Separations Process Research Unit

Waste Type Evaluation For Solidified Waste Tank Sludge

March 2011

U.S. Department of Energy
Office of Environmental Management
Washington, D.C.

This page is intentionally blank.

CONTENTS

CONTENTS

CONTENTS

ACRONYMS, ABBREVIATIONS, AND UNITS

Acronyms and Abbreviations

ALARA	as low as reasonably achievable
Am	americium
CFR	Code of Federal Regulations
C	carbon
Cm	curium
Co	cobalt
Cs	cesium
DOE	Department of Energy
I	iodine
IP	industrial packaging
KAPL	Knolls Atomic Power Laboratory
Ni	nickel
Np	neptunium
NRC	Nuclear Regulatory Commission
Pu	plutonium
REDOX	reduction-oxidation
SPRU	Separations Process Research Unit
PUREX	plutonium uranium extraction
Sr	strontium
Tc	technetium
U	uranium
WVES	West Valley Environmental Services
WVNSCO	West Valley Nuclear Services Company

Units

Bq	Becquerel
Ci	curie
g	gram [mass]
hr	hour
L	liter
m	meter
m^2	square meter
m^3	cubic meter
mR	0.001 Roentgen
mrem	0.001 Roentgen equivalent man
millirem	0.001 Roentgen equivalent man
mSv	millisievert, 0.001 sievert
μCi	microcurie, 0.000001 curie
nCi	nanocurie, 10^{-9} curie
R	Roentgen
rem	Roentgen equivalent man
s	second

EXECUTIVE SUMMARY

This evaluation provides additional information, documentation, and further assurances that the solidified sludge from the underground waste tanks at the Separations Process Research Unit (SPRU) will be low-level radioactive waste. For perspective and additional information, the criteria for waste incidental to reprocessing of Section II.B of Department of Energy (DOE) Manual 435.1-1, *Radioactive Waste Management*, also are considered in the evaluation, even though these criteria do not apply to the solidified tank sludge.

SPRU is located at Knolls Atomic Power Laboratory (KAPL) in New York, which is a Naval Nuclear Propulsion Program Propulsion research facility. SPRU was a pilot plant for development of processes later used for recovering plutonium and uranium from irradiated nuclear materials. It operated from 1950 to 1953 performing laboratory-scale research using non-irradiated and slightly irradiated test specimens. Liquid radioactive waste resulting from research activities was stored in seven underground stainless steel waste tanks located adjacent to the waste processing building. These tanks were comprehensively cleaned out in 1965 and drained in 1978, although sludge remained in the bottom of each tank. This sludge had a total volume of approximately 266 cubic feet.

The SPRU facilities are being decommissioned by a contractor employed by the Department of Energy Office of Environmental Management. In the fall of 2010, this contractor cleaned six of the tanks using high-pressure water spray and consolidated the sludge and water mixture in the seventh tank. This mixture, which has been estimated to contain approximately 68 curies of radioactivity, is planned to be solidified in cement inside industrial packaging Type 2 containers and transported to the Nevada National Security Site, where it has been approved for disposal as low-level radioactive waste. Solidification is a treatment technology used to stabilize the waste to eliminate free liquids for safe transportation and disposal.

This evaluation demonstrates that the sludge cannot be high-level radioactive waste from the reprocessing of spent nuclear fuel based on its origin and characteristics, including: (1) the slightly irradiated and non-irradiated test specimens used for SPRU research and development were not spent nuclear fuel, (2) the sludge did not result from reprocessing, and (3) the waste associated with SPRU operations was not highly radioactive material akin to high-level radioactive waste from reprocessing.[1] Although not required, this evaluation also applies the method and criteria in Section II.B of DOE Manual 435.1-1 to demonstrate that the solidified sludge would meet these criteria had they applied, that is (1) the sludge has been processed to remove key radionuclides to the maximum extent technically and economically practical; (2) will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C; and (3) will be managed in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR 61.55.

[1] This conclusion also applies to the tanks themselves, which are contaminated with small amounts of the same material.

1.0 INTRODUCTION

This section provides introductory information that lays the foundation for the detailed discussions that follow. It describes the evaluation purpose and scope, summarizes the background, and outlines the contents of the rest of the evaluation.

1.1 Purpose

The purpose of this evaluation is to provide additional information, documentation and further assurances concerning the Department of Energy (DOE) management and planned disposal as low-level radioactive waste of the solidified sludge from the underground waste tanks at the Separations Process Research Unit (SPRU) in Niskayuna, New York, by assessing the origin and characteristics of the waste and, although not required, applying the methodology and criteria for "waste incidental to reprocessing" in DOE Manual 435.1-1, *Radioactive Waste Management*.

1.2 Scope

This evaluation focuses on the sludge being removed from the seven underground waste tanks at the SPRU facility, which will be solidified in cement for offsite disposal at the Nevada National Security Site. The sludge from these tanks is being removed in connection with decommissioning of the SPRU facility.[2]

This evaluation considers the waste type by definition of the sludge from the tanks based on its origin and its characteristics after solidification. In so doing, this evaluation takes into account historical and other information previously considered by DOE (DOE 2008) to support DOE's prior conclusion that that the waste should be managed and disposed of as low-level waste. To provide additional information and further assurances, this evaluation also applies the evaluation method and criteria in Section II.B(2)(a) of DOE Manual 435.1-1 pertaining to "waste incidental to reprocessing", even though SPRU did not reprocess spent nuclear fuel and thus did not generate waste "incidental to reprocessing". The evaluation method and criteria are normally used for determining whether waste from reprocessing of spent nuclear fuel is incidental to reprocessing and can be managed as low-level waste. The evaluation method and criteria involve establishing whether the wastes:

(1) Have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical;

(2) Will be managed to meet safety requirements comparable to the performance objectives set out in [Code of Federal Regulations] 10 CFR Part 61, Subpart C, *Performance Objectives*; and

(3) Are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of this Manual [DOE Manual 435.1-1], provided the waste will be incorporated in a solid physical form at a concentration

[2] The tanks and associated piping, although not addressed in this evaluation, are also planned to be removed for offsite disposal as low-level waste during the decommissioning. The empty tanks are expected to contain much less radioactivity than the removed sludge.

that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize.

This evaluation takes into account and reassesses information developed by DOE in an informal waste evaluation completed in 2008 (DOE 2008), which supported the conclusion the sludge in the underground tanks is not high-level waste. This evaluation pertains to radioactivity only. It does not address constituents in the sludge that may be hazardous under the Resource Conservation and Recovery Act[3].

1.3 Background

The main SPRU facilities occupy approximately five acres on the 170-acre Knolls Atomic Power Laboratory (KAPL) site in Niskayuna, New York. KAPL is a U.S. Naval Nuclear Propulsion Program research and development laboratory. Figure 1 shows the location of the laboratory, the SPRU facilities, and the underground waste tanks.

The SPRU facilities were used from 1950 through 1953 for laboratory-scale research and development on the REDOX (reduction-oxidation) and PUREX (plutonium-uranium extraction) chemical separation processes, which were later used at other facilities on a production scale to extract uranium and plutonium from irradiated uranium (EGR 2006a). These activities left radioactive contamination inside the SPRU facilities, which consist primarily of Building G2 where the research and development processes were carried out and Building H2 where the wastes from these processes were managed and stored.

In preparation for the decommissioning, DOE had two historical site assessments prepared (ERG 2006a and ERG 2006b). The facility historical site assessment (EGR 2006a) formed the basis for much of the information in this evaluation.

DOE is using the non-time-critical removal process for the SPRU decommissioning based on its authority under the Comprehensive Environmental Response, Compensation and Liability Act. In support of this process, DOE had an Engineering Evaluation/Cost Analysis (ERG 2007) prepared. Four alternatives were considered and DOE determined that Alternative 4 – removal of the SPRU facilities – best met the removal action objectives (DOE 2009).

A URS Corporation and CH2M Hill team is decommissioning the SPRU facilities under contract to the DOE Office of Environmental Management. This work is being accomplished in accordance with the Decommissioning Plan (URS 2010c) that is required by the contract.

This Decommissioning Plan requires that the sludge in the bottom of the seven underground waste tanks be removed along with the tanks themselves. The plan requires the sludge to be solidified in cement and transported to the Nevada National Security Site for disposal.

[3] It has been determined that the solidified sludge will not be hazardous under the Resource Conservation and Recovery Act and the related New York State Regulation (DK-ES 2010).

Figure 1. Location of the SPRU Facilities and Underground Waste Tanks *(from ERG 2006a)*

By the end of 2010, URS had accomplished a considerable amount of work to this end. This work included removing sludge from six of the seven underground waste tanks and consolidating it in the seventh tank in preparation for solidifying this waste.

1.4 Organization of this Evaluation

Information in the remainder of this evaluation is presented as follows:

Section 2 briefly describes the history of the SPRU facility

Section 3 describes the underground waste tanks, provides additional details on their history and radiological status, and describes the process to be used to solidify the sludge.

Section 4 shows that the sludge from the tanks will be low-level waste after it is stabilized by being solidified for disposal.

Section 5 discusses for information purposes how key radionuclides will have been removed from the waste to the maximum extent technically and economically practical.

Section 6 discusses for information purposes how safety requirements comparable to NRC performance objectives at 10 CFR 61, Subpart C, *Performance Objectives*, will be achieved.

Section 7 explains for information purposes that the radionuclide concentrations in the sludge after solidification will be less than Class C concentration limits, and that this waste will be managed and low-level waste in accordance with Chapter IV of DOE Manual 435.1-1.

Section 8 summarizes DOE's conclusions related to this evaluation.

Section 9 identifies the references cited in the text.

2.0 FACILITY HISTORY

The following information is intended to help place the evaluation into context. It is drawn primarily from the facility historical site assessment (ERG 2006a).

2.1 General Facility History

In May 1946, General Electric (GE) entered into a prime contract with the U.S. Atomic Energy Commission to operate the Hanford Engineering Works in Richland, Washington, and to build and operate a laboratory that would become known as KAPL. The Hanford program involved operating natural uranium-fueled reactors to produce plutonium, with KAPL providing Hanford technical and scientific support. In early 1947, the Atomic Energy Commission launched an urgent program to provide production facilities to recover plutonium and uranium from irradiated fuel. The Hanford fuel recovery plan was based on the REDOX solvent extraction process.

GE Hanford was requested to provide a full-scale REDOX plant at the Hanford site in Richland, Washington by late 1949, and in December 1947, requested a REDOX pilot plant to research the process. The REDOX pilot plant was constructed at KAPL and became known as SPRU, which operated from 1950 to 1953. The SPRU mission consisted of pilot tests on chemical processes to separate plutonium and uranium from unirradiated material or slightly irradiated material encased in aluminum on a laboratory scale; SPRU was never a production plant.

After completion of research and development work on the PUREX process, the SPRU facility decommissioning began in October 1953. All activities associated with continued research on solvent extraction processes ceased once a viable process had been developed and construction began on production facilities at other Atomic Energy Commission sites. Building H2 and the Tank Farm were partially shut down, but some areas remained in use for SPRU decommissioning activities and other KAPL waste management functions.

Figure 2 shows the SPRU facilities before the start of the current DOE decommissioning project. Building G2 and H2 are connected by underground pipe tunnels, which also run in the basements of adjoining KAPL laboratory buildings such as building G1 shown in the figure.

Figure 2. SPRU Facilities Before the Start of the Current Decommissioning Project, Looking North *(from ERG 2006a)*

2.2 Pilot-Scale Test Research

Between 1950 and 1953, SPRU tested chemical processes for separating plutonium and uranium from irradiated materials. These materials, which were encased in aluminum and called slugs, were test specimens irradiated to low levels as discussed in Section 4.4. REDOX chemical test runs were performed until the end of 1950 and PUREX test runs until mid-1953 when the techniques were successfully exported for use at Hanford and the Savannah River Site, respectively. Separated plutonium was transferred to the Los Alamos National Laboratory for test and research purposes. The DuPont Company believed that mixer-settlers would be adopted for the PUREX process planned for the Savannah River Plant it was building, so associated studies were conducted at the SPRU pilot plant in 1951. A series of demonstration runs were performed during the first half of 1952, and alternate PUREX processes were studied and further head-process refinements were developed in the second half of 1952 and the first half of 1953.

Organic waste streams from the pilot-scale test research PUREX process were washed for reuse in the solvent extraction process. These materials were removed from the facility, although associated contamination remained.

Liquid waste generated during SPRU research activities was treated and stored in the tank farm. Radioactive solid waste produced by SPRU research activities included combustible waste, incinerator ash, evaporator sludge, and contaminated equipment and miscellaneous materials. These materials were removed from the facility.

Use of SPRU facilities for REDOX and PUREX research ended in June 1953. Shortly afterwards, Hanford and the Savannah River Site constructed and started up production facilities.

2.3 Tank Farm History

From the beginning, the tank farm received liquid waste from KAPL laboratories as well as from the SPRU research and development work. Between 1950 and 1954, SPRU facilities and KAPL laboratories[4] managed liquid waste using drain lines that traversed the pipe tunnels to the Building H2 waste management process equipment.

Following the initial SPRU decommissioning work in 1953, tank farm use continued for storage of liquid waste until 1978. Processed separations material and waste were accumulated in the tanks from 1950 until 1954, and remained in storage until the mid-1960s. The first comprehensive cleanout of the tanks occurred in 1965, when SPRU and other KAPL waste in the tanks was removed and disposed of offsite. The tanks were drained in 1978.

[4] Building G1, which contains KAPL laboratories, adjoins Building G2 as shown in Figure 2. Other buildings housing KAPL laboratories either are connected to Building G1 or are located nearby.

3.0 TANK CLEANING AND SLUDGE SOLIDIFICATION

This section describes the seven tanks, their history, and their radiological status. It then describes how the sludge was removed from six of the tanks and consolidated in Tank 509E. It briefly describes the process to be used to solidify the sludge for offsite disposal and plans for removal and disposal of the tanks themselves.

