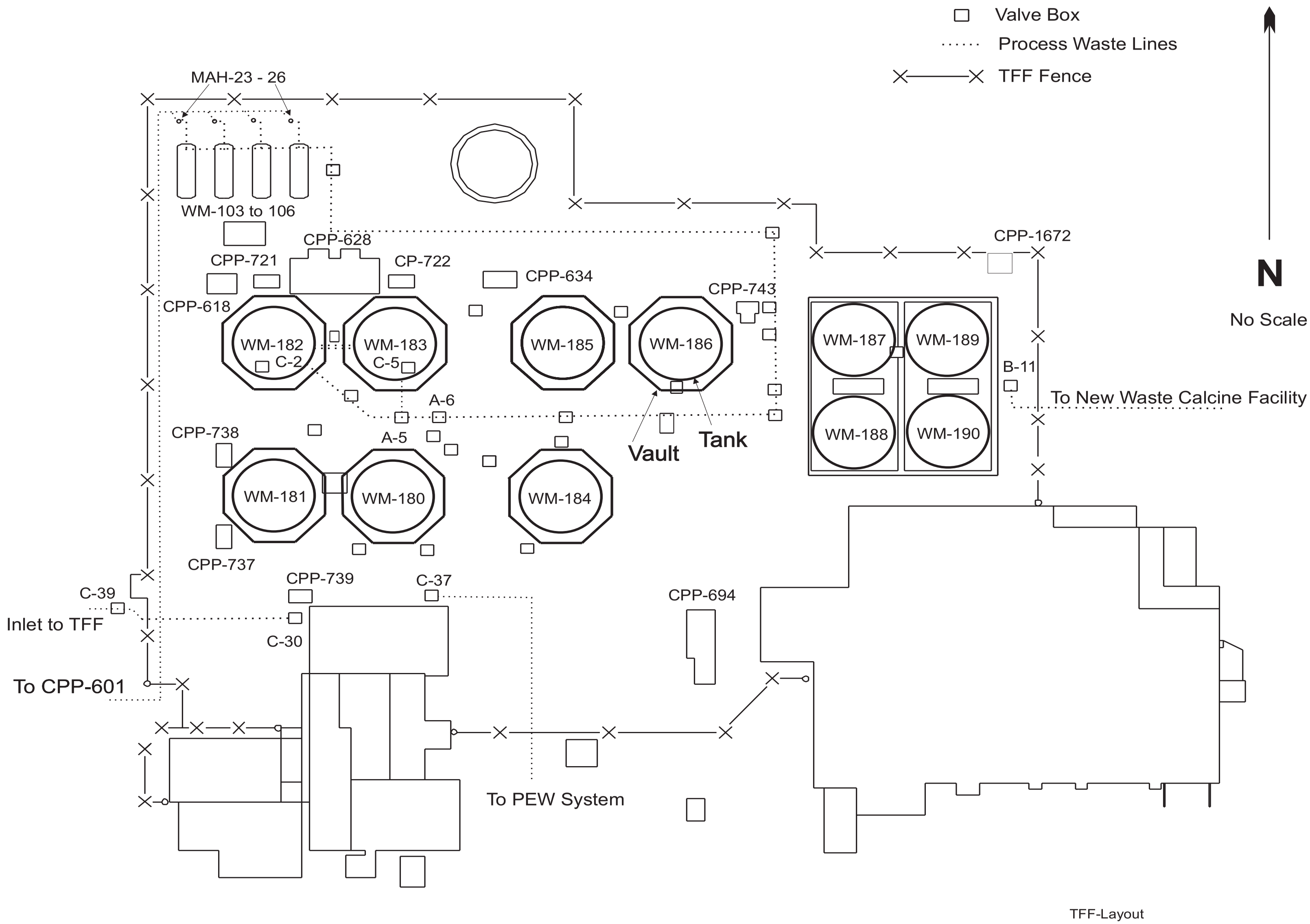


Figure 1-3. Plan view of Tank Farm.

In January 1990, the Idaho Department of Health and Welfare and the U.S. Environmental Protection Agency issued a notice of noncompliance because the large 300,000-gallon liquid waste storage tanks did not meet the secondary containment requirements of RCRA. The Consent Order and subsequent modifications that followed from the notice of noncompliance require INEEL to either upgrade the tank system or permanently cease use[d] of the five 300,000-gallon tanks contained in pillar and panel vaults by June 30, 2003, and to permanently cease use of the remaining 300,000-gallon tanks by December 31, 2012.

Once the Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement (IDAHO HLW & FD FEIS, Reference DOE 2002a) is issued and this WIR determination is approved, a final waste management strategy is expected to be established in a Record of Decision. SBW will be removed from the Tank Farm tanks, treated, and the treatment products will be stored and ultimately disposed of.

The DOE Idaho Operations Office (DOE-ID), through the Idaho HLW & FD FEIS, assessed five SBW treatment alternatives in addition to the No-Action Alternative (DOE 2002a):

1. Continued Current Operations Alternative. SBW would be calcined and added to the bin sets where calcined HLW is stored.

2. Separations Alternative. Several options where SBW would be chemically separated into fractions that can be disposed of differently, depending on the type and level of radioactivity.

3. Nonseparations Alternative. Several Options where SBW would be immobilized for disposal without further separating waste fractions.

4. Direct Vitrification of Sodium Bearing Waste and Vitrification of Calcine With or Without Separations Alternative. SBW would be vitrified directly for disposal without separations and calcine would be vitrified with or without separations depending upon conclusions from future evaluations.

5. Minimum INEEL Processing Alternative. Calcined HLW would be sent to the Hanford site in Washington State for treatment, and SBW would be treated at the INEEL.

It is currently planned the Idaho HLW & FD FEIS will identify as the preferred alternative direct stabilization (solidification) of the SBW, with disposal at WIPP in New Mexico. WIPP disposal as a transuranic waste is planned because the expected concentrations of TRU isotopes in the final product will exceed 100 nCi/g. As described in this WIR determination, direct stabilization alternatives are alternatives that do not remove additional key radionuclides. (For a complete description of SBW treatment alternatives that have been considered, refer to Appendix B.)

## 1.3 Waste Incidental to Reprocessing Requirements

DOE O 435.1, its manual, and implementing guidance state that waste determined to be incidental to reprocessing must be managed under DOE regulatory authority in accordance with the requirements for TRU waste or LLW, as appropriate. These DOE documents present requirements and criteria that must

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d. *Cease use* means to empty the tanks down to their heels, i.e., the liquid level remaining in each tank after lowering to the greatest extent possible by use of existing transfer equipment. Closure plans developed for these tanks will address the remaining heel and vaults, the use of these tanks and equipment for closure including any flushing or other cleaning of the tanks (Second Modification to Consent Order, July 31, 1998).

be satisfied when making WIR determinations. This section of the report discusses these requirements and criteria.

The U.S. Nuclear Regulatory Commission (NRC) established the concept of incidental waste (AEC 1969). The criteria that must be met for the WIR determination are based on NRC correspondence and adjudication relating to HLW definition and regulatory determinations for managing wastes derived from spent fuel reprocessing at other DOE sites. The NRC staff (Bernero 1989) concurred with the methodology proposed by DOE for determining that the Hanford low-activity waste fraction resulting from removal and separation of the inventory of reprocessing wastes stored in underground tanks would be waste incidental to reprocessing (i.e., not HLW) (actions summarized in NRC 1997). In its denial of petition for rulemaking brought by the States of Washington and Oregon (58 FR 12342), the NRC agreed that the residual fraction at Hanford would be incidental waste if three criteria were satisfied (NRC 1993). These three criteria have been incorporated into the DOE WIR determination criteria (DOE 1999c).

A WIR determination is intended to support the proper management, treatment, and disposal of wastes such as SBW and the closure of deactivated HLW facilities such the INTEC Tank Farm. When determining whether SNF reprocessing-plant wastes must be managed as another waste type or as HLW, DOE M 435.1-1, Chapter II (DOE 1999c), states that either (a) the citation or (b) the evaluation process must be used. Because SBW is not consistent with waste types listed for the citation process (DOE M 435.1-1, II, B, 1), the evaluation process is used for the INTEC SBW. Depending on the concentrations of radionuclides in the SBW, DOE O 435.1 allows the final waste to be classified and managed as either TRU waste or LLW if the associated criteria can be satisfied. Additionally, WIR determinations using the evaluation process must be developed using good record-keeping practices, with an adequate quality assurance process, and must be documented to support the determinations.[e] The WIR evaluation criteria for the TRU waste classification are discussed below.

In accordance with DOE M 435.1-1, Section II.B.2(b), wastes to be managed and disposed of as TRU waste must satisfy the following three criteria.

**Criterion 1.** The waste must have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical

**Criterion 2.** The waste will be incorporated in a solid physical form and meet alternative requirements for waste classification and characteristics, as DOE may authorize

**Criterion 3.** The waste is managed pursuant to DOE's authority under the *Atomic Energy Act of 1954* (AEA 1954), as amended, in accordance with the provisions of Chapter III of DOE M 435.1, as appropriate.

## 1.4 INEEL WIR Determinations

The SBW treatment and Tank Farm Facility (TFF) closure involves removing waste from the tanks, treating the waste for disposal, treating and disposing of equipment and materials removed from the TFF, and closing the TFF; including residual waste, tanks, and ancillary equipment that will be stabilized *in situ*. These activities result in various radioactive waste streams that must be classified according to the waste type (i.e., HLW, TRU waste, or LLW). It is currently envisioned that three evaluation WIR determinations will be necessary to support the SBW treatment and Tank Farm closure activities. The

e. DOE 1999d, Chapter II, B.2

final remnants of SBW for offsite disposal will be removed during tank cleaning and closure activities. Additional WIR determinations may be required as INTEC closes other HLW facilities.

1. *SBW WIR Determination Report* (this report). The SBW WIR determination report covers the application of the WIR evaluation process for the SBW currently stored in the TFF 300,000-gallon tanks. These wastes will be removed from the tanks for treatment and disposed as proposed in the Idaho HLW & FD FEIS. The planned treatment for this waste stream is direct stabilization (solidification) and disposal as TRU waste at WIPP. (Direct stabilization processes may generate some secondary waste streams that could be disposed in near-surface LLW disposal sites.)

2. *TFF Residuals WIR Determination Report*. (DOE 2002b) This WIR determination covers the remaining waste residuals, the tanks, vaults, sandpads beneath the tanks, and associated ancillary piping and other systems, structures, and components that will be stabilized *in situ* and meet LLW WIR criteria. The closure of the TFF will be in accordance with the *Idaho Nuclear Technology and Engineering Center Tank Farm Facility Conceptual DOE and HWMA/RCRA Closure Approach* (INEEL 2000a).

3. *Contaminated Equipment and Materials WIR Determination Report*. This WIR determination will cover contaminated equipment and materials removed from INTEC HLW facilities for disposal. The determination will be prepared for the miscellaneous equipment and other related materials that are potentially contaminated by HLW reprocessing streams and have been removed from service.

DOE O 435.1 and accompanying manual and guidance allow using the disposal-site performance objectives to determine key radionuclides. Each INTEC evaluation-WIR determination will use disposal-site specific performance objectives to determine key radionuclides, each will document the technical and economical practicality of removing additional key radionuclides, and each will evaluate proper waste management and disposal.

## 1.5 Report Organization

This report has six chapters. Chapter 1 presents the report's purpose and background information. It explains what a WIR determination is, its basis, and the criteria that must be satisfied for a successful WIR determination. Chapter 2 presents information on the source of SBW and the radionuclide concentrations in the waste. Chapter 3 presents the technical basis for the classification of SBW as TRU waste. Chapter 4 describes the management controls applicable to performing a WIR determination. Chapter 5 presents the report conclusions and summarizes the basis for approving the WIR determination. Chapter 6 lists the references cited in the report. Additional information important to understanding the WIR determination is presented in the appendices.

# 2. WASTE SOURCE DESCRIPTION AND WASTE CHARACTERIZATION

This section discusses the source of the INTEC sodium-bearing waste (SBW) and the current radionuclide profile of the waste. Descriptions of the proposed waste stabilization process are also presented. Because the WIR process focuses on the radiological properties of the waste, the discussion in this chapter is restricted to the radiological characteristics of the SBW. Characterization of the nonradiological, hazardous constituents of the SBW and treated waste streams in accordance with RCRA requirements is deferred to other documentation.

## 2.1 Description of Spent Fuel Reprocessing Wastes

At INTEC, spent fuel was dissolved using various processes, depending on the fuel type (Figure 2-1). Each of the dissolution processes produced an acidic aqueous solution. The solution was processed through the first-cycle extraction system (Phase I in Figure 2-1) with an organic solvent (usually tributyl phosphate in kerosene). The extraction systems used several contactors, including pulsed-plate columns. The uranium was partitioned from the bulk of the fission products and placed in inter-cycle storage to await purification. The aqueous-waste phase from first-cycle extraction contained greater than 99% of the waste radionuclides. Radiation levels in first-cycle extraction wastes were generally 5 to 200 R/hr on contact for 5-ml containers (Wagner 2000). INTEC first-cycle extraction waste was stored in the Tank Farm Facility in below-grade tanks equipped with cooling coils. A small quantity of residual first-cycle waste accounts for about 3 % (by volume) of the SBW inventory (Staiger 2000).

As is typical of liquid-liquid extractions, unwanted radionuclides were carried over with the separated uranium. After sufficient product accumulated in the inter-cycle storage, the uranium was processed through the second- and third-cycle extractions (Phase II in Figure 2-1), where the bulk of the remaining unwanted radionuclides were removed to produce a *clean* uranium product. The uranium was usually shipped to the Oak Ridge National Laboratory in Tennessee for further processing.

The radionuclide curie content in the aqueous intermediate liquid waste in Figure 2-1 from second- and third-cycle extraction was approximately 1% of the initial reprocessing curie-inventory of radionuclides. Radiation levels in second-cycle extraction waste ranged from 20 to 275 mR/hr on contact for 15-ml containers (Wagner 2000). Radiation levels in third-cycle extraction waste were 1 to 5 mR/hr on contact for 15-ml containers (Wagner 2000). Second and third-cycle extraction product accounts for about 17% (by volume) of the SBW inventory (Staiger 1999).

The INTEC reprocessing equipment was designed as *contact maintenance* rather than *remote maintenance*. This means that INTEC personnel had to access the equipment for maintenance. Consequently, the reprocessing equipment was decontaminated frequently, generating large amounts of decontamination waste. At INTEC, the decontamination waste was sent to the Tank Farm and accounts for about 80% (by volume) of the SBW inventory.

## 2.2 Management of Spent Fuel Reprocessing Wastes

In 1988 advice to the DOE Richland Operations Office (Bell 1988), the NRC stated that "… if DOE could demonstrate that the largest practical amount of the total site activity attributable to 'first-cycle solvent extraction' wastes has been segregated for disposal as HLW, then NRC would view the residual as non-HLW."

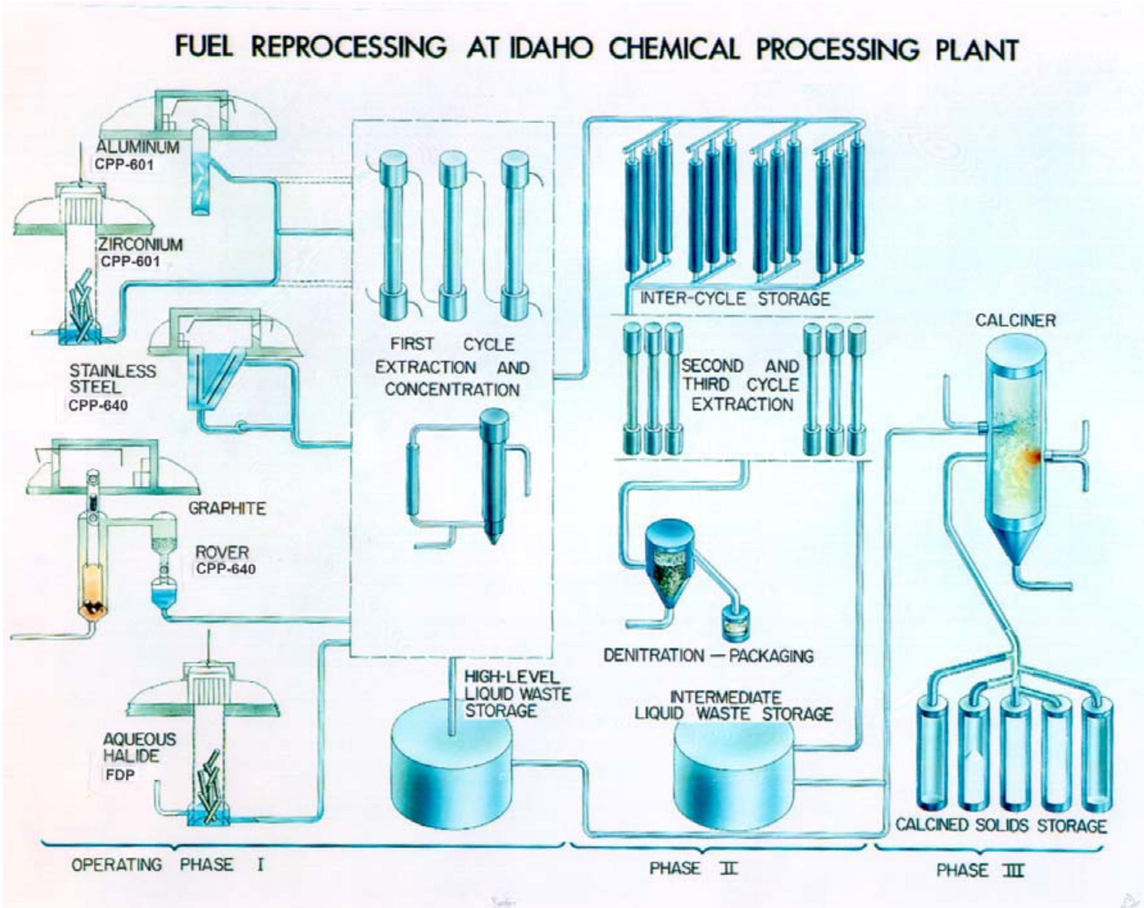


Figure 2-1. Spent fuel reprocessing at INTEC.

INTEC's management of SNF reprocessing wastes segregated the "first-cycle solvent extraction" waste from other reprocessing wastes. As described by Knecht et al. (1997), DOE decided not to neutralize waste or combine the first-cycle extraction waste with other reprocessing wastes, as was the standard practice at Hanford and the Savannah River Site. Instead, INTEC maintained the waste in its original acidic form, stored the waste in stainless steel storage tanks, and physically segregated first-cycle extraction wastes.

Use of stainless steel storage tanks allowed the reprocessing wastes to be managed in its acidic form, reducing the volume of waste. INTEC further reduced waste volume by stabilizing the first-cycle solvent extraction waste and most of the second- and third-cycle extraction waste in a solid form through calcination, achieving a seven-fold volume reduction for this radioactive waste. By evaporating and calcining, and not neutralizing the liquid radioactive waste, INTEC avoided construction of up to 195 additional 300,000-gallon storage tanks (Knecht et al. 1997). The solidified (calcined) extraction waste is currently stored in stainless steel bins with extended design lives. As of March 1998, the last of the liquid first-cycle extraction waste (high-level waste) and most second and third-cycle extraction waste were removed from the Tank Farm tanks. When tanks were emptied, a small (1,000 to 15,000 gallons) heel remained in each tank that the liquid transfer equipment could not remove. Eight of the eleven tanks contained first-cycle extraction waste; they were subsequently used for SBW storage and these small heels were intermingled with SBW. First cycle waste currently represents about 3% (by volume) of the liquid waste.

Calcination began in 1963 when the INEEL started the Waste Calcining Facility. The Waste Calcining Facility operated for 18 years; then was replaced with the New Waste Calcining Facility in 1982.

Beginning in 1978, SBW was blended with high-level waste and processed through the calciner to help reduce the SBW inventory in the Tank Farm. Blending SBW was successful and allowed room for additional SBW storage. SBW continues to be generated from decontamination of reprocessing cells, evaporator equipment, decontamination of miscellaneous equipment, and general decontamination and decommissioning activities. Calcining SBW was effective as long as nonsodium-bearing HLW was available to dilute the high sodium salts in the SBW. The blending of SBW continued until March 1998, when the last of the nonsodium-bearing HLW was calcined. To continue calcining SBW required adding cold chemicals to prevent agglomeration in the calciner vessel. At least three gallons of cold chemicals were added to each gallon of SBW.[f] The resulting mixture allowed SBW to be calcined; however, at a relatively high cost and large increase in the solid radioactive waste volume. The volume of the calcined SBW was still significantly less than the original volume of the liquid SBW.

