2.0 FACILITY OPERATING HISTORY

PURPOSE OF THIS SECTION

The purpose of this section is to describe the facility operating history, thereby providing a foundation for understanding the rest of the plan. Section 2 is also intended to provide information to allow NRC staff to understand (1) the license history, (2) previous decommissioning activities, (3) radioactive spills that have occurred, and (4) onsite burials of radioactive materials.

INFORMATION IN THIS SECTION

This section provides the following information:

- A summary of the license history, including the radionuclides present and how they have been used, addressing both Nuclear Fuel Services (NFS) operations under the license through 1982 and WVDP activities since that time that were not performed under the license;
- A summary of the previous decommissioning and remediation activities and the remediation activities to take place during the period leading up to the interim end state, which will be the point at which Phase 1 proposed decommissioning activities begin;
- A summary of spills of radioactivity that have had the potential to have impacted the environment, both under NFS and during the WVDP; and
- Information on prior onsite burials of radioactive material, except for those in the State-Licensed Disposal Area (SDA) and Waste Management Area 11 (outside the project premises), which are beyond the scope of this plan.

RELATIONSHIP TO OTHER PLAN SECTIONS

To put into perspective the information in this section, one must consider the information in Section 1 on the project background and those facilities and areas within the scope of the DP. Consideration of the information in Section 3 on the facility description and the information in Section 4 on the radiological status of the facility would also help place information in Section 2 into context.

The information in this section serves as the foundation for later sections, such as facility description in Section 3, the radiological status in Section 4, and the decommissioning activities in Section 7.

2.1 License History

Provisional Operating License Number CSF-1 (AEC 1966) was issued on April 19, 1966 by the U.S. Atomic Energy Commission to NFS and the New York State Atomic and Space Development Authority under Section 104b of the Atomic Energy Act of 1954, as amended, to operate a spent fuel reprocessing and radioactive waste disposal facility at the Center. The Atomic Energy Commission was the regulator of this license until 1975 when the NRC was established by passage of the Energy Reorganization Act of 1974.

License CSF-1 provided limits for (1) nuclear fuel (source, special nuclear material and byproduct materials in irradiated or unirradiated solid fuel elements and solutions); (2) unirradiated source material; and (3) material for storage and use for standards, test, measurements, and calibration. The radionuclides and possession limits for these categories are identified in Tables 2-1, 2-2, and 2-3. (See note at the end of Tables 2-1 and 2-2.)

Table 2-1. Limits for Nuclear Fuel in Solid Fuel Elements and Solutions(1)

Category	Pre-irradiation Fuel Compound	Pre-irradiation % U-235 Enrichment in U
1	UO_2	5%
2	UO_2	>5%, but ≤10%
3	ThO_2 + UO_2 Not exceeding 8.5% U	No limitation
4	U-Mo alloy	26.5%
5	U-Zircaloy alloy U-Zr alloy (U content, 10 w/o [wt.%] of alloy)	No limitation
6	U metal or UO_2	5%
7	U-Al alloy	No limitation
8	U-Mo alloy	4.5%
9	U metal	2.5%
10	Plutonium nitrate - In depleted uranyl nitrate solution	250 grams fissile plutonium (Pu-239 and Pu-241) per liter.
The possession limits of the above special nuclear material were 21,000 kg of U-235, 3,200 kg of U-233, and 4,000 kg of plutonium.		

NOTE: (1) The chemical forms of the radionuclides authorized for use changed from solid fuel (elemental metal) to aqueous solutions during reprocessing, with radionuclides used for calibration standards, testing, etc. used primarily in laboratories.

Table 2-2. Limits for Unirradiated Source Material(1)

Material	Possession Limit	Form
Uranium of natural isotopic composition	50,000 pounds	Hanford N-Reactor Fuel
Uranium depleted in the isotope U-235	100,000 pounds	UO_2, metal prototype fuel elements and U_3O_8 granules of depleted uranium
Thorium	50,000 pounds	Thorium nitrate or thorium oxide

NOTE: (1) The chemical forms of the radionuclides authorized for use changed from solid fuel (elemental metal) to aqueous solutions during reprocessing, with radionuclides used for calibration standards, testing, etc. used primarily in laboratories.

Table 2-3. Limits Used for Standards, Test, Measurements, and Calibration(1)

Uranium-235	105 grams	Any
Uranium-233	75 grams	Any
Plutonium(2)	62 grams	Any
Plutonium(2)	14 grams	sealed source
Plutonium-242	6 grams	Any
Plutonium-238	1 gram	Any
Neptunium-237	3.5E-03 curie	Any
Americium-241	1.0E-03 curie	Any
Thallium-204	5.0E-06 curie	Any
Cesium-137	5.0E-03 curie	Any
Cesium-137	33 curies	sealed source
Cesium-134	5.0E-03 curie	Any
Cerium-144	1.0E-02 curie	Any
Iodine-131	6.0E-06 curie	Any
Iodine-129	5.0E-06 curie	Any
Ruthenium-106	1.0E-02 curie	Any
Zirconium-95	5.0E-02 curie	Any
Strontium-90	1.0E-02 curie	Any
Strontium-85	1.0E-02 curie	Any
Krypton-85	3 curies	Any

Table 2-3. Limits Used for Standards, Test, Measurements, and Calibration[(1)]

Material	Possession Limit	Form
Zinc-65	1.0E-02 curie	Any
Cobalt-60	5.0E-02 curie	Any
Cobalt-58	1.0E-02 curie	Any
Manganese-54	5.0E-03 curie	Any
Antimony[(2)]	5.0E-03 curie	Any
Any byproduct material with atomic numbers from 3 to 85	3.0E-06 curie each	Any

NOTES: (1) From Section 3.3 of Appendix A of Provisional License CSF-1, Change 18 (AEC 1966)
(2) Section 3.3 of Appendix A of Provisional License CSF-1, Change 18 (AEC 1966) omitted the mass number of this radionuclide.

From 1966 to 1972, NFS reprocessed under the license more than 600 metric tons of spent fuel in the Process Building (Table 2-4) and generated approximately 600,000 gallons of liquid high-level waste. The facility shut down in 1972 for modifications to increase reliability and to expand capacity. In 1976, without restarting the operation, NFS withdrew from the reprocessing business and returned control of the facilities to the site owner, NYSERDA, the successor to the New York State Atomic and Space Development Authority.

License CSF-1 has been amended by 32 License Amendments. Amendments 1 through 30 allowed operation of the facility with changes to the technical specifications. The changes to the technical specifications were based on changes to facility operations and physical plant modifications. No license amendments were made from 1976 to the start of the WVDP Act implementation in 1981.

License Amendment No. 31 (NRC 1981) transferred the project premises to DOE in accordance with the WVDP Act. The WVDP Act authorized the DOE, in cooperation with NYSERDA, the owner of the site and the holder of NRC license CSF-1, to carry out a high-level radioactive waste management demonstration project for the purpose of demonstrating solidification techniques that could be used for preparing high-level liquid radioactive waste for disposal (DOE and NYSERDA 1981).

On February 11, 1982, the NRC issued License Amendment 32, as requested by NFS, to terminate the authority and responsibility of NFS under the license effective upon DOE assumption of exclusive possession of the project premises. Control of the project premises was formally transferred to DOE effective February 26, 1982 (WVNSCO 1983a). Section 2.1.1 describes NFS activities under the license in more detail. As noted in Section 1, portions of NYSERDA's NRC Part 50 license for the Center, including the technical specifications, have been effectively suspended by NRC since 1981 to facilitate execution of the provisions of the WVDP Act.

2.1.1 Nuclear Fuel Services Operations From 1966 to 1982

Fuel receipt began in 1965, and reprocessing began in April 1966 and ended in 1972.

Receiving Fuel for Reprocessing

Table 2-4 shows the sources of spent nuclear fuel reprocessed at the facility. Additional shipments comprised of 750 spent nuclear fuel assemblies were received between February 1973 and December 1975 in anticipation of facility restart, which never occurred. Of these 750 assemblies, 625 were promptly returned to their original owners and the remaining 125 assemblies remained in storage in the Fuel Receiving and Storage Facility. The final shipment to remove the fuel assemblies from the WVDP was made in 2001.

The spent fuel assemblies were received in casks by rail or truck and placed into the Fuel Receiving and Storage area. The casks were unloaded in the Cask Unloading Pool and the fuel placed in storage canisters, which were then placed in the Fuel Storage Pool awaiting reprocessing. Reprocessing started with moving the canisters by underwater conveyer to the Process Mechanical Cell in the Process Building.

Process Building Arrangements

The Process Building contained the physical and chemical reprocessing operations, which were conducted in specially designed cells, rooms, and aisles. Descriptions of these areas are contained in Section 3. The cells were shielded rooms with concrete walls up to five feet thick where remote spent fuel reprocessing occurred. The rooms in which activities such as chemical preparation and laboratory analysis occurred that did not involve high levels of radioactivity were typically not shielded. The aisles were located adjacent to the shielded cells and provided for remote control of the physical and chemical reprocessing in the cells.

Sectioning and Dissolving the Fuel

The first step in reprocessing operations involved bringing fuel assemblies to the Process Mechanical Cell, where they were remotely disassembled with saws. The fuel rods were chopped into pieces with a shear prior to dissolution. The small pieces of fuel were then loaded into baskets, temporarily stored in the General Purpose Cell, and then transported to one of two dissolvers located in the Chemical Process Cell. The dissolution process consisted of placing the fuel pieces in a dissolver with concentrated nitric acid, which dissolved the irradiated fuel into an aqueous stream containing uranium nitrate, plutonium nitrate, and fission products. Unirradiated fuel went through a similar but abbreviated process.

Table 2-4. Nuclear Fuel Received and Reprocessed[1]

Lot	Source	Reactor	Process Date	Received MTU[2]	Recovered Pu (kg)
2	AEC	N-Reactor	4-22-66	19.7	1.7
1	AEC	N-Reactor	5-20-66	28.8	2.3
3	AEC	N-Reactor	7-15-66	46.7	50.9
4	Commonwealth Edison	Dresden-1	11-12-66	50.0	191.1
5	Yankee Atomic Electric	Yankee Rowe	6-7-67	49.8	285.1
6	AEC	N-Reactor	9-2-67	26.6	52.6
7	AEC	N-Reactor	12-2-67	26.1	47.4
8	AEC	N-Reactor	1-6-68	42.4	75.4
9	AEC	N-Reactor	5-5-68	38.8	79.1
10	AEC	N-Reactor	6-29-68	55.3	115.7
11[3]	Consolidated Edison	Indian Point-1	11-15-68	1.1	-
12	AEC	N-Reactor	2-13-69	48.9	102.5
13	Yankee Atomic Electric	Yankee Rowe	5-14-69	19.6	176.0
14[4]	AEC	N-Reactor	8-16-69	30.3	-
15	Commonwealth Edison	Dresden-1	10-1-69	21.5	104.6
16	Consolidated Edison	Indian Point-1	11-23-69	15.6	107.6
17	Yankee Atomic Electric	Yankee Rowe	6-2-70	9.3	95.6
18	Northern States Power	Pathfinder	8-14-70	9.6	7.1
19	Consumers Power	Big Rock Point	11-26-70	16.4	72.8
20	Consolidated Edison	Indian Point-1	1-11-71	7.6	68.1
21	AEC	N-Reactor	2-25-71	15.8	25.4
22	Puerto Rico Water Resources Authority	Bonus Superheater Bonus Boiler	4-15-71 4-18-71	1.7 2.4	0.9 4.0
23	Pacific Gas and Electric	Humboldt Bay	5-20-71	20.8	87.2
24	Yankee Atomic Electric	Yankee Rowe	7-16-71	9.5	95.7
25	Carolinas-Virginia Nuclear Power Associates	Carolinas-Virginia Tube Reactor	10-4-71	3.5	11.6
26	Consumers Power	Big Rock Point	11-30-71	5.8	27.9
27	NFS, Erwin, Tennessee[5]	SEFOR	12-12-71	0.1	95.5
			Total	625.7	1983.7

NOTES: (1) From DOE 1996.
(2) Metric tons uranium
(3) The lot 11 fuels from Indian Point-1 consisted of highly enriched uranium and thorium but no plutonium.
(4) The lot 14 fuel was unirradiated and therefore contained no plutonium.
(5) This material was a liquid residue generated during fabrication of fuel for the Southwest Experimental Fast Oxide Reactor (SEFOR).

Separating Uranium, Plutonium, and Fission Products

A five-stage solvent extraction process used a tributyl phosphate/n-dodecane solution to separate the fission products from the uranium and plutonium, and then separate the uranium from the plutonium. Following initial separation, the uranium-bearing solution underwent two further solvent extraction purification cycles while the plutonium bearing solutions underwent one additional purification cycle.