3.1 Tank Design

The seven waste collection tanks are made of stainless steel. Six are of 10,000-gallon capacity and one (Tank 578) has a capacity of 5,000 gallons. They are located within individual reinforced concrete vaults. The tops of the vaults are covered with earth and are located about nine feet below the surface. Figure 3 shows the location of the tanks with respect to Building H-2.

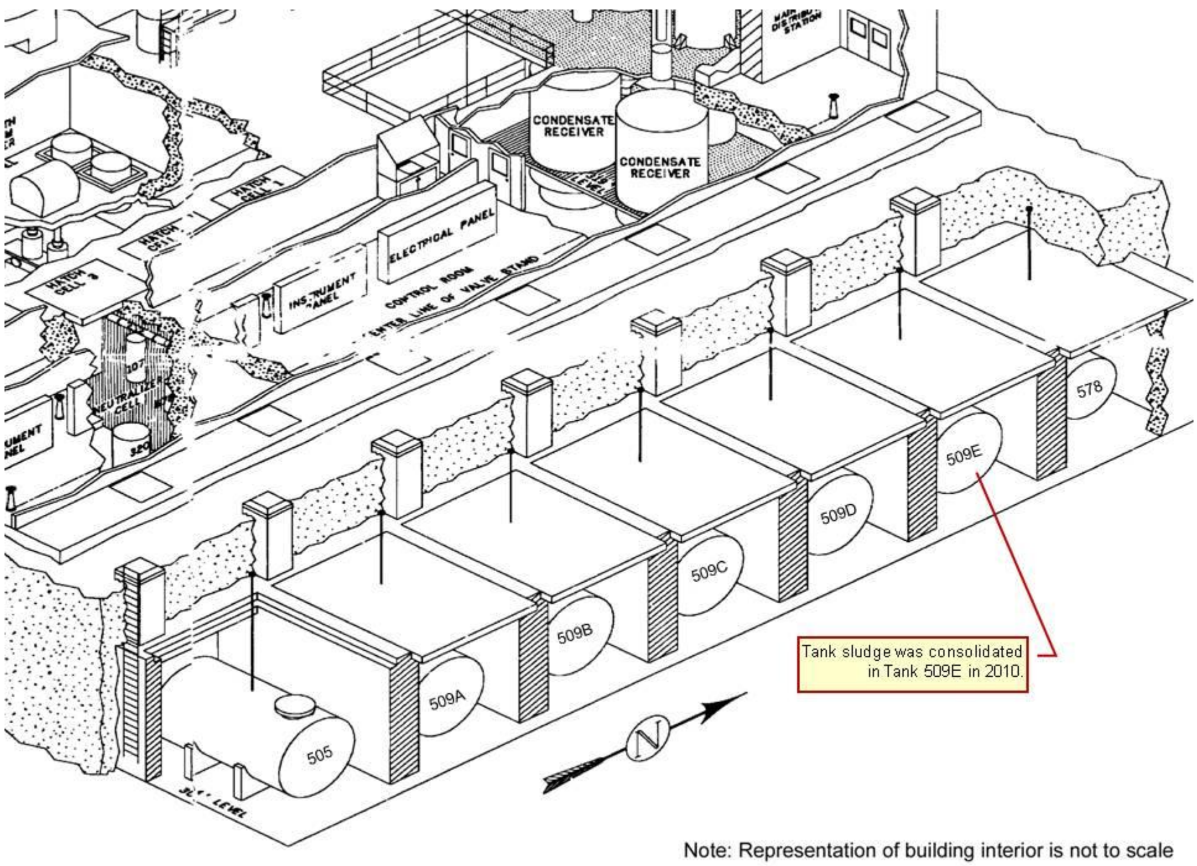

Figure 3. Isometric View of Building H2 and the Waste Tank Vaults *(from ERG 2006a)*

As can be seen in Figure 3, the tanks are numbered 505, 509A, 509B, 509C, 509D, 509E, and 578. Figure 4 shows the top of one of the tanks.

3.2 Tank History

The tank farm was utilized as described in Section 2.3. In 1964 and 1965, more than 20,000 gallons of liquid waste from the tank farm were removed and disposed of offsite. Records show that waste removed in 1965 from Tank 505 and 509A and Tank 320 in neutralizer cell 4 consisted of 2,377 curies of fission products, 903 kilograms of natural uranium, and 539 grams of plutonium (ERG 2006a).

Figure 4. Top of Underground Waste Tank *(from ERG 2006a)*

The tanks continued to receive liquid radioactive wastes from KAPL laboratories operated by the Naval Nuclear Propulsion Program until a new closed-loop radioactive materials laboratory liquid waste reuse system was placed in operation in 1978.

In 1978, liquid waste was emptied from the tanks. This waste, which amounted to a total of approximately 27,000 gallons, was solidified in 55-gallon drums and shipped offsite for disposal as low-level waste at the Chem-Nuclear low-level radioactive waste disposal facility in Barnwell, South Carolina (ERG 2006a).

3.3 Radiological Status

Estimates of the radioactivity in the sludge have been made and dose rates inside the tanks have been measured.

3.3.1 Radioactivity Estimates

In June 2010, an estimate of the activity associated with the sludge in each of the tanks was prepared (URS 2010a). Four different methods were used, including calculations based on analytical data from samples of the sludge collected from each tank in February 2010. The estimates based on analytical data from these samples are shown in Table 1. The estimates using the other methods were in reasonable agreement.

Table 1. Estimated Activity in Underground Waste Tank Sludge (in Curies)[(1)]

Tank	*Sludge Vol. (ft^3)*	*Sr-90*	*Cs-137*	*U-238*	*Pu-239*	*Am-241*
505	31.55	0.36	0.69	0.01	0.49	0.05
509A	45.76	6.08	18.91	0.07	1.68	0.23
509B	8.94	0.41	0.33	0.00	0.11	0.01
509C	106.56	10.71	1.15	0.01	2.75	0.32

Table 1. Estimated Activity in Underground Waste Tank Sludge (in Curies)(1)

Tank	*Sludge Vol. (ft^3)*	*Sr-90*	*Cs-137*	*U-238*	*Pu-239*	*Am-241*
509D	8.94	0.80	1.07	0.00	0.07	0.01
509E	45.76	0.66	1.13	0.01	0.27	0.03
578	18.15	0.95	1.72	0.00	0.49	0.05
Total	265.66	19.97	25.01	0.09	5.86	0.70

NOTES: (1) From URS 2010a. The amounts of other radionuclides in the sludge samples were insignificant, except for Pu-241 (1.30 curies).

Another estimate was made based on analytical data from a composited bulk sample collected in September 2010 from the consolidated sludge in Tank 509E. Table 2 compares the two estimates.

Table 2. Estimated Activity in Consolidated Sludge in Tank 509E (in Curies)

Activity Estimate	*Sr-90*	*Cs-137*	*U-238*	*Pu-239*	*Pu-240*	*Pu-241*	*Am-241*
Individual Tank Totals(1)	19.97	25.01	0.09	5.86	(2)	1.30	0.70
Consolidated Sludge(3)	16.4	43.8	0.05	4.32	1.08	1.12	0.64

NOTES: (1) From Table 1.

(2) Laboratory results were reported as Pu-239/Pu-240.

(3) Preliminary results based on analysis of a composited bulk sample of sludge-water mixture in Tank 509E. The following additional radionuclides were present above the minimum detectable activity but in insignificant concentrations: C-14, Co-60, Ni-63, Tc-99, U-234, U-235, U-236, U-238 Np-237, PU-238, and Pu-242. (URS 2011)

As can be seen in Table 2, the estimates based on analytical data from the consolidated sludge sample are lower than the estimates based on the individual tank sample data for Sr-90 and higher for Cs-137. The totals for alpha-emitting transuranic radionuclides in the tables are in close agreement. The consolidated sludge activity estimates are considered to be the more accurate of the two sets of estimates[5].

For perspective, the amounts of activity in the sludge as shown in Table 2 are small compared to amounts of residual radioactivity in underground waste storage tanks that have stored high-level waste from reprocessing of spent nuclear fuel.[6]

[5] There were several differences between the estimating methods. (1) The consolidated waste in Tank 509E includes sludge and its associated radioactivity from Tank 525-A, a 10,000-gallon evaporator storage tank in the basement of Building H2, as well as the other six underground waste tanks. The sludge in Tank 525-A was estimated to contain approximately 0.021 curie of Sr-90, 3.0 curies of Cs-137, 0.06 curie of Pu-239, and 0.02 curie of Am-241 (URS 2010b). (2) The small heels of sludge-water mix remaining in the six cleaned tanks, which are discussed below, are not included in the consolidated sludge estimate. (3) The consolidated sludge composite sample is likely to be more representative because of the mixing capabilities of the two submersible pumps in Tank 509E than sludge in the bottoms of the seven tanks, which was likely less homogeneous due to stratification. (4) More formal protocols were used in sampling and analysis of the consolidated sludge. Consideration of these differences suggests that the set of estimates based on the consolidated sludge analytical data is more accurate.

[6] The higher estimates in Table 2 for the consolidated sludge total approximately 68 curies. The four underground waste storage tanks at the West Valley Demonstration Project, three of which were exposed to high-level waste from prior reprocessing by Nuclear Fuel Services, Inc., have been estimated to contain approximately 345,000 curies of residual radioactivity after cleaning (WVESCO and Gemini 2005). The West Valley plant was the only commercial production-scale reprocessing facility to operate in the United States.

3.3.2 Dose Rates

Gamma dose rates within the tanks were measured in September 2010 before and after consolidation of the sludge into Tank 509E as shown in Table 3.

Table 3. Gamma Dose Rates Inside Tanks Before and After Sludge Removal (mR/hr)[1]

Tank	*Measurement*	*Location Inside Tank*		
		Top	*Middle*	*Bottom*
505	Before Sludge Removal	65	195	431
	After Sludge Removal	14	20	125
509A	Before Sludge Removal	46	500	1100
	After Sludge Removal	90	175	>200
509B	Before Sludge Removal	5	58	120
	After Sludge Removal	7	10	20
509C	Before Sludge Removal	3	33	140
	After Sludge Removal	7	10	15
509D	Before Sludge Removal	5	60	265
	After Sludge Removal	4	10	20
509E	Before Sludge Removal	100	276	1000
	After Sludge Removal	Sludge not yet removed – see Section 3.4.		
578	Before Sludge Removal	48	91	346
	After Sludge Removal	5	20	140

(1) From URS 2011.

As can be seen from the table, dose rates following sludge removal and tank cleaning were generally significantly lower than before sludge removal.

3.4 Tank Cleaning and Sludge Volumes

Visual observations of the sludge in the tanks in 1989 showed that the sludge was typically dark charcoal to black in color and appeared to be tar-like in consistency. The moisture content of the sludge in the tanks ranged from approximately 47 to 74 percent (DK-ES 2010).

As noted previously, the sludge has been consolidated in Tank 509E. This consolidation was accomplished by removing the sludge from the six other tanks using high-pressure, low-volume water spray to dislodge and mobilize the sludge and pumping the sludge-water mixture into Tank 509E. An articulated nozzle was used to reach all parts of the tank bottoms. Two pumps were used, with one inside the tank. Remote video cameras were used to permit visual observation of the effectiveness of the cleaning process. The same process was used to remove sludge from Tank 525-A.

The resulting total volume was approximately 10,700 gallons, with the sludge moisture content being approximately 95.6 weight percent (DK-ES 2010). As of early March 2011, approximately 10,000 gallons were contained in Tank 509E and the remainder in the day tank[7].

After the sludge is removed from Tank 509E, that tank will be cleaned using the high-pressure water spray process used for the other tanks. This cleanup is expected to generate approximately 200 additional gallons of liquid which will also be solidified for disposal.

Conservative estimates of the contents of the seven tanks as of early March 2011 based on visual observations made following sludge removal (URS 2011) are as follows:

- Tanks 505, 509A, 509B, 509C, and 509D contain heels consisting of approximately 16 gallons (2.14 cubic feet) of a sludge-water mixture; and
- Tank 578 contains a heel consisting of approximately 15 gallons (2.01 cubic feet) of a sludge water mixture.

Tank 509E contains approximately 10,000 gallons of the sludge-water mixture removed from the other tanks as noted previously. The effectiveness of the sludge removal process is discussed in Section 5.2.2 below.

3.5 Sludge Solidification and Disposal Plans

The solidification will take place within 1,270-gallon capacity IP-2 containers[8] generally referred to as waste liners. These steel containers are approximately 74 inches in internal diameter and 68 inches high; Figure 5 shows a typical waste liner.[9] The sludge-water mixture must be solidified to meet the disposal site waste acceptance criterion for free liquid in the waste; this treatment process stabilizes the waste.

Figure 5. Typical Waste Package Suspended Over Shield (*West Valley Demonstration Project Photo)*

In each liner, the waste will be combined with Portland Type 1 cement and calcium polysulfide. The mixing blade installed in the liner, which will remain in place, will be used to ensure homogeneity of the mixture. Approximately 18 or 19 waste liners are expected to be necessary.

[7] The day tank is a 1,270-gallon tank used to support solidification of the sludge. The waste is pumped to the day tank for verification sampling before being solidified.

[8] Industrial packaging Type 2 (IP-2) containers must meet Department of Transportation requirements at 49 CFR 173.411, *Industrial Packaging*, which include passing tests related to preventing loss or dispersal of radioactive contents in an accident.

[9] Decontaminated sodium-bearing wastewater was solidified at the West Valley Demonstration Project using a similar process (Meess and Dundas 2005).

DOE plans to transport the solidified waste to the Nevada National Security Site for disposal in the Area 5 radioactive waste management site as low-level waste[10]. To this end a waste profile sheet (DK-ES 2010) has been prepared for the solidified sludge waste form and submitted to the Nevada National Security Site in accordance with that site's waste acceptance criteria (DOE-NV 2011). This waste profile sheet has been approved by the Nevada National Security Site (DOE-NNSS 2010), which represents approval to ship the solidified sludge to that facility for disposal as low-level waste.[11]

[10] The Nevada National Security Site maintains two radioactive waste disposal facilities known as the Area 3 and Area 5 radioactive waste management sites. Only Area 5 is currently accepting waste for disposal.