The Notice of Non-compliance Consent Order requires that the calciner be permitted under RCRA or be placed in standby. Due to the lack of a hazardous-waste permit, the NWCF was placed in standby in May 2000 awaiting EIS analysis of other treatment options, and no SBW has been calcined since. As described in the Idaho HLW & FD FEIS, calcination is one of the treatment alternatives for SBW and DOE is assessing the potential impacts of upgrading, permitting, and operating the calciner.

## 2.3 Characterization of Waste Streams

SBW currently remains stored in the 300,000-gallon stainless steel tanks. The current inventory of SBW is a mixture that includes waste from various sources, including:

- Decontamination solutions from past spent fuel reprocessing maintenance activities
- Tank heel solids
- Liquid wastes from ongoing maintenance and closure activities at INTEC
- Remaining second- and third-cycle spent fuel reprocessing extraction wastes
- Trace contamination from first-cycle spent fuel reprocessing extraction waste.

The July 1999 SBW inventory of 1.373 million gallons was used to calculate the radionuclide data provided in this report (Kimmitt 2002). Between July 1999 and May 2000 when the calciner was placed in standby, an additional 90,000 gallons were calcined. Between May 2000 and when SBW treatment is completed, additional liquid waste volumes from ongoing INTEC maintenance and closure activities are projected to be about 200,000 gallons at reduced radionuclide concentrations (INEEL 2000b). Therefore radionuclide additions to the Tank Farm would be minor and the data provided in this report should conservatively estimate the radionculide content of the SBW to be treated and disposed of at WIPP. WIPP disposal as a transuranic waste is planned because the expected concentrations of TRU isotopes in the final product will exceed 100 nCi/g. As described, the primary sources of SBW liquid waste are decontamination solutions, second and third-cycle waste, and the small quantity of commingled first-cycle waste. Decontamination solutions are high in sodium salts and because SBW is mostly

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f. Subsequent testing at higher operating temperatures allowed the ratio to be reduced from 3:1 to 1.5:1.

decontamination solutions, it is also high in sodium salts. SBW has been maintained as a >2 molar nitric acid solution.

The sources and quantity of tank solids are estimated from process history, recent tank-heel sampling, and in-tank video inspections. Since only three of eleven tanks have been inspected and sampled (WM-182, WM-183, and WM-188), the estimates are preliminary and will be updated as additional tanks are inspected. Of the tanks that were sampled and inspected, one contained only trace quantities of solids, while the other two contained several inches of flocculent solids on the tank bottom (Poloski and Wilcox 2000 and Patterson 1999). Based upon tank filling history and comparison of inspected tanks with yet-to-be inspected tanks, solids quantities and radiological compositions have been conservatively estimated and should be bounding (Tyson 2002). The quantity of solids sludge from all tanks(25% solids and 75% liquid by volume) is estimated to be about 45,000 gallons containing about 86,000 kg of solids (Poloski and Wilcox 2000). The estimated radionuclide content of the solids is shown in Table 2-1. No definitive studies using actual spent-fuel reprocessing waste have been accomplished to establish the sources of solids. It is estimated that solids result from incomplete fuel dissolution, chemical precipitation and, decontamination activities. Precipitated solids probably resulted from transient conditions in liquids stored in the tanks. A study was completed in 1967 that assessed blending aluminum and zirconium extraction wastes (Newby and Hoffman 1967). A range of stable concentrations was determined and plant practice was to stay within the prescribed range for mixing raffinates. It is likely that transient conditions (outside the prescribed range) existed during the initial stages of changeover from aluminum to zirconium waste storage in a tank and other times when different solutions were added. These transient conditions are likely causes of precipitated-solids formation in tanks WM-182 and WM-183 and probably in other tanks.

### 2.3.1 Radionuclides of Interest in SBW Liquid and Solids

The first criterion for an evaluation WIR determination from DOE M 435.1-1 states that key radionuclides be removed from the waste to the maximum extent that is technically and economically practical. Therefore the first step for waste radionuclide characterization is to identify the key radionuclides. Neither DOE O 435.1, nor the accompanying manual, specifically identifies key radionuclides. However, DOE G 435.1-1 suggests that for a low-level-waste evaluation WIR determination that certain radionuclides (radionuclides of interest) should be considered. Specifically, the groups suggested (see Table 2-1) are:

1. Radionuclides controlled by 10 CFR §61.55 for near-surface land disposal

2. Radionuclides important to satisfying performance objectives of 10 CFR Part 61, Subpart C for land disposal of radioactive waste

3. Other radionuclides DOE has found important for satisfying disposal site performance objectives.

It is clear that DOE O 435.1 allows identification of key radionuclides as radionuclides that are important to the disposal site performance objectives. The guide suggests following similar methodology for transuranic waste and using alternative requirements (see 10 CFR §61.58) for performance objectives.

For WIPP TRU waste disposal, performance objectives similar to those for LLW disposal are found in 40 CFR Part 191. Both 10 CFR Part 61 and 40 CFR Part 191 provide requirements for disposal-site performance objectives. Compliance with performance objectives is documented though a performance assessment (PA) of potential radionuclide releases to the public. The WIPP Compliance Certification Application (DOE 1996b) and supporting PA identify four radionuclides (Am-241, Pu-238, Pu-239, and Pu-240) that contribute to releases from the WIPP repository (Hadgu 2001). These four

Table 2-1. Radionuclides of interest.

| Radionuclide | Long-Term Radiation Hazards[a] | Short-Term Radiation Hazards[b] | Additional DOE Radionuclides[c] | Important to WIPP PA[d] | SBW Liquid (Ci/l)[e] | SBW Solids (Ci/Kg) (dried basis)[e] |
|---|---|---|---|---|---|---|
| *Am-241 | X | | | X | 5.39E-05 | 7.47E-04 |
| Am-242 | X | | | | 1.10E-08 | 1.74E-07 |
| Am-243 | X | | | | 1.56E-08 | 2.47E-07 |
| C-14 | X | | | | 8.66E-11 | 1.37E-09 |
| Cf-249 | X | | | | 1.27E-17 | 2.01E-16 |
| Cf-250 | X | | | | 1.21E-17 | 1.92E-16 |
| Cf-251 | X | | | | 1.96E-19 | 3.11E-18 |
| Cm-242 | X | | | | 9.08E-09 | 1.44E-07 |
| Cm-243 | X | | | | 2.19E-08 | 3.47E-07 |
| Cm-244 | X | | | | 1.44E-06 | 2.29E-05 |
| Cm-245 | X | | | | 2.14E-10 | 3.38E-09 |
| Cm-246 | X | | | | 1.39E-11 | 2.19E-10 |
| Cm-247 | X | | | | 1.56E-17 | 2.47E-16 |
| Cm-248 | X | | | | 1.67E-17 | 2.65E-16 |
| Co-60 | | X | | | 4.83E-05 | 4.25E-04 |
| Cs-137 | | X | | | 3.91E-02 | 6.18E-01 |
| H-3 | | X | | | 1.70E-05 | 4.18E-05 |
| I-129 | X | | | | 4.77E-08 | 6.04E-07 |
| Nb-94 | | | X | | 8.09E-07 | 3.68E-03 |
| Ni-59 | | | X | | 2.73E-06 | 4.33E-05 |
| Ni-63 | | X | | | 3.22E-05 | 3.66E-04 |
| Np-237 | X | | | | 2.00E-06 | 1.75E-06 |
| *Pu-238 | X | | | X | 4.62E-04 | 1.54E-02 |
| *Pu-239 | X | | | X | 5.34E-05 | 1.38E-03 |
| *Pu-240 | X | | | X | 7.19E-06 | 1.19E-04 |
| Pu-241 | X | | | | 2.88E-04 | 1.22E-02 |
| Pu-242 | X | | | | 1.22E-08 | 8.96E-08 |
| Pu-244 | X | | | | 4.85E-16 | 7.68E-15 |
| Se-79 | | | X | | 3.12E-07 | 4.94E-06 |
| Sn-126 | | | X | | 2.95E-07 | 4.66E-06 |
| Sr-90 | | X | | | 3.43E-02 | 4.29E-01 |
| Tc-99 | X | | | | 8.12E-06 | 2.11E-03 |

* Key radionuclide
a. 10 CFR §61.55, Table 1.
b. 10 CFR §61.55, Table 2.
c. DOE G 435.1-1, page II-22
d. Hadgu 2001.
e. Kimmitt 2002, Decayed to July 1999

radionuclides are present in SBW and are considered as the key radionuclides for the SBW WIR determination. Table 2-1 lists the radionuclides of interest from DOE G 435.1-1, Section II, and shows that the four SBW key radionuclides are part of that list.

### 2.3.2 Radionuclide Concentrations of SBW Liquid and SBW Solids Feed Streams

The ability of treated SBW to meet the waste acceptance criteria for WIPP is a function of the radionuclide concentrations in the solidified final waste forms. The radionuclide concentrations in the SBW liquid and SBW solids feed streams are important to calculate the final waste stream characteristics. Consequently, Table 2-1 presents the radionuclide concentrations of both SBW liquid and SBW solids feed streams. Although some direct-stabilization options may not require it, for conservatism the SBW liquid and SBW solids are assumed to be treated separately for all options. This will bound the conditions in the actual treatment options. In Chapter 3, solidified waste product from each treatment option based on feed streams of 100% liquid or 100% solids (dry basis, no occluded water) are evaluated against the WIPP waste acceptance criteria. See Table 3-6.

The radionuclide concentrations of SBW are based on modeling augmented by process knowledge and sampling and analysis. The INTEC Tank Farm tanks were sampled between 1980 and 1994, and a report summarizing these results was prepared in 1994. In 1997, the initial analytical data were supplemented with results from ORIGEN2 modeling to produce a radionuclide inventory of the INTEC Tank Farm tanks. The data were again subjected to the ORIGEN2 model in 1998 to provide an elemental inventory of the SBW tanks. Finally, tanks WM-182, WM-183, and WM-188 were sampled in 1999. Using these analytical and modeling data, INTEC personnel prepared an estimate of current SBW radionuclide concentrations. (See Appendix A for report references and a summary of this process.) This estimate is reflected in Table 2-1.

Much work has been accomplished to define SBW liquid and solid radionuclide concentrations; however, limited information is available for some radionuclides. This uncertainty will be reduced in the future by taking additional samples. The results will be compared with waste characteristics in the WIR determination to ascertain the impact and verify that the WIR-determination conclusions are still valid.

## 2.4 Characterization of Final Waste Forms

This report demonstrates that SBW is waste incidental to reprocessing and can be solidified for TRU waste disposal at the WIPP geologic repository. This approach is consistent with the Idaho HLW & FD FEIS waste management and treatment alternatives (see Appendix B). Several methods can be used to directly treat (solidify) SBW, including evaporation to a granular-solid, incorporation into a grout, or vitrification to a glass. In order to demonstrate that SBW is waste incidental to reprocessing, this report also considers the technical and economical practicality of additional key-radionuclide removal through chemical separation processes. The two separation processes found to be technically practical were Universal Solvent Extraction (UNEX) and Transuranic Solvent Extraction (TRUEX). Section 3 of this report discusses the technical and economical evaluation of these processes. Table 3-5 shows waste quantities and dose rates that would be generated from each SBW treatment option.

### 2.4.1 Description of SBW Solids Treatment

The direct stabilization (solidification) processes produce containerized TRU waste, and secondary wastes. The TRU waste would be disposed of at WIPP; the secondary wastes would be stabilized for appropriate disposal. Evaluation of the projected solidified SBW was performed to determine if the waste forms could meet WIPP waste acceptance criteria. Characterization of the projected SBW forms are based on feasibility-level studies for solidification processes summarized in (Bonnema, et al. 2002) and the SBW radionuclide inventory and mass balance summarized in (Kimmitt 2002). The composition and quantities of the projected waste streams are based on process modeling and, in some cases, process development work using simulated feeds. The concentrations of radionuclides in the final waste forms were projected for conservative conditions based on flowsheets from process feasibility studies (Kimmitt 2002). A range of direct stabilization options has been evaluated for solidifying SBW as follows:

1. Contact-Handled TRU Grout

2. Calcination

3. Steam Reforming

4. Direct Vitrification.

For the purposes of the SBW-WIR analysis, it is assumed for all options that SBW solids are filtered from the process-feed stream and treated separately. The filtered solids (including those from the tank heel removal) would be dried and packaged to meet WIPP waste acceptance criteria as remote-handled (RH) waste between 100 R/hr and 1,000 R/hr. Front-end solids filtration is a required process step for the CH-TRU Grout and the UNEX and TRUEX separation options. It may be required for the other options (calcination, steam reforming, and vitrification) to improve process control and reduce final product variability. However, it is likely that the other options could be designed to handle the normal quantity of entrained solids in SBW feed and filtration equipment would not be necessary. (Feed to past calciner operations was not filtered.) The large quantity of heel solids recovered during tank closure operations may require additional handling equipment for all options; therefore, all SBW treatment options would probably have solid-handling cost. They should be less for calcination, steam reforming, and vitrification because they should not require front-end filtration. Because of this, the cost estimates for these options should be conservative in the SBW WIR economic evaluation.

### 2.4.2 General Description of SBW Direct Stabilization (Solidification) Processes

Contact-Handled TRU Grout with up front cesium separation involves design and construction of a new facility. The process comprises three basic unit operations: solids/liquid filtration, ion exchange through a packed bed to remove cesium, and subsequent grouting and packaging of the cesium-free ion exchange effluent. For this evaluation, crystalline sillicotitanate (CST) is assumed to be the ion exchange media. Each of these unit operations produces a waste stream for WIPP disposal. The filtered solids would be dried and shipped to WIPP as discussed above. The cesium-loaded ion exchange sorbent would be washed, dried, and packaged to meet WIPP waste acceptance criteria as a remote-handled transuranic waste greater than100 R/hr. The cesium free ion exchange effluent (the majority of the waste volume) would be grouted and packaged as contact-handled transuranic waste (less than 200 mrem/hr) for WIPP disposal. Cesium removal results in a significant cost savings due to the reduced quantities of remote-handled TRU waste.

Calcination involves upgrading the existing INTEC New Waste-Calcining facility to meet hazardous-waste permit requirements and to make it capable of controlling off-gas emissions to levels consistent with the EPA Maximum Achievable Control Technology rule. In the calcination process, liquid radioactive waste is injected into a hot (500°C or 600°C) fluidized bed. The water is vaporized, denitration occurs, and material dissolved in the liquid forms dry granular solids. Heat for the process comes from the in-bed combustion of kerosene. Total waste quantities are reduced because the acidic waste is converted to a remote-handled (less than 100 R/hr) dry granular solid for WIPP disposal. The filtered-feed solids would also be sent to WIPP, and secondary waste from off-gas cleanup would be disposed of as appropriate.

Steam Reforming requires construction of new facilities to treat SBW. In Steam Reforming, SBW is mixed with sucrose and/or carbon in a feed makeup tank prior to being fed to a fluidized bed reactor. In the reactor, steam is used as the fluidizing gas and a refractory oxide material is used as the bed media. An organic reductant and other additives are also fed the bed to enhance denitration and prevent particle agglomeration. The reactor vaporizes water in the waste to superheated steam and produces solid fines consisting primarily of inorganic salts. The solid-fines product is filtered from the off-gas and combined with larger particles that are occasionally withdrawn from the bottom of the fluidized bed. Together these solids constitute the primary steam-reformed product that would be packaged as RH-TRU for disposal at WIPP with radiation levels less than 100 R/hr. The filtered-feed solids would also be sent to WIPP, and a secondary waste from off-gas cleanup would be disposed of as appropriate.

Direct vitrification of SBW requires that new facilities be constructed. In the proposed vitrification process, the SBW waste is formed into glass and packaged in metal canisters. This creates a stable leach-resistant waste form, suitable for disposal in a geologic repository. SBW is mixed with glass frit and sucrose solution and fed to a melter that operates between 1000 and 1150°C. The melter is joule-heated with a cold cap and is capable of handling wet-slurried or dry-waste fee. The cold cap on the melter is maintained by the incoming feed pouring on top of the melted solution. In the melter all water evaporates, nitrates react with sugar to form metal oxides, phosphates decompose, and sulfates are dissolved in the glass melt. The melted glass is poured into canisters where it cools and solidifies. Lids are welded on the canisters and they are placed in storage until the vitrified glass is shipped to WIPP as less than 100 R/hr remote-handled transuranic waste. The filtered-feed solids would also be sent to WIPP, and a secondary waste from off-gas cleanup would be disposed of as appropriate.

# 3. TECHNICAL BASIS FOR CONFORMANCE

This chapter demonstrates that SBW conforms to the WIR criteria for TRU waste and should be managed and ultimately disposed of as TRU waste.

DOE M 435.1-1, *Radioactive Waste Management Manual*, lists three criteria that must be satisfied to demonstrate through a WIR determination that a waste associated with spent nuclear fuel (SNF) reprocessing can be managed as TRU waste:

1. The waste must have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical

2. The waste must be incorporated into a solid physical form and meet alternative requirements for waste classification and characteristics, as DOE may authorize

3. The waste must be managed pursuant to DOE's authority under the Atomic Energy Act of 1954 (AEA 1954), as amended, in accordance with the provisions of Chapter III of DOE M 435.1-1, as appropriate.

The historical INTEC management of reprocessing waste streams maintained a segregation of first-cycle extraction waste from other reprocessing wastes. DOE has removed the first-cycle extraction waste, most second- and third-cycle waste, and some SBW, and converted it to calcine. Information in the following sections shows: (1) that it is not technically practical to remove the additional key radionuclides from the SBW solids prior to disposal and (2) that even though technology exists to remove additional key radionuclides from SBW liquid, it is not economically practical. It is also concluded that removing additional key radionuclides for disposal at a HLW geologic repository would not significantly increase protection of public health or safety compared to direct stabilization and disposal of SBW at the WIPP geologic repository. Therefore, DOE's planning baseline is to pursue direct stabilization (solidification) of liquid SBW and disposal of SBW dried solids and solidified liquid as TRU waste at WIPP.

## 3.1 Criterion 1. Removal of Key Radionuclides

The first WIR criterion in DOE's Radioactive Waste Management Manual (DOE M 435.1-1, II(B)(2)(b), pg. II-2) is that key radionuclides be removed to the maximum extent technically and economically practical. This section discusses how INTEC waste management practices have removed and calcined the largest practical amount of initial key radionuclide activity (96%). In addition 99% of the initial reprocessing inventory of all radionuclides has been removed (through proper management and radioactive decay) from INTEC tank farm waste. This section also demonstrates that although further separation of the liquid SBW into HLW, LLW, and TRU waste fractions (i.e., key radionuclide removal) is technically feasible, it is not economically practical. The cost for removing additional key radionuclides was not justified for the negligible decrease in potential public risk.

### 3.1.1 Identification of Key Radionuclides

As discussed in DOE G 435.1-1, Section 2.3.1 the radionuclides of interest are those in 10 CFR §61.55, Tables 1 and 2, plus additional radionuclides DOE considers important to disposal site performance objectives. SBW contains all of these radionuclides (see Table 2-1). In order to determine which SBW radionuclides were key to the WIPP disposal site performance objectives, DOE-ID evaluated the WIPP RH-TRU Inventory Impact Report (Hadgu 2001) and the WIPP Compliance Certification Application (DOE 1996b) and supporting PA. These documents list four radionuclides (Am-241, Pu-238, Pu-239, and Pu-240) that impact WIPP disposal site performance objectives because they account for most of the radionuclide release.

These documents also state that the impact of Sr-90 and Cs-137 (the source of most SBW radioactivity) on repository performance is not significant because of the relatively short half-lives of these elements and the small contribution to the total inventory. All of the fission products decay very rapidly and can be excluded from further consideration. After 350 years even if the entire inventory from the repository were released, the contribution from the Sr-90 and Cs-137 would be 3 orders of magnitude below the regulatory limit. As demonstrated in the WIPP Compliance Certification Application (DOE 1996b) the only mechanism for release of fission products to the accessible environment is during human intrusion, an event extremely unlikely prior to 350 years. Meeting the WIPP waste acceptance criteria minimizes the short-term health effects due to Sr-90 and Cs-137 (Hadgu 2000). The SBW treatment facility will be designed to minimize exposure of the workers at INTEC and the WIPP WAC will ensure that the waste packages are within the transportation safety and WIPP RH-TRU worker exposure requirements.