After leaving the extraction columns, the uranium-bearing solutions underwent an additional purification step that consisted of silica gel bed sorption. An ion-exchange process further purified the plutonium bearing solutions. The product solutions were concentrated, packaged, stored, and shipped off site. The NFS West Valley product was a nitrate solution (uranyl nitrate or plutonium nitrate) that was shipped to another out-of-state facility for purification and conversion to oxide. A representation of the fuel reprocessing operation is shown in Figure 2-1. The process used was the PUREX[1] process.

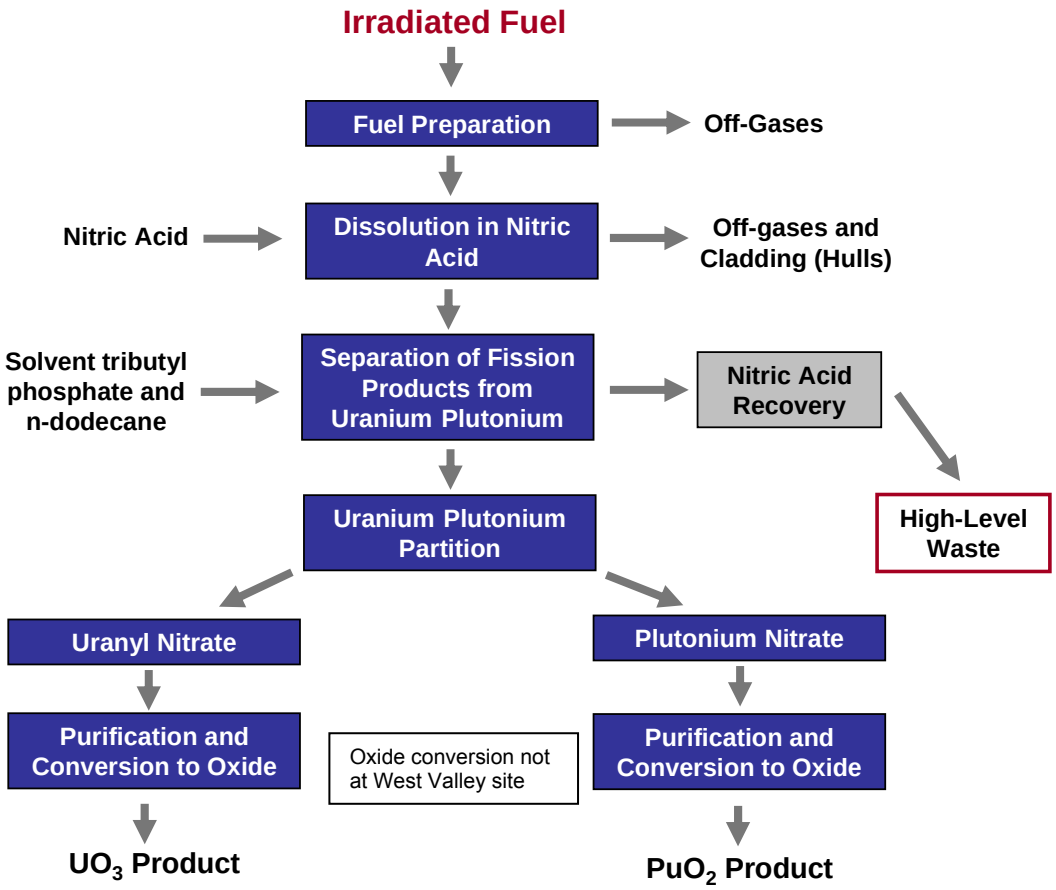

Figure 2-1. Spent Fuel Reprocessing Diagram (PUREX Process)

[1] Plutonium uranium refining by extraction.

Two systems, the HLW Evaporator and the LLW Evaporator were used to reduce the volume of aqueous waste generated during fuel reprocessing operations. The HLW Evaporator reduced the volume of aqueous waste generated during the partition cycle of the solvent extraction process. Both evaporators were used to reduce the volume of aqueous waste generated in the other four solvent extraction cycles.

Use of HLW Tanks 8D-1, 8D-2, 8D-3, and 8D-4

Approximately 580,000 gallons of liquid HLW was produced from the normal operation of the plant in reprocessing uranium fuel using the PUREX process (Duckworth 1972a). This waste was neutralized by the addition of sodium hydroxide before transfer to Tank 8D-2, a 750,000-gallon HLW storage tank. (Tank 8D-1, a spare 750,000-gallon tank identical to 8D-2 was designed for storing excess liquid from Tank 8D-2, but was never used by NFS to store HLW.)

Neutralizing the acidic high-level waste prior to transfer caused most of the fission product elements (the major exception was cesium) to precipitate out and form sludge at the bottom of Tank 8D-2. Therefore, the waste was not homogeneous but was comprised of supernatant liquid and solids (sludge).

Approximately 12,000 gallons of acidic high-level radioactive liquid waste were produced in reprocessing thorium-enriched uranium fuel using the THOREX[2] process. This waste was not neutralized because the thorium would have precipitated out of solution. This acidic waste was stored in Tank 8D-4, a 15,000-gallon capacity stainless steel tank. (Spare Tank 8D-3 is identical to Tank 8D-4 but was never used by NFS to store HLW.)

The radionuclide content of the HLW stored in Tanks 8D-2 and 8D-4 at the completion of reprocessing is given in Table 2-5. The chemical compositions of the supernatant and sludge in Tank 8D-2 at the completion of reprocessing are provided in Tables 2-6 and 2-7, respectively. The chemical composition of Tank 8D-4 at the completion of reprocessing is provided in Table 2-8. The radioactivity content is indexed to the start of HLW processing activities in 1988.

The spent tributyl phosphate/n-dodecane solvent solution used in each of the five solvent extraction cycles was cleaned in the extraction cells after each use. Following solvent wash, the clean solvent was transferred to the solvent storage tank. The spent wash solutions were then sent to tanks in the Liquid Waste Cell.

The Solvent Waste Catch Tank received the spent sodium carbonate and dilute nitric acid wash solutions that were used in the solvent cleanup system. The sodium carbonate and nitric acid washes used in the solvent cleanup were also collected in the Waste Catch Tank and then transferred to the Solvent Waste Hold Tank where they were sampled and subsequently sent through normal plant waste processing (Tank 8D-2 or LLW treatment) depending on their radioactivity concentration.

[2] Thorium reduction extraction.

Other liquid waste from Process Building operations (i.e., acid fractionator condensate, floor drains in various cells, chemical makeup areas, analytical laboratory, wash solutions from decontamination operations, etc.) were either treated in the Low-Level Waste Treatment Facility or routed to the underground waste tanks depending on their radioactivity level.

Use of the Low-Level Waste Treatment Facility

During initial NFS operations prior to construction of the Low-Level Waste Treatment Facility in 1971, low-level wastewater was routed through the Neutralization Pit, the Interceptor, and Lagoons 1, 2, and 3 in series before being discharged to Erdman Brook.

Following construction of the Low-Level Waste Treatment Facility and Lagoons 4 and 5, wastewater containing low levels of radionuclides (<0.005 µCi/mL) was treated in that facility by clarification, filtration, and ion exchange. This wastewater was collected from the Process Building, the Laundry, and the Fuel Receiving and Storage Facility and transported by underground drain lines sequentially to the Neutralization Pit, interceptors, and Lagoon 1, Lagoon 2, and to the Low-Level Waste Treatment Facility for treatment. Treated wastewater was piped to Lagoons 4 or 5, then to Lagoon 3 before batch discharge to Erdman Brook. (NFS 1973). See Figure 2-3 for the location of the Low-Level Waste Treatment Facility.

Radionuclides removed from the water were confined in a sludge that was packaged in drums and disposed of as radioactive waste. Much of the sludge was buried in the NRC-Licensed Disposal Area (NDA), mostly after closure of the SDA in 1975. While NFS used the State-Licensed Disposal Area (SDA) for LLW disposal, the WVDP did not use the SDA for radioactive waste disposal (DOE 1978, Wild 2000).

Table 2-5. Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Reprocessing(1)

Radionuclide	Half-Life (Year)(2)	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
H-3	1.23E+01	9.5E+1	~0	<2.0E+00	<9.7E+01
C-14	5.73E+03	1.4E+02	~0	(3)	1.4E+02
Fe-55	2.7E+00	(3)	1.0E+03	(3)	1.0E+03
Ni-59	7.5E+04	(3)	8.2E+01	(3)	8.2E+01
Co-60	5.27E+00	~0	4.7E+00	1.2E+03	1.2E+03
Ni-63	1.00E+02	8.9E+02	6.4E+03	(3)	7.3E+03
Se-79	6.5E+04	3.7E+01	~0		3.7E+01
Sr-90	2.86E+01	2.9E+03	6.9E+06	5.0E+05	7.4E+06
Y-90[3]	7.31E-03	2.9E+03	6.9E+06	5.0E+05	7.4E+06
Zr-93	1.53E+06	(3)	2.3E+02	(3)	2.3E+02
Nb-93m	1.46E+01	(3)	2.3E+02	(3)	2.3E+02

Table 2-5. Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Reprocessing[1]

Radionuclide	Half-Life (Year)[2]	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
Tc-99	2.13E+05	1.6E+03	[3]	8.0E+01	1.7E+03
Ru-106	1.01E+00	[3]	1.3E+02	<3.1E-01	1.3E+02
Rh-106	9.48E-07	[3]	1.3E+02	<3.1E-01	1.3E+02
Pd-107	6.5E+06	[3]	1.2E+00	[3]	1.2E+00
Sb-125	2.77E+00	4.8E+01	4.5E+03	[3]	4.5E+03
Te-125m	1.59E-01	1.1E+01	1.0E+03	[3]	1.0E+03
Sn-126	1.00E+05	[3]	4.0E+01	[3]	4.0E+01
Sb-126m	3.61E-05	[3]	4.0E+01	[3]	4.0E+01
Sb-126	3.39E-02	[3]	5.6E+01	[3]	5.6E+01
I-129	1.57E+07	2.1E-01	[3]	<1.5E-01	<3.6E-01
Cs-134	2.06E+00	1.4E+04	[3]	2.9E+02	1.4E+04
Cs-135	2.3E+06	1.6E+02	[3]	[2]	1.6E+02
Cs-137	3.02E+01	7.3E+06	[3]	5.1E+05	7.8E+06
Ba-137m[3]	4.85E-06	6.8E+06	[3]	4.8E+05	7.3E+06
Ce-144	7.78E-01	2.9E-05	1.4E+01	<2.0E-02	1.4E+01
Pr-144	3.29E-05	2.9E-05	1.4E+01	<2.0E-02	1.4E+01
Pm-147	2.62E+00	1.7E+02	3.1E+05	4.5E+03	3.1E+05
Sm-151	9.0E+01	1.1E+00	2.1E+05	1.5E+01	2.1E+05
Eu-152	1.36E+01	4.2E-02	4.2E+02	5.8E+00	4.3E+02
Eu-154	8.8E+00	1.4E+01	1.3E+05	2.6E+03	1.3E+05
Eu-155	4.96E+00	2.3E+00	2.3E+04	3.1E+02	2.3E+04
Th-232	1.41E+10	[3]	[3]	1.6E+00	1.6E+00
U-233	1.59E+05	4.9E-01	6.9E+00	2.6E+00	1.0E+01
U-234	2.45E+05	2.9E-01	4.0E+00	3.0E-01	4.6E+00
U-235	7.04E+08	6.4E-03	8.9E-02	4.9E-03	1.0E-01
U-236	2.34E+07	1.9E-02	2.7E-01	1.0E-02	3.0E-01
U-238	4.47E+09	5.7E-02	7.9E-01	6.1E-04	8.5E-01
Np-237	2.14E+06	[3]	1.1E+01	[3]	1.1E+01
Np-239	6.45E-03	[3]	2.4E+03	[3]	2.4E+03
Pu-238	8.78E+01	1.3E+02	6.5E+03	5.3E+02	7.2E+03
Pu-239	2.41E+04	2.5E+01	1.7E+03	1.7E+01	1.7E+03

Table 2-5. Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Reprocessing(1)

Radionuclide	Half-Life (Year)(2)	Tank 8D-2 Supernatant	Tank 8D-2 Sludge	Tank 8D-4	Total
Pu-240	6.57E+03	1.9E+01	1.3E+03	9.0E+00	1.3E+03
Pu-241	1.44E+01	1.5E+03	8.5E+04	9.3E+02	8.7E+04
Pu-242	3.76E+5	2.5E-02	1.7E+00	1.3E-02	1.7E+00
Am-241	4.32E+02	(3)	7.2E+04	2.7E+02	7.2E+04
Am-242	1.83E-03	(3)	2.1E+01	(3)	2.1E+01
Am-242m	1.52E+02	(3)	2.1E+01	(3)	2.1E+01
Am-243	7.38E+03	(3)	2.4E+03	8.8E+00	2.4E+03
Cm-242	4.47E-01	(3)	2.2E+00	<1.1E-03	2.2E+00
Cm-243	2.85E+01	(3)	1.7E+02	5.0E-02	1.7E+02
Cm-244	1.81E+01	(3)	2.2E+04	1.6E+01	2.2E+04
Cm-245	8.50E+03	(3)	1.0E+01	1.2E-03	1.0E+01
Cm-246	4.75E+03	(3)	4.3E+00	(3)	4.3E+00

NOTES: (1) From Eisenstatt 1986, fission and activation products decay-corrected to July 1987.
(2) Half-life values from Grove Engineering 2003.
(3) Not present or undetectable.
(4) The progeny of Sr-90 and Cs-137 are included here counter to normal practice because they were reported in Table 6 of Eisenstatt 1986.