[11] In connection with the decommissioning work, the seven underground waste tanks will be removed and transported offsite for disposal as radioactive waste. Preparation of a tank for removal will involve removing the piping connections, preparing the tank as required to meet transportation and disposal requirements, and transporting it to the radioactive waste disposal facility for disposal as low-level waste.

4.0 SOLIDIFIED SLUDGE WASTE FORM TYPE

This section discusses the origin of the sludge and DOE waste type requirements. It then demonstrates that the solidified sludge waste form will be appropriately classified as low-level waste.

4.1 Origin of the Sludge

As discussed previously, the SPRU facility was used for laboratory-scale research and development of chemical separation processes that were later used at other facilities on a production scale to extract uranium and plutonium from irradiated uranium slugs and spent nuclear fuel. The underground waste tanks received radioactive wastes from these laboratory-scale processes that were carried out in Building G-2 from 1950 through 1953, a period of approximately four years.

As discussed previously, the underground waste tanks also received various liquid radioactive wastes from laboratories in other buildings that were operated by the Naval Nuclear Production Program. Receipt of such wastes began in 1950 and ended in 1978[12]. This waste material was not associated with SPRU operations.

Sludge was also known to be associated with waste from the Naval Nuclear Propulsion Program laboratories. For example, in the 1980s liquid radioactive waste from the radioactive materials laboratory in Building E4 was unable to be transported to Building H2 through the piping because a section of piping leading from this laboratory to Building H2 was approximately 90 percent full of solids and sludge (ERG 2006a).

Given this history, it is reasonable to conclude that the sludge in the tanks came from waste associated with both SPRU operations and Naval Nuclear Propulsion Program laboratories, but the relative amounts from these two sources cannot be quantified using available records.

4.2 DOE Waste Types and Low-Level Waste

Under DOE Manual 435.1-1, DOE categorizes radioactive waste into three types: high-level waste, transuranic waste, and low-level waste. DOE Manual 435.1-1, Section IV.A, defines low-level waste as follows:

> "Low-level radioactive waste is radioactive waste that is not high-level radioactive waste, spent nuclear fuel, transuranic waste, byproduct material (as defined in section 11e.(2) of the *Atomic Energy Act of 1954*, as amended), or naturally occurring radioactive material."

[12] The facility historical site assessment (ERG 2006a) describes large liquid waste volumes processed in Building H2 in fiscal year 1952 and controls implemented on installation of new waste transfer lines from other buildings (page 6-12). It states that between 1950 and 1954 KAPL laboratories in Buildings E1, E2, E3, E4, D3, D4, and G1 managed liquid waste by transferring it through the piping tunnels to Building H2 for treatment (page 6-14). It indicates that laboratories in Building E1 and G1 continued to use tunnel waste transfer lines to transport liquid waste to Building H2 until 2001 (page 6-14). However, in 1977, a new radioactive material laboratory radioactive liquid waste treatment system was placed into service, after which the underground waste tanks were no longer needed for storage of liquid waste from the laboratories in other buildings. In 1978, all liquid waste was removed from the underground waste tanks (ERG 2006a, page 6-15).

The waste type is determined after the waste is packaged in its final form for disposal. Radioactive waste is, by definition, low-level waste if it is not (1) spent nuclear fuel, (2), high-level waste, (3) transuranic waste, (4) by-product material, or (5) naturally occurring radioactive material.

4.3 Waste Type By Definition

The waste type of the solidified sludge is determined based on the definition of low-level waste. In applying this definition, the following points and characteristics of the waste bear emphasis:

(1) The slightly irradiated and non-irradiated test specimens used in the SPRU research and development work were not spent nuclear fuel and the associated waste therefore did not result from the reprocessing of spent nuclear fuel.

(2) The laboratory-scale research and development performed at the SPRU facility would not have constituted reprocessing even if spent nuclear fuel had been involved in the process.

(3) Radioactive materials in the underground waste tanks did not exhibit the properties of highly radioactive material associated with high-level waste.

(4) Fission products in the underground waste tanks before the first comprehensive tank cleanout effort were present in much smaller amounts than fission products associated with high-level waste.

(5) Consideration of the above factors leads to the conclusion that the sludge does not meet the definition of high-level waste.

(6) The approved waste profile for disposal of the solidified sludge at the Nevada National Security Site shows that the solidified waste form will not be transuranic waste.

(7) The sludge is not byproduct material by definition.

(8) The sludge is not naturally occurring radioactive material.

(9) Because the solidified sludge is not high-level waste, spent nuclear fuel, transuranic waste, byproduct material, or naturally occurring radioactive material, the waste is low-level waste by definition.

Each waste type is discussed in the following sections to demonstrate that the waste is low-level waste by definition.

4.4 Spent Nuclear Fuel

Spent nuclear fuel is generally considered to be nuclear fuel that is irradiated in a nuclear reactor to the point where it is no longer useful in sustaining a nuclear reaction. The NRC in 10 CFR 72.3 defines spent nuclear fuel as follows:

> *"Spent Nuclear Fuel* or *Spent Fuel* means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements

> by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies."

DOE Manual 435.1-1, *Radioactive Waste Management*, defines spent nuclear fuel as follows:

> "Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Test specimens of fissionable material irradiated for research and development only, and not production of power or plutonium, may be classified as waste, and managed in accordance with the requirements of this Order [DOE Order 435.1] when it is technically infeasible, cost prohibitive, or would increase worker exposure to separate the remaining test specimens from other contaminated material."

The materials used in the SPRU process included uranium slugs irradiated at the Hanford site for use in laboratory-scale research and development work. Hanford shipping records show that four-inch-long and two-inch-long slugs were sent to SPRU. One record indicates that some were "samples cut from irradiated slugs" that were 1-5/8 inches in outside diameter and four inches long (Bragg 1949). The primary slug used in the Hanford reactors for separation[13] was eight inches in length (Hanford 1956). SPRU records show some 843 slugs were received for research and development work (KAPL 1992).

These test specimens would not meet the NRC definition of spent nuclear fuel because they were not designed to serve as source of energy in a power reactor and would have been incapable of doing so. They were not designed to sustain a chain reaction.

These test specimens were irradiated to very low levels compared to spent nuclear fuel. Available records show irradiation in the 100 to 1000 megawatt days per ton range. For example, one report showed a group of slugs destined for SPRU being irradiated to 400 to 600 megawatt days per ton (Hanford 1952). In comparison, spent fuel from nuclear power reactors is typically irradiated to 40,000 megawatt days per ton (IAEA 2005). Each irradiated slug would therefore contain a small fraction of the fission products present in an equal amount of spent nuclear fuel. This comparison shows that the irradiated test specimens used in SPRU research and development work were more like unused nuclear fuel than spent nuclear fuel. In fact, 22 percent of the test specimens used in the SPRU process were not irradiated at all (KAPL 1998) and therefore contained no fission products.

This information demonstrates that the slightly irradiated and non-irradiated uranium slugs used in the SPRU research and development work were not spent nuclear fuel.

4.5 High-Level Waste

DOE Manual 435.1-1, Section II.A, defines high-level waste as follows:

> "High-level waste is the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel [defined in Section 4.4], including liquid waste produced directly in

[13] Separation processes include aqueous separation processes, e.g., the REDOX and the PUREX processes, and nonaqueous processes, e.g., pyrometallurgical and pyrochemical processes (DOE G 435.1-1). They are used to separate desired radioisotopes from unwanted radioactive materials.

reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation."

The solidified sludge waste form is not high-level waste for the following reasons:

***It did not result from reprocessing of spent nuclear fuel*.** As discussed above, the slightly irradiated uranium slugs used in the SPRU research were not spent nuclear fuel. Even if they had been, the SPRU facility did not possess the characteristics of a reprocessing facility. As a laboratory-scale facility used for experimental or analytical purposes, SPRU was not a production facility designed or used for the separation of the isotopes of plutonium as defined by the Nuclear Regulatory Commission (NRC) at 10 CFR 50.2[14]. Instead, the research and development work involved testing and refining various aspects of the separations process activities, which would have included activities such as varying pH and performing valance studies.

One key to understanding the research-related nature of the SPRU facility is that as soon as an appropriate technology for actinide separations was identified, the SPRU facility stopped work and large production separations facilities were constructed and operated at Savannah River Site, Hanford, and, later, at West Valley. SPRU activities were clearly not undertaken for production purposes.

The Department considers reprocessing "to be those actions necessary to separate fissile elements (U-235, Pu-239, U-233, and Pu-241) and/or transuranium elements (e.g., Np, Pu, Am, Cm, Bk) from other materials (e.g., fission products, activated metals, cladding) contained in spent nuclear fuel for the purposes of recovering desired materials" (DOE Guide 435.1-1). The SPRU laboratory-scale research was performed to support development of processes to separate and recover such materials, rather than to recover the desired materials.

The waste associated with SPRU operations was not highly radioactive material akin to high-level waste from reprocessing. The highly radioactive nature of high-level waste from reprocessing is associated with radiation from fission products, primarily Sr-90 and Cs-137. The amounts of fission product activity in the underground waste tanks were much less than those

[14] The definition at 10 CFR 50.2 states that: "*Production facility* means:

(1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; or

(2) Any facility designed or used for the separation of the isotopes of plutonium, <u>except laboratory scale facilities designed or used for experimental or analytical purposes only</u>; or

(3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, except (i) <u>laboratory scale facilities designed or used for experimental or analytical purposes</u>, (ii) facilities in which the only special nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U-235 and plutonium produced by the irradiation, if the material processed contains not more than 10^{-6} grams of plutonium per gram of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235, and (iii) facilities in which processing is conducted pursuant to a license issued under parts 30 and 70 of this chapter, or equivalent regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear material, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to not more than 100 grams of uranium enriched in the isotope 235 and not more than 15 grams of any other special nuclear material." [emphasis added]

in underground waste tanks that contain high-level waste at other DOE sites. Table 4 shows estimated Sr-90 and Cs-137 activity in the underground waste tanks in 1963 before comprehensive cleanup of the tanks compared to activity in two underground high-level waste tanks at the West Valley Demonstration Project at the conclusion of reprocessing at that site.

Table 4. Estimated Fission Product Activity Comparison Before Cleaning (in Curies)

Radionuclide/Tank	*SPRU 505*(1)	*SPRU 509A*(1)	*WVDP 8D-2*(2)	*WVDP 8D-4*(2)
Sr-90	354	704	6,900,000	450,000
Cs-137	377	636	7,400,000	480,000

LEGEND: WVDP = West Valley Demonstration Project
NOTES: (1) From EGR 2006a. Data are available for only two of the seven tanks.
(2) From Rykken 1986. Tank 8D-2 is a 750,000-gallon capacity carbon steel tank. Tank 8D-4 is a 15,000-gallon stainless steel tank. Both tanks contained high-level waste.

The comparison in Table 4 is not exact because the SPRU fission product estimates were made ten years after research and development ended and the tanks received liquid radioactive waste from various KAPL laboratories as well as from SPRU operations. But it is clear from information in the table the amounts of fission product activity in the tanks were much less than in tanks used to store high-level waste from reprocessing.

Although dose rate data associated with the SPRU underground waste tanks in the early years are limited, it is clear that dose rates associated with the SPRU tanks were much less than those associated with high-level waste tanks. Gamma dose rates inside the SPRU tank vaults in 1971 were typically in the 100 to 150 mR/hr range (ERG 2006a). As shown in Table 3, the maximum gamma dose rates inside the tanks in 2010 before sludge removal ranged from 15 to 1000 m/R/hr. For perspective, a gamma dose rate of 400 R/hr (400,000 mR/hr) was measured inside Tank 8D-1 at the West Valley Demonstration Project (Fazio 2002).

Similar comparisons can be made after the SPRU tanks have been cleaned. Table 5 shows a comparison of fission product content between representative SPRU underground waste tanks before sludge removal and typical underground waste tanks that have contained high-level waste.

Table 5. Estimated Fission Product Activity Comparison After Cleaning (in Curies)

Radionuclide/Tank	*SPRU 505*(1)	*SPRU 509A*(1)	*SRS Tank 18*(2)	*WVDP Tank 8D-4*(3)
Sr-90	0.36	6.08	1100	2400
Cs-137	0.69	18.91	9700	1800

LEGEND: SRS = Savannah River Site, WVDP = West Valley Demonstration Project
NOTES: (1) From Table 1 prior to sludge removal.
(2) From SRR 2010.
(3) From WVNSCO and Gemini 2005.

This table shows that even before sludge was removed from the SPRU tanks, they contained orders of magnitude less radioactivity than high-level waste tanks that had been cleaned.

4.6 Transuranic Waste

DOE Manual 435.1-1, Section III.A, defines transuranic waste as follows:

> "Transuranic waste is radioactive waste containing more than 100 nanocuries (3700 becquerels) of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years, except for:
>
> (1) High-level radioactive waste;
>
> (2) Waste that the Secretary of Energy has determined, with the concurrence of the Administrator of the Environmental Protection Agency, does not need the degree of isolation required by the 40 CFR Part 191 disposal regulations; or
>
> (3) Waste that the Nuclear Regulatory Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61."

The solidified sludge waste form will not be transuranic waste because it will not contain more than 100 nanocuries of alpha-emitting transuranic isotopes with half-lives greater than 20 years per gram of waste. This matter has been directly addressed in the waste profile sheet. The solidified sludge waste form is expected to contain 30 to 40 nanocuries per gram of alpha-emitting transuranic isotopes with half-lives greater than 20 years and it will be verified that all of the waste containers have less than 95 nanocuries per gram of these radionuclides (DK-ES 2010).

As explained in the waste profile sheet, the concentrations of alpha-emitting transuranic radionuclides with half-lives exceeding 20 years in the consolidated sludge-water mixture in Tank 509E exceed 100 nanocuries per gram. However, it is necessary to solidify this material for disposal purposes because of the free liquids in the waste in order to meet the disposal facility waste acceptance criteria (DOE-NV 2011)[15,16,17]

Solidifying the sludge-water mixture to meet the disposal facility waste acceptance criteria is consistent with Department guidance in DOE Guide 435.1-1, *Implementation Guide for Use with DOE M 435.1*.