Based upon the guidance document for DOE O 435.1, these documents, and consultation with Sandia National Laboratory and DOE-Carlsbad Field Office (CBFO), it was concluded that for SBW disposal at WIPP; Am-241, Pu-238, Pu-239, and Pu-240 are key radionuclides and Sr-90, Cs-137, and other isotopes are not key radionuclides.

### 3.1.2 Previous Segregation of Radionuclides

In correspondence concerning the Hanford HLW tanks (noted in Chapter 2), the NRC stated that "… if DOE could demonstrate that the largest practical amount of the total site activity attributable to 'first-cycle solvent extraction' wastes has been segregated for disposal as HLW, then NRC would view the residual as non-HLW."

INTEC's management of SNF reprocessing wastes segregated the "first-cycle solvent extraction" waste from other reprocessing wastes. INTEC maintained the waste in its original acidic form, stored the waste in stainless steel storage tanks, and segregated first-cycle extraction waste from the bulk SBW. As of March 1998, this segregated waste and most second and third-cycle waste was removed from the Tank Farm tanks leaving the blend of wastes that constitutes the SBW currently in storage. Between March 1998 and the last calciner run in 2000, over 200,000 gallons additional SBW was calcined. Because INTEC segregated and calcined the most highly radioactive waste and other waste as practical before the calciner was placed on standby, almost the entire radionuclide inventory has been removed from the tank farm. Figure 3-1 illustrates the mass balance for INTEC reprocessing waste management. Appendix A presents additional information and references for tank waste-inventory sources and the mass-balance calculations. Of the approximately 44 million curies (9.4 million gallons) of radioactive waste generated by spent fuel reprocessing, 81 thousand curies were sent to other DOE national laboratories, 19 million curies were reduced through radioactive decay, and 24 million curies were removed from the Tank Farm and calcined. The total curies removed, including decay, represents 99% of the total INTEC curie inventory generated through spent fuel reprocessing. The current tank inventory (479 thousand curies) represents about 1% of the initial spent fuel waste inventory (Tyson 2002). Table 3-1 provides: total estimated curies generated at INTEC, estimated curies remaining in the tanks, and percent reduced by radionuclide. Greater than 96% of the four key radionuclides important to WIPP have been removed.

### 3.1.3 Additional Removal of Key Radionuclides

DOE evaluated options for removing additional key radionuclides from SBW solids and liquids. The evaluations focused on methods for separating key radionuclides that were in solution since any practical methods of key radionuclide removal involved liquid-separation processes. DOE considered a number of options, discussed in Appendix B, for treating SBW. These included alternatives assessed in

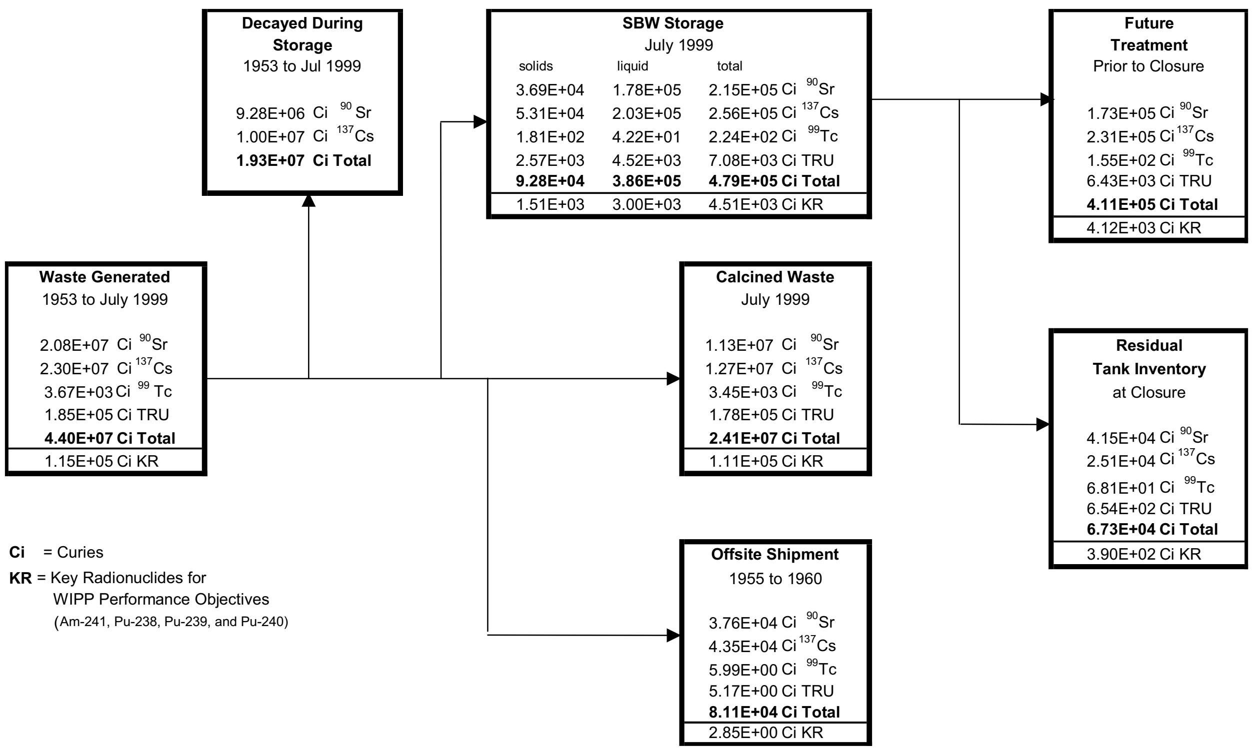


Figure 3-1. Mass balance for INTEC spent nuclear fuel processing (Tyson 2002).

Table 3-1. Percent curies removed for radionuclides – Decayed to July 1999 (Tyson 2002).

| Radionuclides | Total Curies Generated at INTEC | Total Curies Remaining in Tank (Liquids) | Total Curies Remaining in Tank (Solids) | Total Curies Remaining in Tank (Liquids & Solids) | Percent (in Liquids & Solids) of Initial Curies Removed |
|---|---|---|---|---|---|
| *Am-241 | 8.12E+03 | 2.81E+02 | 6.42E+01 | 3.45E+02 | 95.7% |
| Am-242 | 1.78E+00 | 5.71E-02 | 1.49E-02 | 7.20E-02 | 96.0% |
| Am-243 | 1.26E+01 | 8.11E-02 | 2.12E-02 | 1.02E-01 | 99.2% |
| C-14 | 3.00E-02 | 4.50E-04 | 1.18E-04 | 5.68E-04 | 98.1% |
| Cf-249 | 1.86E-09 | 6.61E-11 | 1.73E-11 | 8.34E-11 | 95.5% |
| Cf-250 | 1.36E-09 | 6.31E-11 | 1.65E-11 | 7.96E-11 | 94.1% |
| Cf-251 | 3.11E-11 | 1.02E-12 | 2.67E-13 | 1.29E-12 | 95.9% |
| Cm-242 | 1.49E+00 | 4.72E-02 | 1.23E-02 | 5.95E-02 | 96.0% |
| Cm-243 | 7.34E-01 | 1.14E-01 | 2.99E-02 | 1.44E-01 | 80.4% |
| Cm-244 | 4.42E+01 | 7.51E+00 | 1.96E+00 | 9.47E+00 | 78.6% |
| Cm-245 | 6.60E-03 | 1.11E-03 | 2.91E-04 | 1.40E-03 | 78.8% |
| Cm-246 | 4.60E-04 | 7.21E-05 | 1.89E-05 | 9.10E-05 | 80.2% |
| Cm-247 | 5.45E-10 | 8.11E-11 | 2.12E-11 | 1.02E-10 | 81.2% |
| Cm-248 | 6.19E-10 | 8.71E-11 | 2.28E-11 | 1.10E-10 | 82.2% |
| Co-60 | 8.88E+03 | 2.51E+02 | 3.65E+01 | 2.88E+02 | 96.8% |
| Cs-137 | 2.30E+07 | 2.03E+05 | 5.31E+04 | 2.56E+05 | 98.9% |
| H-3 | 2.01E+04 | 8.82E+01 | 3.59E+00 | 9.18E+01 | 99.5% |
| I-129 | 6.39E+00 | 2.48E-01 | 5.19E-02 | 3.00E-01 | 95.3% |
| Nb-94 | 3.36E+02 | 4.20E+00 | 3.17E+02 | 3.21E+02 | 4.4% |
| Ni-59 | 6.79E+01 | 1.42E+01 | 3.72E+00 | 1.79E+01 | 73.6% |
| Ni-63 | 5.72E+03 | 1.67E+02 | 3.14E+01 | 1.98E+02 | 96.5% |
| Np-237 | 4.10E+01 | 1.04E+01 | 1.50E-01 | 1.06E+01 | 74.3% |
| *Pu-238 | 1.03E+05 | 2.40E+03 | 1.32E+03 | 3.72E+03 | 96.4% |
| *Pu-239 | 3.10E+03 | 2.77E+02 | 1.18E+02 | 3.95E+02 | 87.3% |
| *Pu-240 | 1.47E+03 | 3.74E+01 | 1.02E+01 | 4.76E+01 | 96.8% |
| Pu-241 | 6.99E+04 | 1.50E+03 | 1.05E+03 | 2.55E+03 | 96.4% |
| Pu-242 | 3.13E+00 | 6.32E-02 | 7.70E-03 | 7.09E-02 | 97.7% |
| Pu-244 | 3.79E-08 | 2.52E-09 | 6.60E-10 | 3.18E-09 | 91.6% |
| Se-79 | 1.07E+02 | 1.62E+00 | 4.24E-01 | 2.04E+00 | 98.1% |
| Sn-126 | 9.52E+01 | 1.53E+00 | 4.01E-01 | 1.93E+00 | 98.0% |
| Sr-90 | 2.08E+07 | 1.78E+05 | 3.69E+04 | 2.15E+05 | 99.0% |
| Tc-99 | 3.74E+03 | 4.22E+01 | 1.81E+02 | 2.23E+02 | 94.0% |
| **TOTAL** | 4.40E+07 | 3.86E+05 | 9.31E+04 | 4.79E+05 | 98.9% |
| **TOTAL Key Radionuclides** | 1.16E+05 | 3.00E+03 | 1.51E+03 | 4.51E+03 | 96.1% |

* Key radionuclides

the Idaho HLW & FD FEIS, recommendations from the National Academy of Sciences, and INTEC evaluations. DOE's evaluation and review process resulted in two categories of SBW separation-options being considered for additional key-radionuclide removal:

- Precipitation Options
  - Hydroxide precipitation
  - Modified Hydroxide precipitation
  - Low-temperature precipitation
  - High-temperature evaporation and precipitation.
- Solvent Extraction Options
  - Universal extraction (UNEX)
  - TRU extraction (TRUEX)
  - Modified UNEX separations.

### 3.1.4 Technical Practicality of Further Key Radionuclide Removal from SBW Solids

For additional key-radionuclide removal from SBW solids to be technically practical, solids dissolution would be required in order to make chemical separation possible. Physical separation was not practical since SBW solids contained the same radionuclide assortment as the liquids with no way of physically separating key radionuclides. SBW liquid is $\geq$ 2 molar nitric acid solution and because the solids have had long-term exposure to this acid, further dissolution in order to separate additional key radionuclides would be difficult if not impossible. Strong acid mixtures at elevated temperatures could possibly dissolve some constituents in the SBW solids; however no production scale technologies exist and because of the relatively small solids quantity and the severe conditions anticipated for dissolution, it was not technically practical to develop dissolution processes. As a result, no technologies have been demonstrated for dissolving SBW solids and additional key-radionuclide removal is considered not technically practical.

### 3.1.5 Technical Practicality of Further Key Radionuclide Removal from SBW Liquids

It was determined that it is possible to separate and remove additional key radionuclides from SBW liquids through various precipitation and solvent extraction options. These options were retained for further evaluation to determine technical practicality as described below.

***3.1.5.1 Precipitation Options.*** The precipitation methods for SBW separation use either chemical or temperature manipulation to precipitate metals and other constituents. Following solid/liquid separations, the separated fractions would go through additional treatment and be disposed of as appropriate. All of the precipitation methods were eliminated from continued consideration for performing further key radionuclide separation and removal due to various technical difficulties in maintaining an operational system under both normal and off-normal conditions (see Appendix B).

***3.1.5.2 Solvent Extraction Options.*** Key radionuclides can be removed from the SBW using various organic solvents. The two most common processes are transuranic solvent extraction (TRUEX), which removes actinides including transuranics, and the universal solvent extraction (UNEX), which removes actinides, cesium, and strontium. These two solvent extraction processes and modifications to the processes were assessed by the Idaho HLW & FD FEIS. Removing additional key radionuclides from SBW liquid through solvent extraction results in a low-level waste fraction and a high-level waste fraction. The LLW fraction would be treated and disposed of as appropriate. The HLW fraction containing the key radionuclides would be vitrified for disposal at the HLW National Repository (see Table 3-5). These two technically practical options are described below.

UNEX: In the universal solvent extraction (UNEX) process, the waste feed is filtered and then contacted with the UNEX solvent to extract actinides, cesium, and strontium at high efficiency with a four-component solvent. The actinides are stripped from the solvent with a 0.5 molar solution of guanidine carbonate containing 10 g/l diethylenetriamine pentaacetic acid (DTPA). The separation efficiency for key radionuclides is greater than 99.9% for both plutonium isotopes and americium isotopes. The removal efficiency for other radionuclides of interest (major contributors to the radiation field) is 99.5% for cesium isotopes and greater than 99.9% for both strontium and europium isotopes. The remainder of the process is very similar to the TRUEX process (except that mercury is not separated out; it is retained in the low-side grout) and the volumes of secondary wastes from off-gas treatment are assumed to be the same as TRUEX. Because of the high separation efficiencies, it is assumed that UNEX would remove all 3,000 curies of key radionuclides from SBW liquids. The separated radionuclides (including key radionuclides) would be disposed of at the HLW National Repository and the remaining waste would be suitable for Class-A disposal. For a more complete description of the UNEX process see Bonnema, et al. 2002.

TRUEX: In the TRU Separations process, the waste feed is filtered and then contacted with the TRUEX solvent to extract actinides at high efficiency. The actinides are stripped from the solvent using a solution of 1-hydroxyethane-1,1 diphosphonic acid (HEDPA). The separations efficiency for plutonium isotopes is 99.8% and for americium isotopes is 99.9%. The separated actinides are combined, concentrated by evaporation, and vitrified. The raffinate (bulk waste) from the TRUEX process is concentrated by evaporation and grouted to produce a remote-handled low activity waste (RH-LAW) waste. The TRUEX solvent is washed with a sodium carbonate solution and mercury is precipitated from the carbonate wash effluent as mercury sulfide. The washed TRUEX solvent is recycled. Nitric acid and water are recovered from evaporator condensates and recycled to the process. Because of the high separation efficiencies, it is assumed that TRUEX would remove all 3,000 curies of key radionuclides from SBW liquids. The fraction containing the separated radionuclides (including key radionuclides) would be suitable for disposal at the HLW National Repository and the other fraction would be suitable for Class-C low-level waste (LLW) disposal. For a more complete description of the TRUEX process see Bonnema et al. (2002).

The UNEX and TRUEX options were retained for economic evaluation.

### 3.1.6 Economic Practicality of Further Radionuclide Separation and Removal

An evaluation was performed to assess the economic practicality of further key-radionuclide removal. Costs for key radionuclide removal at other DOE sites were evaluated to help determine what was a reasonable expenditure. DOE Hanford determined that even though it was technically practical to remove their additional key radionuclides (Cs-137, Sr-90, TRU radionuclides, and Tc-99), that it was not economically practical. Cost ranged from $982 million ($30/curie) for Cs removal to $7.9 billion ($790,000/curie) for TRU radionuclide removal (WHC 1996). DOE Savannah River determined that it was technically practical to remove additional key radionuclides from their F & H tank farm tanks. However they determined that a cost of $10.5 million to reduce the dose to member of the public from 1.9 to 1.7 mrem/year and to reduce the drinking water dose to an inadvertent intruder from 130 mrem/year to 110 mrem/year was not economically practical (NRC 1999).

The economic practicality of removing additional key radionuclides from INTEC SBW was evaluated by determining removal costs and the effect of reducing radionuclide releases to the public. The economical practicality of further separating the SBW to perform key radionuclide removal was evaluated using several key cost parameters. When evaluating alternatives, total treatment and disposal costs were considered. Certain costs were judged to be essentially the same for each alternative and therefore were not used to compare options. Therefore costs presented in this document do not represent total costs and should not be used for life-cycle budget planning. Other costs were different for each treatment process and were used to compare alternatives. Examples of costs that were judged to be essentially the same were: utility costs, costs for facilities common to all options, analytical support costs, management costs, maintenance costs, and the cost of inflation. Major costs that were different and used to discriminate among alternatives were facility design and construction costs (total project costs); operation costs; facility decontamination, demolition, and disposal (DD&D) costs; and transportation and waste disposal costs. Only discriminatory costs were considered when performing the economic evaluations for comparing various SBW treatment alternatives. Costs were presented in year-2001 dollars. Details of this evaluation are presented in Bonnema et al. (2002). The estimated key radionuclide removal and treatment costs and the amount of total key-radionuclide activity removed were used to calculate a cost per curie attributable to each option evaluated. These factors are summarized in Table 3-2 and discussed below. The effect on the performance assessment at the WIPP disposal site is also evaluated.

When evaluating the economical practicality of removing additional key radionuclides, SBW solids were not considered since additional key-radionuclide removal from SBW solids is not technically practical. Solids contain an estimated 1,500 curies of key radionuclides (see Table 3-1).

***3.1.6.1 Solidification with Disposal at WIPP (Base Case).*** Based on protection of the public health and the environment, technical feasibility, cost, and other relevant factors; it is anticipated that DOE will present direct stabilization (solidification) of SBW as part of the preferred treatment alternative for the Idaho HLW & FD FEIS.

DOE established SBW solidification with WIPP disposal as the base case for evaluating the economic practicality of further separating SBW to perform key radionuclide removal. Using discriminatory costs DOE calculated a base cost of $566 million for the CH TRU Grout option, $992 million for the Calcination option, $1,054 million for the Steam Reforming option, and $1,359 million for the Vitrification option (see Table 3-2). Actual costs will be higher than this, since the purpose was not to capture total costs but to count costs that discriminated among alternatives. Based on technical practicality, the key radionuclide separation options chosen for economic comparison with the base case were UNEX, and TRUEX.

***3.1.6.2 Universal Extraction (UNEX)***. The UNEX process option removes essentially all key radionuclides from SBW liquid and many other radionuclides (see Table 2-1). This is about 2.6% of the initial key radionuclide inventory, or 3,000 curies (see Table 3-1). The discriminatory cost of this key-radionuclide removal option is approximately $1,732 million (see table 3-2). This is $373 million to $1,166 million above the cost of direct stabilization options, or $124 thousand to $389 thousand per curie more for additional key radionuclide removal (see Table 3-3). The relatively high additional cost are mainly attributable to the construction and operation of facilities to separate SBW and treat the high-level waste fraction.

Table 3-2. SBW treatment alternative estimated costs (see Bonnema et al. 2002).

| | Direct Stabilization Alternatives (Cost x millions) | | | | Key-Radionuclide Separation Alternatives | |
|---|---|---|---|---|---|---|
| Alternative | CH-TRU Grout * | Calcination | Steam Reforming | Direct Vitrification | UNEX* * | TRUEX ** |
| Total Project Cost | $242 | $301 | $485 | $746 | $989 | $1,663 |
| Operations | $89 | $212 | $163 | $189 | $416 | $399 |
| DD&D | 67 | 84 | 136 | 209 | 277 | 466 |
| Transportation & Disposal | | | | | | |
| Class-A Disposal at Hanford | | | | | $15 | |
| Class-C Disposal at Hanford | | $0.1 | $0.1 | $0.3 | | $87 |
| CH-TRU Disposal at WIPP | $105 | | | | | |
| RH TRU disposal at WIPP | $63 | $325 | $270 | $215 | $21 | $21 |
| HLW Disposal at National Rep. | | | | | $14 | $143 |
| | | | | | | |
| Total | $566 | $922 | $1,054 | $1,359 | $1,732 | $2,779 |
| | *Includes up-front removal of Cs to allow the bulk waste to be disposed of as CH grout. (Grouting without Cs removal total cost would be greater than $1 billion including disposal costs for over 5,000 $m^3$ of RH grouted product.) | | | | **Includes the cost of vitrification & HLW disposal. | |

Table 3-3. Cost per curie for additional key radionuclide removal.

| Removal Option | Curies Removed | Cost above Grout | Cost above Calcination | Cost above Steam Reforming | Cost above Direct Vitrification | Additional $/Curie Range |
|---|---|---|---|---|---|---|
| UNEX | 3,000 | $1,166 million | $810 million | $678 million | $373 million | $389,000–$124,000 |
| TRUEX | 3,000 | $2,213 million | $1,857 million | $1,725 million | $1,420 million | $738,000–$473,000 |

The solidified waste from the UNEX option would be disposed of at two locations, the HLW geologic repository and a near-surface LLW facility.[g] Both disposal sites are designed to meet environmental safety standards for protection of human health and the environment.