Table 2-6. Chemical Composition of Tank 8D-2 Supernatant at the Completion of Reprocessing(1)

Compound	% (weight of compound/total weight of supernatant) Wet Basis	% (weight of compound/total weight of compounds) Dry Basis	Total Weight of compounds in Supernatant (Kg)
$NaNO_3$	21.10	53.38	602,659
$NaNO_2$	10.90	27.57	311,326
Na_2SO_4	2.67	6.76	76,261
$NaHCO_3$	1.49	3.77	42,557
KNO_3	1.27	3.21	36,274
Na_2CO_3	0.884	2.24	25,249
NaOH	0.614	1.55	17,537
K_2CrO_4	0.179	0.45	5,113
NaCl	0.164	0.42	4,684
Na_3PO_4	0.133	0.34	3,799

Table 2-6. Chemical Composition of Tank 8D-2 Supernatant at the Completion of Reprocessing[1]

Compound	% (weight of compound/total weight of supernatant) Wet Basis	% (weight of compound/total weight of compounds) Dry Basis	Total Weight of compounds in Supernatant (Kg)
Na_2MoO_4	0.0242	0.06	691
Na_3BO_3	0.0209	0.05	597
$CsNO_3$	0.0187	0.05	534
NaF	0.0176	0.04	503
$Sn(NO_3)_4$	0.00859	0.02	245
$Na_2U_2O_7$	0.00808	0.02	231
$Si(NO_3)_4$	0.00806	0.02	230
$NaTcO_4$	0.00620	0.02	177
$RbNO_3$	0.00416	0.01	119
Na_2TeO_4	0.00287	0.007	82
AlF_3	0.00271	0.007	77
$Fe(NO_3)_3$	0.00152	0.004	43
Na_2SeO_4	0.00054	0.001	15
$LiNO_3$	0.00048	0.001	14
H_2CO_3	0.00032	0.0008	9
$Cu(NO_3)_3$	0.00022	0.0005	6
$Sr(NO_3)_2$	0.00013	0.0004	4
$Mg(NO_3)_2$	0.0008	0.0002	2
Compound Totals	**39.53 %**	**100.00 %**	**1,129,038**
Total H_2O (100% - 39.53%)	**60.47 %**	**NA**	**1,727,164**

NOTE: (1) From Eisenstatt 1986.

Table 2-7. Chemical Composition of Tank 8D-2 Sludge at the Completion of Reprocessing[1]

Compound	Total Mass in Sludge (kg)	Compound	Total Mass in Sludge (kg)
$Fe(OH)_3$	66,040	$Cu(OH)_2$	376
$FePO_4$	6,351	$Zr(OH)_2$	159
$Al(OH)_3$	5,852	$Sm(OH)_3$	143
MnO_2	4,581	$Zn(OH)_2$	128
$CaCO_3$	3,208	$Cr(OH)_3$	65
$UO(OH)_2$	3,087	$Hg(OH)_2$	23
$Ni(OH)_2$	1,088	$Eu(OH)_3$	7.5
SiO_2	1,263	$Gd(OH)_3$	1.7
$MgCO_3$	826	$Pm(OH)_3$	1.5
AlF_3	536		
Fission Products		**Fission Products**	
$Zr(OH)_4$	805	$Y(OH)_3$	103
$Nd(OH)_3$	621	$Rh(OH)_4$	79
$Ru(OH)_4$	458	$Pd(OH)_2$	34
$Ce(OH)_3$	354	$Sn(OH)_4$	2.5
$BaSO_4$	303	$Cd(OH)_2$	1.7
$SrSO_4$	217	AgOH	0.7
$La(OH)_3$	185	$Sb(OH)_3$	0.7
$Pr(OH)_3$	170	$In(OH)_3$	0.3
Transuranics		**Transuranics**	
PuO_2	37	AmO_2	28
NpO_2	35	CmO_2	0.4
Total Chemical Composition = 97,172 kg			

NOTE: (1) From Eisenstatt 1986, with fission products reported separately, unlike other tables, consistent with Eisenstatt 1986.

Table 2-8. Chemical Composition of Tank 8D-4 Waste at the Completion of Reprocessing(1)

Compound	% (Mass of Compound/Mass of Solution)	Total Solution Mass in Tank (kg)	Compound	% (Mass of Compound/Mass of solution)	Total Solution Mass in Tank (kg)
$Th(NO_3)_4$	26.69	12,997	$Ce(NO_3)_3$	0.0387	19
$Fe(NO_3)_3$	19.41	9,452	$Zr(NO_3)_4$	0.0288	14
$Al(NO_3)_3$	9.57	4,660	$Sm(NO_3)_3$	0.0286	14
HNO_3	4.88	2,376	$La(NO_3)_3$	0.0269	13
$Cr(NO_3)_3$	4.40	2,143	$Pr(NO_3)_3$	0.0267	13
$Ni(NO_3)_2$	1.81	881	$Zn(NO_3)_2$	0.0226	11
H_3BO_3	1.10	536	$Rh(NO_3)_4$	0.0222	11
$NaNO_3$	0.759	370	Na_2TcO_4	0.0206	10
Na_2SO_4	0.414	202	$UO_2(NO_3)_3$	0.0156	8
KNO_3	0.294	143	$Y(NO_3)_3$	0.0134	7
Na_2SiO_3	0.290	141	Na_2SeO_4	0.00767	4
K_2MnO_4	0.281	137	$RbNO_3$	0.00619	3
$Nd(NO_3)_3$	0.146	71	$Co(NO_3)_2$	0.00505	2
$Mg(NO_3)_3$	0.131	64	$Pd(NO_3)_4$	0.00469	2
NaCl	0.115	56	NaF	0.00244	1
Na_2MoO_4	0.114	56	$Cu(NO_3)_2$	0.00177	0.9
$Ca(NO_3)_2$	0.0700	34	$Pu(NO_3)_4$	0.00152	0.7
$Ba(NO_3)_2$	0.0697	34	$Eu(NO_3)_3$	0.00142	0.7
$Ru(NO_3)_4$	0.0643	31	$Gd(NO_3)_3$	0.00037	0.2
$CsNO_3$	0.0502	24	$^1X(NO3)_4$	0.00035	0.2
Na_2TeO_4	0.0410	20	$Pm(NO_3)_2$	0.00034	0.2
$Sr(NO_3)_2$	0.0407	20			
Total Weight % in Solution = 71.02 % (total mass of compounds/total mass of solution) or 34,583 kg in Tank. Total weight % of H_2O (100% - 71.02%) = 28.98 % or 14,114 kg in Tank					
Solids					
Compound	Total Solids Mass (kg)		Compound	Total Solids Mass (kg)	
$Th(NO_3)_4$	18,958		Insolubles	39	

NOTE: (1) From Eisenstatt 1986. LEGEND: X = Am-241, Am-243, Cm-242, Cm-243, Cm-244, Cm-24

2.1.2 West Valley Demonstration Project From 1982 to 2008

To meet its objective of solidifying HLW at the site, the WVDP developed the Integrated Radwaste Treatment System and built the Vitrification Facility.

Integrated Radwaste Treatment System

The Integrated Radwaste Treatment System was designed for supernatant and sludge wash solution processing, solidification, and storage. The Integrated Radwaste Treatment System was comprised of four components:

- The Supernatant Treatment System, which decontaminated solutions from the HLW tanks through an ion-exchange process;
- The Liquid Waste Treatment System, which employed an evaporator to concentrate solutions received from the Supernatant Treatment System and byproduct solutions received from vitrification operations;
- The Cement Solidification System that was used to solidify Liquid Waste Treatment System concentrates; and
- The Drum Cell, which provided storage for solidified wastes received from the Cement Solidification System.

The Integrated Radwaste Treatment System pretreatment process is illustrated in Figure 2-2. The initial objective of this system was successfully attained in 1995, resulting in nearly 20,000 drums of solidified waste stored in the Drum Cell. In 2007 those drums were shipped to an offsite LLW disposal facility, leaving the Drum Cell empty of stored radioactive waste in 2008.

Vitrification Facility

This facility was designed for the stabilization and packaging of HLW sludge and contaminated ion-exchange resin (zeolite) generated as a byproduct of Supernatant Treatment System operations. It stabilized the following waste streams in a borosilicate glass matrix: (1) the HLW sludge in Tank 8D-2 that had been generated during PUREX reprocessing by NFS, (2), spent Supernatant Treatment System zeolite, and (3) acidic THOREX waste from Tank 8D-4 generated by the reprocessing of thorium fuel.

The former reprocessing facilities were modified to accommodate the vitrification system and ancillary waste treatment and storage systems. Modifications included removing the reprocessing equipment and decontaminating a number of process cells so that workers could enter the cells for extended periods without respiratory protection. After cleaning the former reprocessing cells, equipment was installed to process gaseous and liquid waste streams. Risers were remotely installed in the HLW tanks, and equipment and pumps were installed for processing HLW supernatant and washing HLW sludge.

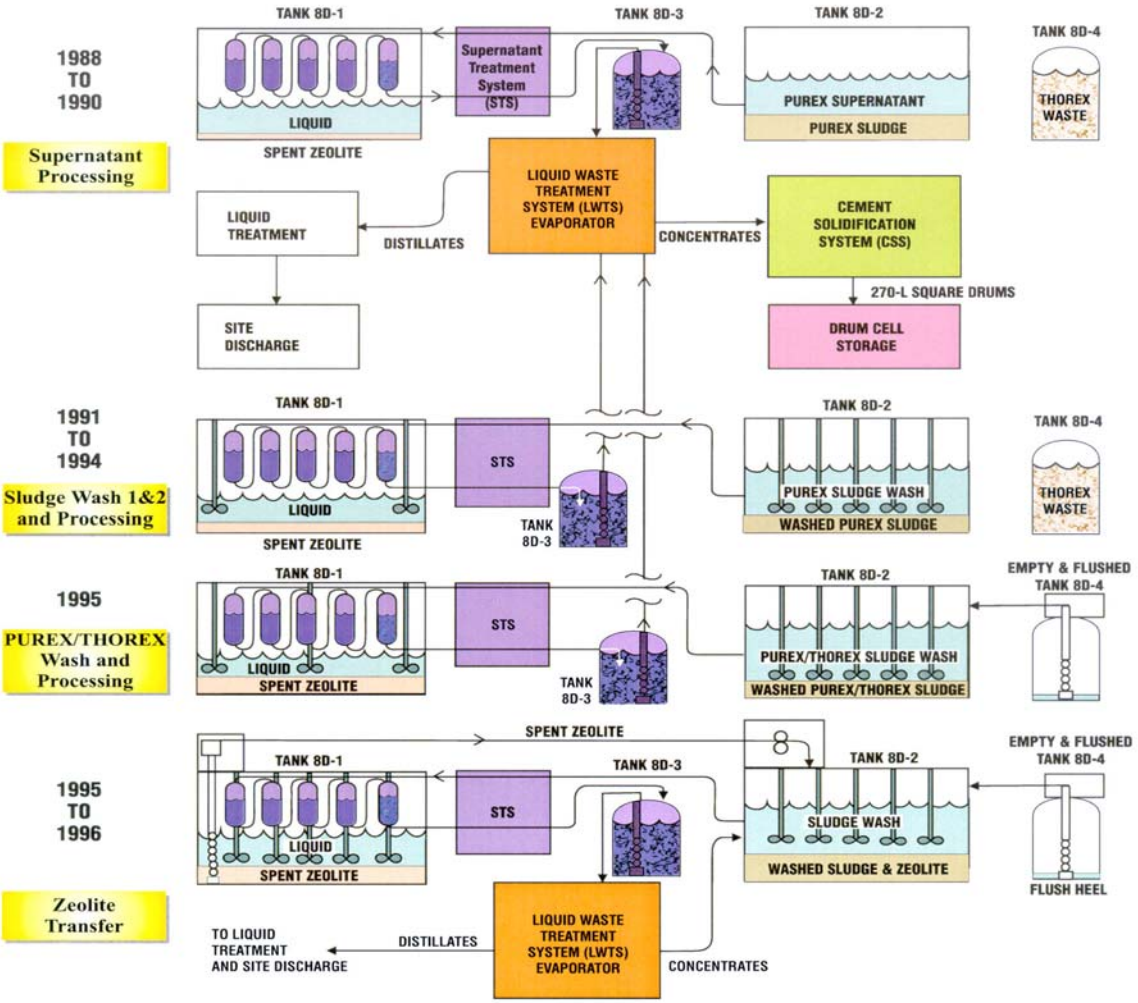

Figure 2-2. Simplified HLW Pretreatment Process Diagram

Underground Waste Tanks 8D-1, 8D-2, 8D-3, and 8D-4

Pre-Vitrification HLW tank usage by the WVDP is outlined in Section 2.1.2 under Integrated Radwaste Treatment System. Tank 8D-1 was used to house the Supernatant Treatment System treatment columns used to remove radioactivity from the Tank 8D-2 supernatant, sludge wash, and PUREX/THOREX wash processing campaigns. The treated liquid was transferred to Tank 8D-3 and then volume-reduced in the Liquid Waste Treatment System, and solidified in the Cement Solidification System for offsite disposal as LLW. The zeolite resin used to treat the supernatant, sludge wash, and PUREX/THOREX wash remained in Tank 8D-1, and was added to the feed mixture to be vitrified. The thorium-bearing HLW from tank 8D-4 was mixed with the contents of tank 8D-2 and washed to remove soluble salts before being readied for vitrification.