---

[15] Section 3.1.5 of the waste acceptance criteria (DOE-NV 2011) states that "Liquid waste and waste containing free liquids ***shall*** be converted into a form that contains as little free-standing and noncorrosive liquid as is reasonable achievable. Liquid waste and waste containing free liquids should be processed to a solid form or packaged in sufficient absorbent for twice the volume of liquid." It would not be feasible to use absorbent with the sludge-water mixture, so solidification is required.

[16] Solidification would have also been required if the waste had contained higher concentrations of alpha-emitting transuranic radionuclides so the final waste form would be transuranic waste in order to comply with the waste acceptance criteria for the Waste Isolation Pilot Plant.

[17] The solidification process does not amount to dilution of the waste. DOE Guide 435.1-1 states that "Dilution of a transuranic waste stream to reclassify the waste as low-level waste (i.e., reducing the concentration to less than or equal to 100 nCi (3700 Bq) per gram) is not permitted by the Department. While it is recognized that in the course of stabilizing a waste stream some changes in waste concentration may occur, actions to dilute a waste stream below the concentration limits for transuranic waste are prohibited. <u>It is also recognized that actions taken to process a waste stream for safety or technological reasons that are justified, may result in the waste being reclassified after processing as low-level waste</u>." [emphasis added]

4.7 Byproduct Material

The *Atomic Energy Act*, as revised in 1978 and in 2005 by the *Energy Policy Act*, defines byproduct material in Section 11e. The sludge is not byproduct material because it does not meet the statutory definition.

4.8 Naturally Occurring Radioactive Material

The radioactivity in the sludge is obviously not naturally occurring radioactive material since it was produced in connection with laboratory-scale research and development work and other laboratory procedures.

4.9 Definition-Based Waste Type of the Solidified Sludge

Based on the definitions of the various waste types described above, the solidified sludge will be low-level waste because it is not high-level waste, spent nuclear fuel, transuranic waste, byproduct material, or naturally occurring radioactive material. This conclusion has been substantiated by the Nevada National Security Site's approval of the waste profile sheet for disposal of the solidified sludge as low-level waste at that facility (DOE-NV 2010)[18].

[18] The rationale for establishing the waste type of the stabilized sludge just described also applies to the tanks themselves because the residual contamination in the tanks is the same material as the removed sludge. That is, the tanks will be low-level waste like the solidified sludge.

5.0 KEY RADIONUCLIDE REMOVAL

The purpose of this section is to evaluate, for additional information, whether the first evaluation criterion of Section II.B2(a) of DOE Manual 435.1-1, *Radioactive Waste Management*, concerning waste incidental to reprocessing, would be satisfied if SPRU had generated waste from reprocessing of spent nuclear fuel such that this criterion were to apply to the waste. This criterion reads as follows:

> "[The subject wastes] have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical."

This section begins with a discussion of key radionuclides. It then discusses methods used to remove key radionuclides, and whether those methods removed or would remove key radionuclides to the maximum extent technically and economically practical. It ends with the conclusion about meeting this criterion.

5.1 Key Radionuclides

5.1.1 Introduction

The key radionuclides in this evaluation are based on consideration of the following information:

- Guidance in DOE Guide 435.1-1 on identification of key radionuclides;
- NRC requirements for classification of radioactive waste for near-surface disposal that appear in 10 CFR 61.55;
- Radionuclides known to be present in the sludge;
- The relationship between DOE disposal site waste acceptance criteria and the performance of DOE LLW disposal sites in meeting objectives for protecting individuals and the environment; and
- The radionuclides of importance in the performance assessment of the Nevada National Security Site Area 5 LLW disposal area, although such consideration is not required by DOE Manual 435-1 or DOE Guide 435.1-1.

Consideration of this information will ensure that those radionuclides in the solidified sludge that could contribute significantly to radiological risks to workers, the public, and the environment are identified and taken into account.

5.1.2 DOE Guidance on Key Radionuclides

DOE guidance on selection of key radionuclides is provided in Section II.B of DOE Guide 435.1-1, with the applicable portion reading as follows:

> "... it is generally understood that [the term] key radionuclides applies to those radionuclides that are controlled by concentration limits in 10 CFR 61.55. Specifically these are: long-lived radionuclides, C-14, Ni-59, Nb-94, Tc-99, I-129, Pu-241, Cm-242, and alpha emitting transuranic nuclides with half-lives greater than five years and; short-lived radionuclides, H-3, Co-60, Ni-63, Sr-90, and Cs-137. In addition, key radionuclides are those that are important to satisfying the performance objectives of 10 CFR Part 61, Subpart C [for near-surface radioactive waste disposal facilities]."

This guidance considers both the waste classification requirements in 10 CFR 61.55 for radioactive waste destined for near-surface disposal and achieving the waste disposal site performance objectives.

5.1.3 Requirements of 10 CFR 61.55

The radionuclides listed in the guidance found in DOE Guide 435.1 appear in 10 CFR 61.55 in the form of two tables, which are reproduced here as follows.

Table 6. 10 CFR 61.55, Table 1 (Long-Lived Radionuclides)

Radionuclides	*Concentration (Ci/m^3)*
C–14	8
C–14 in activated metal	80
Ni–59 in activated metal	220
Nb–94 in activated metal	0.2
Tc–99	3
I–129	0.08
Alpha-emitting transuranic nuclides with half-life greater than 5 years	100(1)
Pu–241	3,500(1)
Cm–242	20,000(1)

NOTE: (1) These values are in units of nanocuries per gram.

Table 7. 10 CFR 61.55, Table 2 (Short-Lived Radionuclides)

Radionuclides	*Concentration (Ci/m^3)*		
	Column 1 [Class A]	*Column 2 [Class B]*	*Column 3 [Class C]*
Total of all nuclides with less than 5 year half-life	700	(1)	(1)
H–3	40	(1)	(1)
Co–60	700	(1)	(1)
Ni–63	3.5	70	700
Ni–63 in activated metal	35	700	7000
Sr–90	0.04	150	7000
Cs–137	1	44	4600

NOTE: (1) There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in the table determine the waste to be Class C independent of these nuclides.

The concentrations given in these tables are used for low-level waste classification purposes. The class of low-level waste is determined by concentrations of long-lived radionuclides, by concentrations of short-lived radionuclides, or by both in those cases where the waste contains both types of radionuclides.

5.1.4 Radionuclides in the Sludge

The sludge contains a mixture of both long-lived and short-lived radionuclides. The classification concentration limits in 10 CFR 61.55 for waste containing both long-lived and short-lived radionuclides are as follows:

(1) If the concentration of a nuclide listed in Table 1 does not exceed 0.1 times the value listed in Table 1, the class shall be that determined by the concentration of nuclides listed in Table 2.

(2) If the concentration of a nuclide listed in Table 1 exceeds 0.1 times the value listed in Table 1 but does not exceed the value in Table 1, the waste shall be Class C, provided the concentration of nuclides listed in Table 2 does not exceed the value shown in Column 3 of Table 2.

For mixtures of radionuclides, 10 CFR 61.55 specifies that the sum-of-fractions rule will be used in determining the class of low-level waste. This rule entails dividing each radionuclide's concentration by the appropriate limit, adding the resulting fractions, and comparing their sum to 1.0. A sum of fractions less than 1.0 indicates compliance of the radionuclide mixture with the relevant classification criteria.

As noted previously, DOE Guide 435.1-1 indicates that one criterion for determining key radionuclides in waste is their importance in satisfying safety requirements comparable to the performance objectives of 10 CFR Part 61, Subpart C for the waste disposal facility. These performance objectives are described in Section 6.2 below.[19]

5.1.5 Radionuclides Important to the Disposal Site Performance Assessment

Because meeting the waste acceptance criteria for a given disposal facility ensures that the facility performance objectives will be achieved, those radionuclides that are of particular importance in the disposal site performance analyses are considered in identifying key radionuclides. These

[19] In practice, meeting the waste acceptance criteria for the disposal facility ensures that the facility performance objectives will be achieved. The rationale for this conclusion for a DOE low-level waste disposal facility such as Area 5 at the Nevada National Security Site may be briefly summarized as follows:

- DOE performance objectives for its low-level waste disposal facilities are comparable with those of 10 CFR 61, Subpart C;
- Disposal site performance in compliance with the performance objectives is determined by a performance assessment of the facility and by a composite analysis that considers other radioactivity sources in the area along with the radioactivity in the disposal site;
- These analyses are based on a projected total radionuclide inventory for the full, closed disposal site;
- This projected total inventory is based on the waste acceptance criteria, thus linking these criteria directly to the calculated disposal site performance;
- The subject low-level waste stream (the solidified sludge) will meet the waste acceptance criteria; and
- Meeting the waste acceptance criteria will therefore ensure that the performance objectives will be achieved, because waste meeting these criteria would not increase the assumed waste inventory used in the performance assessment analyses.

These matters are addressed in more detail in Section 6.2.

radionuclides are Tc-99, Th-229, U-233, U-234[20], and U-238 for the Nevada National Security Site Area 5 waste disposal area (NST 2010). The results of the latest performance assessments of the Nevada National Security Site low-level waste disposal areas are discussed in Section 6.2 of this evaluation.

5.1.6 Conclusions About Key Radionuclides in the Sludge

Based on consideration of the factors discussed above, DOE considers all radionuclides listed in Tables 1 and 2 of 10 CFR 61.55 to be key radionuclides for the purposes of this evaluation, with the caveat that some are of lesser importance due to their low concentrations in the waste, their small dose conversion factors, or both. Table 8 shows these radionuclides.

Table 8. Key Radionuclides for this Evaluation

Radionuclide	*10 CFR 61.55 Long-Lived Radionuclides*	*10 CFR 61.55 Short-Lived Radionuclides*	*Radionuclides Important to PA*
H-3		X	
C-14	X		
Co-60		X	
Ni-59	X		
Ni-63		X	
Sr-90		X	
Nb-94	X		
Tc-99	X		X(1)
I-129	X		
Cs-137		X	
Th-229			X(1)
U-233			X(1)
U-234			X(1)
U-238			X(1)
Np-237(3)	X		
Pu-238(3)	X		
Pu-239(3)	X		
Pu-240(3)	X		
Pu-241	X		
Pu-242(3)	X		
Am-241(3)	X		

[20]U-234 present at the time of disposal is the predominant source of Pb-210 (NST 2010). Pb-210 was not identified as a key radionuclide because its presence at the time of estimated maximum dose is due to U-234 in the disposed of waste, rather than Pb-210 in the waste.

Table 8. Key Radionuclides for this Evaluation

Radionuclide	*10 CFR 61.55 Long-Lived Radionuclides*	*10 CFR 61.55 Short-Lived Radionuclides*	*Radionuclides Important to PA*
Am-243	X		
Cm-242	X		
Cm-243(2)	X		
Cm-244(2)	X		

LEGEND: PA = performance assessment

NOTES: (1) Radionuclides important to the performance assessment of the Area 5 radioactive waste management site.
(2) Alpha emitting transuranic radionuclides with half-life greater than five years.

5.2 Removal to the Maximum Extent Technically and Economically Practical

This section begins with a brief discussion intended to provide perspective on the application of this criterion to the SPRU underground waste tank sludge.

5.2.1 Introduction

Removal to the maximum extent "technically and economically practical" is not removal to the extent "practicable" or theoretically "possible."[21] Nor does the criterion connote removal which may be notionally capable of being done.[22] Rather, the adverbs "technically" and "economically" modify and add important context to that which is contemplated by the criterion. Moreover, a "practical" approach as specified in the criterion is one that is "adapted to actual conditions" (Fowler 1930); "adapted or designed for actual use" (Random House 1997); "useful" (Random House 1997); selected "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure" (Random House 1997); fitted to "the needs of a particular situation in a helpful way" (Cambridge 2004); "effective or suitable" (Cambridge 2004). Therefore, the evaluation as to whether a particular key radionuclide has been or will be removed to the "maximum extent that is technically and economically practical" will vary from situation to situation, based not only on reasonably available technologies but also on the overall costs and benefits of deploying a technology with respect to a particular waste stream. The "maximum extent that is technically and economically practical" standard contemplates, among other things: consideration of expert judgment and opinion; environmental, health, timing, or other exigencies; the risks and benefits to public health, safety, and the environment arising from further radionuclide removal as compared with countervailing considerations that may ensue from not removing or delaying removal; life cycle costs; net social value; the cost (monetary as well as environmental and human health and safety

[21] In evaluating whether key radionuclides have been removed to the maximum extent that is "technically and economically practical", DOE has considered the guidance in DOE Guide 435.1-1 as well as the plain meaning of the phrase "technically and economically practical." NRC staff guidance for NRC consultation activities related to DOE waste determinations (NRC 2007), and the approach taken pursuant to the similar criterion in Section 3116(a) of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (see e.g., Basis for Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center Tank Farm Facility (DOE 2006)).

[22] Compare the meaning of practical with the meaning of *practicable*. As commonly understood, *practicable* refers to something that can be put into effect. *Practical* refers to something that is also sensible and worthwhile. Thus, it might be *practicable* to transport children to school by balloon, but it would not be *practical*. (Webster's 1984)

costs) per curie removed; radiological removal efficiency; the point at which removal costs increase significantly in relationship to removal efficiency; the service life of equipment; the reasonable availability of proven technologies; the limitations of such technologies; the usefulness of such technologies; and the sensibleness of using such technologies. What may be removal to the maximum extent technically and economically practical in a particular situation or at one point in time may not be that which is technically and economically practical, feasible, or sensible in another situation or at a prior or later point in time. In this regard, it may not be technically and economically practical to undertake further removal of certain radionuclides because further removal is not sensible or useful in light of the overall benefit to human health and the environment.[23]

NOTE

For perspective, it should be noted that most of the waste and the associated radioactivity had been removed from the seven underground waste tanks before the sludge removal activities of the current decommissioning program began. Section 3.2 notes that the waste removed in 1965 contained 2,377 curies of fission products, 903 kilograms of natural uranium, and 539 grams of plutonium (539 grams of Pu-239 would have contained about 36 curies of activity). Section 3.2 also notes that liquid waste was emptied from the tanks in 1978.