***3.1.6.3 Transuranic Extraction (TRUEX).*** The TRUEX process option removes essentially all key radionuclides from SBW liquid. This is about 2.6% of the initial key radionuclide inventory, or 3,000 curies (see Table 3-1). The discriminatory cost of this key-radionuclide removal option is approximately $2.78 billion (see Table 3-2). This is $1.42 billion to $2.21 billion above the cost of direct stabilization options, or $473 thousand to $738 thousand per curie more for additional key radionuclide removal (see Table 3-3). The high additional cost are mainly attributable to the construction and operation of facilities to treat the high level waste fraction separated from the SBW.

g. The Hanford RCRA Part B permit does not currently allow acceptance of offsite waste for disposal.

The solidified waste from the TRUEX option would be disposed of at two locations, the HLW geologic repository and a near-surface LLW facility.[g] Both disposal sites are designed to meet government environmental safety standards.

#### 3.1.6.4 Effect of Key Radionuclide Removal on Worker Occupational Dose.

Occupational dose to workers during SBW treatment and disposal was generally found to be low (Reference the Idaho HLW & FD FEIS, Chapter 5). Allowable worker radiation exposures are set by DOE regulations. Additional shielding was added to facilities that handle more highly radioactive; so that in the final analysis, worker exposure was about the same for all options. Likewise; shipping, handling, and disposal facilities all have equipment and procedures to handle waste product safely. Therefore, increased radiation levels for various waste types were reflected in increased costs for additional shielding, shipping, and handling requirements.

#### 3.1.6.5 Effect of Key Radionuclide Removal on WIPP Repository Performance.

The WIPP Land Withdrawal Act (LWA) of 1992 provides capacity limits based upon radioactive doses for CH and RH TRU waste. The LWA also contains guidance on the RH TRU capacity limit for waste with doses $\geq$ 100 R/hr (see Table 3-4). The quantity of waste $\geq$ 100 R/hr from some INTEC SBW treatment options may exceed the guidance limit. Also, tank heel solids may be > 100 R/hr and quantities may be more than estimated (see Table 3-5). However, if these wastes exceed the LWA > 100 R/hr volume limits, then part or all of the waste could be treated to meet the WIPP waste acceptance criteria of < 100 R/hr.

The WIPP Compliance Certification Application (DOE 1996b) and supporting PA conservatively modeled TRU components from all waste planned for disposal. The total TRU-source term from SBW liquid and solid treatment products constitutes about 7,000 curies (see Figure 3-1).[h] The key radionuclides in treated SBW that contribute to potential release (see Figure 3-1) at WIPP is conservatively estimated to be about 4,500 curies. The UNEX or TRUEX treatment options would remove about 3,000 curies of key radionuclides from the SBW liquid. The estimated total is 4,070,000 curies for radionuclides important to WIPP performance objectives from all sources at closure (reference DOE 1996b, Table WCA-5). The 3,000 curies that could be separated by removing additional key radionuclides from the INTEC SBW is less than 0.1% of the total WIPP curies at closure.

Sandia National Laboratories, at the request of the DOE-ID, has performed a preliminary investigation for the potential impact to the Performance Assessment of the Waste Isolation Pilot Plant due to an additional waste form (Sanchez 2002). The waste form was assumed to consist of SBW with a total TRU activity (alpha emitting transuranic radionuclides with half-lives greater than 20 years) of 7,000 curies. The following paragraphs summarized the finding of the preliminary investigation.

The EPA regulations for WIPP disposal (40 CFR 191, EPA 1996) govern the projected cumulative release of radioactive waste to the accessible environment. The release limits for isolation or containment are based on long-term (post-closure) human health risks expressed in "EPA Units" (also termed "normalized releases"). An "EPA Unit" is the amount of waste containing 1,000,000 curies of alpha-emitting transuranic radionuclides with half-lives greater than 20 years and is used to calculate a conditional scenario of groundwater contamination and the resulting ingestion pathways (Hadgu 2001).

The application of the normalizing process for cumulative releases (40 CFR 191,Appendix A, Table 1) to the WIPP repository yields a bounding source term of approximately 10,000 EPA Units with

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h. Some residual solids will remain after the SB W storage tanks are cleaned for closure. Since the amount will vary depending on cleaning effectiveness, for conservatism, all of the solids are assumed to go to the WIPP repository.

the expectation that this source term would never be exceeded. It follows that the WIPP repository upper bound performance, based upon a source term of 10,000 EPA Units and a release limit of 1,000 curies per EPA Unit, are sufficiently large to be insensitive to the INTEC SBW source term of 7,000 curies (or 0.007 EPA Units).

The small volumes of SBW (see Table 3-5) would not have a significant impact on waste matrix properties (solubilities, consolidation strength, etc.), when compared to the waste matrix and total volume of TRU waste scheduled for WIPP disposal (see Table 3-4). Nor would they have a substantial impact on either indirect releases (subsurface release to biosphere) or the direct human intrusion (drilling activities that penetrate the waste region). Thus long- term human health risks (post-closure risks for as identified via 40 CFR 191) are not impacted by any possible waste treatment to remove additional radionuclides from SBW (Sanchez 2002).

Table 3-4. WIPP disposal capacities (DOE 2001c).

| Disposal Type | Capacity ($m^3$)[a] | Estimated Available (Uncommitted) Capacity ($m^3$)[b] |
|---|---|---|
| Contact Handled TRU <200 mR/hr | 168,520 total | 64,276 |
| Remote Handled TRU >200 mR/hr <100 R/hr | 7,080 total | 4,979 |
| Remote Handled TRU >100 R/hr < 1000R/hr | 350 | ~290 |
| | 175,600($m^3$) Total Capacity | |

a. Column 2 in presents disposal capacity for CH and RH-TRU waste given in the Land Withdrawal Act.
b. Column 3 presents the remaining disposal capacity based upon projected disposal volumes (does not include INEEL SBW TRU waste).

Table 3-5. Waste generation quantities for SBW disposal options -contact dose rates in July 1999 (Kimmitt 2002).

| Option | Solids RH TRU ($m^3$)[a] | RH TRU ($m^3$) | CH TRU ($m^3$) | CH-Mixed LLW ($m^3$) | RH-Mixed LLW ($m^3$) | HLW ($m^3$) |
|---|---|---|---|---|---|---|
| Direct stabilization | | | | | | |
| CH-TRU grout | 81 (130 R/hr) | 168[b] (251 R/hr) | 4600 (190 mr/hr) | | | |
| Calcination | 81 (130 R/hr) | 1,201 (46 R/hr) | | 50[c] (<5 mr/hr) | | |
| Steam reforming | 81 (130 R/hr) | 981 (57 R/hr) | | 50[c] (<5 mr/hr) | | |
| Direct vitrification | 81 (130 R/hr) | 764[b] (35 R/hr) | | 110[c] (<5 mr/hr) | | |
| Separation of key radionuclides | | | | | | |
| TRUEX | 81 (130 R/hr) | | | | 6,763 (8 R/hr) | 210 (36 R/hr) |
| UNEX | 81 (130 R/hr) | | | 6,664 (93 mr/hr) | | 20 (1,130 R/hr) |

a. Values for dried solids. The remaining table values are for SBW liquid treatment.
b. Volumes increased to meet shipping requirements.
c. Secondary waste streams.

Therefore based upon the large expense for additional key radionuclide removal for a negligible decrease in public risk, it is determined that additional removal of key radionuclide from liquid SBW is not economically practical. This is understandable since WIPP is a deep geological repository specifically designed for disposal of TRU waste.

### 3.1.7 Summary of Criterion 1 Evaluation

Greater than 96% of the key radionuclides and 99% of all radionuclides generated from spent fuel reprocessing have been removed from the INTEC tank-farm waste. The remaining SBW was evaluated using the WIR determination process to see if the waste should be classified and managed as other than high-level waste. The proposed SBW disposal is at the WIPP geological repository as a TRU waste. The benefit of removing additional key-radionuclides from SBW liquids and solids was evaluated. This evaluation process involved: determining which SBW radionuclides were important to WIPP performance objectives, determining technically viable options for removing additional important (key) radionuclides, and determining if the technically viable options were economically practical when compared to the reduction in radionuclide releases at WIPP. It was determined that directly stabilized SBW (without additional key radionuclide removal) would meet the WIPP waste acceptance criteria and waste acceptance requirements. It was determined that there are no technically viable options for removing additional key radionuclides from SBW solids. It was also determined that even though there are technically viable options for removing additional key-radionuclides firm liquid SBW, that it is not economically practical since it would cost, depending on the option chosen, an estimated $1.73 billion to $2.78 billion to remove a relatively small number of curies. The small reduction of key radionuclides from additional separations has no significant effect on reducing radionuclide releases at WIPP. Key radionuclide removal from liquid SBW does not significantly impact the source term at closure for the long-term hazards analysis or near-term safety requirements since the WIPP facility has been designed to handle and dispose of transuranic waste. Therefore, it was concluded that additional key-radionuclide removal from SBW solids is not technically practical and from SBW liquid is not economically practical and that SBW should be managed as TRU waste.

## 3.2 Criterion 2. Incorporate into a Solid Form and Meet Alternative Classification Requirements

This section discusses how the proposed management of SBW will meet the second WIR criterion specified in DOE's Radioactive Waste Management Manual: "The waste will be incorporated in a solid physical form and meet alternative requirements for waste classification and characteristics, as DOE may authorize" (DOE M 435.1-1, II(B)(2)(b)(2), pg. II-2, DOE 1999c).

The SBW will be incorporated into a solid waste form through one of several solidification processes: CH-TRU Grout, Calcination, Steam Reforming, or Vitrification.

The alternative classification requirements pertains to waste that cannot meet the limits of 10 CFR §61.55 for low-level waste. The solidified SBW is a TRU waste and meets the alternative classification requirements by compliance with the WIPP waste acceptance criteria (WAC) being developed for remote-handled transuranic (RH-TRU) waste (DOE 2000). The solidified waste produced from treatment of SBW liquids and solids must meet the WIPP WAC for CH-TRU and RH-TRU waste. Table 2-1 provides the key radionuclide concentrations for the 100% SBW liquid and 100% SBW solids feed streams considered for this evaluation. The actual SBW feed stream to the stabilization process will vary depending on the process chosen to treat the SBW liquid and solids. The individual SBW liquid and SBW solids feed stream radionuclide concentrations were used in models (Kimmitt 2002) to project bounding characteristics of the solidified waste. Table 3-6 compares the solidified waste characteristics to the proposed WIPP WAC for RH-TRU waste. The solidified SBW is expected to meet the WIPP WAC and therefore the alternative classification requirements.

Secondary wastes would be solidified to meet WIPP or LLW disposal facilities waste acceptance criteria as appropriate.

## 3.3 Criterion 3. Manage Pursuant to Chapter III of DOE M 435.1-1

This section discusses how the proposed management and disposal of the SBW will meet the third criterion for an evaluation WIR: "Managed pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, in accordance with the provisions of Chapter III of DOE M 435.1-1." Solidified sodium-bearing waste would be managed and disposed of as transuranic waste in accordance with DOE M 435.1-1. WIPP is a permitted disposal site for contact-handled mixed transuranic wastes and is expected to be permitted for remote-handled waste by 2003, long before sodium-bearing waste is shipped from the Idaho Nuclear Technology and Engineering Center.

The WIPP Land Withdrawal Act of 1992, as amended, defines transuranic waste and limits disposal at WIPP to transuranic waste resulting from atomic energy defense activities. DOE General Counsel (Nordhaus 1996) interpreted the definition of atomic energy defense activities as stated in the Nuclear Waste Policy Act of 1982. This interpretation was used to document that the SBW qualifies as defense waste (Bergholz 2002) and is eligible for disposal at WIPP.

The solidified SBW will have an activity greater than 100 nCi of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years. Kimmitt (2002) provides the estimated partitioning of the radionuclides, volume of waste streams, and concentration by treatment option. Table 3-6 evaluates each CH-TRU and RH-TRU waste stream for compliance with the WIPP waste acceptance criteria

Transuranic waste shall be disposed in accordance with the requirements of 40 CFR Part 191. Table 2-1 presents the key radionuclide concentrations for the 100% SBW liquid and 100% SBW solids feed streams considered for this evaluation. The actual SBW feed stream to the solidification process may be a blend of SBW liquid and solids slurry depending on the technology selected. The individual SBW liquid and SBW solids feed stream radionuclide concentrations were used in the solidification models (Kimmitt 2002) to project bounding characteristics of the solidified waste.

The solidified SBW can be treated to meet the waste acceptance criteria and performance assessment requirements of 40 CFR Part 191 for the WIPP geologic repository. By meeting the WIPP WAC for CH and RH-TRU, the solidified SBW will not adversely affect the performance objectives of the WIPP repository, since the SBW key radionuclides represent less than 0.1% of the total source term used for the WIPP PA. Solidified SBW will be managed and disposed of as TRU waste in accordance with provisions of Chapter III, DOE M 435.1-1, "TRU Waste Requirements."

For secondary wastes, meeting the disposal site waste acceptance criteria would satisfy Criterion 3 requirements.

Table 3-6. Comparison of projected SBW CH-TRU and RH-TRU waste characteristics to WIPP waste acceptance criteria.

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| Filter Vents (RH-TRU 72B and CNS 10-160B transport casks) | Each waste payload container and any sealed secondary containers greater than 4 liters overpacked in the payload container must have one or more filter vents. Filter vents are optional on metal secondary containers containing solid inorganic waste only. | Records of visual inspection.<br>Site procurement specifications and QA acceptance reports, or manufacturers' fabrication documentation. | Meets Criterion.<br>• Acceptable filters can be installed, or<br>• Document that the waste is inorganic waste. |
| Payload container description/weight (RH-TRU 72B transport cask) | Payload container shall be DOT Type A or equivalent, and it must meet the requirements of the RH-TRU 72-B Cask SAR.<br><br>Weight of loaded DOT Type A container must not exceed the tested values; 5,250 lb, when direct loaded, or 5,980 lb when loaded in three 55-gal drums or 30-gal drums prior to placement in the RH canister. Higher weight limits will be allowed upon appropriate testing. | Procurement records and visual inspection.<br><br>Calculate gross weight of payload container or weight payload container(s) on calibrated scale. | Meets Criterion.<br>All containers meet criterion. |
| Payload container description/weight (CNS 10-160B) | Payload containers shall be DOT Type A or equivalent, and the must meet the requirements of the CNS 10-160B Cask SAR.<br>Weight of contents, shoring, secondary containers, and optional shield insert not to exceed 14,500 lb. | | Not Applicable.<br>Project will use RH-TRU 72 B Cask. |
| Payload container condition (RH-TRU 72B and CNS 10-160B) | Payload container shall be in good condition. | Procurement controls and visual inspection. | Meets Criterion. |
| Payload container identification (RH-TRU 72B and CNS 10-160B) | Payload containers have a unique identification number. | Visual inspection prior to shipment | Meets Criterion. |
| Secondary containers (CNS 10-160B) | Secondary containers or components must be shored to prevent movement during accident conditions. | Visual inspection prior to shipment | Not Applicable.<br>Project will use RH-TRU 72 B Cask. |

Table 3-6. (continued).

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| Sharp or heavy objects (RH-TRU 72B and CNS 10-160B) | Sharp or heavy objects in the waste shall be blocked, braced, or suitably packaged as necessary to provide puncture protection for the payload containers packaging these objects. | Visual inspection prior to shipment | Meets Criterion. |
| Residual liquids (RH-TRU 72B and CNS 10-160B) | Aggregate amount of residual liquid < 1 volume percent of payload container.<br>< 1 inch or 2.5 cm in bottom of internal containers. | Use one of the following:<br>Acceptable Knowledge, or<br>Radiography, or<br>Visual examination | Meets Criterion. |
| Compressed gases (RH-TRU 72B and CNS 10-160B) | Compressed gases are prohibited. | Use one of the following:<br>Acceptable Knowledge, or<br>Radiography, or<br>Visual examination | Meets Criterion. |
| Sealed containers (RH-TRU 72B and CNS 10-160B) | Sealed containers > 4 liters are prohibited except for metal containers packaging solid inorganic waste. | For newly generated waste, use one of the following:<br>Process Knowledge, or<br>Radiography, or<br>Visual examination<br><br>For retrievably stored waste, develop a sampling program using visual examination, radiography, or Process Knowledge | Meets Criterion.<br>SBW final waste form is a solid, inorganic waste. |
| Waste form (RH-TRU 72B and CNS 10-160B) | Only waste forms belonging to S3000 (homogeneous solids), S4000 (soils and gravel), and S5000 (debris) summary category groups are acceptable. | Use one of the following:<br>Acceptable Knowledge, or<br>Radiography, or<br>Visual examination<br><br>Report in the WIPP waste information systems (WWIS) | Meets Criterion.<br>All waste forms are S3000. Acceptable knowledge will be used based on analysis of process feeds. |
| Waste type and content code (RH-TRU 72B) | Must meet remote-handled transuranic content code (RH TRUCON) description code. | Classification of waste based on physical form and chemicals/materials present in the waste | TRUCON waste specific number will be obtained from WIPP after the WIR and Idaho HLW & FD FEIS ROD are approved if the documents specify that the waste is going to WIPP. |

Table 3-6. (continued).

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| Waste type and content code (CNS 10-160B) | Must meet content code descriptions in CNS 10-160B SAR. | Classification of waste based on physical form and chemicals/materials present in the waste | Not Applicable.<br>Project will use RH-TRU 72 B Cask. |
| Flammable volatile organic compounds (VOCs)<br><br>(RH-TRU 72B and CNS 10-160B) | ≤ 500 ppm total flammable VOCs in the payload container headspace. | Use one of the following:<br>Process Knowledge to show no flammable VOCs are present, or<br>Process Knowledge to show that potentially flammable VOCs present would be less than 500 ppm in the headspace if all of the potentially flammable VOCs vaporized into the headspace of the payload container, or<br>PK may include headspace gas sampling and analysis | Meets Criterion.<br>Based upon process knowledge and tank sampling, SBW would not generate 500 ppm VOCs |
| Hazardous waste codes<br>(RH-TRU 72B and CNS 10-160B) | Hazardous wastes are limited to those having hazardous waste codes listed in Attachment O of the WIPP Hazardous Waste Facility Permit | Use Acceptable Knowledge | Meets Criterion.<br>Assigned Waste Codes: F001, F002, F005, and U134.<br><br>U134 is not currently in the WIPP permit; however, in June 2002, WIPP submitted a permit modification request to add U134 to their permit. |
| PCBs<br>(RH-TRU 72B and CNS 10-160B) | PCBs < 50 ppm. | Acceptable Knowledge | Meets Criterion.<br>There are no PCBs in SBW. |
| Explosives<br>(RH-TRU 72B and CNS 10-160B) | Explosives are prohibited. | Use Acceptable Knowledge. | Meets Criterion.<br>There are no explosives in SBW. |
| Corrosives<br>(RH-TRU 72B and CNS 10-160B) | Corrosives are prohibited. | Use Acceptable Knowledge. | Meets Criterion.<br>Waste forms are not corrosive. |
| Ignitables<br>(RH-TRU 72B and CNS 10-160B) | Ignitables are prohibited. | Use: Acceptable Knowledge. | Meets Criterion.<br>There are no ignitables in SBW. |

Table 3-6. (continued).