Solidification Activities

During the vitrification process, the mobilized sludge and cesium-loaded zeolite resin (which was transferred from Tank 8D-1 to Tank 8D-2) were transferred to the Concentrator Feed Makeup Tank in the Vitrification Cell, where excess water was removed and glass formers added. The resulting mixture was then transferred to the Melter Feed Hold Tank. From this tank, the feed was delivered to the Slurry-Fed Ceramic Melter, where it was heated to form a molten, waste-loaded, borosilicate glass.

The molten glass was then poured into a stainless steel canister located in and positioned by a rotating turntable. Once a canister was filled, it remained on the turntable for initial cooling, then it was removed from the turntable for further cooling, canister lid welding, and external decontamination. The borosilicate glass matrix filled each canister to more than 80 percent of its volume as required by the Waste Acceptance Product Specifications established by DOE (DOE 1993).

After decontamination, the canister was loaded onto a transfer cart that moved on rails through the transfer tunnel and into the High Level Waste Interim Storage Facility (the former Chemical Process Cell) in the Process Building, where the canisters were loaded into racks for storage. The canisters will remain there until they are transported to an alternate storage location.

A total of 275 canisters of HLW were produced. Two additional canisters were filled with materials which remained in the melter. The solidification of the liquid HLW waste was completed in September 2002 and the Vitrification Facility was radiologically characterized in November 2002 (Lachapelle 2003)[3].

Table 2-9 provides the major chemical components of the glass waste form, and Table 2-10 describes the radionuclide content of a typical vitrified HLW canister processed during the HLW vitrification campaign (WVNSCO 2007a).

Sodium-Bearing Waste

As a component of tank management over time, sodium salts were added to the HLW tanks to limit corrosion of the carbon steel tanks. More recently, clean utility water used to cool the in-tank mobilization pumps added excess fluids to the HLW tanks before and during vitrification. Since sodium is a limiting ingredient in a qualified glass recipe, the high-sodium water was segregated from the HLW feed mixture. A process was developed to volume-reduce the waste water containing high levels of sodium and solidify the 11,500 gallons of concentrate into a form suitable for LLW land disposal. The solidification was completed within the O1-14 building in 2004, and the sodium-bearing waste was shipped for disposal in 2007. (Rowell 2001, WVNSCO and URS 2005, Bower 2008)

The amount of residual radioactivity in the HLW tanks is discussed in Section 4.1.

[3] This characterization took place before decontamination of the Vitrification Cell, which entailed removing the slurry-fed ceramic melter, tanks, and other equipment.

Liquid LLW Streams

Under the WVDP, the Low-Level Waste Treatment Facility included the Neutralization Pit, the interceptors, Lagoons 2-5, and the LLW2 Building, which replaced the NFS O2 Building. The wastewater is collected in one of the interceptors. After radiological analysis, the wastewater is transferred to Lagoon 2 and is then treated in the LLW2 Building. Following treatment, the wastewater is transferred to Lagoons 4 and 5. If the treated wastewater in Lagoons 4 and 5 meets specifications, it is transferred to Lagoon 3 for eventual release through a State Pollutant Discharge Elimination System-permitted outfall to Erdman Brook. Out-of-specification wastewater is returned to Lagoon 2 and is re-treated.

In summary, under the WVDP the Vitrification Facility, the Integrated Radwaste Treatment System, the Sludge Mobilization System, and a new low level waste treatment facility (LLW2 Building) were developed and operated. The waste (supernatant and sludge) in the HLW tanks was vitrified and solidified in stainless steel canisters that are stored in the High-Level Waste Interim Storage Facility in the Process Building.

Table 2-9. Chemical Composition of Glass Waste Form[(1)]

Component	Nominal Weight %	Range Weight %		Component	Nominal Weight %	Range Weight %	
AgO	0.0001			Nd_2O_3	0.1209	0.08	0.19
Al_2O_3	2.8295	1.19	7.15	NiO	0.3358	0.22	0.52
AmO_2	0.0073			NpO_2	0.0224	0.01	0.03
BaO	0.0540	0.04	0.08	P_2O_5	2.5084	0.21	3.16
B_2O_3	9.9516	9.33	10.66	PdO	0.0062		
CaO	0.5993	0.39	0.93	Pm_2O_3	0.0003		
CdO	0.0003			Pr_6O_{11}	0.0321	0.02	0.05
CeO_2	0.0670	0.04	0.10	PuO_2	0.0076		
CmO_2	0.0001			Rb_2O	0.0005		
CoO	0.0002			RhO_2	0.0136	0.01	0.02
Cr_2O_3	0.3112	0.21	0.48	RuO_2	0.0759	0.05	0.12
Cs_2O	0.0826	0.05	0.13	SO_3	0.2164	0.14	0.33
CuO	0.0001			Sb_2O_3	0.0001		
Eu_2O_3	0.0014			SeO_2	0.0005		
Fe_2O_3	12.1573	8.32	18.50	SiO_2	44.8770	42.08	48.10
Gd_2O_3	0.0003			Sm_2O_3	0.0267	0.02	0.04
In_2O_3	0.0001			SnO_2	0.0006		
K_2O	3.5733	3.36	3.84	SrO	0.0269	0.02	0.04

Table 2-9. Chemical Composition of Glass Waste Form(1)

Component	Nominal Weight %	Range Weight %		Component	Nominal Weight %	Range Weight %	
La_2O_3	0.0337	0.02	0.05	Tc_2O_7	0.0021		
Li_2O	3.0315	2.84	3.25	ThO_2	3.5844	1.83	6.56
MgO	1.3032	1.22	1.39	TeO_2	0.0028		
MnO_2	1.3107	0.84	1.96	TiO_2	0.9800	0.92	1.05
MoO_2	0.0088		0.01	UO_2	0.5605	0.37	0.87
NaCl	0.0183	0.01	0.03	Y_2O_3	0.0177	0.01	0.03
NaF	0.0013			ZnO	0.0010		
Na_2O	10.9335	10.25	11.71	ZrO_2	0.2943	0.19	0.45
Insolubles	0.0080						

NOTE: (1) From Eisenstatt 1986.

Table 2-10. Typical HLW Canister Radionuclide Content(1)

Radionuclide	Estimated Activity (Ci/canister)	Radionuclide	Estimated Activity (Ci/canister)
Ni-63	3.5E+01	Pu-240	4.0E+00
Sr-90	1.36E+04	Pu-241	1.75E+02
Sm-151	1.89E+02	Am-241	1.53E+02
Cs-137	2.34E+04	Cm-243	1.0E+01
Pu-238	1.9E+01	Cm-244	3.5E+01
Pu-239	5.0E+00		

NOTE: (1) From WVNSCO 2007a

2.2 Site Decontamination Activities (1966 – 2011)

This section summarizes remediation activities[4] performed by NFS, those that have been performed by the WVDP, and those that will be performed by the WVDP to establish the interim end state before the beginning of activities under this plan. Although the WVDP remediation activities have generally been performed in connection with cleanup, modifications, or deactivation work, they are relevant to the starting point for the decommissioning.

[4] For purposes of this section, the terms *remediation* and *decontamination* are roughly equivalent. Each is defined as the removal of undesired residual radioactivity from facilities, soil, or equipment prior to release (NRC 2006). The term *remediation* may also be used in the context of preparing facilities to conform to specific requirements using fixatives or other treatments.

2.2.1 NFS Remediation Activities (1966 – 1981)

During the 1960s, NFS remediation efforts were limited to those actions needed to maintain production, such as spill cleanup and equipment replacement. In the 1970s, NFS initiated decontamination activities initially in preparation for extensive in-cell reliability and expansion work to increase production. Decontamination procedures were prepared for decontamination of the partition cycle, uranium cycle, plutonium cycle, solvent recovery systems, acid recovery system and acid storage tanks, and the dissolver off-gas system (Riethmiller 1981).

Gross decontamination was accomplished by flushing process tanks and piping and removing loose contamination from the cells and process equipment. In some cases, fixatives were applied to contamination that could not be readily removed.

Changes in mixed fission product activity levels were determined from measurements obtained by lowering dosimeters, strung at various levels, into Extraction Cells 1, 2, and 3 through holes drilled in the Extraction Chemical Room floor. Activity removed by decontamination activities from 1972 through 1977, including amounts of uranium and plutonium, is summarized in Table 2-11. No extensive decontamination activities are documented from 1977 until commencement of DOE operations in 1982.

Table 2-11. Activity Removed by NFS for the Period 1972 Through 1977(1)

Year	Mixed Fission Products (curies)	Uranium (grams)	Plutonium (grams)
1972	182,758.1	47,700	1550
1973	886.2	3,722	24
1974	659.6	5,099	229
1975	15	572	12
1976	22.3	282	18
1977	6.8	718	1
Total	184,348	58,093	1,834

NOTES: (1) From Riethmiller 1981.

Radioactive material generated during the NFS remediation work was disposed of as radioactive waste in the NDA and SDA.

2.2.2 WVDP Remediation Activities (1982 – 2011)

After 1982, remediation activities included decontamination, waste removal, equipment removal, and the application of fixatives. Procedures were developed by West Valley Nuclear Services Company (WVNSCO) as part of the remediation project for each facility. Radioactive material and waste generated or removed as part of remediation activities were packaged for offsite shipment or temporary storage, with some waste disposed of in the NDA prior to 1987.

Figures 2-3 and 2-4 show those WVDP facilities that have had a history of radiological contamination. Figure 2-5 shows locations of planned remediation activities for site facilities before Phase 1 of the proposed decommissioning. Table 2-12 that follows these figures provides a legend for the acronyms and abbreviations in the figures. This table also identifies the functions of the facilities.

List of Facilities Remediated or to be Remediated

Table 2-13 that follows these figures lists those facilities (in alphabetical order) that have been or will be remediated (or partially remediated) before the start of the Phase 1 of the proposed decommissioning. The type and form of contamination are specified, as well as information on the radiological conditions before and after remediation based on available data. The activities that caused the facility to become contaminated are also summarized. Facilities that have been removed as of 2008 are identified as “Removed.” More-detailed descriptions of these facilities appear in Section 3, along with layout drawings showing their locations. Section 3 also contains photographs of many of these facilities.

Note that Table 2-13 does not list non-radiological facilities that have been or will be removed as part of the work to establish the interim end state, such as the Cold Chemical facility, the Vehicle Repair shop, and the Vitrification Test Facility (as shown on Figure 2-5). The table also does not address facilities outside of the project premises since the scope of the Phase 1 proposed decommissioning activities is limited to the project premises.