5.2.2 Sludge Removal

The method used to remove the sludge from the six tanks was technically and economically practical. High-pressure, low-volume waste spray effectively dislodged and mobilized the sludge and the pumps used removed approximately 99 percent of the sludge that was in the tanks.[24] This method was generally similar to the high-pressure water spray method used to effectively clean the stainless steel underground waste tanks at the Idaho National Laboratory (DOE-ID 2006).

Table 9 shows other representative methods and identifies whether they would be technically and economically practical for removal of the sludge in the previously cleaned tanks. Note that Table 9 identifies the method already used as method number 1.

[23] As a general matter, such a situation may arise if certain radionuclides are present in such extremely low quantities that they make an insignificant contribution to potential dose to workers, the public, and the hypothetical human intruder.

[24] The estimated volume of sludge in the six tanks before tank cleaning from Table 1 was approximately 220 cubic feet. The estimated volume of sludge-water mixture remaining in the six tanks based on the estimates in Section 3.4 is approximately 95 gallons or 12.7 cubic feet. This sludge originally had moisture content in the 46.9 to 83.5 weight percent range and the consolidated sludge in tank 509E has a moisture content of 95.6 weight percent (DK-ES 2010). Considering the volume reduction and the difference in moisture content, assuming that the moisture content of the sludge-water mixture remaining in the six tanks in the same as the sludge-water mixture in Tank 509E, the process removed approximately 99 percent of the sludge.

Table 9. Tank Sludge Removal Methods, Technical and Economic Practically

No.	*Method*	*TP*	*EP*	*Basis for Technical and Economic Practicality*
1	Remotely operated high-pressure, low-volume water spray with pumps to remove sludge	yes	yes	Visual observations and reductions in dose rates showed that this method to be effective. It removed approximately 99 percent of the sludge from the six tanks.
2	Worker entering tanks to remove sludge using manual tools and standard cleaning methods.[(1)]	yes	no	Commonly used method to clean tanks that are not radioactively contaminated. However, the hazards involved would far outweigh the benefits. The hazards would include confined space dangers, additional personnel radiation exposure, and hazards associated with work in high radioactive contamination levels. Use of this method would not result in overall benefit to human health and the environment and would be inconsistent with as reasonably achievable (ALARA) principles of DOE Policy 441.1, *Department of Energy Radiological Health and Safety Policy*.
3	Robotic waste retrieval device called sand mantis.[(2)]	no	-	This eight-foot long, 800 pound robotic device uses a high-pressure water jet to create vacuum to remove material from a waste tank. It is designed for use with large, flat-bottomed tanks. It would not be technically practical to use it in smaller, curved-bottom tanks such as the SPRU underground waste tanks.
4	Fire hose sluicing.[(2)]	yes	no	This process offers no advantages over the method used. It would generate an excessive amount of water containing sludge that would have to be solidified, greatly increasing the solidified waste volume and the associated transportation and disposal costs.
5	Vacuum retrieval.[(3)]	no	-	This complex equipment uses a venturi system to remove waste from large underground waste tanks. It could not be readily adapted for use in the small SPRU tanks.

LEGEND: EP = economically practical TP = technically practical.

NOTES: (1) Workers entered a 300,000-gallon underground waste tank during the decommissioning of the Barnwell Nuclear Fuel Plant to stabilize a mixed waste heel containing natural uranium, but the dose rates inside the tank were insignificant (ASME 2004, Chapter 28).

(2) Used effectively in underground waste tank cleaning at the Savannah River Site (SRNL 2010).

(3) Developed for use in underground waste tank cleaning at the Hanford site (SRNL 2010, Berriochoa 2011).

5.2.3 Additional Treatment of the Removed Sludge

Consideration was given to methods that could be used to remove key radionuclides from the consolidated sludge in Tank 509E. For example, the dominant radionuclide in the sludge is cesium 137 and a wide range of technologies to remove cesium 137 from a liquid waste stream are available (ASME 2004). For example, an ion exchange decontamination process could be used for this purpose[25].

Treating the consolidated sludge by ion exchange would require obtaining and setting up the necessary equipment, operating it to treat the waste stream, dismantling it, and disposing of the associated waste, including radioactively contaminated filters and spent ion exchange resin. This process would therefore produce a liquid waste stream with a lower concentration of Cs-137 along with additional solid radioactive waste to be disposed of, including the spent resin which would typically have to be solidified in cement because of its free liquid content.

This process would be technically practical. However, it would not be economically practical or sensible for a number of reasons including the following:

- The waste will meet waste acceptance criteria for a federal low-level waste disposal facility after it is solidified to stabilize the liquids.
- It does not require further processing that would result in additional worker radiation exposure and additional industrial hazards that would be inconsistent with the DOE ALARA requirements.
- It is a low specific activity waste form that meets transportation requirements without special packaging and shielding.
- Treating the waste form to further remove key radionuclides would produce a second waste stream that would be more dangerous to workers, the public and the environment because the technology employed would most likely be ion exchange. Ion exchange, by concentrating the key radionuclides such as Cs-137 and Sr-90, would result in a waste form that would have significant dose implications during operation, packaging, transportation, and disposal. The end result of any additional treatment would still be a conditioned sludge that would have to be solidified and disposed as radioactive waste, along with a concentrated resin that would also require solidification and possibly shielding for disposal. A concentrated source term would typically require additional action by the disposal facility to ensure that the performance assessment results would not be affected by its disposal.

Consideration of such factors leads to the conclusion that removal of the key radionuclide Cs-137 using a process such as ion exchange would produce no net benefit to human health and the environment, would be inconsistent with ALARA principles, and would not be economically practical. Similar logic would apply to any other process that might be used to remove key radionuclides from the waste stream. For example, fibrous activated carbon has been found to be effective in removing plutonium from a radioactive waste stream (Tajiri, et al. 2000). Use of this process would not be economically practical for the same reasons as ion exchange, because the solidified sludge requires no further treatment for transportation and disposal as low-level waste and substantial disadvantages would be associated with any additional treatment.

[25] Ion exchange involves a reversible chemical reaction between an insoluble solid and a solution during which ions may be interchanged. This process is commonly used in the separation of radionuclides and for water softening.

5.3 Conclusion

The information in Table 9 shows that two of the four representative alternate methods for removing additional sludge from the tanks would be technically practical but neither of these would be economically practical. The discussion about additional treatment of the consolidated sludge shows that this would not be economically practical. Consequently, the waste has been processed to remove key radionuclides to the maximum extent that is technically and economically practical, meeting the criterion.

6.0 MEETING SAFETY REQUIREMENTS

6.1 Introduction

Because the solidified sludge waste packages will not exceed concentration limits for Class C low-level waste as explained in Section 7, the second criterion of section II.B.2(a) of DOE Manual 435.1-1 applies to this evaluation:

> "[The waste] will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, *Performance Objectives*."

This section discusses different elements of the DOE safety requirements for disposal of low-level waste – which are comparable with NRC performance objectives at 10 CFR Part 61, Subpart C – and shows that this criterion will be met in solidification of the sludge and its disposal at the Nevada National Security Site.

6.2 DOE Safety Requirements

DOE has established in DOE Manual 435.1-1 requirements for management of radioactive waste to ensure protection of workers, the public, and the environment, and complies with applicable Federal, State, and local laws and regulations. DOE has also established specific requirements for its radioactive waste disposal facilities, including the Area 5 radioactive waste management site at the Nevada National Security Site where the solidified sludge is planned to be disposed of. These requirements include:

(1) Performance objectives set forth in Chapter IV of DOE Manual 435.1-1, which include maximum dose limits;

(2) DOE regulations at 10 CFR Part 835, *Occupational Radiation Protection*, and DOE Order 5400.5, *Radiation Protection of the Public and the* Environment[26], which are cross referenced in Chapters I and IV of DOE Manual 435.1-1;

(3) Waste acceptance requirements, which, among other things, establish limits on radionuclides that may be disposed of based on a performance assessment of the facility;

(4) A performance assessment (and updates) of the facility, to provide reasonable expectation that DOE's performance objectives will not be exceeded;

(5) A composite analysis that considers other radioactivity sources in the area as well as the disposal facility;

(6) A performance assessment and composite analysis maintenance plan,

(7) A preliminary closure plan; and

(8) A monitoring plan (DOE Manual 435.1-1).

For wastes to be disposed of at DOE facilities, DOE establishes waste acceptance criteria, based upon an independently reviewed and accepted low-level waste performance assessment, which

[26] DOE Order 458.1, *Radiation Protection of the Public and the Environment*, has superseded DOE Order 5400.5 except for except for Chapter III (Derived Concentration Guides for Air and Water) and Figure IV-1 (Surface Contamination Guidelines). The dose standards in the two orders are identical.

also include provisions for maintenance and updating. Acceptability of the low-level waste performance assessment is verified against the performance objectives of Section IV.P of DOE Manual 435.1-1, as well as other requirements in DOE Manual 435.1-1, through an independent review process. This review serves as the basis for DOE to issue a Disposal Authorization Statement, which specifies any additional conditions that the site may need to impose to ensure the performance objectives of DOE Manual 435.1-1, IV.P are met.

These performance objectives, regulations, and Orders are set forth or cross referenced in DOE Manual 435.1-1. They provide safety requirements comparable to the NRC performance objectives of 10 CFR 61, Subpart C (Wilhite 2001). The relevant performance objectives are described below.

6.2.1 General Safety Requirement

The general requirement in DOE Manual 435.1-1, Section IV.P(1), is expressed as follows:

> “Low-level waste disposal facilities shall be sited, designed, operated, maintained, and closed so that a reasonable expectation exists that the following performance objectives will be met for waste disposed of after September 26, 1988.”

The four relevant DOE performance objectives are addressed in Subsections 6.2.2 through 6.2.5.

6.2.2 Protection of the General Population from Releases of Radioactivity

DOE requirements in DOE Manual 435.1-1, Section IV.P(1), read as follows:

“(a) Dose to representative members of the public shall not exceed 25 millirem (0.25 mSv) in a year total effective dose equivalent from all exposure pathways, excluding the dose from radon and its progeny in air.

(b) Dose to representative members of the public via the air pathway shall not exceed 10 millirem (0.10 mSv) in a year total effective dose equivalent, excluding the dose from radon and its progeny.

(c) Release of radon shall be less than an average flux of 20 pCi/m^2/s (0.74 Bq/m^2/s) at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/L (0.0185 Bq/L) of air may be applied at the boundary of the facility.”

To address this requirement, the projected performance of the planned disposal site and the potential impact of disposal of the solidified sludge are briefly discussed.

Assessment of Area 5 Performance

The report of the most recent annual review of performance assessments and composite analyses for the Area 3 and Area 5 waste disposal facilities at the Nevada National Security Site was issued in 2010 (NST 2010). This report addresses matters such as new or revised waste streams, monitoring results, research and development, the inventory estimates at planned closure, updated performance assessment results, and updated composite analysis results. It also identifies special analyses that were performed in the previous year[27].

As explained in the report of the annual review (NST 2010), the updated Area 5 performance assessment provides reasonable expectation that DOE's performance objectives will be achieved. This report summarizes the results of probabilistic analyses for Area 5 for the following scenarios:

- All pathways dose for members of the public, with the predicted peak annual dose estimated to occur at 1000 years after planned closure (i.e., in 3028, the end of the compliance period) for the resident farmer scenario;
- The air pathway dose for members of the public, with the predicted peak annual dose to a resident farmer at 1000 years after facility closure; and
- The Radon 222 flux density at the surface of the disposal units, which is predicted to reach a peak 1000 years after facility closure.

This report shows that the predicted potential doses to representative members of the public to be less than the performance objective dose limits. Table 10 shows the specific results of the performance assessment, including the inadvertent intruder scenarios that are discussed below.

Table 10. Area 5 Performance Assessment Results[(1)]

Criterion (Controlling Exposure Scenario or Disposal Unit)	***Limit***	***Mean***	***95^{th} Percentile***	***Time of Maximum***
Air Pathway Dose to Member of the Public (mrem/yr) (Resident Farmer)	10	0.046	0.19	1000 yr
All Pathways Dose to Member of Public (mrem/yr) (resident farmer)	25	1.3	4.2	1000 yr
Rn-222 Flux Density ($Bq/m^2/s^2$) (Pit 13)	0.74	0.72	2.1	1000 yr
Acute Drilling Intruder(mrem/yr) (Pit 6)	500	2.8	5.2	1000 yr
Acute Construction Intruder(mrem/yr) (Shallow Land Burial Unit)	500	250	510	1000 yr

NOTE: (1) From NST 2010. Note that the mean dose estimates are used for determining compliance.

[27] A low-level waste disposal facility performance assessment involves detailed analyses of potential radiation doses to those who may be affected in future years to ensure that the closed facility will meet its performance objectives. These performance objectives include dose limits for a member of the public and for a hypothetical person who, unaware of the buried radioactivity, might drill a well into the buried waste, referred to as the post-drilling scenario, or establish a farm on the site, known as the intruder-agriculture scenario.

Special analyses and composite analyses use similar methodologies, with the focus on the waste stream of interest and all relevant radioactivity sources at the site, respectively. Special analyses are performed for waste streams with a sum of fractions of radionuclide action levels greater than one or where preliminary screening indicates that disposal of a new waste stream has a potential to alter performance assessment assumptions or conceptual models. A composite analysis is required for all DOE sites that manage radioactive waste.

Both performance assessments and composite analyses are required to be maintained in accordance with Section IV.P.(4) of DOE Manual 435.1-1.