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| Reactives (RH-TRU 72B and CNS 10-160B) | Reactives are prohibited. | Use Acceptable Knowledge. | Meets Criterion.<br>AK that there are No reactives in SBW |
| Pyrophorics (RH-TRU 72B and CNS 10-160B) | < 1 % radionuclide pyrophorics by weight of the payload container.<br>Nonradionuclide pyrophorics are prohibited. | Use Acceptable Knowledge. | Meets Criterion.<br>There are no pyrophorics in SBW. |
| Hydrogen gas concentration and total gas generation (RH-TRU 72B and CNS 10-160B) | Option 1(Test): Hydrogen and total gas generation rate ≤ limit in RH-TRU Waste Shipping Package SAR for applicable content code must ensure hydrogen gas ≤ 5% by volume. | 1. Gas generation testing as per Attachment 2 of Appendix 1.3.7 of the 72-B Cask SAR Hydrogen and total gas generation rate ≤ rate limit.<br>2. For inorganic waste forms and organic waste forms ≤ 5.92 watts, compliance with total gas generation rates is demonstrated by analysis. | Meets Criterion for Option 2 (analysis).<br>See gas generation estimates (Table 3-7) |
| Hydrogen gas concentration and total gas generation (RH-TRU 72B and CNS 10-160B) (continued) | Option 2 (Analysis): Decay heat within each payload container ≤ limit in RH-TRU 72-B Waste Shipping Package SAR for applicable content code | Acceptable methods for characterizing the isotopes (all but trace) necessary for calculation of decay heat are demonstrated using any of the following methods:<br>• Non-Destructive Analysis<br>• Radiochemical assay<br>• Material Accountability & Tracking (MA&T) Process Knowledge<br>• Gamma dose measurement.<br>Use value plus measurement error for decay heat Trace isotopes (limited to 5%) may be demonstrated by Process Knowledge and waste management databases. | Calculated estimates are within limits. A content code of OR311A (ORIGEN code used to calculate radioactive decay) has been assumed. |
| Waste compatibility (RH-TRU 72B and CNS 10-160B) | No chemicals or materials that are incompatible | Use applicable information from Acceptable Knowledge and/or radiography and/or visual examination (VE) to demonstrate compliance with Permit Application Appendix C-1 analyses and content code description. | Meets Criterion.<br>The waste forms are compatible with the containers. |

Table 3-6. (continued).

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| Radiation dose rate (including neutron contribution) (RH-TRU 72B and CNS 10-160B) | > 200 mrem/hr and ≤1000 rem/hr at the surface of the payload container | Measure values from calibrated beta/gamma and neutron dose rate instruments.<br><br>Report radiation dose rate at the surface of the payload container in the WWIS. | Meets criterion.<br>Calculated surface dose rates:<br>See Table 3-8. |
| Radiation dose rate (including neutron contribution) (RH-TRU 72B and CNS 10-160B) (continued) | ≤ 200 mrem/hr at the surface of the shipping cask | Measure values from calibrated beta/gamma and neutron dose rate instruments.<br><br>Report radiation dose rate at the surface of the payload container in the WWIS. | Meets criterion.<br>Calculated dose rates:<br>See Table 3-8. |
| Radiation dose rate (including neutron contribution) (RH-TRU 72B and CNS 10-160B) (continued) | ≤ 10 mrem/hr at 1 meter from the surface of the shipping cask | Measure values from calibrated beta/gamma and neutron dose rate instruments.<br><br>Report radiation dose rate at the surface of the payload container in the WWIS. | Meets criterion.<br>Calculated dose rates:<br>See Table 3-8 |
| Radiation dose rate (including neutron contribution) (RH-TRU 72B and CNS 10-160B) (continued) | < 1 rem/hr at 1 meter based on hypothetical accident conditions (HAC) | Use one of the following:<br>Process Knowledge or<br>Calculations based on radionuclide composition. | Meets criterion.<br>Calculated dose rates:<br>See Table 3-9 |
| Removable surface contamination (RH-TRU 72B and CNS 10-160B) | Limits from Table 2.2 of DOE-STD-1098-99, *Radiological Control*, must be met.<br>< 20 disintegrations per minute (dpm)/ 100 $cm^2$ for alpha<br><br>< 200 dpm/ 100 $cm^2$ for beta-gamma<br><br>Fixing of surface contamination is prohibited. | Records of surface contamination surveys taken on RH-TRU waste payload containers prior to release from a radiological contamination area.<br><br>Records of surface contamination surveys taken of cask prior to shipment. | Meets Criterion.<br>Surface contamination will be removed. |

Table 3-6. (continued).

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| Radionuclide composition (RH-TRU 72B and CNS 10-160B) | Description on shipping papers of radionuclide composition comprising 95% or more of the radioactive hazard must be reported in the WWIS for each payload container.<br><br>Curie content in each payload container will be reported for tracking total repository curie inventory. | Process Knowledge<br>If PK is not available, use one or more of the following methods:<br>• Nondestructive Analysis<br>• Radiochemical assay<br>• Material Accountability & Tracking (Process Knowledge)<br>• Gamma dose measurement<br>• Waste management database (Process Knowledge)<br>Applied to one of the following:<br>1. payload container<br>2. packages in a payload container<br>3. sample of a waste stream.<br>Report in the WWIS. | Meets Criterion.<br><br>Process knowledge will be used. |
| Pu-239 fissile gram equivalent (FGE) (RH-TRU 72B) | ≤ payload container limit identified in the RH-TRU 72-B Waste Shipping Package SAR (<325 grams FGE for the 72-B Cask).<br><br>≤ shipping cask limit identified in the RH-TRU 72-B Waste Shipping Package SAR.*<br><br>*These limits apply to the calculated FGE value plus its associated propagated error expressed as one standard deviation. | Use radionuclide composition information and calculate in accordance with Sec. 9.3 of App. 1.3.7 of the SAR. | Meets criterion.<br><br>Calculated $^{239}$Pu FGE per waste container:<br>See Table 3-10 |
| Fissile materials (CNS 10-160B) | Not to exceed mass limits of 10 CFR §71.53. | Use radionuclide composition information to calculate in accordance with 10 CFR §71.53. | Not Applicable.<br>Project will use RH-TRU 72 B Cask |
| Pu-239 equivalent activity (PE-Ci) (RH-TRU 72B) | ≤ 80 PE-Ci/ RH-TRU 72-B canister if waste is direct loaded.<br><br>≤ 240 PE-Ci/ RH-TRU 72-B canister if waste is loaded into three 30-gallon or 55-gallon drums prior to placement in the RH canister. | Use radionuclide composition information and calculate in accordance with methodology described in Appendix A of WIPP RH TSR, Sec. 5.9.12. | Meets criterion.<br><br>Calculated PE-Ci per waste container:<br>See Table 3-11 |

Table 3-6. (continued).

| Waste Attribute | Waste Acceptance Criterion | Methods of Compliance | Solidified SBW |
|---|---|---|---|
| TRU alpha activity concentration (RH-TRU 72B and CNS 10-160B) | > 100 nanocuries of alpha-emitting TRU isotopes with half lives > 20 years per gram of waste (nCi/g)<br><br>This limit applies to the calculated TRU alpha concentration without its associated propagated error. | Use radionuclide composition information and calculate the nCi/g value for the payload container. TRU alpha concentration determination may be performed on a waste stream basis and reported on a container basis.<br><br>Report in the WWIS. | Meets criterion.<br><br>Calculated TRU per waste container:<br>See Table 3-12 |
| Radionuclide activity (RH-TRU 72B) | ≤ 23 Curies per liter averaged over the volume of the RH-TRU 72-B canister (Ci/l)*<br>* This limit applies to the total activity of the RH canister. | Use radionuclide composition information and calculate the Ci/l value for the RH canister.<br><br>Report in the WWIS. | Meets criterion.<br><br>Calculated activity per waste container:<br>See Table 3-12 |
| Radionuclide activity (CNS 10-160B) | ≤ 20 curies of plutonium content for the CNS 10-160B cask*<br>*This limit applies to the total activity of the cask. | | Not Applicable.<br><br>Project plans to use RH-TRU 72 B Cask. |
| Decay heat (RH-TRU 72B) | ≤ 50 W/RH-TRU 72-B cask*<br>≤ decay heat limit per payload container, as specified in applicable content code<br>* These limits apply to the calculated decay heat value plus its associated error expressed as one standard deviation. | Acceptable methods for characterizing the isotopes (all but trace) necessary for calculation of decay heat are demonstrated using any of the following methods:<br>• Process Knowledge<br>• Nondestructive Analysis<br>• Radiochemical assay<br>• Material Accountability & Tracking<br>• Gamma dose measurement.<br>Trace isotopes (limited to 5%) may be demonstrated by Process Knowledge and waste management databases | Meets Criterion.<br><br>Calculated watts per waste container:<br>See Table 3-8. |
| Decay heat (CNS 10-160B) | ≤ 100 W/CNS 10-106B cask*<br>≤ decay heat limit per payload container, as specified in applicable content code<br>* These limits apply to the calculated decay heat value plus its associated error expressed as one standard deviation | | Not Applicable.<br><br>Project plans to use RH-TRU 72 B Cask. |
| Waste origin | Must be generated from defense-related activities | Process Knowledge | Meets Criterion. |

Table 3-7. Estimated Gas generation rates for treated SBW.

| Waste Stream | Rate. G-moles per second per canister (1999 Activity Values) |
|---|---|
| Dried heel solids (blended with sand) | 4.54E-09 |
| Calcine | 1.18E-09 |
| CH TRU Grout Process CST (blended with sand) | 4.04E-09 |
| Direct Verification Glass | 1.90E-09 |
| Steam Reforming Primary solids | 1.57E-09 |
| Direct Grouting Grout | 1.78E-08 |
| CH TRU Grout Process Grout (3 drums/canister) | 8.31E-09 |

Table 3-8. Dose rate and decay heat.

| Process | Waste Stream | Container | Decay Date | Dose-Rate, mR/hr Contact | Dose-Rate, mR/hr 1 Meter from Cask | Heat Generation, Watts |
|---|---|---|---|---|---|---|
| Solids Drying | Dried SBW Heel Solids | Canister | 7/1999 | 129,000 | | 5.5 |
| | | | 7/2016 | 77,800 | | 3.7 |
| | | | 7/2032 | 53,600 | | 2.6 |
| | Dried SBW Heel Solids | 72-B Cask | 7/1999 | 1.23 | 0.41 | 5.5 |
| | | | 7/2016 | 0.30 | 0.10 | 3.7 |
| | | | 7/2032 | 0.14 | 0.05 | 2.6 |
| NWCF (calcination) | Calcine | Canister | 7/1999 | 45,720 | | 1.5 |
| | | | 7/2016 | 26,040 | | 1.0 |
| | | | 7/2032 | 17,920 | | 0.7 |
| | Calcine (heel solids included in calciner feed) | Canister | 7/1999 | 47,100 | | 1.56 |
| | Calcine (heel solids included in calciner feed) | 72-B Cask | 7/1999 | 0.64 | 0.21 | 1.56 |
| | Spent Carbon | 55-gal Drum | 7/1999 | 0.00 | | 0.0000043 |
| | | | 7/2016 | 0.00 | | 0.0000018 |
| | | | 7/2032 | 0.00 | | 0.0000008 |
| UNEX (separations) | Glass | Canister | 7/1999 | 1,130,000 | | 88.4 |
| | | | 7/2016 | 754,000 | | 59.9 |
| | | | 7/2032 | 520,000 | | 41.9 |
| | Stabilized Solvent | 55-gal Drum | 7/1999 | 0.57 | | 0.000002 |
| | | | 7/2016 | 0.33 | | 0.000001 |
| | | | 7/2032 | 0.23 | | 0.000001 |
| | Grout | 55-gal Drum | 7/1999 | 93 | | 0.000324 |
| | | | 7/2016 | 29 | | 0.000111 |
| | | | 7/2032 | 17 | | 0.000067 |
| CH-TRU Grout | Grout | 55-gal Drum | 7/1999 | 190 | | 0.056347 |
| | | | 7/2016 | 42 | | 0.038323 |
| | | | 7/2032 | 14 | | 0.026997 |
| | CST | Canister | 7/1999 | 1,299,000 | | 23.9 |
| | | | 7/2016 | 747,900 | | 14.4 |
| | | | 7/2032 | 516,300 | | 9.9 |
| | CST | 72-B Cask | 7/1999 | 13 | 4 | 23.9 |
| | | | 7/2016 | 2 | 0.5 | 14.4 |
| | | | 7/2032 | 1 | 0.4 | 9.9 |
| | CST (mixed with sand) | Canister | 7/1999 | 250,900 | | 4.9 |
| | | | 7/2016 | 144,300 | | 2.9 |
| | | | 7/2032 | 99,590 | | 2.0 |
| Direct Vit | Glass | Canister | 7/1999 | 34,700 | | 2.4 |
| | | | 7/2016 | 19,720 | | 1.6 |
| | | | 7/2032 | 13,570 | | 1.1 |
| | Grout | 55-gal Drum | 7/1999 | 5 | | 0.0005966 |
| | | | 7/2016 | 3 | | 0.0005420 |
| | | | 7/2032 | 3 | | 0.0005101 |
| | Carbon | 55-gal Drum | 7/1999 | 0.00 | | 0.0000047 |
| | | | 7/2016 | 0.00 | | 0.0000019 |
| | | | 7/2032 | 0.00 | | 0.0000008 |
| Steam Reforming | Primary Solids | Canister | 7/1999 | 56,510 | | 1.9 |
| | | | 7/2016 | 32,210 | | 1.2 |
| | | | 7/2032 | 22,100 | | 0.9 |
| | Primary Solids (blended with heel solids in process feed) | Canister | 7/1999 | 57,700 | | 2.1 |
| | Primary Solids (blended with heel solids in process feed) | 72-B Cask | 7/1999 | 0.74 | 0.2 | 2.1 |
| | Carbon | 55-gal Drum | 7/1999 | 0.00 | | 0.000004 |
| | | | 7/2016 | 0.00 | | 0.000002 |
| | | | 7/2032 | 0.00 | | 0.000001 |
| TRUEX (separations) | Stabilized Solvent | 55-gal Drum | 7/1999 | 0.0397 | | 0.000002 |
| | | | 7/2016 | 0.0138 | | 0.000001 |
| | | | 7/2032 | 0.0064 | | 0.000001 |
| | Glass | Canister | 7/1999 | 36,260 | | 3.3 |
| | | | 7/2016 | 11,110 | | 2.0 |
| | | | 7/2032 | 7,501 | | 1.4 |
| | Grout | 55-gal Drum | 7/1999 | 8,418 | | 0.065918 |
| | | | 7/2016 | 4,835 | | 0.041886 |
| | | | 7/2032 | 3,335 | | 0.028748 |
| Direct Grouting | Grout | Canister | 7/1999 | 8,209 | | 0.360080 |
| | | | 7/2016 | 4,662 | | 0.232080 |
| | | | 7/2032 | 3,207 | | 0.161070 |

Table 3-9. Summary of hypothetical accident conditions (HAC) calculations for treated SBW.

| Material | HAC Value |
|---|---|
| Calcine | 0.126 |
| Dried Heel Solids | 0.502 |
| CST (from CH-TRU Grout Process) | 0.784 |
| Direct Vitrification Glass | 0.199 |
| Steam Reforming Primary Solids | 0.156 |
| Direct Grout | 0.029 |
| Grout (from CH-TRU Grout Process) | 0.003 |

Table 3-10. Summary of fissile gram equivalents per container for treated SBW.

| Material | Fissile Gram Equivalents |
|---|---|
| Calcine | 26.62 |
| Dried Heel Solids | 111.32 |
| CST (from CH-TRU Grout Process) | 0.01 |
| Direct Vitrification Glass | 41.86 |
| Steam Reforming Primary Solids | 32.94 |
| Direct Grout | 6.20 |
| Grout (from CH-TRU Grout Process) | 1.72 |

Table 3-11. Summary of Pu-239 activity for treated SBW.

| Material | Pu-239 Activity, Ci |
|---|---|
| Calcine | 1.880 |
| Dried Heel Solids | 14.001 |
| CST (from CH-TRU Grout Process) | 0.003 |
| Direct Vitrification Glass | 2.956 |
| Steam Reforming Solids | 2.326 |
| Direct Grout | 0.438 |
| Grout (from CH-TRU Grout Process) | 0.122 |

Table 3-12. Radionuclide content per waste container.

| Process -> | | NWCF (calcination) | CH-TRU Grout | | Direct Vit | Steam Reforming | Direct Grouting |
|---|---|---|---|---|---|---|---|
| Waste Stream -> | Dried SBW Heel Solids | Calcine | Grout | CST | Glass | Primary Solids | Grout |
| Material Type | Dried Granular Solids (salts and oxides) | Dried Granular Solids (oxides) | Grout | Dried Granular Solids (see note 2) | Glass | Dried Granular Solids (oxide) | Grout |
| Material Specific Gravity | 1.6 | 1.2 | 1.65 | 1.5 | 2.6 | 1.2 | 1.6 |
| Total Waste Stream Volume, $m^3$ | 81 | 1,200 | 4,600 | 21 | 678 | 970 | 5,160 |
| Type of Container (see notes) | Canister | Canister | 55-gallon Drum | Canister | Canister (see note 5) | Canister | Canister |
| # of Containers | 101 | 1,500 | 23,000 | 207 | 955 | 1,213 | 6,450 |
| Activity in Each Container, Ci: | | | | | | | |
| Am-241 | 6.34E-01 | 1.87E-01 | 1.21E-02 | 9.00E-06 | 2.94E-01 | 2.31E-01 | 4.35E-02 |
| Am-243 | 2.10E-04 | 5.40E-05 | 3.50E-06 | 0.00E+00 | 8.49E-05 | 6.68E-05 | 1.26E-05 |
| Cm-242 | 1.22E-04 | 3.14E-05 | 2.03E-06 | 0.00E+00 | 0.00E+00 | 3.89E-05 | 7.32E-06 |
| Cm-244 | 1.94E-02 | 5.00E-03 | 3.24E-04 | 0.00E+00 | 7.86E-03 | 6.19E-03 | 1.16E-03 |
| Np-237 | 1.48E-03 | 6.93E-03 | 4.49E-04 | 1.37E-06 | 1.09E-02 | 8.57E-03 | 1.61E-03 |
| Pu-238 | 1.31E+01 | 1.60E+00 | 1.04E-01 | 2.67E-03 | 2.52E+00 | 1.98E+00 | 3.72E-01 |
| Pu-239 | 1.17E+00 | 1.85E-01 | 1.20E-02 | 1.93E-04 | 2.91E-01 | 2.29E-01 | 4.30E-02 |
| Pu-240 | 1.01E-01 | 2.49E-02 | 1.61E-03 | 2.90E-05 | 3.91E-02 | 3.08E-02 | 5.79E-03 |
| Pu-241 | 1.03E+01 | 9.99E-01 | 6.47E-02 | 1.93E-03 | 1.57E+00 | 1.24E+00 | 2.33E-01 |
| Pu-242 | 7.61E-05 | 4.21E-05 | 2.73E-06 | 9.61E-08 | 6.62E-05 | 5.21E-05 | 9.80E-06 |
| Th-230 | 7.68E-06 | 1.98E-06 | 1.28E-07 | 0.00E+00 | 3.11E-06 | 2.45E-06 | 4.61E-07 |
| U-232 | 1.86E-05 | 4.80E-06 | 3.11E-07 | 0.00E+00 | 7.55E-06 | 5.94E-06 | 1.12E-06 |
| U-233 | 3.10E-07 | 8.00E-08 | 5.18E-09 | 0.00E+00 | 1.26E-07 | 9.90E-08 | 1.86E-08 |
| U-234 | 5.84E-03 | 2.15E-03 | 1.39E-04 | 9.62E-07 | 3.38E-03 | 2.66E-03 | 5.00E-04 |
| U-235 | 2.02E-04 | 5.15E-05 | 3.33E-06 | 1.96E-08 | 8.09E-05 | 6.37E-05 | 1.20E-05 |
| U-236 | 3.45E-04 | 9.48E-05 | 6.14E-06 | 9.13E-09 | 1.49E-04 | 1.17E-04 | 2.21E-05 |
| U-238 | 4.00E-05 | 4.15E-05 | 2.68E-06 | 2.32E-09 | 0.00E+00 | 5.13E-05 | 9.65E-06 |
| | | | | | | | |
| Ba-137m | 4.97E+02 | 1.28E+02 | 8.29E-03 | 9.26E+02 | 2.01E+02 | 1.58E+02 | 2.98E+01 |
| Ce-144 | 1.40E-01 | 2.99E-02 | 1.93E-03 | 0.00E+00 | 4.70E-02 | 3.70E-02 | 6.95E-03 |
| Co-60 | 3.60E-01 | 1.67E-01 | 1.08E-02 | 3.54E-05 | 2.63E-01 | 2.07E-01 | 3.89E-02 |
| Cs-134 | 6.63E+00 | 2.74E-01 | 1.78E-05 | 1.99E+00 | 4.31E-01 | 3.39E-01 | 6.38E-02 |
| Cs-135 | 8.54E-03 | 2.20E-03 | 1.42E-07 | 1.59E-02 | 3.46E-03 | 2.72E-03 | 5.12E-04 |
| Cs-137 | 5.25E+02 | 1.35E+02 | 8.76E-03 | 9.79E+02 | 2.13E+02 | 1.67E+02 | 3.15E+01 |
| Eu-152 | 2.87E-02 | 7.40E-03 | 4.79E-04 | 0.00E+00 | 1.16E-02 | 9.16E-03 | 1.72E-03 |
| Eu-154 | 1.19E+00 | 5.57E-01 | 3.61E-02 | 3.26E-05 | 8.76E-01 | 6.89E-01 | 1.30E-01 |
| Eu-155 | 2.56E+00 | 4.53E-01 | 2.93E-02 | 0.00E+00 | 7.12E-01 | 5.60E-01 | 1.05E-01 |
| Pm-147 | 4.11E+00 | 1.06E+00 | 6.86E-02 | 0.00E+00 | 1.67E+00 | 1.31E+00 | 2.47E-01 |
| Pr-144 | 1.40E-01 | 0.00E+00 | 2.33E-03 | 0.00E+00 | 5.66E-02 | 0.00E+00 | 8.38E-03 |
| Ni-63 | 3.10E-01 | 1.11E-01 | 7.22E-03 | 0.00E+00 | 1.75E-01 | 1.38E-01 | 2.60E-02 |
| Ru-106 | 1.01E-01 | 2.26E-02 | 1.46E-03 | 0.00E+00 | 3.56E-02 | 2.80E-02 | 5.27E-03 |
| Sb-125 | 4.97E+01 | 1.12E-01 | 7.25E-03 | 2.87E-03 | 1.76E-01 | 1.39E-01 | 2.61E-02 |
| Sm-151 | 3.34E+00 | 8.60E-01 | 5.57E-02 | 0.00E+00 | 1.35E+00 | 1.06E+00 | 2.00E-01 |
| Sr-90 | 3.64E+02 | 1.19E+02 | 7.68E+00 | 3.70E-01 | 1.87E+02 | 1.47E+02 | 2.76E+01 |
| Tc-99 | 1.79E+00 | 2.81E-02 | 1.82E-03 | 1.52E-04 | 4.28E-02 | 3.48E-02 | 6.54E-03 |
| Y-90 | 3.64E+02 | 1.19E+02 | 7.68E+00 | 3.70E-01 | 1.87E+02 | 1.47E+02 | 2.76E+01 |
| H-3 | 3.55E-02 | 5.87E-02 | 3.80E-03 | 0.00E+00 | 0.00E+00 | 7.27E-02 | 1.37E-02 |
| I-129 | 5.12E-04 | 1.65E-04 | 1.07E-05 | 2.27E-05 | 1.37E-04 | 2.04E-04 | 3.84E-05 |
| | | | | | | | |
| TRU Activity Concentration, nCi/gram | 11,700 | 2,087 | 393 | See note 2 | 1,706 | 2,582 | 364 |
| Activity in Container, total Ci/liter of waste | 2.31 | 0.63 | 0.08 | 2.38 | 1.00 | 0.78 | 0.15 |