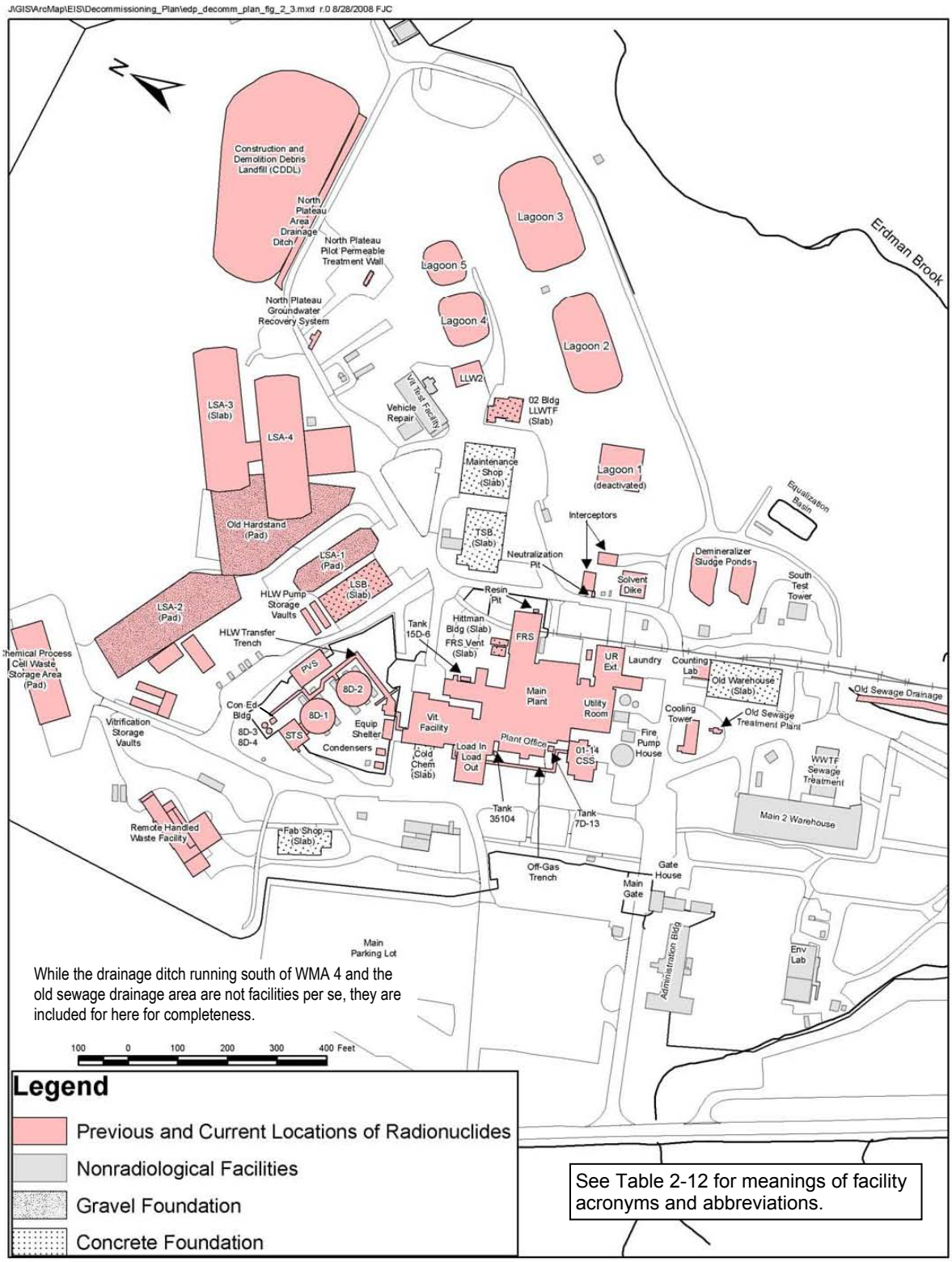

Figure 2-3. Previous and Current Locations of Radionuclides in North Plateau Facilities at the WVDP

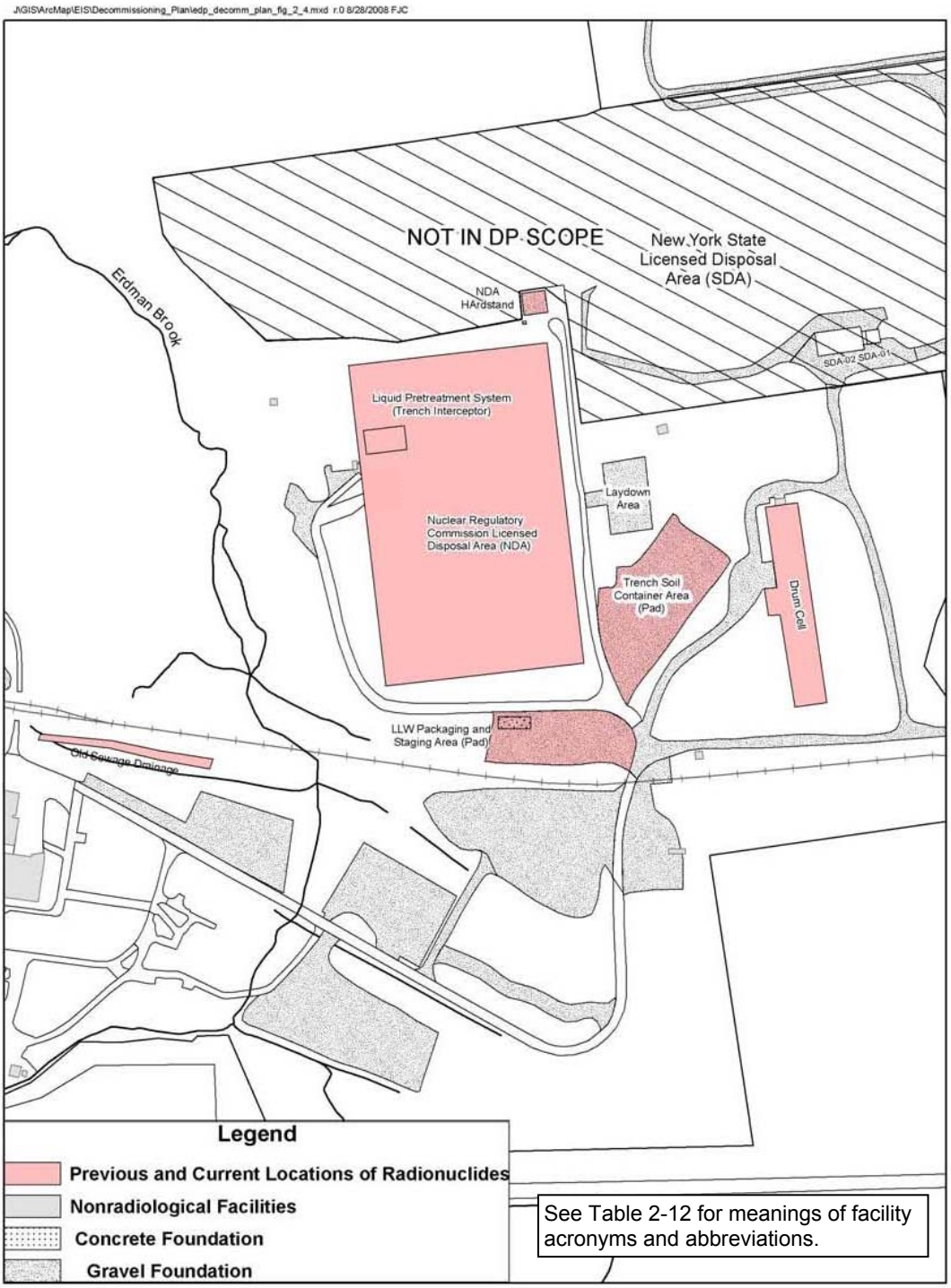

Figure 2-4. Previous and Current Locations of Radionuclides in South Plateau Facilities at the WVDP

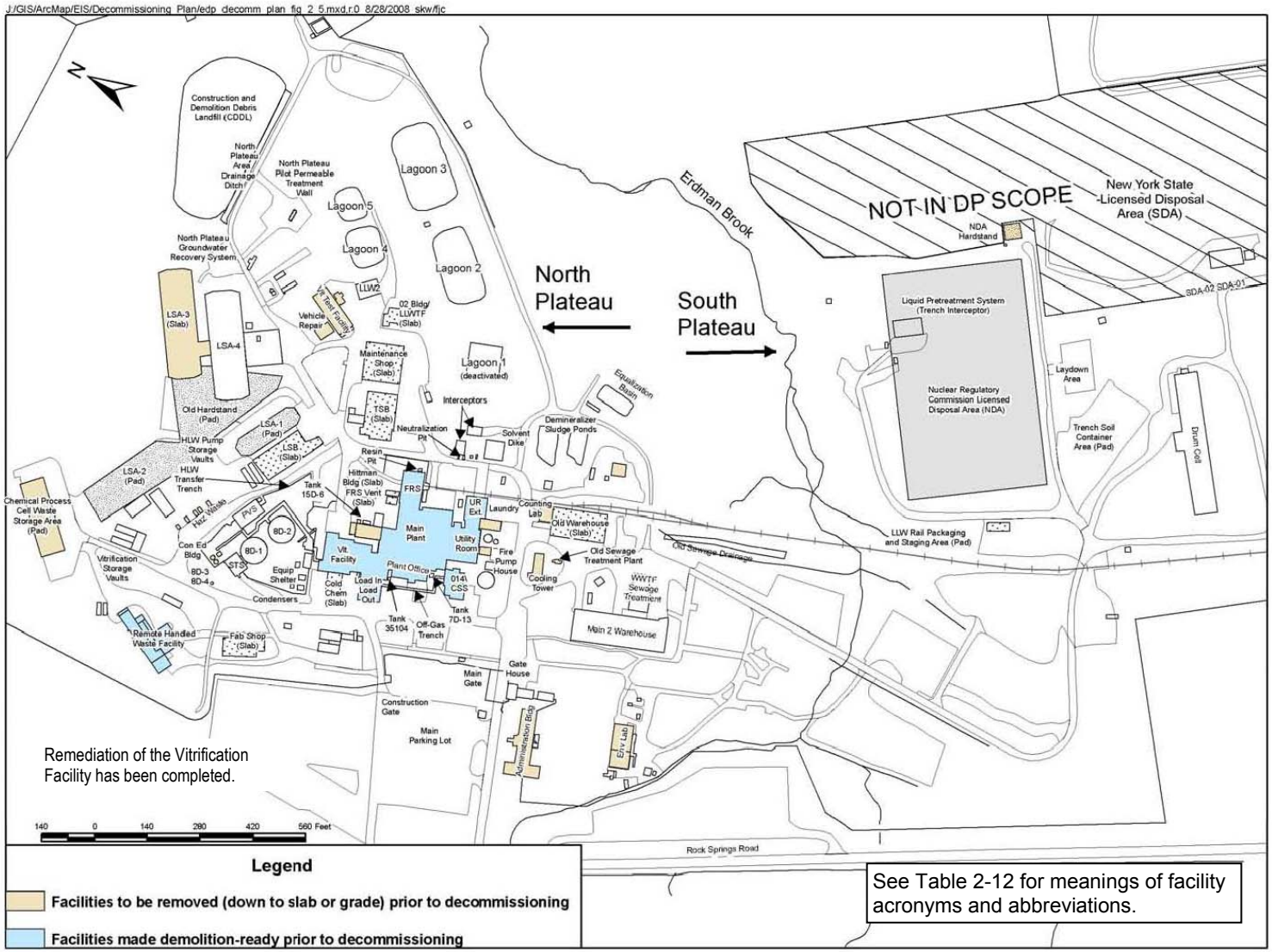

Figure 2-5. Locations of Planned Remediation Activities for Site Facilities Prior to Phase 1 of the Proposed Decommissioning

Table 2-12. Facilities Shown in Figures 2-3 through 2-5

Designation	Facility	Function
8D-1, -2, -3, -4	Underground waste tanks	Designed to store HLW; 8D-1, 8D-2, and 8D-4 have contained HLW.
01-14	The Cement Solidification System building	Facility housed the Cement Solidification System and the vitrification off-gas treatment equipment.
CDDL	Construction & Demolition Debris Landfill	Non-radioactive waste burial area.
Cold Chem	Cold Chemical facility	Housed containerized non-radioactive chemicals.
Con Ed Bldg	Consolidated Edison Building	Houses HLW tank instrumentation and equipment.
CPC-WSA	Chemical Process Cell Waste Storage Area	Storage for equipment and waste from the CPC (now HLW Interim Storage Facility).
CSS	Alternate designation for the 01-14 building	Facility housed the Cement Solidification System and the vitrification off-gas treatment equipment.
Env Lab	Environmental Laboratory	Houses environmental testing equipment and instrumentation.
Equip. Shelter	Equipment Shelter	Houses HLW tank instrumentation and equipment.
Fab Shop	Fabrication Shop	Non-radioactive metal fabrication shop – demolished, slab remaining.
FRS	Fuel Receiving and Storage Facility	Formerly used to store spent nuclear fuel.
FRS Vent	Fuel Receiving and Storage Ventilation Building	Housed cooling system equipment for the FRS pool water – demolished, slab remaining.
LLW2	Low Level Waste 2	Houses low level radioactive liquid treatment system currently in use.
LLWTF	Low Level Waste Treatment Facility	Housed low level radioactive liquid treatment system – demolished, slab remaining.
LSA 1	Lag Storage Area 1 (also, LSA2, LSA3 and LSA4)	Containerized radioactive waste storage. LSA1 and LSA2 have been removed, gravel pads remain.
LSB	Lag Storage Building	Containerized radioactive waste storage building – demolished, slab remaining.
NDA	NRC-Licensed Disposal Area	Radioactive waste burial area.
O2 Bldg	An alternate designator for the LLWTF	Housed low level radioactive liquid treatment system – demolished, slab remaining.
PVS	Permanent Ventilation System [Building]	Provides ventilation for the Supernatant Treatment System and the underground waste tanks.
STS	Supernatant Treatment System [Building]	Facility used primarily for treatment of HLW supernatant.
TSB	Test and Storage Building	Non-radioactive fabrication and testing shop – demolished, slab remaining.
UR Expan	Utility Room expansion facility	Houses utility systems equipment.
Vit. Facility	Vitrification Facility	Housed systems for solidifying HLW.
WWTF	Waste Water Treatment Facility	Sewage Treatment Plant.