Estimated Impact of the Solidified Sludge

Because it meets the waste acceptance criteria (DOE-NV 2011) as demonstrated in the approved waste profile (DK-ES 2010), disposal of the solidified sludge will have an insignificant impact on performance of the Area 5 radioactive waste management site. Before acceptance, the waste profile was reviewed by the Nevada National Security Site Waste Acceptance Review Panel, a group of waste management specialists who review new and revised waste streams planned for disposal in the Area 5 Radioactive Waste Management Site.

6.2.3 Protection of Individuals from Inadvertent Intrusion

DOE requirements of DOE Manual 435.1-1, Section IV.P(2)(h), for protection of individuals from inadvertent intrusion read as follows:

> "For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 millirem (1 mSv) in a year and 500 millirem (5 mSv) total effective dose equivalent excluding radon in air."

Assessment of Area 5 Performance

The report of the most recent annual review of performance assessments and composite analyses for the Area 3 and Area 5 radioactive waste management sites at the Nevada National Security Site (NST 2010) demonstrates that there is a reasonable expectation that the Area 5 Radioactive Waste Management Site will meet the DOE intruder dose criteria. The scenarios evaluated as described in this report were as follows:

- The drilling worker intruder scenario, with the predicted peak annual acute dose 1000 years after facility closure, the end of the compliance period; and
- The home construction intruder scenario, with the predicted peak annual acute dose 1000 years after facility closure.

Table 10 shows the results of evaluation of these scenarios to be well under the 500 mrem/yr llimit.

Chronic intruder scenarios are no longer reported in the Annual Summary Report for the Area 5 Radioactive Waste Management Site because chronic intrusion would be unlikely due to a change in the institutional control policy made in 2008. The planned land-use restrictions will prohibit public access to groundwater for 1,000 years within the compliance boundary negotiated with the State of Nevada, which is to include the Area 5 Radioactive Waste Management Site.[28] (NST 2010)

Estimated Impact of the Solidified Sludge

Because it meets the waste acceptance criteria (DOE-NV 2011) as demonstrated in the approved waste profile (DK-ES 2010), disposal of the solidified sludge will have an insignificant impact on compliance with the requirements for protection of individuals from inadvertent intrusion

6.2.4 Protection of Individuals During Operations

The DOE requirements in DOE Manual 435.1-1, Section I.E(13), for protection of individual during operations read as follows:

> "Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR Part 835, *Occupational Radiation Protection,* and DOE 5400.5, *Radiation Protection of the Public and the Environment*."

The DOE requirements apply to the workers at SPRU who will be involved with solidifying the sludge and preparing the waste packages for disposal, as well as to the public at the site. The DOE performance requirements also apply to the workers at the Nevada National Security Site who will handle disposal of the solidified sludge waste packages and to the public at that site.

Both SPRU and the Nevada National Security Site maintain radiation protection programs based on the requirements of 10 CFR 835. These programs also comply with various DOE directives (including DOE Order 5400.5, other orders, policies, guides, and manuals), and supplemental technical standards.

The SPRU radiological protection program and ALARA measures are described in the SPRU Radiological Controls Manual (URS 2009). The Nevada National Security Site radiological protection program and ALARA measures are described in the Nevada National Security Site Radiological Control Manual (NST 2009)

Compliance with the radiological control program requirements and the associated ALARA processes will provide reasonable expectation that SPRU worker doses will be well below the 500 mrem per year administrative limit, which is 10 percent of the annual DOE occupational dose limit of 5000 mrem per year in 10 CFR 835, Subpart C. Compliance with the SPRU radiological control program requirements and the associated ALARA processes will also ensure that potential exposures to the public from onsite work related to solidifying the sludge and preparing the waste packages for shipment are well below the applicable limit[29].

[28] Chronic intruder doses continue to be calculated by the performance assessment model but are no longer reported in the Annual Summary Report for reasons specified in that report (NST 2010). This practice is consistent with Section IV.P(2) of DOE Manual 435.1-1, which provides for considering the likelihood of inadvertent intruder scenarios in interpreting the results of the analyses if adequate justification is provided.

[29] The applicable limit is 10 mrem per year for exposure to a member of the public from air emissions (DOE Order 5400.5), which is the same as the U.S. Environmental Protection Agency requirement in 40 CFR 61.92, with which DOE complies.

Doses to workers at the Nevada National Security Site who will be involved with handling the solidified sludge waste packages will be minimized by compliance with that site's radiological control program and the associated ALARA processes. Compliance with the radiological control program requirements, following ALARA processes, and the short duration of the work to place the waste packages in the disposal facility provides reasonable expectation that worker doses will be ALARA.

Potential exposures to members of the public associated with onsite handling of the solidified sludge waste packages at the Nevada National Security Site are also expected to be very low. Operations to dispose of the solidified sludge waste packages would be of short duration, would take place in a radiologically controlled area with no routine public access, and would take place at the isolated government-controlled Nevada National Security Site.

6.2.5 Stability of the Disposal Site After Closure

The DOE requirements in DOE Manual 435.1-1, Sections IV.Q(1)(a) and (b) and IV.Q(2)(c), for stability of the disposal site after closure are expressed as follows:

> "Disposal Site Stability (DOE Manual 435.1, Section IV.Q(1)(a) and (b)). A preliminary closure plan shall be developed and submitted to Headquarters for review with the performance assessment and composite analysis. The closure plan shall be updated following issuance of the disposal authorization statement to incorporate conditions specified in the disposal authorization statement. Closure plans shall:
>
> (a) Be updated as required during the operational life of the facility.
>
> (b) Include a description of how the disposal facility will be closed to achieve long-term stability and minimize the need for active maintenance following closure and to ensure compliance with the requirements of DOE 5400.5, *Radiation Protection of the Public and the Environment*."
>
> "Disposal Facility Closure (DOE Manual 435.1, Section IV.Q(2)(c)). Institutional control measures shall be integrated into land use and stewardship plans and programs, and shall continue until the facility can be released pursuant to DOE Order 5400.5, *Radiation Protection of the Public and the Environment*."

DOE has developed a preliminary closure plan for the Area 5 radioactive waste management site in accordance with the DOE requirements. The plan will ensure that the applicable requirements of DOE Order 5400.5 will be met following closure of the Area 5 radioactive waste management site, which is currently planned for 2028. The applicable requirements of DOE Order 5400.5 include the public dose limit of 100 mrem per year effective dose equivalent (Chapter II, Section 1.a(1)) and the airborne emissions limit of 10 mrem per year effective dose equivalent (Chapter II, Section 1.b and 10 CFR Part 61).

6.2.6 Additional Information

Appendix A and Appendix B provide additional details on the comparability of DOE and NRC safety requirements and dose standards, respectively.

6.3 Conclusion

The foregoing discussion shows that the solidified sludge will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, *Performance Objectives*, thus meeting the subject criterion.

7.0 CLASS C CONCENTRATION LIMITS AND MANAGEMENT AS LOW-LEVEL WASTE

The purpose of this section is to demonstrate that the solidified sludge waste packages will be in a solid physical form, will not exceed Class C concentration limits, and will be managed in accordance with DOE requirements as low-level radioactive waste.

The third and final criterion of DOE Manual 435.1-1, Section II.B(2)(a) to be demonstrated is:

> "[The wastes] are to be managed, pursuant to DOE's authority under the Atomic Energy Act of 1954, as amended, and in accordance with the provisions of Chapter IV of DOE Manual 435.1-1, provided the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize."

As explained previously, the sludge will be solidified in cement in IP-2 shipping containers to comply with disposal site waste acceptance criteria. Hence, it will be in a solid physical form.

Because the solidified sludge waste form will contain a mixture of radionuclides, the total concentration is determined by the sum of the fractions rule, as specified in NRC's regulations at 10 CFR 61.55(a)(7). Additionally, because the radionuclide mixture contains some long-lived radionuclides that are listed on Table 1 of 10 CFR 61.55 (reproduced in Table 6 of this evaluation), and some short-lived radionuclides that are listed on Table 2 of 10 CFR 61.55 (reproduced in Table 7 of this evaluation), waste classification would be determined as specified in 10 CFR 61.55(a)(5), which states:

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

Radiological characterization of the sludge before solidification was as described in Section 3.3. Using the results of the characterization based on analytical data from the composite sample collected from the consolidated sludge-water mixture in Tank 509E and the estimated volume and mass of the final waste form, Table 11 below shows that the solidified sludge waste form will not exceed Class C concentration limits.

Table 11. Solidified Sludge Waste Concentration Results[1]

Isotope	*Estimated Activity (Ci)*	*Class C Limit (Ci/m³)*	*Class C Limit (nCi/g)*	*Waste Stream Concentration (Ci/m3)*	*Waste Stream Concentration (nCi/g)*	*Table 1[2] Fraction*	*Table 2[2] Fraction*
C-14	6.79E-04	8.00E+00		9.07E-06		1.14E-06	
Ni-63	5.66E-02	7.00E+02		7.57E-04			1.08E-06
Sr-90	1.64E+01	7.00E+03		2.19E-01			3.13E-05
Tc-99	2.09E-03	3.00E+00		2.80E-05		9.32E-06	
I-129	0.00E+00	8.00E-02		0.00E+00		0.00E+00	
Cs-137	4.38E+01	4.60E+03		5.86E-01			1.27E-04
Np-237	1.86E-04		1.00E+02		1.39E-03	1.39E-05	
Pu-238	7.33E-02		1.00E+02		5.47E-01	5.47E-03	
Pu-239	4.32E+00		1.00E+02		3.22E+01	3.22E-01	
Pu-240	1.08E+00		1.00E+02		8.06E+00	8.06E-02	
Pu-241	1.12E+00		3.50E+03		8.36E+00	2.39E-03	
Am-241	6.42E-01		1.00E+02		4.79E+00	4.79E-02	
Am-243	0.00E+00		1.00E+02		0.00E+00	0.00E+00	
Cm-242	0.00E+00		2.00E+04		0.00E+00	0.00E+00	
Cm-243	0.00E+00		1.00E+02		0.00E+00	0.00E+00	
Cm-244	0.00E+00		1.00E+02		0.00E+00	0.00E+00	
					Sums of fractions[3]	**4.59E-01**	**1.60E-04**

NOTE: (1) From estimated activity based on analytical data from the composited sample from Tank 509E (from URS 2011) using a total solidified waste volume of 74.76 m^3 and total mass of 134,158,000 grams (from DK-ES 2010).

This table shows that the sum of fractions for both the 10 CFR 61.55 Table 1 (long-lived) and Table 2 (short-lived) radionuclides is below 1.0, and thus the solidified sludge waste form will not exceed concentration limits for Class C low-level waste. The calculations were performed for an average waste package. Due to the homogeneity of the waste in Tank 509E and the day tank considering their mixing capabilities, variations from these values for individual waste packages are expected to be minor.

However, an upper bound calculation for the Class C sum of fractions was also performed using radionuclide activity estimates consistent with high-activity range estimates in the waste profile (DK-ES 2010) corresponding to a total transuranic radionuclide concentration in the final waste form of 95 nCi/g. This calculation produced a total Class C sum of fractions of 9.55E-1, still less than 1.0. (URS 2011)

As discussed previously, this waste will be transported to the Nevada National Security Site Area 5 radioactive waste management site for disposal. At the Nevada National Security Site, the solidified sludge waste packages will be disposed of as low-level waste and managed in accordance with DOE requirements for low-level waste disposal in Chapter IV of DOE Manual 435.1-1. The required waste profile (DK-ES 2010) has been developed in accordance with the Nevada National Security Site waste acceptance criteria (DOE-NV 2011). This waste profile has been formally approved by the Nevada National Security Site, as noted previously (DOE-NNSS 2010).

The foregoing information shows that the solidified sludge waste packages will meet the third criterion of DOE Manual 435.1-1, Section II.B.2(a).

8.0 CONCLUSIONS

Based on information provided in the preceding sections of this evaluation, DOE re-affirms its prior conclusion that the SPRU solidified sludge is not high-level waste, confirms that it is appropriately classified as low-level waste and, for additional information and further assurance, concludes that this waste meets the evaluation criteria in Section II.B(2)(a) of DOE Manual 435.1-1 for waste incidental to processing if these criteria were to apply.

9.0 REFERENCES

Federal Statutes

Atomic Energy Act of 1954, as amended.

Resource Conservation and Recovery Act of 1976, as amended.

Comprehensive Environmental Response, Compensation, and Liability Act of 1980.

Energy Policy Act of 2005.

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

Code of Federal Regulations and Federal Register Notices

10 CFR 50.2, *Definitions.*

10 CFR 61, *Licensing Requirements for Land Disposal Of Radioactive Waste.*

10 CFR 72.3, *Definitions.*

10 CFR 835, *Occupational Radiation Protection.*

40 CFR 61, *National Emission Standards for Hazardous Air Pollutants.*

49 CFR 173.411, *Industrial Packaging*.

DOE Orders, Policies, and Standards

DOE Order 435.1, Change 1, *Radioactive Waste Management*.

DOE Order 458.1, *Radiation Protection of the Public and the Environment.*

DOE Order 5400.5, *Radiation Protection of the Public and the Environment.*

DOE Policy 441.1, *Department of Energy Radiological Health and Safety Policy.*

DOE Manual 435.1-1, *Radioactive Waste Management.*

DOE Guide 435.1-1, *Implementation Guide for Use with DOE M 435.1*.

Other References

ASME 2004, *The Decommissioning Handbook*. Toboas, A.L., et al., ed., ASME Press, New York, 2004.

Berriochoa 2011, "Everything Old is New Again," Berriochoa, M.V., article in *Radwaste Magazine*, American Nuclear Society, vol. 18, no. 1, January – April 2011.

Bragg 1949, *Proposal for an Experimental Unit for Operation at Full Hanford Activity*. Bragg, L.B, Hanford Works, Richland, Washington, October 7, 1949.

DK-ES 2010, *Waste Profile Sheet, Duratek/Energy Solutions SPRU Solidified Sludge, DRTK-000000025, Rev 0.* Duratek/Energy Solutions, Oak Ridge, Tennessee, November 26, 2010.