Notes:

1. A canister is assumed to be a 72-B container. Design information for the inner container and the cask are available in the SAR at: http://www.wipp.carlsbad.nm.us/library/RHsar/rhsar/01_03_04.pdf

2. Because of the high activity concentration in waste CST, only a limited amount of the material may be placed in a canister and meet requirements for transportation and disposal at WIPP. The activity figures for waste CST in this table reflect the activity **per canister.** It is expected that small vessels containing the waste CST would be placed in canisters along with an appropriate packing material. The activity concentrations in the CST itself will be about 7.9 times higher than the values shown in this table for CST. **CST is expected to be a TRU waste.**

3. A 55-gallon drum is assumed to contain 53 gallons (0.2 m3) of material. The rest is void space.

4. Activity levels have a base date of July, 1999

5. 72-B Canister Specs: L=10 ft. 1 inch; ID = 26 inches; Carbon Steel construction; wall thickness = 0.25 in.

6. For waste in canisters going to WIPP, it is assumed that material occupies 0.8 cubic meters. The exception to this is Direct Vit glass waste, which, because of weight limits, must be contain no more than 0.71 cubic meters. Other glass canisters, not going to WIPP, are based on 0.8 cubic meters per canister.

# 4. MANAGEMENT CONTROL SYSTEMS

This chapter summarizes the management control systems applicable to the INEEL WIR evaluation process.[i] The management controls are those systems that ensure that both the primary project objectives and an optimum margin of safety for protection of personnel, the public, and the environment are met. The following elements are addressed:

- Organizations and responsibilities
- Procedures
- Quality assurance
- Document and record control
- Training and qualifications.

These elements were adopted by DOE (1999d) as good practices for performing and documenting WIR determinations. The management controls implemented by the field element managers for these elements ensure that the following requirements applicable to the WIR determination process are met at the INEEL.

- WIR determinations are made by either the *citation* or *evaluation* process described in Chapter II of DOE M 435.1-1 [DOE 1999c, I.2.F.(18)].
- The Office of Environmental Management (DOE-EM) is consulted for WIR determinations using the evaluation process [DOE 1999c, I.2.F.(18)].
- WIR determinations by the evaluation process must be developed under good record-keeping practices, with an adequate quality assurance process, and are documented to support the determinations [DOE 1999c, II.B.(2)].

The management controls also ensure appropriate consultation with the NRC staff for WIR determinations using the evaluation process [DOE 1999d, II.B.]

## 4.1 Organization and Responsibilities

This section describes the DOE-ID and management and operating contractor (M&O) organizations responsible for developing, reviewing, and approving WIR determinations. The relationships between the DOE-ID and M&O with organizations outside of INEEL are also discussed.

### 4.1.1 DOE-Idaho Operations Office

The DOE-ID organization as it relates to the WIR determination development and approval process is illustrated in Figure 4-1. The shaded positions in the figure are those with organizational responsibilities for portions of the WIR determination process, as described below.

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i. Organizational structures of both DOE and the M&O Contractor are constantly changing. This document presents the organizations at the time this WIR document was published.

The manager of DOE-ID is the approval authority for all INEEL WIR determinations. The manager of DOE-ID may delegate the authority and responsibility for the WIR determination to the assistant manager for Environmental Management. Such delegation must be documented in accordance with DOE M 435.1-1, I.1.A. DOE retains ultimate responsibility for compliance with the conditions of DOE O 435.1, including decisions related to managing residual wastes to meet the criteria defined in the WIR determination process. Additionally, the manager of DOE-ID is responsible for ensuring consultation and coordination with the DOE-EM, for waste determined to be incidental to reprocessing through the "evaluation process," to ensure consistency across the DOE complex.[j]

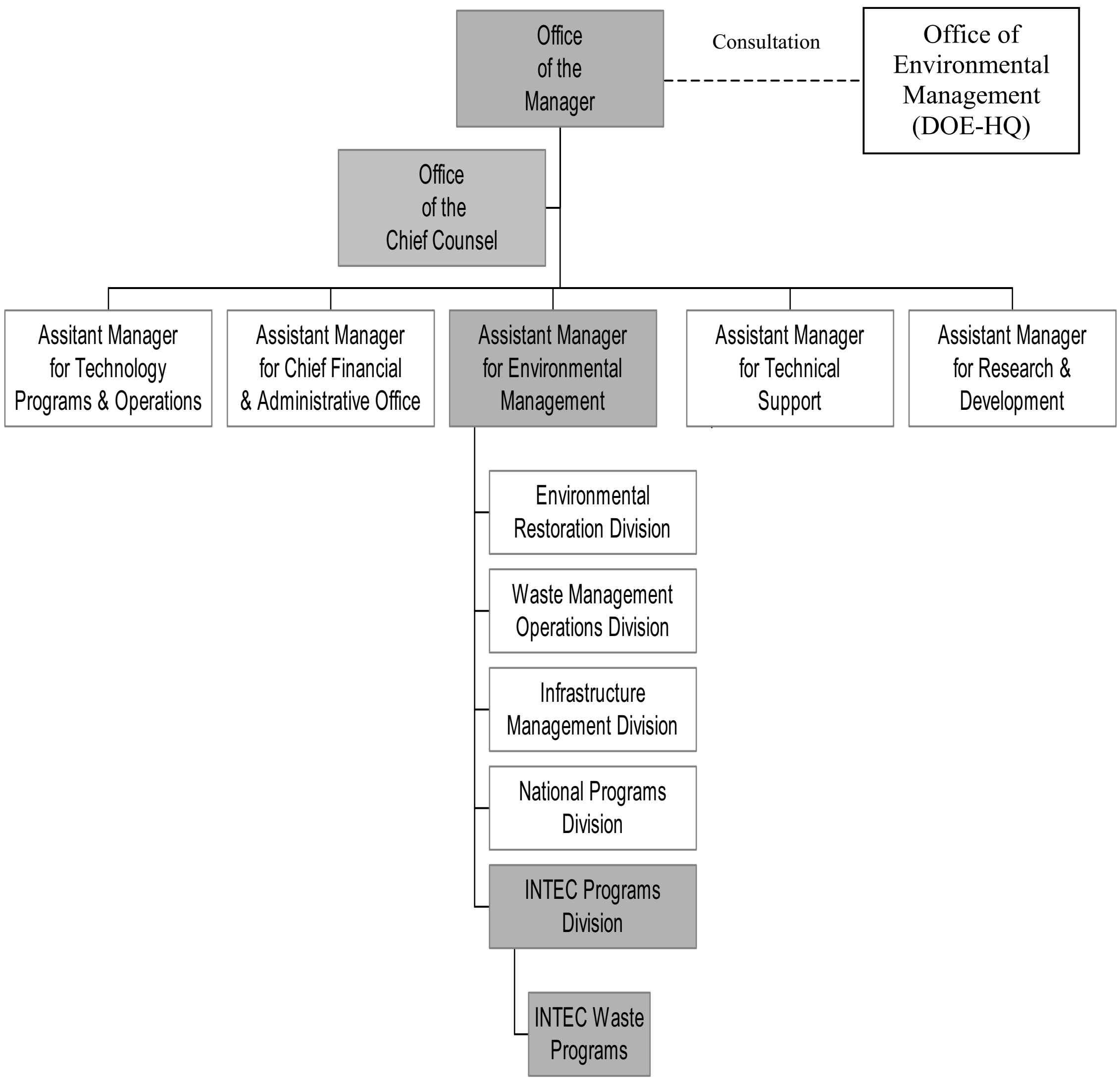


Figure 4-1. DOE-Idaho Operations Office.

j. DOE 1999d, Section I, 2F, (18).

The responsibilities of the assistant manager for Environmental Management include the overall execution of environmental restoration activities, waste management operations, infrastructure management, INTEC programs, and national programs at the INEEL. With regard to WIR determination activities, the assistant manager for Environmental Management is responsible for ensuring coordination and integration of WIR determination activities across DOE-ID waste management programs. The assistant manager for Environmental Management is responsible for determining to what extent the NRC will be involved in the review of WIR determinations. The assistant manager for Environmental Management is also responsible for coordination and consultation with DOE-EM on WIR determination decisions. The assistant manager for Environmental Management and staff provide management direction and oversight of M&O performance associated with WIR determination development.

The Field Office Chief Counsel reviews proposed WIR determinations and provides legal counsel as requested by the manager of DOE-ID. The Field Office Chief Counsel also acts as the interface with the DOE-Headquarters, Office of General Counsel.

The manager, INTEC Waste Programs, is responsible for preparing and submitting the WIR determination (including any necessary revisions thereto) and supporting analyses, timely response to communications and inquiries from Headquarters, and other interface support activities.

### 4.1.2 INEEL Management & Operations

The Bechtel BWXT Idaho, LLC (BBWI) M&O organization as it relates to the development and approval of the WIR determination is illustrated in Figure 4-2. The shaded positions in the figure are those with organizational responsibilities for portions of the WIR determination process, as described below.

The M&O president and general manager is responsible for overall management of contractor activities and is accountable for complying with the INEEL M&O contract conditions, including development of an M&O quality assurance program. The general manager is ultimately responsible to DOE-ID for ensuring the WIR determination has been developed in accordance with all applicable requirements, and that the information is true, accurate, and complete.

The vice president, Environmental Management Programs, is responsible for ensuring integration of the HLW Program WIR activities with the other INEEL Waste Management Programs, including LLW and TRU Waste Programs.

The manager of projects for the High-Level Waste Program ensures that cost-effective and fully compliant nuclear operations programs are in-place and operating at the INEEL. This person also has the authority, responsibility, and accountability for establishing and maintaining the necessary programs and procedures to ensure consistent implementation of the WIR determination process. INTEC TFF closure activities and the WIR determination, including any supporting analyses, performance assessments, and sampling activities, are the responsibility of the High-Level Waste manager of projects.

The Tank Farm Project develops WIR determination reports in accordance with the plan, *Performing Waste Incidental to Reprocessing Determination* (BBWI 2002). The Tank Farm Project organization selects and supervises appropriate subcontractors as required for developing the WIR determination report. The Tank Farm Project organization also develops the WIR determination report review criteria and chairs the review team.

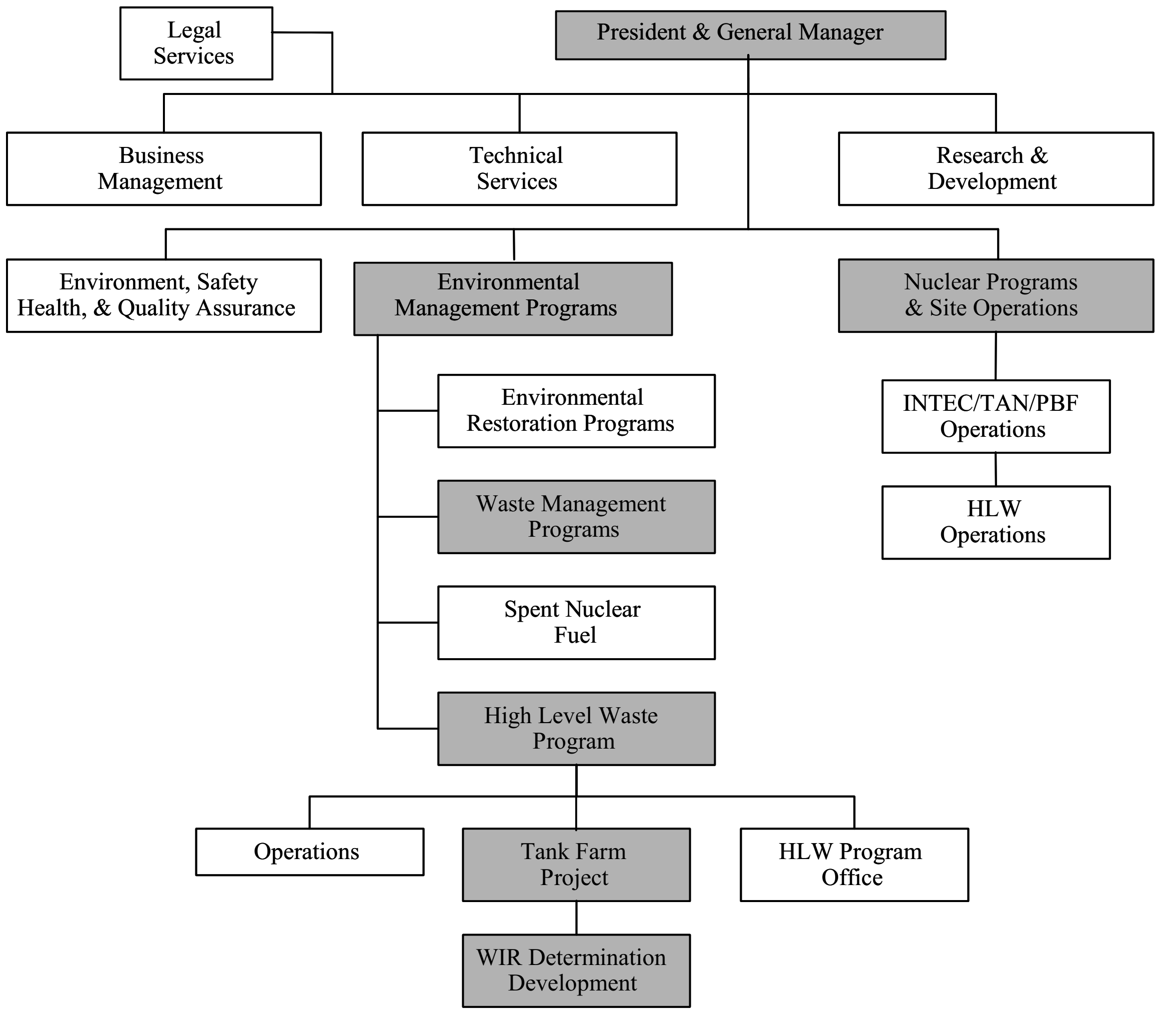


Figure 4-2. Management & Operating Contractor organization.

The manager of projects for the Waste Management Program ensures that cost-effective and fully compliant waste management programs are in-place and operating at the INEEL. This person also has the authority, responsibility, and accountability for establishing and maintaining the systems, policies, and procedures for waste management and packaging and transportation for the INEEL.

The vice president of Nuclear Programs and Site Operations is responsible for all INTEC HLW operations and provides review of the WIR determination for all INTEC facilities.

The M&O uses INEEL site personnel and subcontractors as required to accomplish TFF closure and development of the supporting analyses, including the WIR determination reports. The M&O has established technical and administrative requirements for its subcontractors through procurement subcontracts.

### 4.1.3 WIR Determination Development Interfaces

In accordance with the Idaho Operations Office Order on waste incidental to reprocessing, the SBW WIR determination report was developed in coordination with various external organizations. The following discusses the interfaces and technical support provided by these organizations.

***4.1.3.1 DOE, Office of Environmental Management (DOE-EM).*** DOE M 435.1-1 states that field element managers are responsible for ensuring consultation and coordination with DOE-EM for WIR determinations using the evaluation process. To meet this requirement, DOE-ID has involved DOE-EM staff in the development and review of the SBW WIR determination report. The methodology for this report was reviewed by DOE-EM prior to report preparation. Comments provided by DOE-EM staff have been resolved and incorporated into the document as appropriate.

***4.1.3.2 U. S. Nuclear Regulatory Commission (NRC).*** The NRC has statutory authority for the licensing of facilities authorized for permanent storage or disposal of HLW after 1972 and the passage of the Energy Reorganization Act. The WIR determination process, described in DOE M 435.1-1, was established in close coordination with the NRC staff. The DOE WIR process is consistent with the guidance provided in previous NRC reviews of WIR determinations. Experience and lessons-learned from the NRC review of the WIR determinations developed by the Hanford and Savannah River Site have been incorporated into the INEEL SBW WIR determination report.

The NRC does not have regulatory authority or jurisdiction over the SBW at the INEEL; however, DOE requested and received NRC independent technical review and comment on the SBW WIR determination report in accordance with DOE M 453.1-1 guidance (see Appendix C).

## 4.2 Procedures

The SBW WIR determination report was developed using formal processes and methods. Existing INEEL policies, programs, and procedures were used to manage and implement many of the INEEL activities that support the WIR determination process. Implementing documents and procedures were also used for tank inventory sampling, data collection, analyses, and other activities performed in support of the WIR determination process. This section discusses the key documents used for the management and performance of the WIR determination activities.

The INEEL Document Management Control System provides written instructions for preparing, reviewing, approving, maintaining, and distributing documents and changes to documents. The Document Management Control System applies to controlled documents that are developed to prescribe processes, specify requirements, and establish design as it relates to Tank Farm closure activities, including development of the SBW WIR determination report. The various controlled documents pertaining to the SBW WIR determination report are described below.

As shown in Figure 4-3, the flow down of requirements that are applicable to the WIR process is traceable to several higher-tier documents:

- DOE O 435.1, Radioactive Waste Management (DOE 1999b), and its supporting documents, DOE M 435.1-1, Radioactive Waste Management Manual (DOE 1999c), and DOE G 435.1-1, Implementation Guide for use with DOE M 435.1-1 (DOE 1999d), set forth the requirements and guidance applicable to WIR determinations.

- ID O 435.A, Waste Incidental to Reprocessing (DOE 2001a), provides specific requirements for WIR determination processes applicable to DOE-ID and the INEEL.