Table 2-13. Facilities Remediated or to be Remediated by the WVDP Before Decommissioning[1]

Facility	Location and Function	Principal Radionuclides			Expected Status at the Start of Phase 1 of the Decommissioning
		Type	Form	Initial Activity and Cause of Contamination	
01-14 Building	WMA-1 Radioactive waste processing system facility	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination, fixed contamination	Contamination from previous solidification system operations, and filtration/treatment of vitrification off-gas.[3]	Deactivated and prepared for demolition. Partially decontaminated, radiation area in some cells, significant contamination in filters (if still in place).
Chemical Process Cell Waste Storage Area	WMA-5 Containerized LLW storage	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination	~275 Ci Cs-137 in packaged equipment as of 1996.[4] 15 mR/h from stored waste, removable contamination below detection limits.[6] Incidental contamination possible from radioactive waste container storage activities.	Removed to grade. No contamination above 10 CFR 835 control limits.[5]
Contact Size Reduction Facility	WMA-1 Radioactive waste size reduction system facility	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination	5 mR/h, removable contamination below detection limits.[6] Incidental contamination possible from radioactive waste size reduction activities.	Removed to concrete slab. No contamination above 10 CFR 835 control limits.[5]
Cooling Tower	WMA-6 Utility water cooling system	Radionuclide mix typical of feed and waste contamination[2]	Fixed surface contamination	< 0.1 mR/h, removable contamination below detection limits.[6] Coil leaks from contaminated cooling water.	Removed to concrete basin. Contamination above 10 CFR 835 control limits, posting required.[5]
FRS Ventilation Building	WMA-1 Cooling system for fuel pool water	Fission products and transuranics from spent fuel	Surface contamination	1.3 mR/h, removable contamination below detection limits.[7] Spent nuclear fuel pool water contamination.	Removed October 2006, slab remains. No contamination above 10 CFR 835 control limits.[5]
Lag Storage Addition 1 (LSA 1)	WMA-5 Radioactive waste container staging area	Radionuclide mix typical of feed and waste contamination[2]	Low-level fixed contamination in some areas	< 0.1 mR/h, removable contamination below detection limits.[7] Incidental contamination from containerized LLW staging and sorting activities.	Removed 2006, slab remains. No contamination above 10 CFR 835 control limits.[5]
Lag Storage Addition 2 (LSA 2 hardstand)	WMA-5 Radioactive waste container staging area	Radionuclide mix typical of feed and waste contamination[2]	Low-level fixed contamination in some areas	15 mR/h from stored waste, removable contamination below detection limits.[6] Incidental contamination from containerized LLW staging and sorting activities.	Slab remains. No contamination above 10 CFR 835 control limits.[5]

Table 2-13. Facilities Remediated or to be Remediated by the WVDP Before Decommissioning[1]

Facility	Location and Function	Principal Radionuclides			Expected Status at the Start of Phase 1 of the Decommissioning
		Type	Form	Initial Activity and Cause of Contamination	
Lag Storage Addition 3 (LSA 3)	WMA-5 Radioactive waste container staging area	Radionuclide mix typical of feed and waste contamination[2]	Low-level fixed contamination in some areas	50-100 mR/h from stored waste, removable contamination below detection limits.[6] Incidental contamination from containerized LLW staging & sorting activities.	Slab remains. No contamination above 10 CFR 835 control limits.[5]
Lag Storage Building	WMA-5 Radioactive waste container staging area	Radionuclide mix typical of feed and waste contamination[2]	Low-level fixed contamination in some areas	< 0.1 mR/h, removable contamination below detection limits.[7] Incidental contamination from containerized LLW staging & sorting activities.	Removed October 2006, slab remains. No contamination above 10 CFR 835 control limits.[5]
Laundry Room	WMA-1 Contaminated clothing cleaning facility	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination, fixed contamination	0.4 mR/h, 2,000 dpm/100 cm^2 beta.[8] Incidental contamination from sorting and handling of contaminated laundry.	To be removed to concrete slab. Contamination above 10 CFR 835 control limits, posting required.[5]
LLWTF (O2 Building)	WMA-2 Radioactive material processing system facility	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination, fixed contamination	0.12 mR/h, 3,700 dpm/100 cm^2 beta.[7] Contamination from previous radioactive water treatment system operations.	Removed October 2006, slab remains. Contamination above 10 CFR 835 control limits, posting required. [5]
Maintenance Shop	WMA-2 Tool crib and non-radiological equipment maintenance.	Radionuclide mix typical of feed and waste contamination[2]	Incidental surface contamination	< 0.1 mR/h, removable contamination below detection limits.[8] Incidental contamination from mud nests (bird and wasp) and tools.	Removed June 2007, slab remains. No contamination above 10 CFR 835 control limits.[5]
Master Slave Manipulator Repair Shop	WMA-1 Radioactive equipment repair	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination	2.4 mR/h. [6] Disassembly and repair of radiologically contaminated equipment.	To be removed to concrete slab. No contamination above 10 CFR 835 control limits[,5]
NDA Hardstand/ Staging Area	WMA-7 Radioactive waste container staging area	Fission products and transuranics from spent fuel	Surface contamination, soil contamination	6 mR/h, 6,300 dpm/100 cm^2 beta.[7] Storage of waste containers prior to disposal.	Above-grade structure removed September 2006, gravel pad remains. Contamination above 10 CFR 835 control limits, posting required. [5]

Table 2-13. Facilities Remediated or to be Remediated by the WVDP Before Decommissioning[1]

Facility	Location and Function	Principal Radionuclides			Expected Status at the Start of Phase 1 of the Decommissioning
		Type	Form	Initial Activity and Cause of Contamination	
Old/New hardstand	WMA-5 Radioactive transport vehicle staging area	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination, soil contamination	~10 Ci beta, ~2 Ci alpha prior to transfer to Lagoon 1 for stabilization.[9] Storage of radioactive material transport containers prior to disposition.	Removed contaminated asphalt and peripheral biomass in 1984, gravel pad remains. Contamination above 10 CFR 835 control limits, posting required. [5]
Old Sewage Treatment Facility	WMA 6 Sanitary waste treatment until 1985	Radionuclide mix typical of feed and waste contamination[2]	Possible surface contamination	Low level radioactivity may be present from sewage lines running from the Process Building.	Possible low level contamination in concrete basins and other remaining equipment.
Old (Main 1) Warehouse	WMA-6 Receipt and storage of non-radiological materiel	Radionuclide mix typical of feed and waste contamination[2]	Incidental surface contamination	< 0.1 mR/h with removable contamination below detection limits.[8] Incidental contamination from wasp, bird, and rodent nests.	Removed May 2006, slab remains. No contamination above 10 CFR 835 control limits.[5]
Process Building	WMA-1 Spent nuclear fuel reprocessing facility	Radionuclide mix typical of feed and waste contamination[2] in most areas (see Table 4-3)	Surface contamination, some contamination in depth	Residual contamination ~6,200 Ci (see Tables 4-5, 4-6, and 4-7) from operations associated with reprocessing of spent nuclear fuel. (This does not include radioactivity in the 275 vitrified HLW canisters temporarily stored in the HLW Interim Storage Facility as shown in Table 2-10.)	Partially decontaminated, high radiation area in some cells, vitrified HLW canisters stored in the HLW Interim Storage Facility.
Radwaste Process (Hittman) Building	WMA-1 Radiological material processing	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination	8 mR/h, 3,700 dpm/100 cm^2 beta[7] Stabilizing radiologically contaminated materials	Removed October 2006, slab remains. Contamination above 10 CFR 835 control limits, posting required.[5]
Remote-Handled Waste facility	WMA-5 Size-reduction and packaging of highly radioactive waste	Radionuclide mix typical of feed and waste contamination[2]	Surface contamination	~4,800 Ci aged mixed fission products (max annual waste estimate).[10] Contamination of facility cell systems from size-reduction of highly radioactive waste	Deactivated and prepared for demolition. Partially decontaminated, low levels of contamination, may be significant contamination in Work Cell.

Table 2-13. Facilities Remediated or to be Remediated by the WVDP Before Decommissioning[1]

Facility	Location and Function	Principal Radionuclides			Expected Status at the Start of Phase 1 of the Decommissioning
		Type	Form	Initial Activity and Cause of Contamination	
Test and Storage Building (TSB)	WMA-2 Testing & process development, equipment fabrication, office space	Radionuclide mix typical of feed and waste contamination[2]	Incidental surface contamination	< 0.1 mR/h, removable contamination below detection limits.[8] Incidental contamination from wasp and bird nests	Removed May 2006, slab remains. No contamination above 10 CFR 835 control limits[5]
Vitrification Facility	WMA-1 High-temperature process system for HLW vitrification	See Table 4-4.	Surface contamination	~1900 Ci, see Table 4-8. Contamination from HLW vitrification process	Deactivated and prepared for demolition. Partially decontaminated, high radiation levels in Vitrification Cell.

NOTES:
(1) The list of facilities is from DOE 2006 and includes only contaminated facilities. Section 3 describes these facilities.
(2) Feed and waste contamination is described in Section 4.1 and Table 4-3 shows typical relative fractions of the dominant radionuclides in this type of contamination.
(3) No meaningful initial activity estimate is available. The vitrification off-gas system contains significant residual activity as indicated in Section 4.1.5, but most is located outside the building in the off-gas line. Approximately 3000 curies of decontaminated supernatant and sludge wash solutions were solidified in steel drums in the Cement Solidification System (Marschke 2006).
(4) WVNSCO 2007a.
(5) Removable and fixed slab/soil contamination per 10 CFR 835 control levels Listed radioactivity values for surface contamination within a controlled area are shown in Table 2-13. Radioactivity levels inside a radiological area within a controlled area may be higher, depending upon the controls imposed, per Table 2-14.
(6) WVES 2008.
(7) WVNSCO 2006.
(8) WVNSCO 2007b.
(9) Derived from WVNSCO 1995.
(10) URS 2001.

Information in Table 2-13

Radiological survey data for 2006 through mid-2008 were used to identify recent radiological conditions for most facilities. Section 4 addresses the radiological status of various areas of the Process Building and other facilities within plan scope in more detail.

Discussion of WVDP Remediation Efforts

Historical remediation activities are summarized in Section 2.2. As of 2008, remediation of WVDP facilities remained a work in progress. Areas in which initial deactivation work was completed in late 2004 include three cells in the Process building: the General Purpose Cell, the Process Mechanical Cell, and Extraction Cell 2. Additional decontamination is planned for the floors and walls of the General Purpose Cell and the Process Mechanical Cell.

Deactivation of the Vitrification Cell in the Vitrification Facility was completed in 2005. In late 2008, the cell was being used for sorting and packaging of radioactive waste so conditions in this area are subject to change and additional decontamination may be performed before Phase 1 of the proposed decommissioning.