DOE 2008, *SPRU Tank Waste Classification*, staff draft white paper. U.S. Department of Energy, Office of Environmental Management, March 10, 2008.

DOE 2009, *Revised Action Memorandum for the Separations Process Research Unit (SPRU) Disposal Project*. U.S. Department of Energy, Office of Environmental Management, SPRU Field Office, Niskayuna, New York, August 2009.

DOE-ID 2006, *Basis for Section 3116 Determination for the Idaho Nuclear Technology and Engineering Center Tank Farm Facility*, DOE-NE-11226, Revision 0. U.S. Department of Energy, Idaho Falls, Idaho, November 2006.

DOE-NNSS 2010, *Approval to Ship Duratek/Energy Solutions DRTK Low-Level Radioactive Waste to the Nevada National Security Site (NNSS)*. DOE memorandum, U.S. Department of Energy, Nevada Site Office, Las Vegas, Nevada, November 4, 2010.

DOE-NV 2011, *Nevada Test Site Waste Acceptance Criteria*, DOE/NV-325-Rev. 8-01. U.S. Department of Energy, National Nuclear Security Administration, Nevada Site Office, Waste Management Project, Las Vegas, Nevada, January 2011.

ERG 2006a, *Nuclear Facility Historical Site Assessment for the Separations Process Research Unit (SPRU) Disposition Project.* Environmental Resource Group, LLC, April 2006.

ERG 2006b, *Land Area Historical Site Assessment for the Separations Process Research Unit (SPRU) Disposition Project.* Environmental Resource Group, LLC, December 2006.

ERG 2007, *Nuclear Facility Engineering Evaluation/Cost Analysis for the Separations Process Research Unit (SPRU) Disposition Project,* final. Environmental Resource Group, LLC, July 2007.

Fazio 2002, *High-Level Waste Tank 8D-1 and 8D-2 Radionuclide Inventory Report as of September 1, 2002*, WVDP-364-001, Revision 0. Fazio, J.M., West Valley Nuclear Services Company, West Valley, New York, September 30, 2002.

Hanford 1952, Memorandum regarding 24 irradiated slugs shipped from Hanford to SPRU. Hanford Works, Richland Washington, May 6, 1952.

Hanford 1956, *Fuel Element Technical Manual*. Hanford Works, Richland Washington, August 1956.

IAEA 2005, *Current Trends in Nuclear Fuel for Power Reactors.* International Atomic Energy Agency, Vienna, Austria, 2005. [This document is undated but the data extend through 2005]

KAPL 1992, *Preliminary Evaluation of the Status of the Separation Process Research Unit (SPRU),* Draft. Knolls Atomic Power Laboratory, Niskayuna, New York, October 1992.

KAPL 1998, *Separation Process Research Unit (SPRU) History and Current Status*. Knolls Atomic Power Laboratory, Niskayuna, New York, 1998.

Meess and Dundas 2005, "Sodium-Bearing Wastewater Processing at the West Valley Demonstration Project," Meess, D.C and J.M. Dundas. Waste Management 2005, Tucson, Arizona, February 27 – March 3, 2005.

NFS 2011, National Science Foundation website, http://www.nsf.gov/statistics/randdef/fedgov.cfm, visited in March 2011.

NRC 2007, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations*, Draft Final Report for Interim Use, NUREG-1854. U.S. Nuclear Regulatory Commission, Washington, D.C., August 2007.

NST 2009, *Nevada Test Site Radiological Control Manual*, DOE/NV/25946—801, Revision 0. National Security Technologies LLC, Las Vegas, Nevada, October 2009.

NST 2010, *2009 Annual Summary Report for the Area 3 and Area 5 Radioactive Waste Management Sites at the Nevada Test Site Nye County, Nevada: Review of the Performance Assessments and Composite Analyses*, DOE/NV/25946—6893. National Security Technologies, LLC, Las Vegas, Nevada, March 2010.

SRNL 2010, *EM 31 Retrieval Knowledge Center Meeting Report: Mobilize and Dislodge Tank Waste Heels*, SRNL-STI-2009-00535, Revision 0. Savannah River National Laboratory, Aiken, South Carolina, February 2010.

SRR 2010, *Performance Assessment for the F-Tank Farm at the Savannah River Site*, SRS-REG-2007-00002, Revision 1. Savannah River Remediation, LLC, Aiken, South Carolina, March 31, 2010.

Tajiti, et al. 2000, "Experience of Test Operation for Removal of Fission Product Nuclides in TRU-Liquid Waste and Concentrated Nitric Acid Using Inorganic Ion Exchangers," Tajiti, H., et al., Waste Management 2000, Tucson, Arizona, February 27 – March 2, 2000.

URS 2009, *SPRU DP Radiological Control Manual* SPRU-RC-001, Revision 4. URS Corporation, Niskayuna, New York, March 4, 2009.

URS 2010a, *Radiological Engineering Calculation, 578, 509A-E & 505 Tank Source Term*, REC-10-010. URS Corporation, Niskayuna, New York, June 16, 2010.

URS 2010b, *Radiological Engineering Calculation, Total DP Source Term*. REC-10-015, Revision 0. URS Corporation, Niskayuna, New York, July 7, 2010.

URS 2010c, *SPRU DP Decommissioning Plan*, SPRU-DD-004, Revision 2. URS Corporation, Niskayuna, New York, November 12, 2010.

URS 2011, E-mails from Terry Hissong, URS, to Hugh Davis, DOE-SPRU, providing information on tank heels and the preliminary SPRU Sludge Characterization Assessment Report, March 18 and 19, 2011.

Wilhite 2001, *Comparison of LLW Disposal Performance Objectives, 10 CFR 61 and DOE 435.1*, WSRC-RP-2001-00341. Wilhite, E.L., Westinghouse Savannah River Company, Aiken, South Carolina, 2001.

WVNSCO and Gemini 2005, *West Valley Demonstration Project, Residual Radionuclide Inventory Estimate for the Waste Tank Farm.* West Valley Nuclear Services Company and Gemini Consulting Company, West Valley, New York, February 7, 2005.

APPENDIX A
COMPARABILITY OF DOE AND NRC REQUIREMENTS FOR LOW-LEVEL WASTE DISPOSAL

Appendix Purpose

The purpose of this appendix is to show that Department of Energy and Nuclear Regulatory Commission requirements for disposal of low-level waste are comparable.

Appendix Content

This appendix identifies applicable Department of Energy performance objectives and the similar Nuclear Regulatory Commission performance objectives and discusses their comparability.

Key Points

- Requirements for low-level waste disposal are embodied in sets of performance objectives for the waste disposal facility.
- The Department of Energy performance objectives are described in DOE Manual 435.1-1, *Radioactive Waste Management*.
- The Nuclear Regulatory Commission performance objectives are described in Subpart C, *Performance Objectives*, of 10 CFR Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste*.
- Department of Energy and Nuclear Regulatory Commission performance objectives for low-level waste disposal are comparable.

1.0 Introduction

This appendix identifies performance objectives for disposal of LLW by the DOE and the NRC. It then compares these performance objectives.

Information in this appendix is based in part on previous detailed comparison studies of DOE and NRC performance objectives for LLW disposal (Cole, et al. 1995 and Wilhite 2001).

2.0 Applicable Performance Objectives

DOE Manual 435.1-1, *Radioactive Waste Management*, describes DOE requirements for disposal of LLW. The comparable NRC requirements appear in Subpart C of 10 CFR Part 61, which lists one general requirement and four performance objectives, which are reproduced below.

Section 61.40, General Requirement

"Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44."

Section 61.41, Protection of the General Population from Releases of Radioactivity

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

Section 61.42, Protection of Individuals from Inadvertent Intrusion

"Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

Section 61.43, Protection of Individuals During Operations

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

Section 61.44, Stability of the Disposal Site After Closure

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

3.0 Comparability of the General Requirements

3.1 DOE

The general requirement in DOE Manual 435.1-1, Section IV.P(1), is expressed as follows:

"Low-level waste disposal facilities shall be sited, designed, operated, maintained, and closed so that a reasonable expectation exists that the following performance objectives will be met for waste disposed of after September 26, 1988."

3.2 NRC

The NRC regulations in 10 CFR 61.40 provide in relevant part:

"Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44."

3.3 Discussion

The statement of NRC requirements in 10 CFR 61.40 is nearly identical to that of the DOE general requirement. The DOE requirement adds the concept of maintenance, which is implicit in the NRC requirement. The DOE requirement does not mention control after closure, but this concept is embodied in the DOE requirements for closure, specifically DOE Manual 435.1, Section IV.Q (2)(c), which requires DOE control until it can be shown that release of the disposal site for unrestricted use will not compromise DOE requirements for radiological protection of the public.

The DOE general requirement for LLW disposal and the NRC general requirement of 10 CFR 61.40 are therefore comparable.

4.0 Comparability Regarding Protection of the General Population from Releases of Radioactivity

4.1 DOE

DOE requirements of DOE Manual 435.1-1, Section IV.P(1), read as follows:

"(a) Dose to representative members of the public shall not exceed 25 millirem in a year total effective dose equivalent from all exposure pathways, excluding the dose from radon and its progeny in air.

(b) Dose to representative members of the public via the air pathway shall not exceed 10 millirem in a year total effective dose equivalent, excluding the dose from radon and its progeny.

(c) Release of radon shall be less than an average flux of 20 pCi/m^2/s at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/L of air may be applied at the boundary of the facility."

4.2 NRC

NRC regulations in 10 CFR 61.41 are expressed as follows:

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

4.3 Discussion

DOE uses more current radiation protection methodology, consistent with that used in NRC's radiation protection standards in NRC's 10 CFR 20, *Standards for Protection Against Radiation*. Because NRC has not revised 10 CFR 61.41 to reflect the more current methodology in 10 CFR 20, DOE's requirements and those in 10 CFR 20 differ slightly from those in 10 CFR 61.41. However, the resulting allowable doses are comparable, as NRC has acknowledged (NRC 2005). Both NRC and DOE use a performance assessment to assess whether the dose limit will be met.

The DOE requirements go beyond this NRC performance objective by specifying an assessment of the impacts of LLW disposal on water resources (i.e., DOE Manual 435.1, Section IV.P (2)(g)). The NRC requirement includes maintaining releases to the environment ALARA. Although this requirement is not included in the DOE performance objective, it is included in the performance assessment requirements (i.e., DOE Manual 435.1-1, Section IV.P (2)(f)).

5.0 Comparability Regarding Protection of Individuals from Inadvertent Intrusion

5.1 DOE

DOE requirements of DOE Manual 435.1-1, Section IV.P(2)(h), for protection of individuals from inadvertent intrusion read as follows:

> "For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 millirem (1 mSv) in a year and 500 millirem (5 mSv) total effective dose equivalent excluding radon in air."

5.2 NRC

NRC requirements of 10 CFR 61.42 are expressed as follows:

> "Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

5.3 Discussion

The DOE LLW disposal requirement that the performance assessment include an assessment of the impacts on a person inadvertently intruding into the disposal facility is more stringent than the NRC requirement. The NRC waste classification system is based on intruder calculations using a 500 millirem per year dose limit (NRC 1982). The DOE requirement uses a 100 millirem per year limit for chronic exposures and a 500 millirem limit for acute exposures.

6.0 Comparability Regarding Protection of Individuals During Operations

6.1 DOE

The DOE requirements in DOE Manual 435.1-1, Section I.E(13), for protection of individual during operations read as follows:

> "Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR Part 835, *Occupational Radiation Protection,* and DOE 5400.5, *Radiation Protection of the Public and the Environment*."

6.2 NRC

The NRC requirements of 10 CFR 61.43 are expressed as follows:

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

6.3 Discussion

The ALARA concept is an integral part of DOE radiation and environmental protection programs, as expressed in DOE Policy 441.1, *Department of Energy Radiological Health and Safety Policy.* DOE requirements for occupational radiological protection are addressed in 10 CFR 835, and similar requirements for radiological protection of the public and the environment are addressed in DOE Order 5400.5. The NRC 10 CFR 61.43 requirement references 10 CFR 20, *Standards for Protection Against Radiation*, which contains similar radiological protection standards for workers and the public.

Appendix B provides additional information on the comparability of DOE and NRC radiation dose standards that apply to protection of individuals during operations.

7.0 Comparability Regarding Stability of the Disposal Site After Closure

7.1 DOE

The DOE requirements of DOE Manual 435.1-1, Sections IV.Q(1)(a) and (b) and IV.Q(2)(c), for stability of the disposal site after closure are expressed as follows:

> "Disposal Site Stability (DOE Manual 435.1, Section IV.Q(1)(a) and (b)). A preliminary closure plan shall be developed and submitted to Headquarters for review with the performance assessment and composite analysis. The closure plan shall be updated following issuance of the disposal authorization statement to incorporate conditions specified in the disposal authorization statement. Closure plans shall:
>
> (a) Be updated as required during the operational life of the facility.
>
> (b) Include a description of how the disposal facility will be closed to achieve long-term stability and minimize the need for active maintenance following closure and to ensure compliance with the requirements of DOE 5400.5, *Radiation Protection of the Public and the Environment*."
>
> "Disposal Facility Closure (DOE Manual 435.1, Section IV.Q(2)(c)). Institutional control measures shall be integrated into land use and stewardship plans and programs, and shall continue until the facility can be released pursuant to DOE Order 5400.5, *Radiation Protection of the Public and the Environment*."

7.2 NRC

The NRC requirements of 10 CFR 61.44 state that:

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

7.3 Discussion

The DOE LLW disposal requirements address long-term stability of the site by requiring a description of how closure will achieve stability in the closure plan, and by a description of how closure will minimize the need for active maintenance following closure (DOE Manual 435.1, Section IV.Q (1)(b)). Additionally, one of the performance assessment requirements (DOE Manual 435.1, Section IV.P (2)(c)) states: "Performance assessments shall address reasonably foreseeable natural processes that might disrupt barriers against release and transport of radioactive materials."

Thus, the performance assessment will include a projection of the long-term stability of the site, considering reasonably foreseeable natural processes such as erosion, degradation of waste packages, etc.