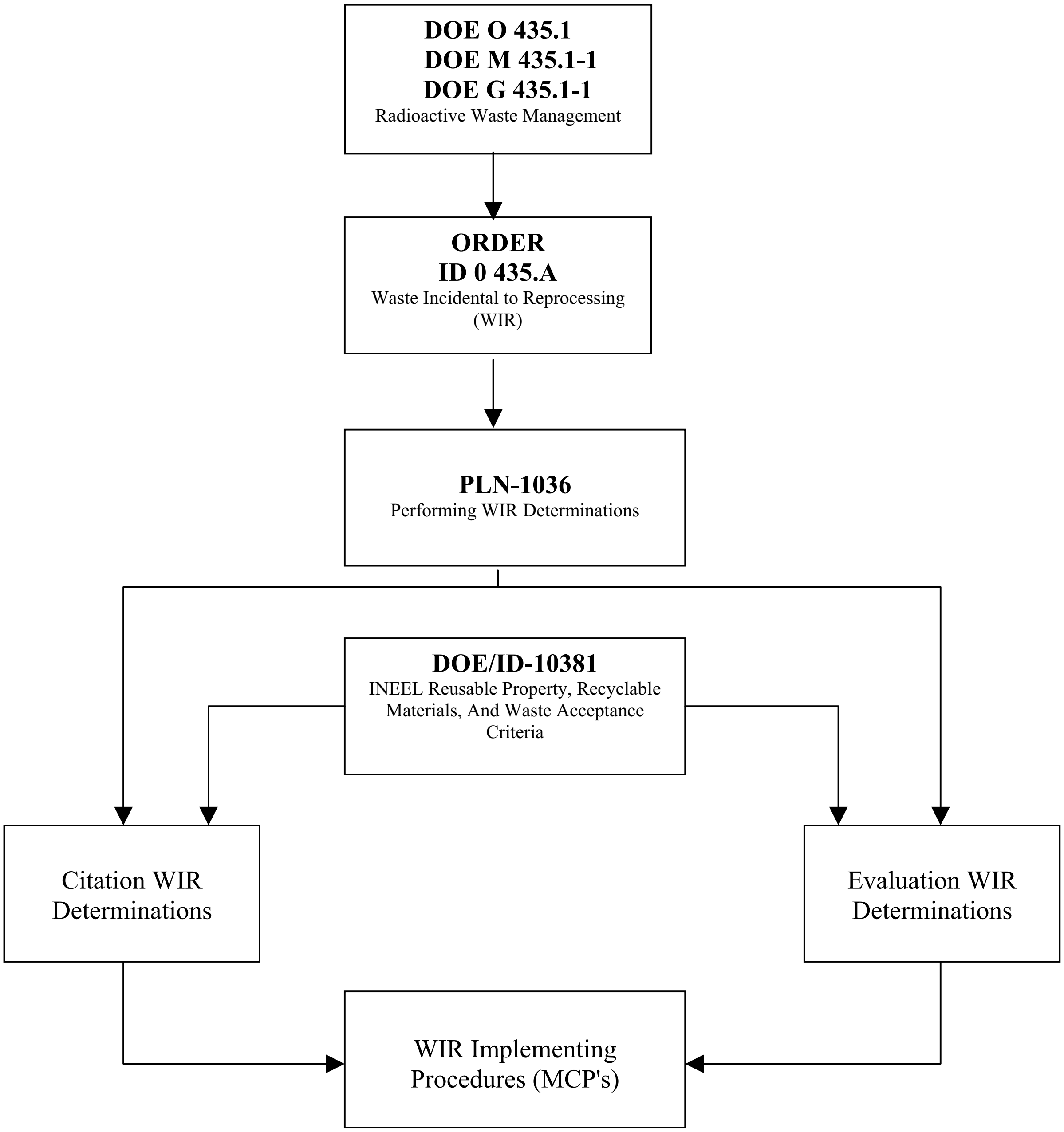


Figure 4-3. Document hierarchy.

- DOE/ID-10381, INEEL Reusable Property, Recyclable Materials, and Waste Acceptance Criteria (RRWAC) (DOE 2001b) compiles DOE-ID requirements for characterizing, packaging, and documenting reusable property, recyclable materials, and waste to be received by INEEL. The scope of the RRWAC includes requirements applicable to the following radioactive waste classifications: LLW, TRU waste, HLW. The RRWAC also specifies requirements for identifying and managing hazardous and nonhazardous wastes under the Solid Waste Act and RCRA. The RRWAC requires that each generator of radioactive waste provide assurance that appropriate sections of the acceptance criteria and applicable requirements are met.

- PLN-1036, Performing Waste Incidental to Reprocessing Determination (BBWI 2002), describes the methods used by the INEEL M&O Contractor to ensure waste that is incidental to reprocessing is properly classified as either TRU waste or LLW. The applicable elements necessary for waste classification are included in PLN-1036.

- Management Control Procedures are controlled, implementing documents that prescribe administrative processes to be performed to support Tank Farm closure and development of the WIR determination reports.

## 4.3 Quality Assurance

Pursuant to DOE O 435.1 (DOE 1999b), Chapter 1, General Requirements and Responsibilities, DOE and its contractors shall develop and maintain a Quality Assurance (QA) program for radioactive waste management facilities, operations, and activities that meet the requirements of 10 CFR 830, Subpart A, Quality Assurance Requirements, and DOE (1999a), as applicable.

The SBW WIR determination report was developed under a QA program that ensures the validity of the information used to make the determination. This section describes the QA programs applicable to the SBW WIR determination to ensure compliance with DOE M 435.1-1, Section II.B.(2). Figure 4-4 illustrates the relationship of the various QA programs and documents discussed below.

DOE-ID established QA requirements for the INEEL M&O through the INEEL M&O contract with BBWI. The INEEL M&O *Quality Assurance Program Requirements Documents* (QAPRD) (INEEL 2001a) describes the Quality Assurance Program of BBWI. DOE O 414.1A, *Quality Assurance* (DOE 1999a) and 10 CFR 830, Subpart A, *Quality Assurance Requirements*, which are the bases for this document. The QAPRD applies to M&O organizations responsible for achieving, maintaining, and verifying the quality of items and activities in support of facilities, programs, and projects. The QAPRD also applies to companies performing work for BBWI, as specified in procurement contracts.

Table 4-1 identifies the specific American Society of Mechanical Engineers NQA-1 elements (ASME 1997) that BBWI determined applicable to the WIR process. In addition to the WIR process QA requirements, the solidified SBW and associated waste certification documents will be evaluated against the final WIPP RH-TRU WAC to ensure appropriate QA program requirements are met for waste storage, transportation, and disposal.

Details for implementing the ten NQA-1 elements listed in Table 4-1 can be found in the M&O contractors *Quality Program Plan for the High Level Waste Program Office* (INEEL 2001c). It describes the quality assurance program for HLW activities managed by the INEEL HLW Program, including TFF closure and WIR determination activities. BBWI implements its QA program using a graded approach. The graded approach is implemented by the use of quality levels that identify the relative importance of

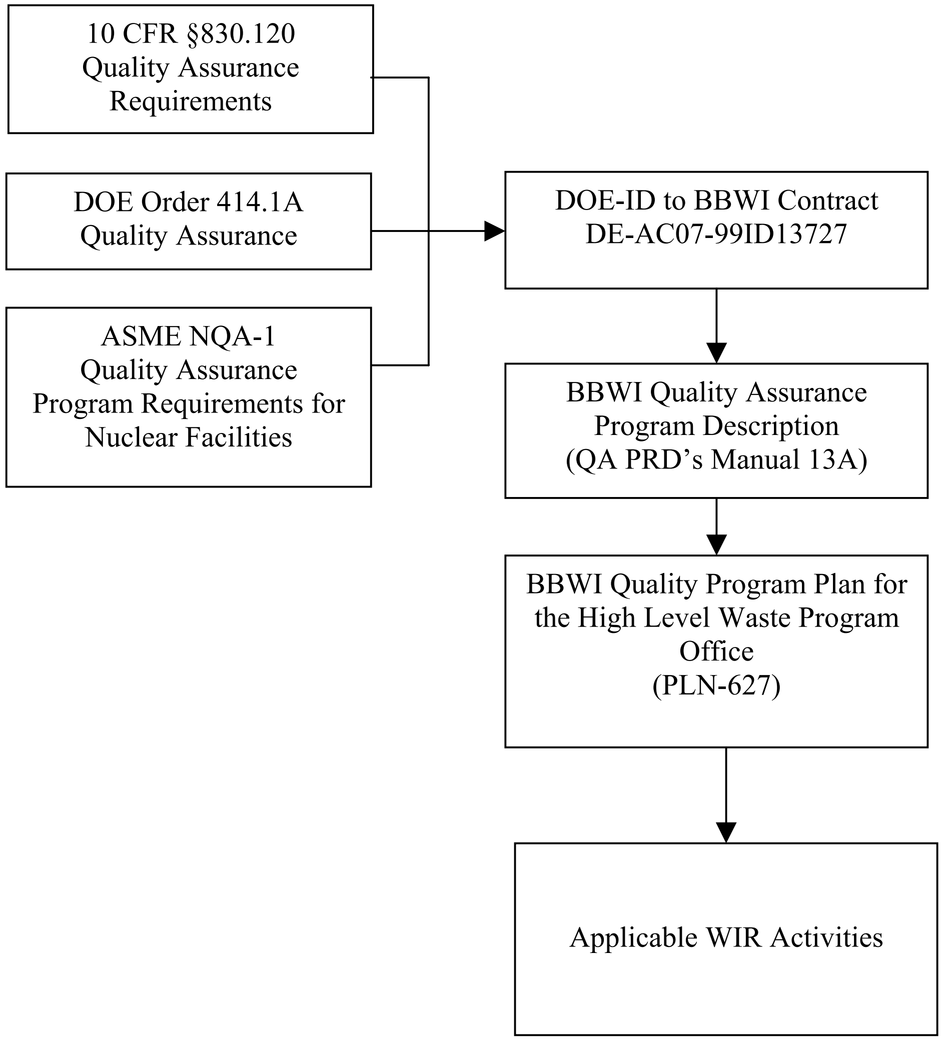


Figure 4-4. Quality Assurance program hierarchy.

Table 4-1. ASME NQA-1 applicability.

| ASME NQA-1 1997 Element No. | Description |
|---|---|
| 1 | Organization |
| 2 | Quality Assurance Program |
| 3 | Design Control |
| 4 | Procurement Document Control |
| 5 | Instructions, Procedures, and Drawings |
| 6 | Document Control |
| 7 | Control of Purchase Items and Services |
| 16 | Corrective Action |
| 17 | Quality Assurance Records |
| 18 | Audits |

an item or activity to the consequence of failure, should failure of the item or activity occur. A quality level list (Q-list) will be developed for the Tank Farm closure and SBW treatment projects that identifies the quality level assigned to items and activities in accordance with the *Quality Program Plan for the High-Level Waste Program Office*.

Management control procedures are used to facilitate implementation of the QA program's graded approach and assignment of quality levels to systems, structures, components, and activities.

## 4.4 Document and Record Control

Records management systems ensure that records important to safety and quality are generated, reviewed, approved, collected, and maintained. The management system provides controls so that records accurately reflect completed work and facility conditions and comply with applicable statutory or contractual requirements.

INEEL management control procedures incorporate the requirements of DOE O 200.1, *Information Management Program* (DOE 1996a), and DOE O 414.1A, *Quality Assurance* (DOE 1999a). Schedules for retention and disposition of records are in accordance with the General Records Schedule of the National Archives and Records Administration and other approved records schedules. Management Control Procedures include instructions for retention, protection, preservation, changes, traceability, accountability, and retrievability of records, and provide controls to ensure records are legible, accurate, complete, retrievable, and validated by authorized personnel.

Records are stored and maintained in a manner that minimizes the risk of damage, larceny, vandalism, or deterioration. Active records are not sent to records-holding facilities but are stored in a facility where the records may be readily accessed.

## 4.5 Training and Qualifications

The INEEL training program focuses on providing employees with the knowledge and skills necessary to perform tasks that meet acceptance criteria. Site training, with the assistance of subject matter experts, is responsible for analyzing, designing, developing, implementing, and evaluating training programs and processes. Management control procedures detail the instructional processes used, including self-study, computer or video-based training, instructor-led training, or on-the-job training.

DOE-ID INTEC Programs Division and the BBWI Waste Program Offices, define training and qualification requirements for selected positions or job categories by considering the level of knowledge and skills required to perform tasks. Training plans are developed to guide the development of skills and knowledge necessary for employees to meet requirements of specific job categories. Training and qualification requirements are established and periodically reviewed to ensure that requirements continue to reflect training needs.

INEEL management control procedures incorporate the requirements of DOE O 5480.20A, *Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities* (DOE 1994). Management control procedures describe personnel selection requirements and training, qualification, certification, and continued training processes. Procedures specify the frequency for which training is needed. DOE-ID INTEC Programs Division and the BBWI High-Level Waste Program office determine and document when personnel are suitably qualified to accomplish assigned tasks.

Personnel that perform WIR determination development, review, approval, and revision functions receive training in the applicable scope, purpose, and objectives of the WIR process and the specific quality assurance objectives of the assigned task before performing WIR process activities. Personnel also receive training on applicable implementing procedures used in the performance of the task.

This training includes appropriate subject material from the following documents:

- DOE O 435.1, Radioactive Waste Management (DOE 1999b)
- DOE M 435.1-1, Radioactive Waste Management Manual (DOE 1999c)
- DOE G 435.1-1, Implementation Guide for use with DOE M 435.1-1 (DOE 1999d)
- 10 CFR Part 61, Licensing Requirements for Land Disposal of Radioactive Waste, Subpart C Performance Objectives; and Subpart D, Technical Requirements for Land Disposal Facilities - 61.55 - Waste Classification
- ID O 435.A, Wastes Incidental to Reprocessing (DOE 2001a)
- DOE/ID-10381, INEEL Reusable Property, Recyclable Materials, and Waste Acceptance Criteria (RRWAC) (DOE 2001b)
- PLN-1036, Performing Waste Incidental to Reprocessing Determination (BBWI 2002)
- PLN-627, Quality Program Plan for the High-Level Waste Program Office. (BBWI 2001)

Training is documented and training records are maintained in accordance with INEEL management control procedures. Qualifications of BBWI personnel supporting the WIR development process are documented, and appropriate records are maintained. Subcontractors supporting tank closure and WIR determination activities are selected based on company and personnel qualifications and experience. All training records are available for review.

# 5. CONCLUSIONS AND BASIS FOR APPROVAL

It has been determined that sodium-bearing waste (SBW) meets the requirements of the *Radioactive Waste Management Manual*, Section II.B.2(b) (DOE 1999c), as waste incidental to reprocessing; that it meets criteria as TRU waste; and therefore will be managed and disposed of in accordance with DOE TRU waste requirements. This determination was based on the following criteria:

**Criterion 1.** The waste must have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical [DOE M 435.1-1, II(B)(2)(b)(1)].

- DOE M 435.1-1 provides flexibility for DOE to determine which radionuclides are important for meeting disposal-site performance objectives. Therefore, DOE uses a disposal site-specific risk-based approach for determining key radionuclides.

- The planned disposal location for INTEC sodium-bearing waste is the Waste Isolation Pilot Plant (WIPP) in New Mexico. The important radionuclides for WIPP performance objectives that account for most radionuclide release and therefore the most risk are Am-241, Pu-238, Pu-239, and Pu-240. These were evaluated as key radionuclides for meeting the SBW WIR determination criterion 1 requirements.

- The Idaho Nuclear Technology and Engineering Center segregated, removed, and converted the first-cycle extraction waste and most of the second- and third-cycle extraction waste (representing 96% of the key radionuclide curie inventory from reprocessing) to a stable solid waste form (calcine). It is planned that this solidified extraction waste will be further treated and disposed of as high-level radioactive waste.

- Additional key radionuclide removal from the remaining sodium-bearing waste would incur an additional $373 million and, depending upon the treatment process selected, up to $2.21 billion to remove about 3,000 curies. It was determined that the large expenditure for this relatively small reduction in radionuclide release (risk) was not economically practical.

**Criterion 2.** The waste will be incorporated in a solid physical form and meet alternative requirements for waste classification and characteristics, as the Department of Energy may authorize [DOE M 435.1-1, II(B)(2)(b)(2)].

- The Department of Energy plans to remove, solidify, and dispose of the sodium-bearing waste remaining in the 300,000-gallon storage tanks as mixed transuranic waste at the WIPP geologic repository. The solidified waste would meet WIPP waste acceptance criteria.

**Criterion 3.** The waste is managed pursuant to Department of Energy's authority under the *Atomic Energy Act of 1954*, as amended, in accordance with the provisions of Chapter III of Department of Energy Manual 435.1-1, as appropriate [DOE M 435.1-1, II(B)(2)(b)(3)].

- The solidified sodium-bearing waste would meet the waste acceptance criteria for the WIPP geologic repository as contact-handled and/or remote-handled mixed transuranic waste. Solidified sodium-bearing waste would be managed and disposed of as transuranic waste in accordance with DOE M 435.1-1. WIPP is a permitted disposal site for contact-handled mixed transuranic wastes and is expected to be permitted for remote-handled waste by 2003, long before sodium-bearing waste is shipped from the Idaho Nuclear Technology and Engineering Center.

- Disposal of the sodium-bearing waste as a mixed transuranic waste in the WIPP geologic repository would provide public health and safety protection and meet the applicable environmental protection standard of 40 CFR Part 191.

This WIR determination was made in accordance with the management and QA protocols described in Chapter 4. The mixed TRU waste will be managed in accordance with the provisions of applicable federal and state laws and DOE Orders. As recommended in DOE G 435.1-1, the Nuclear Regulatory Commission has provided a technical review of this waste-incidental-to-reprocessing determination document. The NRC recommendations (See Appendix C) have been incorporated into this document.

It was concluded that sodium-bearing waste meets the criteria for disposal at the WIPP geological repository as a transuranic waste.

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# Appendix A

# Summary of the Basis for the INTEC Radionuclide Inventory and Mass Balance

# Appendix A

# Summary of the Basis for the INTEC Radionuclide Inventory and Mass Balance

## INTRODUCTION

Inasmuch as the U.S. Department of Energy's (DOE) mission changed at the Idaho Nuclear Technology and Engineering Center (INTEC) from spent nuclear fuel (SNF) management through reprocessing for enriched uranium recovery and radioactive waste management to SNF storage and environmental remediation, more detailed inventory information has been required. A general discussion of how the inventories were derived is presented below.

Other than for a few radionuclides measured for process control, analytical inventories of radionuclides stored at INTEC are generally unavailable. Analytical methods were developed that facilitated measuring the concentration of the key process control parameters. However, no systematic effort was made to collect and retain analytical data more than a few years.

In 1993, a diligent effort was made to reconstruct information relative to past chemical analyses, waste volumes, and other pertinent information and to establish a reasonable pedigree for existing waste stored in the Calcined Solids Storage Facilities and liquid waste stored in the Tank Farm Facility (TFF). This effort generated information that gave scientists and engineers conservative estimates of radionuclide contents necessary for risk evaluations and for process and equipment designs. Refinements in estimates continue as additional information is gathered.

Data were gathered from published information such as reports and letters and from microfilm notes and log information. Process knowledge was obtained from the personal files of key employees and phone interviews with former employees. Reprocessing information was collected, and a fairly complete list of reprocessed fuels types was assembled.

An estimation method was developed that relied on fuel/waste type (aluminum, zirconium, electrolytic, etc.), a measured radionuclide concentration, and time of interest to generate a probable distribution of radionuclide concentrations. This method used the following assumptions:

1. *Radionuclides in INTEC wastes are the result of reprocessing operations.* This is a realistic assumption because there are only minor sources of radioactivity stored at INTEC other than reprocessing (these include contaminated water from INTEC fuel handling and storage operations and contaminated water from reactor operations and cleanup at other Idaho National Engineering and Environmental Laboratory [INEEL] facilities).

2. *Radionuclides were partitioned from usable uranium during first-cycle extraction.* It is realistic to assume that the bulk of the activity comes from first-cycle operations because the dissolution/extraction processes employed for uranium recovery removed a significant fraction of the fission products during first-cycle extraction. In fact, historical records indicate partitioning is very effective, as evidenced in the high decontamination factors, which average $7.4 \times 10^3$ for gross beta and $3.2 \times 10^3$ for gross gamma. These account for greater than 99% of the radioactivity. The uranium-bearing product stream from first-cycle extraction was sent to second- and third-cycle to remove plutonium contamination, after which the plutonium waste stream was sent to the tank farm. Because of this, the estimated quantity of plutonium in the sodium-bearing waste (SBW) is based upon actual tank samples rather than upon estimates from first-cycle extraction waste.

3. *Fuel inventories are represented by historical averages for the particular fuel type*. This is realistic because the bulk of the fuels processed came from a small number of reactor types with similar operating histories.