The Interim Waste Storage Facility and the Lag Storage Building, as well as the Lag Storage Area 1 weather shelter were decontaminated and demolition completed in 2006. The Interim Waste Storage Facility concrete slab was removed. Support facilities and structures demolished and removed by the end of 2006 included the north Waste Tank Farm Test Tower, the O2/LLWTF Building, the Maintenance Storage Area, the Sample Storage and Packaging Facility, the Fabrication Shop, the Radwaste Process (Hittman) Building, and the Cold Chemical Facility. In 2007 the Test and Storage Building, the Maintenance Shop, and the Main 1 Warehouse were demolished and removed. (WVNSCO and URS 2005, WVNSCO and URS 2006, WVNSCO and URS 2007, WVES and URS 2008)

The facilities being removed are being taken down to their concrete floor slabs and foundations. Facilities inside the fenced controlled area may already be below the surface contamination levels for materials in a controlled non-radiological area per 10 CFR 835, as shown in Table 2-14. Those facility locations will have few, if any, access restraints imposed. Other remaining floor slabs and foundations within the controlled fenced area may be posted to restrict personnel access, per 10 CFR 835 requirements for radiological control area restrictions as shown in Table 2-15.

Table 2-14. DOE 10 CFR 835 Surface Contamination Guidelines (in dpm/100 cm^2)[(1)]

Radionuclide Contaminant[(2),(4),(6)]	Removable[(2),(4)]	Total (Fixed + Removable)[(2),(3)]
U-natural, U-235, U-238, and associated decay products	1,000[(7)]	5,000[(7)]
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	20	500
Th-natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above[(5)]	1,000	5,000
Tritium and STCs[(6)]	10,000	See note (6).

NOTES:
(1) The values in this table, with the exception noted in note (6) below, apply to radioactive contamination deposited on, but not incorporated into the interior or matrix of, the contaminated item. Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides apply independently.
(2) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.
(3) The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the value specified. For purposes of averaging, any square meter of surface shall be considered to be above the surface contamination value if: (1) from measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds three times the applicable value.
(4) The amount of removable radioactive material per 100 cm^2 of surface area should be determined by swiping the area with dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note - The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area shall be based on the actual area and the entire surface shall be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.
(5) This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.
(6) Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply. In certain cases, a “Total” value of 10,000 dpm/100 cm^2 may be applicable either to metals of the types from which insoluble special tritium compounds (STCs) are formed, that have been exposed to tritium, or to bulk materials to which insoluble special tritium compound particles are fixed to a surface.
(7) These limits apply only to the alpha emitters within the respective decay series.

Table 2-15. Radiological Areas and Radioactive Material Areas(1)

Area Name	Posting	Reference Value
Radiation Area	"Caution, Radiation Area"	Radiation area means any area, accessible to individuals, in which radiation levels could result in an individual receiving an equivalent dose to the whole body in excess of 0.005 rem (0.05 mSv) in 1 hour at 30 centimeters from the source or from any surface that the radiation penetrates
Contamination Area	"Caution, Contamination Area"	Contamination area means any area, accessible to individuals, where removable surface contamination levels exceed or are likely to exceed the removable surface contamination values specified in Table 2-14, but do not exceed 100 times those values.
Radioactive Material Area	"Caution, Radioactive Material(s)"	Radioactive material area means any area within a controlled area, accessible to individuals, in which items or containers of radioactive material exist and the total activity of radioactive material exceeds the applicable values provided in appendix E of 10 CFR 835.(2)

NOTES: (1) From 10 CFR 835, with only those areas likely to be applicable to a foundation slab or other open area listed.

(2) Appendix E of 10 CFR 835 lists individual radionuclide radioactivity levels below which radiological controls are not required.

During the deactivation activities, equipment is being removed using conventional segmenting and handling techniques. The structures are being removed using conventional dismantlement and demolition methods. Waste generated is being shipped off site. Radiological surveys, which are discussed further in Section 9, would document the radiological conditions at the conclusion of deactivation. The radionuclide most significant from the standpoint of radiation protection during this work is Cs-137.

As a major facility undergoing preparation for demolition during decommissioning, most Process Building areas are being deactivated during work to achieve the interim end state, with piping and equipment removed and piping cut off flush with facility surfaces. The Vitrification Facility has undergone a similar deactivation and the Remote-Handled Waste Facility will be deactivated in the same manner. However, some radioactive equipment and significant amounts of residual radioactivity will remain in the Process Building and Vitrification facility at the beginning of Phase 1 proposed decommissioning work as detailed in Section 4.1.

2.3 Spills and Uncontrolled Release of Radioactivity

This section describes spills and uncontrolled releases of radioactivity that have impacted the environment or had the potential to do so. Most of the numerous spills of radioactivity that occurred during NFS operations were contained within the Process Building and these are not detailed here. However, the radioisotope inventory reports generated by the Facility Characterization Project (Michalczak 2004) have documented conditions resulting from significant spills contained within the facilities.

There were two major spills considered to be significant to the site that occurred during licensed reprocessing operations, producing areas of contamination known today as the north plateau groundwater plume and cesium prong. Table 2-16 provides information about the radioactivity associated with the north plateau groundwater plume. More details on radioactivity associated with these two areas appear in Section 4.2.

2.3.1 North Plateau Groundwater Plume

The north plateau groundwater plume is a 540-foot wide by 1,300-foot long (in 2007) zone of groundwater contamination that extends northeastward from the Process Building in WMA 1 to the Construction and Demolition Debris Landfill in WMA 4, where it splits into western and eastern lobes. Lagoon 1 is also a possible contributor of gross beta activity in part of the plume, at least in this lagoon's immediate vicinity (Figure 2-6) (WVES and URS 2008).

Strontium-90 and its decay product, Y-90, are the principal radionuclides in this plume, with both radionuclides contributing equal amounts of beta activity. In 1994 it was determined that Sr-90 concentrations were as high as 1.2 µCi/L in groundwater on the east side of the Process Building. Results of the latest core area investigation in 1998 determined that the highest Sr-90 concentration was 0.705 µCi/L beneath the Uranium Loadout Room near the southeast end of the Process Building (Hemann and Steiner 1999). More information about the plume appears in Section 4.2.

The presumed primary source of the plume was an acid recovery line that leaked in the southwest corner of the Process Building during the late 1960's. The leak released an estimated 200 gallons of radioactive nitric acid from the Off-Gas Operating Aisle down to the underlying Off-Gas Cell and the adjacent southwest stairwell (Carpenter and Hemann 1995).

The leakage apparently flowed through an expansion joint in the concrete floor of the Off-Gas Cell and migrated into the sand and gravel underlying the Process Building (Westcott 1998). This leak also contributed to sewage treatment system contamination (Duckworth 1972b).

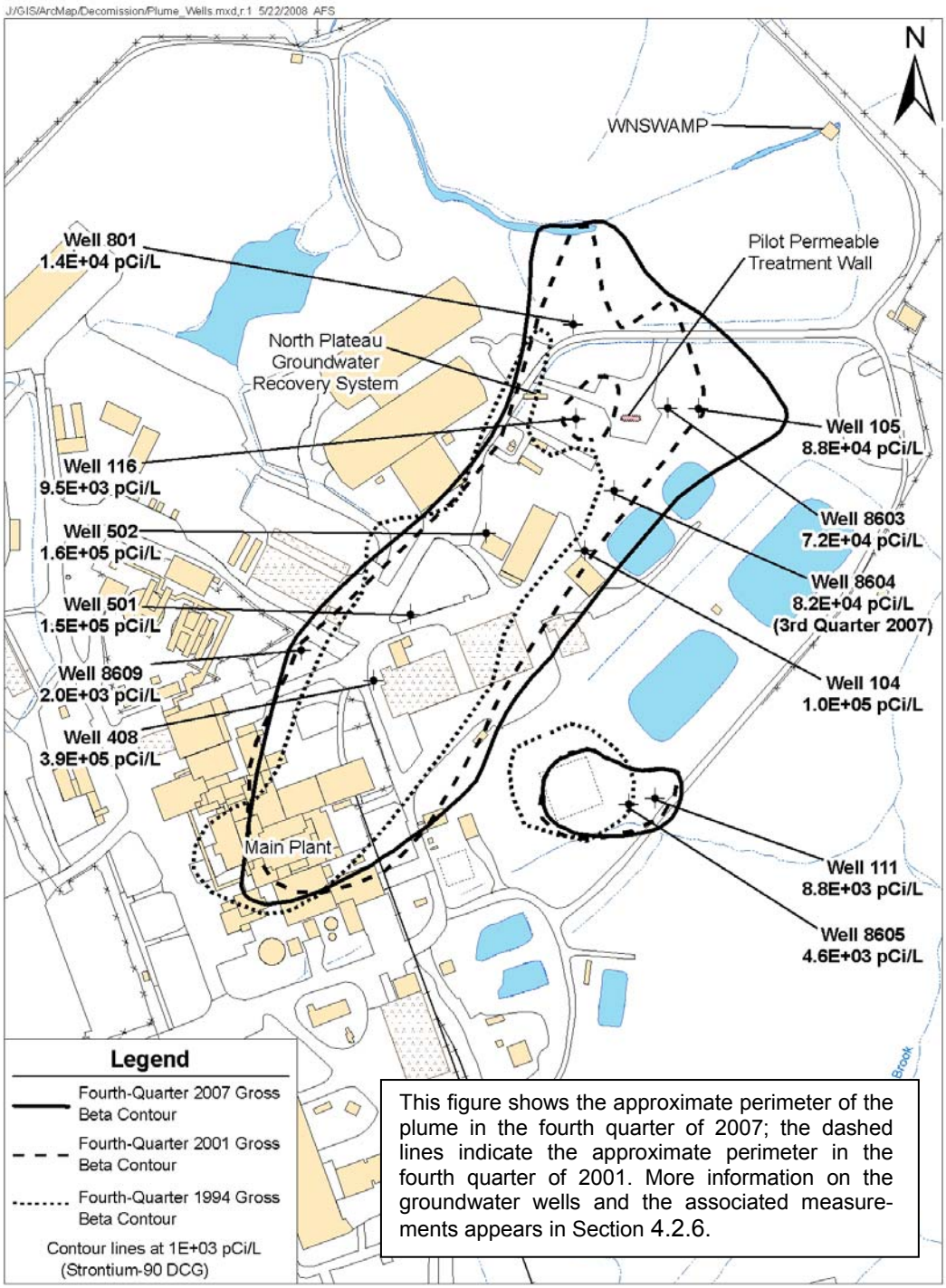

Figure 2-6. Sr-90 Groundwater Plume on the North Plateau

Mobile radionuclides such as H-3, Sr-90, and Tc-99 have migrated with groundwater along the northeast groundwater flow path in the north plateau. The Lagoon 1 design (to allow liquid to seep from the impoundment while retaining sediment and non-aqueous

contaminants inside the basin) allowed tritiated water, originally containing about 6,000 curies of tritium in leachate pumped from the SDA for treatment, to infiltrate areas of the north plateau groundwater in the mid-1970s (Smokowski 1977). These conditions were an unintended consequence of the lagoon design, and resulted in an extensive investigation by NFS, extending through the transfer of operational control to DOE in the early 1980s (Marchetti 1982).

The potential dose effects of the tritium are, however, small in comparison to the potential effects from the Sr-90 plume of present interest. Currently, the highest Sr-90 concentrations in groundwater exist at the closest Geoprobe™ sampling point downgradient from the original release point beneath the Off-Gas Cell in the Process Building. Less mobile radionuclides such as Cesium-137 are expected to have remained beneath the immediate source area due to the high cesium sorption capacity of the minerals in the sand and gravel.

An order-of-magnitude estimate of the radionuclides and amounts released by the acid leak, and the estimated remaining amount in 2011, are presented in Table 2-16. These estimates totaled approximately 200 curies in 1972 and will total approximately 77 curies in 2011.

Table 2-16. Released Radionuclide Activity Estimates for the North Plateau Plume[(1)]

Radionuclide	Plume Activity in 1972 (Ci)	Plume Activity in 2011 (Ci)
H-3	2.4E-03	2.6E-04
C-I4	1.3E-03	1.3E-03
Co-60	3.8E-05	2.3E-07
Sr-90	9.3E+01	3.6E+01
Tc-99	1.5E-02	1.5E-02
Cd-113m	4.1E-02	5.7E-03
Sb-I25	1.8E+00	1.1E-04
Sn-126	3.8E-04	3.8E-04
1-129	2.0E-06	2.0E-06
Cs-137	9.8E+01	4.0E+01
Eu-154	4.1E+00	1.9E-01
Ra-226	0.0E+00	1.2E-10
Ac-227	1.4E-08	6.2E-09
Ra-228	2.7E-13	5.7E-14
Th-229	6.1E-11	2.5E-07
Pa-231	2.7E-09	3.4E-09
Th-232	5.5E-14	5.5E-14

Table 2-16. Released Radionuclide Activity Estimates for the North Plateau Plume(1)

Radionuclide	Plume Activity in 1972 (Ci)	Plume Activity in 2011 (Ci)
U-232	4.8E-05	3.3E-05
U-233	6.9E-05	6.9E-05
U-234	4.0E-05	4.6E-05
U-235	8.9E-07	8.9E-07
Np-237	2.4E-04	2.5E-04
U-238	7.9E-06	7.9E-06
Pu-238	6.9E-02	5.1E-02
Pu-239	1.6E-02	1.6E-02
Pu-240	1.2E-02	1.3E-02
Pu-241	1.7E+00	2.5E-01
Am-241	6.6E-01	6.6E-01
Cm-243	4.2E-04	1.6E-04
Cm-244	3.3E-01	7.4E-02

NOTE: (1) From Westcott 1998.