8.0 References

Code of Federal Regulations

10 CFR 20, *Standards for Protection Against Radiation*.

10 CFR 61, Subpart C, *Licensing Requirements for Land Disposal of Radioactive Waste, Performance Objectives*.

10 CFR 835*, Occupational Radiation Protection*.

DOE Orders, Policies, and Manuals

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., January 7, 1993.

DOE Policy 441.1*, Department of Energy Radiological Health and Safety Policy*. U.S. Department of Energy, Washington, D.C., April 26, 1996.

DOE Manual 435.1-1, *Radioactive Waste Management Manual*, Change 1. U. S. Department of Energy, Washington, D.C., June 19, 2001.

Other References

Cole, et al. 1995, *Comparison of Selected DOE and Non-DOE Requirements, Standards, and Practices for Low-Level Radioactive Waste Disposal*, DOE/LLW-225, Revision 0. Cole, L., D. Kudera, and W. Newberry, Idaho National Engineering Laboratory, Lockheed Idaho Technologies Company, Idaho Fall, Idaho, December 1995.

NRC 1982, *Final Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste," Summary and Main Report*, NUREG-0945, Volume 1. U.S. Nuclear Regulatory Commission, Washington, D.C., November 1982.

NRC 2005, *U.S. Nuclear Regulatory Commission Technical Evaluation Report for the U.S. Department of Energy Savannah River Site Draft Section 3116 Waste Determination for Salt Waste Disposal*. U.S. Nuclear Regulatory Commission, Washington, D.C., December 2005.

Wilhite 2001, *Comparison of LLW Disposal Performance Objectives, 10 CFR 61 and DOE 435.1*, WSRC-RP-2001-00341. Wilhite, E.L., Westinghouse Savannah River Company, Aiken, South Carolina, 2001.

APPENDIX B
COMPARABILITY OF DOE AND NRC DOSE STANDARDS

Appendix Purpose

The purpose of this appendix is to compare Department of Energy and Nuclear Regulatory Commission radiation dose standards that apply to individual workers and to members of the public.

Appendix Content

This appendix identifies applicable Department of Energy dose standards and the similar Nuclear Regulatory Commission dose standards and discusses their comparability.

Key Points

- The Department of Energy radiation dose standards appear in 10 CFR 835, *Occupational Radiation Protection*, and in DOE Orders.
- The Nuclear Regulatory Commission radiation dose standards appear in 10 CFR 20, *Standards for Protection Against Radiation.*
- Department of Energy and Nuclear Regulatory Commission radiation dose standards are comparable.

1.0 Introduction

The purpose of this appendix is to compare the DOE and NRC dose standards that apply to protection of the public and the workers from radiation during operations associated with preparing the solidified sludge at the SPRU facility and handling of the solidified sludge when it is received at the Nevada National Security for disposal.

Section 6.2.4 of the body of this evaluation briefly addressed protection of individuals during these operations at SPRU and the Nevada National Security Site. Appendix A also addressed this matter. This appendix provides a more detailed treatment of the dose standards used.

Requirements in NRC's regulations at 10 CFR 61.43 state:

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

This requirement references 10 CFR 20, *Standards for Protection Against Radiation*, which contains radiological protection standards for workers and the public. The DOE requirements for occupational radiological protection are provided in 10 CFR 835, *Occupational Radiation Protection*, and those for radiological protection of the public and the environment are provided in DOE Order 5400.5, *Radiation Protection of the Public and the Environment*.

The NRC standards for radiation protection in 10 CFR 20 that are considered in detail in this evaluation are the dose limits for the public and the workers during disposal operations set forth in 10 CFR 20.1101(d), 20.1201(a)(1)(i), 20.120 1(a)(1)(ii), 20.120 1(a)(2)(i), 20.120 1(a)(2)(ii), 20.1201(e), 20.1208(a), 20. 1301(a)(1), 20.1301(a)(2), and 20.1301(b).[30] These NRC dose limits correspond to the DOE dose limits in 10 CFR 835 and relevant DOE orders that establish DOE regulatory and contractual requirements for DOE facilities and activities. As discussed in Section 5.2.4 of this evaluation, operations related to disposal of the solidified sludge will meet these dose limits and doses will be maintained ALARA.

2.0 Air Emissions Limit for Individual Member of the Public

2.1 DOE

DOE limits doses from air emissions to the public to 10 mrem/yr in DOE Order 5400.5 (1993). The DOE is also subject to and complies with the U.S. Environmental Protection Agency requirement in 40 CFR 61.92, which has the same limit.[31]

2.2 NRC

The NRC regulation in 10 CFR 20.1101(d) provides in relevant part:

> "[A] constraint on air emissions of radioactive material to the environment, excluding radon-222 and its daughters, shall be established ... such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv)/yr from these emissions."

2.3 Discussion

The DOE and NRC requirements are comparable.

3.0 Total Effective Dose Equivalent Limit for Adult Workers

3.1 DOE

DOE's regulation in 10 CFR 835.202(a)(1) requires that the occupational dose per year for general employees shall not exceed a total effective dose equivalent of 5 rem[32].

[30] The "standards for radiation protection" in 10 CFR 20 (as cross-referenced in the performance objective in 10 CFR 61.43), which are relevant to this evaluation, are the dose limits for radiation protection of the public and the workers during disposal operations, and not those which address general licensing, administrative, programmatic, or enforcement matters administered by NRC for NRC licensees. Accordingly, this evaluation addresses in detail the radiation dose limits for the public and the workers during disposal operations that are contained in the provisions of 10 CFR 20 referenced above. Although 10 CFR 20.1206(e) contains limits for planned special exposures for adult workers, there will not be any such planned special exposures for work related to the sludge solidification. Therefore, this limit is not discussed further in this evaluation. Likewise, 10 CFR 20.1207 specifies occupational dose limits for minors. However, there will not be minors working at SPRU or the Nevada National Security Site who would receive an occupational dose. Therefore, this limit is not discussed further in this evaluation.

[31] 40 CFR 61.92 provides as follows: "Emissions of radionuclides to the ambient air from DOE facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr. It is assumed that the individual is an adult living at the site perimeter that is exposed to the maximum yearly radioactive atmospheric release and maximum radiation concentration in food for 365 days per year. For the airborne pathway, the dose is developed by the input of atmospheric release data, vegetation consumption data, milk consumption data, and beef consumption data."

3.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

"(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures...to the following dose limits.

(1) An annual limit, which is the more limiting of –

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv)."

3.3 Discussion

The DOE and NRC requirements are comparable.

4.0 Any Individual Organ or Tissue Dose Limit for Adult Workers

4.1 DOE

The DOE regulation in 10 CFR 835.202(a)(2) provides in relevant part:

". . . the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:

(2) The sum of the deep dose equivalent for external exposures and the committed dose equivalent to any organ or tissue other than the lens of the eye of 50 rems (0.5 sievert)"

4.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

"(a) [C]ontrol the occupational dose to individual adults, except for planned special exposures...to the following dose limits.

(1) An annual limit, which is the more limiting of –...

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv)."

4.3 Discussion

The DOE and NRC requirements are comparable.

[32] The DOE's regulations at 10 CFR 835.202(a)(1) and (a)(2) require that the occupational dose per year for general employees shall not exceed both a total effective dose equivalent of 5 rem and the sum of the deep-dose equivalent for external exposures and the committed dose equivalent to any other organ or tissue other than the lens of the eye of 50 rem. The NRC's regulation specifies that either of these two limits shall be met by NRC licensees, whichever is more limiting. Thus, DOE imposes stricter, separate requirements. The provisions of DOE's requirements at 10 CFR 835.202(a)(1) and (a)(2), which correlate to NRC requirements at 10 CFR 20.1201(a)(1) and (a)(2), are discussed in separate subsections in this evaluation.

5.0 Annual Dose Limit to the Lens of the Eye for Adult Workers

5.1 DOE

The DOE regulation in 10 CFR 835.202(a)(3) provides in relevant part:

> ". . . the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:
>
> (3) A lens of the eye dose equivalent of 15 rems (0.15 sievert)"

5.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposure the following dose limits. ...
>
> (2) The annual limits to the lens of the eye, to the skin of the whole body or to the skin of the extremities, which are:
>
> (i) A lens dose equivalent of 15 rems (0.15 Sv).

5.3 Discussion

The DOE and NRC requirements are comparable.

6.0 Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers

6.1 DOE

The DOE regulation in 10 CFR 835.202(a)(4) provides in relevant part:

> ". . . the occupational dose received by general employees shall be controlled such that the following limits are not exceeded in a year:
>
> (4) A shallow dose equivalent of 50 rems (0.5 sievert) to the skin or any extremity."

6.2 NRC

The NRC regulation in 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures...to the following dose limits. ...
>
> (2) The annual limits to the lens of the eye, the skin of the whole body, or to the skin of the extremities, which are: ...
>
> (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

6.3 Discussion

The DOE and NRC requirements are comparable.

7.0 Limit on Soluble Uranium Intake

7.1 DOE

Requirements in DOE Order 440.1A for soluble uranium intake are the more restrictive of the concentrations in the American Conference of Governmental Industrial Hygienists threshold limit values (0.2 mg/m^3, which is the same as noted in 10 CFR 20, Appendix B) or the Occupational Safety and Health Administration permissible exposure limit (0.05 mg/m^3). The permissible exposure limit for soluble uranium, which equates to a soluble uranium intake of 2.4 mg/week, is the more restrictive of the two.

7.2 NRC

The NRC regulation in 10 CFR 20.1201(e), concerning occupational dose limits for adults, provides in relevant part: "in addition to the annual dose limits, ... limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity."

7.3 Discussion

The DOE requirements are more restrictive.

8.0 Dose Equivalent to an Embryo/Fetus

8.1 DOE

The DOE regulation in 10 CFR 835.206(a) provides in relevant part:

> "The dose equivalent limit for the embryo/fetus from the period of conception to birth, as a result of occupational exposure of a declared pregnant worker, is 0.5 rem (0.005 sievert)."

After declaration of pregnancy, DOE provides the option of a mutually agreeable assignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure during the remainder of the gestation period is unlikely. In addition, personnel dosimetry[33] is provided and used to track exposure carefully.

8.2 NRC

The NRC regulation in 10 CFR 20.1208(a), concerning the dose equivalent to an embryo/fetus, provides in relevant part:

> "ensure that the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv)."

8.3 Discussion

The DOE and NRC requirements are comparable.

9.0 Dose Limits for Individual Members of the Public (Total Annual Dose)

9.1 DOE

Provisions in DOE Order 5400.5 limit public doses to 0.1 rem per year.

---

[33] The term *dosimetry* or *personnel dosimetry* refers to a device carried or worn by an individual working near radiation for measuring the amount of radiation to which he or she is exposed.

9.2 NRC

The NRC regulation in 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

(a) "[C]onduct operations so that –

> (1) The total effective dose equivalent to individual members of the public ...does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released..., from voluntary participation in medical research programs, and from the ...disposal of radioactive material into sanitary sewerage."

9.3 Discussion

The DOE and NRC requirements are comparable.

10.0 Dose Limits for Individual Members of the Public (Dose Rate in Unrestricted Areas)

10.1 DOE

DOE's regulation in 10 CFR 835.602 establishes the expectation that the total effective dose equivalent in controlled areas will be less than 0.1 rem per year. In accordance with 10 CFR 835.602, radioactive material areas have been established for accumulations of radioactive material within controlled areas that could result in a radiation dose of 100 millirem per year or greater. Averaged over a work year, this yields a constant average dose rate of 0.00005 rem per hour. In addition, training and dosimetry are required for individual members of the public for entry into controlled areas, as well as signs at each access point to a controlled area.

10.2 NRC

The NRC regulation in 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

(a) "[C]onduct operations so that –

> (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and
>
> (2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with § 35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.
>
> (b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals."

10.3 Discussion

The DOE and NRC requirements are comparable.

11.0 Dose Limits for Individual Members of the Public With Access to Controlled Areas[34]

11.1 DOE

The DOE regulation in 10 CFR 835.208 provides:

> "The total effective dose equivalent limit for members of the public exposed to radiation and/or radioactive material during access to a controlled area is 0.1 rem (0.001 sievert) in a year."

DOE requires training for individual members of the public before entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem in a year

11.2 NRC

The NRC regulation in 10 CFR 20.1301(b), concerning dose limits for individual members of the public, provides in relevant part:

> "if ... members of the public [are permitted] to have access to controlled areas, the limits for

11.3 Discussion

The DOE and NRC requirements in this area are comparable.

12.0 As Low As Reasonably Achievable

12.1 DOE

The DOE regulation in 10 CFR 835.2 defines ALARA as "the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations." The DOE regulation in 10 CFR 835.2 also specifies: "ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits as is reasonably achievable."

12.2 NRC

The NRC regulation in 10 CFR 20.1003 defines ALARA in relevant part: "ALARA . . . means making every reasonable effort to maintain exposures to radiation as far below the dose limits . . . as is practical consistent with the purpose for which the . . . activity is undertaken."

12.3 Discussion

The DOE and NRC definitions of ALARA are comparable.

---

[34] DOE defines a controlled area in 10 CFR 835.2 as "any area to which access is managed by or for DOE to protect individuals from exposure to radiation and/or radioactive material." NRC in 10 CFR 20.1003 defines restricted areas as "an area, access to which is limited ... for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials." The two definitions are essentially the same.

12.0 References

Code of Federal Regulations

10 CFR 20, *Standards for Protection Against Radiation*.

10 CFR 835*, Occupational Radiation Protection*.

40 CFR 61, *National Emission Standards for Hazardous Air Pollutants*.

DOE Orders and Policies

DOE Order 440.1A, *Worker Protection Management for DOE Federal and Contractor Employees*. U.S. Department of Energy, Washington, D.C., March 27, 1998.

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, Change 2. U.S. Department of Energy, Washington, D.C., January 7, 1993.

DOE Policy 441.1*, Department of Energy Radiological Health and Safety Policy*. U.S. Department of Energy, Washington, D.C., April 26, 1996.