4. *Computer simulation using ORIGEN2 gives an accurate distribution of the radionuclides present in INTEC reprocessing waste.* ORIGEN2 is a nationally recognized code used to simulate reactor operating histories and decay processes. It provides reasonably accurate models of the radionuclide inventories present in the fuel types processed at INTEC.

5. *SBW waste can be modeled as a fuel type.* This is appropriate because almost all radionuclides present in the SBW came directly or indirectly from reprocessing operations at INTEC. The SBW model was built from weighted averages of the different fuel types processed.

6. *Cs-137 is an acceptable predictor for the movement of other radionuclides in INTEC waste processes.* This is a reasonable assumption because Cs-137 chemistry is not complicated. It has a high decontamination factor in first-cycle solvent extraction and is neither diluted nor concentrated preferentially in INTEC processes. Gross precipitation of radionuclides has not been experienced at INTEC. No unit operations downstream of the extraction equipment are designed or operated in a manner to concentrate specific radionuclides. This is generally true except for iodine. Since iodine does not follow the other fission products during extraction cleanup, adjustments were made to reflect actual measured iodine distributions during extraction and waste processing. In addition, Cs-137 has a relatively high heat generation component and is a significant direct radiation hazard. Therefore, it was of particular interest for safety reasons. Since Cs-137 is easy to sample and inexpensive to measure, it was almost always analyzed and reported. This makes it an ideal indicator for other INTEC radionuclides.

7. *Waste submitted for calcination is not returned to liquid storage.* This is a reasonable assumption because only a very small fraction of the waste sent for calcination was returned to the tank farm. Historically, only 14% of the volume of waste submitted for calcination was returned to the tank farm. The concentration of radionuclides in this returned waste was approximately one tenth that in the normal feed coming from the Tank Farm. Approximately 50% of the returned waste contained a fluoride concentration that prevented evaporative concentration of the waste. This waste was subsequently calcined. It is estimated that these factors give a return of nuclides from the calciners in the range of 2%.

8. All solution transfers in and out of the tank farm system were considered. Transfers between tanks were ignored. The calculations were based on a July 1999 decay date.

9. Tanks WM-103 through WM-106 are each estimated to have 0.1 kg of solids in the bottom of each tank. This is supported by video inspections made in 1990 that show little if any accumulation of solid material.

10. Tank WM-190 is essentially free of both liquid and solid waste accumulation. The tank is probably best described as being slightly contaminated.

11. For tanks WM-180 through WM-189, sludge volumes are assumed to follow those outlined in EDF-TST-001.

12. The sludge is assumed to consist of 75 vol% interstitial liquid and 25 vol% solid particles. The interstitial liquid is assumed to have a composition equal to the bulk liquid in each tank.

13. Waste compositions from light-duty utility arm (LDUA) sampling for tanks WM-182, WM-183, and WM-188 were used to represent all tanks. For species not measured in LDUA samples, historical reporting for entrained solids was used, or when no analyses are available, an estimate was made.

# METHODOLOGY

The basic methodology used to obtain the radionuclide inventory and mass balance is described by O'Brien, et al (2002). The total INTEC radionuclide inventory was built using a backend approach. Since radionuclide input to the waste tanks was not measured, it was estimated from known information, historical output, present liquid waste and calcine accumulation and future processing. Accurate records exist for volumes stored and processed and the inventories of those streams. Liquid samples of wastes from the 300,000-gallon tanks were analyzed for fission products in the early 1960s. Supplemental information was obtained from analysis of calcine samples retrieved from the Calcined Solids Storage Facility. Following appropriate adjustment for radioactive decay, a reasonable initial inventory for measured isotopes was generated.

The inventories of the unmeasured isotopes were estimated using modeling. Fuel from many different reactors was processed at INTEC in campaigns of similar fuel types. The principal fuel types, aluminum, zirconium, stainless steel, and graphite, were processed in a manner to generate a uniform recovery product. Consequently, fuel enrichments and cladding types were matched. Each type had experienced approximately similar burn-ups. Tyson (2002) described how simplifications permitted the use of ORIGEN2 to estimate the inventory of important isotopes not measured. The projected inventory from the ORIGEN2 model was compared to the available measured concentrations. The ORIGEN2 input parameters were then adjusted to coincide with the measured concentrations. This validation process provided a reasonable estimation of the concentrations of unmeasured isotopes. Using the estimated radionuclide profiles and the volume of waste processed for each campaign, INEEL staff generated estimates of the radionuclide inventory processed to calcine. The decay isotopes Sr-90, Tc-99, and Cs-137 were back calculated to an appropriately conservative date. Transuranics have longer half-lives and were assumed to equal the estimate of Tyson (2002) normalized to the Cs-137 in 1999.

Using data from a 1994 inventory of the liquid waste tanks the inventory of radionuclides remaining in the waste tanks was estimated. The process also employed the ORIGEN2 model to estimate unmeasured isotopes.

Details concerning the mass balance for the INTEC TFF are provided by Tyson (2002). The estimate of total waste generated from 1953 to the present was made by determining the volume of liquid added to and transferred from each tank. Basic data were obtained from the TFF monthly volume database and weekly production reports. Estimates for each tank were prepared using available data or using estimates for missing data, as described above. These data were supplemented with information concerning waste shipped offsite.

The TFF residual radionuclide inventory estimate consists of a SBW liquid component and a SBW solids component. For calculation purposes, the liquid volume was assumed to be 1,318 gallons for each of the 300,000-gallon tanks and 400 gallons for the 30,000-gallon tanks, based on the depth of the jet pumps. Residual solids for each of the 300,000-gallon tanks were assumed to be reduced to 2,323 kg by preclosure flushing and transfer operations. Video inspection of the 30,000-gallons indicated very little solids, and, therefore, heel solids were estimated at 0.1 kg per tank.

The radiochemical analysis of the solids in each tank was derived from analytical measurements. When analytical measurements were not available, isotope concentrations were estimated using the adjusted ORIGEN2 ratios normalized to appropriate Cs-137 concentrations.

The estimates for each tank were summed to yield an estimate for the TFF at the time of closure. The data used to estimate the inventory and mass balance were verified by Tyson (2002)

## REFERENCES

Tyson, D. R., 2002. *Validation of the Radionuclide Mass Balance Used in the INTEC SBW WIR Determination Report,* EDF-1920, Rev 4, INEEL/EXT–2001-534, August 29, 2002.

# Appendix B

# Summary of the Technical Practicality of Sodium-Bearing Waste Treatment Options

# Appendix B

# Summary of the Technical Practicality of Sodium-Bearing Waste Treatment Options

## PURPOSE

This appendix summarizes the selection process that U.S. Department of Energy (DOE) employed to identify a reasonable range of waste processing alternatives for the sodium bearing waste - waste incidental to reprocessing (WIR) determination. The selection process evaluated the available data and reach consensus on the sodium-bearing waste (SBW) treatment options- stabilization and key radionuclide separation. These treatment options were retained for economic evaluation in the SBW WIR determination process. The SBW WIR determination activities are based on and support the considerations for the Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement (Idaho HLW & FD FEIS). See reference DOE 2002.

## BACKGROUND

In 1995, DOE and the State of Idaho entered into a settlement agreement which, in part, set enforceable milestones for the treatment of 3,800 cubic meters of solid high-level radioactive waste (HLW) calcine and the then existing 1.9 million gallons of liquid SBW stored at the Idaho Nuclear Technology and Engineering Center (INTEC).

In 1997, DOE took an important step toward meeting those milestones by filing a notice of intent to complete an environmental impact statement (EIS) in accordance with the National Environmental Policy Act. The EIS process evaluates the environmental impacts of, and ultimately make decisions regarding, the alternatives for treating the HLW and SBW, as well as newly generated liquid waste (NGLW), and the alternatives for the disposition of related HLW Program facilities at INTEC.

The State of Idaho subsequently agreed to participate as a cooperating agency in the development of the EIS as a means to support the settlement agreement and to facilitate the EIS review process.

From 1997 to 1999, DOE assessed over 100 potential options for treating SBW and calcine and selected the most promising technologies for a bounding environmental impact analysis in the draft EIS. Most of the 100 potential options were eliminated because of nonviable technologies, complicated process operations, or unacceptable regulatory risks. DOE assessed several options for further key-radionuclide removal and also for SBW stabilization.

In January 2000, DOE issued the draft EIS, but did not identify a preferred alternative, to allow for the consideration of public comment as a part of the preferred alternative selection process.

From January to April 2000, the preferred alternative selection process commenced with development of a decision management plan that defined the management approach. Key to this approach was establishment of a decision management team (DMT), who were assigned the responsibility for overseeing the evaluation of relevant data, reaching consensus, and recommending the preferred alternative to senior DOE management. The plan also defined the roles and responsibilities of several subteams supporting the DMT. Support work (i.e., initial data gathering and evaluation) commenced in January 2000, and the DMT was formally established April 28, 2000. The DMT considered many treatment options including those from the INEEL Citizens Advisory Board, subject matter experts, the National Research Council (NAS 1999), the INEEL operating contractor, and others. An independent review was conducted by the DOE Tank Focus Area Group within EM-50 to assist DOE in narrowing the SBW treatment options (PNNL 2000).

In December 2001, DOE-HQ held a top to bottom assessment of all DOE site waste disposal and cleanup plans. A conclusion from the review was that a broad general class of treatments such as stabilization or separations whose bounding environmental impacts were analyzed in the Idaho HLW & FD FEIS should be used for selecting the preferred SBW treatment alternative. It was also decided to evaluate SBW treatment independent of calcine treatment. As a result, the SBW-WIR determination is structured to allow selection of any of the wide range of SBW treatment options in the EIS including grouting, calcination, steam reforming, and others.

## Alternatives Evaluation for SBW Treatment

The purpose of the EIS alternative evaluation is to identify a reasonable set of treatment alternatives for the Idaho radioactive waste (see DOE 1999). The primary selection process for SBW treatment alternatives identification are (see DOE 2002, Appendix B):

- Review previous HLW and SBW management studies, DOE environmental impact statements, technical literature, industry recommendations, and stakeholder comments
- Identify an initial list of candidate alternatives
- Review engineering studies and public input
- Revise the initial set of candidate alternatives based on recent studies and stakeholder inputs from scoping meetings
- Conduct additional engineering studies for newly identified treatment alternatives and clarification studies for the initial treatment alternatives, as required
- Identify screening criteria to evaluate the candidate alternatives
- Describe the criteria that were used to assess each alternative
- Apply the screening criteria to each candidate alternative
- Review alternative screening process by independent subject matter experts
- Select the recommended set of candidate alternative for the Idaho HLW & FD FEIS.

DOE identified a no-action alternative, separation alternatives, non-separation alternatives, and minimum INEEL processing alternatives for initial EIS screening. Several candidate alternatives were eliminated for initial EIS analysis. These alternatives were not considered for one or more of the following reasons: (1) did not meet the purpose and need of the EIS, (2) required significantly more development work to achieve technical maturity, (3) are very similar to or are bounded by other selected alternatives, or (4) judged to be impractical or too costly for consideration. Alternatives eliminated from detailed analysis for technical reasons included such process e as *in-situ* vitrification, tank upgrades for long-term storage, and homogenization and mixing of various wastes. Other alternatives were eliminated from detailed analysis because they did not support the EIS purpose and need such as bringing in waste from outside of INTEC for treatment or using old INTEC facilities as a second HLW repository.

This systematic process resulted in the selection of potential treatment alternatives that will allow DOE greater programmatic flexibility in implementing the SBW alternative and coordinating programs and technologies with other DOE sites. Based on the studies, input from the public, and input from independent subject matter experts, DOE evaluated the data and reached consensus on two sets of candidate treatment options for SBW waste separation and solidification (see Table B-1). These two sets of options were categorized as direct stabilization (solidification) options and separation options (for additional key radionuclide removal).

## Conclusions

Two categories of treatment technologies meet the technical criteria to be considered for the preferred for SBW treatment. The SBW-WIR determination reflects the direction from the DOE top-to-bottom review and the risked based approach for determining key radionuclides. The SBW-WIR determination uses a broad general class of treatment technologies to develop bounding volumes and radionuclide concentrations. The bounding conditions demonstrate that each treatment option generate final waste forms that comply with disposal-site waste-acceptance criteria and performance objectives. For the SBW WIR determination, the following radionuclide treatment options from the Idaho HLW & FD FEIS selection process were retained for economic evaluation:

- Direct Stabilization (Solidification) Options
  - CH TRU Grout
  - Calcination
  - Steam Reforming
  - Direct Vitrification
- Separation Options for Additional Key-Radionuclide Removal
  - UNEX separation
  - TRUEX separation.

# REFERENCES


DOE (U.S. Department of Energy) 2002. *Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement*, DOE/EIS-0287, September 2002**.**

DOE (U.S. Department of Energy) 1999. *Process for Identifying Potential Alternatives for the Idaho High-Level Waste and Facilities Disposition Draft EIS*, DOE-ID 10627, Rev. 1, March 2, 1999.

NAS (National Academy of Sciences/National Research Council), 1999, *Alternative High-Level Waste Treatments at the Idaho National Engineering and Environmental Laboratory,* National Academy Press, Washington D.C., December.

PNNL (Pacific Northwest National Laboratory) 2000. *Assessment of Selected Technologies for the Treatment of Idaho Tank Waste and Calcine*, PNNL-13268, July 2000.

Table B-1. Evaluation of SBW treatment options.

| SBW TREATMENT OPTION | SUMMARY OF REASONS FOR ACCEPTANCE OR ELIMINATION FROM FURTHER CONSIDERATION |
|---|---|
| **SBW Stabilization Options**[k] | |
| Direct vitrification to WIPP | Retained for economic evaluation in the WIR. DOE retained these technologies from the EIS decision making process because of their technical maturity and ability to produce waste forms that meet the disposal site waste acceptance criteria. In these technologies direct solidification processes would be used to stabilize the waste for disposal. |
| Calcination | |
| Steam Reforming | |
| TRU Grout/CsIX | |
| **Precipitation Options** | |
| Hydroxide precipitation | Not Retained - All precipitation options were eliminated from further consideration because of technical difficulties with maintaining an operational system under both normal and abnormal conditions. These technologies employed either a first stage of evaporation or neutralization to produce a precipitate and a second stage of filtration and drying to produce a remote handled transuranic product that would be shipped the Waste Isolation Pilot Plant. |
| Modified hydroxide precipitation | |
| Low-temperature precipitation | |
| High-temperature evaporation and precipitation | |
| **Solvent Extraction Options**[k] | |
| UNEX to HLW Repository, grout to LLW site | UNEX and TRUEX were retained for economic evaluation for additional key radionuclide removal. Based on DOE experience with solvent extraction systems, these technologies were considered to be viable options for removing key radionuclides and placing them into final waste forms that would be acceptable for disposal at the National HLW repository or for near-surface landfill disposal as required. These processes use chemical separations to produce a high-level waste fraction containing separated radionuclides (including key radionuclides) that would be vitrified and shipped to the HLW Repository. The low-level waste fraction from separations would be grouted and disposed in a low-level waste disposal site. |
| TRUEX separations with class C grout | |

k. Option retained for economic evaluation.

# Appendix C

# Resolution of NRC Review Recommendations

# Appendix C

# Resolution of NRC Review Recommendations

## PURPOSE

As recommended in DOE G 435.1-1, the Nuclear Regulatory Commission has provided a technical review of this waste-incidental-to-reprocessing determination document. The review occurred between September 2001 and August 2002. Because the SBW will be treated to meet WIPP repository requirements and disposed under DOE & EPA jurisdiction, the NRC did not review for compliance with WIR Criteria 2 and 3, rather they focused on Criterion 1 – the assessment of whether the waste has been processed, or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical. Accordingly, the NRC only provided comments and observations on the methodology for meeting Criteria 2 and 3 that were identified during the review. The NRC's final report was documented in Greeves (2002).

## ASSUMPTIONS

The NRC made the following assumptions in assessing conformance with Criterion 1:

- NRC is providing technical assistance, only. NRC is not a regulatory authority for SBW stabilization and disposal.
- NRC focused its review on Criterion 1. Any comments made with regard to Criteria 2 and 3 were observed during the Criterion 1 review, and do not indicate a thorough review of Criteria 2 and 3.
- Cost estimates associated with the different options are reasonable.
- Identifying key radionuclides based on predicted performance is reasonable. NRC staff review focused on key radionuclides that could affect health and safety after disposal.
- The characterization of the radionuclide composition of the SBW liquids and solids is a reasonable representation of the actual composition.

## CONCLUSIONS

The NRC made the following conclusions with respect to Criterion 1:

- The NRC agrees that it is not technically practical to remove additional key radionuclides from the SBW solids prior to disposal.
- The NRC agrees that even though the technology exists to remove additional key radionuclides from SBW liquid, it is not economically practical to do so.

## RECOMMENDATIONS

The NRC made the following recommendations with respect to meeting Criterion 1:

- Although a significant amount of work has been completed in an attempt to define the SBW liquid and solid radionuclide concentrations, limited information is available in some key areas or to support some key assumptions. The residual uncertainty can likely be reduced through the collection of additional information during future activities (e.g., solid and liquid sampling). As additional information is collected, an impact assessment on the SBW WIR determination should be completed.

  DOE Response: *This comment was addressed in Section 2.3.2 of this document. The uncertainty will be reduced in the future by taking additional samples. The results will be compared with waste characteristics in the WIR determination to ascertain the impact and verify that the WIR-determination conclusions are still valid.*

- As it is important to assess operational exposures in reference to Criterion 1, NRC's request for additional information (RAI) requested DOE-ID to provide a brief analysis describing impacts to workers resulting from the options evaluated. In the response to the RAI, DOE-ID noted that dose to the worker was generally found to be insignificant, and that allowable worker radiation exposures are set by DOE regulations. Additional shielding was added to facilities (that would be required for the various SBW treatment options) that handle more highly radioactive waste, so that the worker exposure is projected to be about the same for all options. Likewise, shipping, handling, and disposal facilities all have equipment and procedures to handle waste product safely. Therefore, increased radiation levels for various waste types were reflected in increased costs for additional shielding, shipping, and handling requirements. DOE-ID noted that this discussion would be included in the revised SBW WIR determination; however, it appears that it was not included. DOE should provide this discussion in the final WIR determination.

  DOE Response: *The discussion on worker exposure has been added to Section 3.1.6 of this document.*

## ADDITIONAL NOTES

Although NRC staff review of the SBW WIR determination focused on Criterion 1, the staff also noted the following during its review:

- DOE-ID noted that WIPP is currently not permitted to accept RH-TRU waste, but it is expected to be permitted in 2003. DOE-ID also noted that the draft waste acceptance criteria (WAC) for RH-TRU are not expected to change. NRC staff suggests that if there are changes to the plans to permit WIPP to accept RH-TRU waste or if the draft WAC for RH-TRU changes, DOE-ID should revisit the WIR determination before final decisions regarding the SBW treatment process and final waste forms are made.

  DOE Response: *DOE-Carlsbad has submitted permit modifications and waste acceptance criteria changes for review and approval. It is expected that these modifications and changes will be approved soon and DOE can finalize SBW-treatment decisions. DOE recognizes that final SBW treatment decisions cannot be made until WIPP is fully permitted and authorized to accept the treated waste.*

- The residual uncertainty regarding the radionuclide inventory is expected to have a greater impact on DOE-ID's WIR determination for tank closure. NRC plans to document its concerns regarding the current tank inventories in a future RAI on the tank closure WIR determination.

  DOE Response: *Source term estimates for the tank farm radionuclide inventory are based upon actual samples and scientifically sound models. DOE has added conservatism to ensure that the estimated radionuclide inventory is bounding; therefore, DOE is reasonably confident that WIPP disposal and INEEL tank-closure performance objectives can be met. But more importantly, before SBW treatment and tank closure actions are finalized, DOE will confirm through sampling actual waste materials that all performance objectives are met and that public health and safety are assured.*

# REFERENCES


Bernero, R.M. (NRC) 1993. Letter to J. Lytle (DOE), March 2, 1993.

Greeves, J. T. (NRC), 2002. Letter to J. T. Case (DOE-ID), "NRC Review of *Idaho National Engineering and Environmental Laboratory Draft Waste Incidental to Reprocessing Determination for Sodium-Bearing Waste*, Conclusions and Recommendations," August 2, 2002.

NRC (U.S. Nuclear Regulatory Commission) 2002. 67 FR 5003, *Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement, Federal Register,* February 1, 2002, pp. 5003.

NRC (U.S. Nuclear Regulatory Commission) 1999. *Classification of Savannah River Residual Tank Waste as Incidental,* SECY-99-284, December 15, 1999.