In 1995, a pump and treat system was installed to slow the migration and lower the water table in the western lobe of the plume. A pilot-scale permeable treatment wall was installed in 1999 to provide some plume migration control for the eastern lobe of the plume. These facilities are described in Section 3.

In addition to the known acid spill affecting the north plateau, during NFS operations several incidents such as inadvertent transfers of higher-than-intended activity occurred in the interceptor basin system upstream of the lagoon system (Lewis 1967, Taylor 1967, Wischow 1967). Documented accounts of leakage and spills in the area (Lewis 1967, Carpenter and Hemann 1995) corroborate the generally elevated observed subsurface soil contamination in the area west of Lagoon 1 to the vicinity of the Process Building. Such localized subsurface soil contamination can be attributed to these unintended operational releases.

2.3.2 Old Sewage Plant Drainage

The old sewage treatment plant outfall drainage extends approximately 650 feet to the south of a culvert near the Old Warehouse location, flowing into the first culvert under the railroad tracks on the south plateau. In the 1960s and 1970s, the old sewage treatment plant experienced several contamination events, some of which were expressed as radioactivity increases in the treated effluent (DOE 1978). Figures 2-3 and 2-4 show where the drainage is located.

Actions were taken to find and repair the suspected sewage line leak, but when excavation of the line neared the south side of the Process Building, radiation levels from soil contamination hampered the project (Duckworth 1972b). Direct radiation levels of several mR/h were measured on containers of sludge removed from the sewage treatment plant for disposal in the 1980s.

A 1982 gamma radiation survey of the drainage channel showed levels three feet above the surface ranging from 110 to 500 μR/h on a section of the channel extending approximately 200 feet south of the sewer outfall (Marchetti 1982). The contaminated portion of the area was about 15 feet wide and 600 feet long, the northern 200 feet of which exhibited significant contamination in sediments represented by an 800 pCi/g Cs-137 result on the sample collected at that location, and up to 1 mR/hr near the surface of the drainage ditch. The sediment layer is estimated to be at least a foot thick.

In order to prevent further contaminant transport downstream, a new drainage channel was excavated to the west of the contaminated drain, and the spoil was placed over the old channel. At least three feet of soil covers the old drainage channel sediment. Some drainage near the old outfall exhibits residual surface contamination. (See Section 4.)

2.3.3 The Cesium Prong

The cesium prong is an airborne deposition plume resulting from a series of Process Building ventilation system air filter failures during licensed operations starting in March 1968, and culminating in a main ventilation system filter failure that occurred on September 4, 1968 (Urbon 1968a, Urbon 1968b). These airborne releases contaminated a portion of the West Valley site as shown in Figure 2-7. The primary contaminant is Cs-137.

A study that focused on the portion of the cesium prong outside of the Center boundary showed that contamination concentrations decrease with depth. Seventy-five percent of the activity was determined to be in the upper two inches of soil, 20 percent in the layer between two inches deep and four inches deep, and five percent in the four to six inch layer (Luckett 1995). Therefore, 95 percent of the activity in the affected area outside of the Center lies in the upper four inches of soil. It is probable that similar conditions exist on the Center property closer to the source of the contamination, but data from this area are not available. Surface soil within the project premises would be characterized during Phase 1 of the proposed decommissioning as described in Section 9.

2.3.4 Summary of Spills During NFS Operations

Table 2-17 provides a summary listing of major spills that impacted the environment during the period when NFS was operating the reprocessing plant.

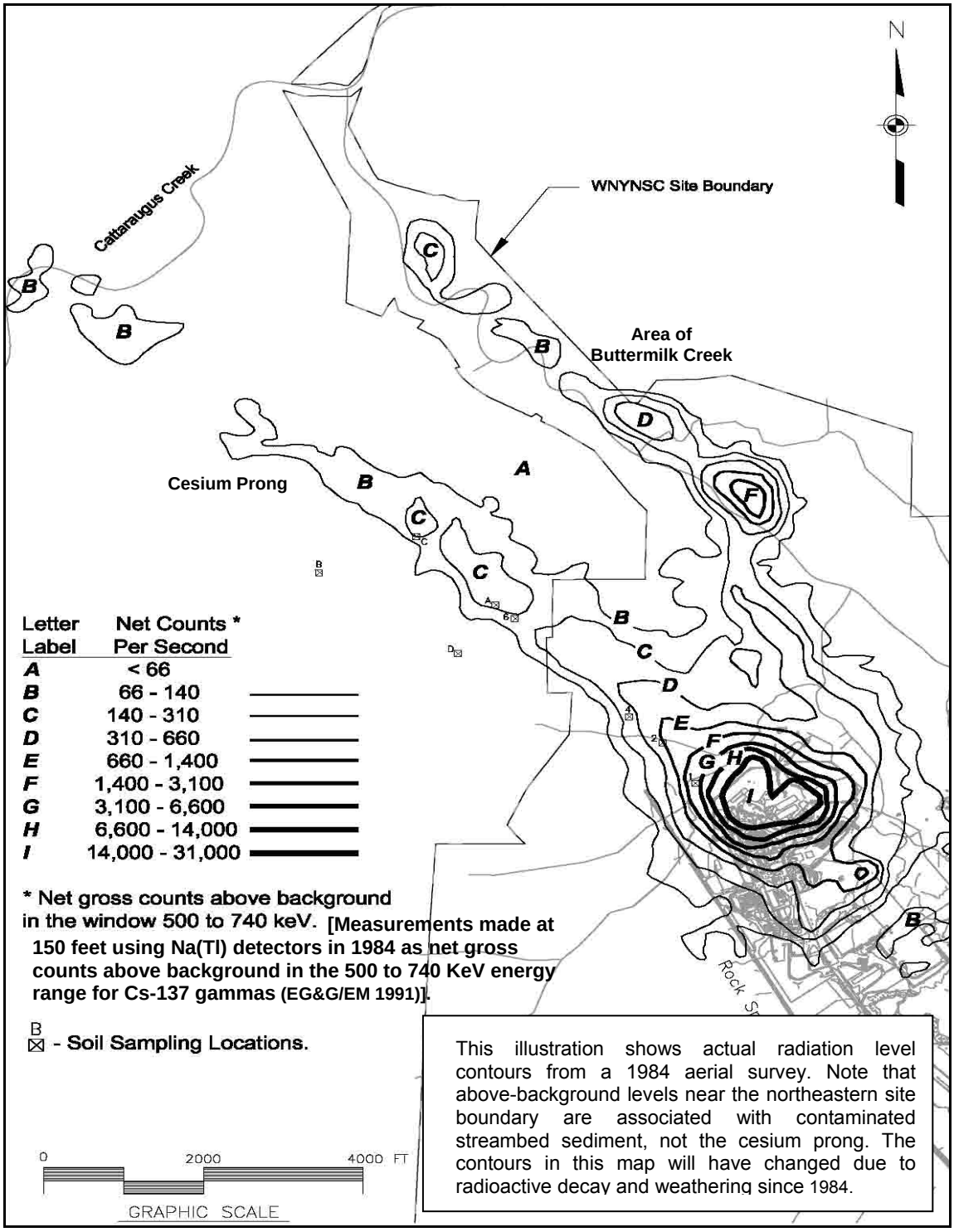

Figure 2-7. 1984 Aerial Radiation Survey Isopleths of the WVDP and Surrounding Area

Table 2-17. Principal Radionuclides in Major Spills Occurring During NFS Operations

Release Event and Origin Location	Principal Radionuclides			
	Type	Form	Activity or Concentration	Documentation Notes
1968 radioactive acid spill that produced the major contribution to the north plateau groundwater plume. WMA 1: from southwest corner of the Process Building.	Sr-90 (predominant mobile contaminant)	Liquid to groundwater, soil	0.705 µCi/L (maximum)[1] [Original spill volume estimated at 200 gallons, ~93 Ci Sr-90][2]	Line 7P-240-1-C failed inside the OGA in January 1968, and leakage drained from the OGA through the ARPR to the underlying soil.[3], [4]
Wastewater Line to Tank 7D-13 contribution to north plateau groundwater plume. WMA 1: near the south side of the Process Building.	Radionuclide mix typical of feed and waste contamination	Liquid to groundwater, soil	Unknown amount and activity At levels ~ 5E-03 µCi /mL, the interceptor release limit.	Line 7P-160-2-C leaked an unknown amount of radioactive wastewater in February 1967 during transfer from Tank 7D-13.[5]
Contaminated groundwater noted during new interceptor construction. WMA 2: south of Old Interceptor at site of New Interceptors.	Radionuclide mix typical of feed and waste contamination	Liquid to groundwater, soil	Unknown amount and quantity; evidently not sufficient to cause worker dose constraints.	Evidence of earlier leakage, but not a spill reported by NFS[6]
Resin Pit spills during Fuel Receiving and Storage spent nuclear fuel pool water filtration system maintenance. WMA 1: east of FRS.	Cs-137, Sr-90	Solid and liquid to groundwater, soil	Unknown amount and quantity. Some effect on groundwater noted.	Incidental small spills of resin and fluid during maintenance. Information from subsurface probing investigation [3]
Tank 8D-2 ventilation condensate line (operates under vacuum) was noted to be breached. WMA 3: one leak noted between HLW tanks and southwest side of Process Building (in WMA 1) at ARPR, other leaks thought to exist in WMA 3.	Cs-137, H-3	Liquid to groundwater, soil	No evidence of out-leakage, but possibility exists of localized groundwater effects.	Line 8P-46-6-A5 failed integrity test. NFS evaluation in 1977.[4]
A line from the in-cell LLW Evaporator to acid recovery failed in-cell during waste transfer to Tank 8D-2. WMA 1: ARPR in southwest corner of Process Building.	Fission products and transuranics from spent fuel	Liquid to groundwater, soil	Leakage resulted in 555 gallons of liquid waste sent to the interceptor (sufficient to read >~ 100 mR/hr at the interceptor), and requiring pumpout back to the Process Building for treatment.	Line 7P-170-2A failed in-cell on 2/14/67. Reported by NFS [7], [5] Leakage did not result in any known release to the environment.

Table 2-17. Principal Radionuclides in Major Spills Occurring During NFS Operations

Release Event and Origin Location	Principal Radionuclides			
	Type	Form	Activity or Concentration	Documentation Notes
Sanitary sewer line leak near Process Building allowed contaminated groundwater to affect Sewage Treatment Plant. WMA 1: in-leakage near southwest side of Process Building.	Cs-137, Sr-90, I-129	Liquid to groundwater, soil	Estimated 0.052 Ci Sr-90 released: sewage treatment outfall area soil contaminated to 1 mR/h.	Sewage Treatment Plant and outfall drainage were contaminated to low levels, effluent concentrations subsiding after leak was repaired. Reported by NFS (4),(8)
Overflow of Lagoons 4 and 5: treated water released to local soil and groundwater. WMA 2: northeast of the O2 Building.	Cs-137, Sr-90	Liquid to groundwater, soil	Unknown amount and activity: probably close to free release level of < 3E-7 µCi /mL.	Temporary loss of Lagoon 3 capacity allowed overflow of releasable treated effluent to occur at an unplanned location. Reported by NFS (9)
Leakage from waste containers or fuel casks contaminated asphalt "Old Hardstand" north of the Process Building. WMA 5: footprint located west of LSA 3 and LSA 4.	Fission products and transuranics from spent fuel	Liquid to groundwater, soil	Unknown amount and activity of leaks: maximum surface reading was 100 mR/hr on localized surfaces. Material was removed and placed in Lagoon 1 in 1984. Approximately 1,700 cubic yards of removed material, <10,000 dpm/g beta-gamma, <2,000 dpm alpha. (11)	Leakage from waste transport trailers parked on the hardstand contaminated the asphalt surface. Runoff contaminated the adjacent soil and drainage ditch. Noted, but not detailed during 1982 environmental characterization.(8) Significant contamination was noted in 1983.(10)
Cesium prong created by particulate deposition following 1968 dissolver off-gas HEPA filter failure. WMA 1, 3, 4, 5, 10: general deposits to the north-northwest of the Process Building. Detectable deposits extend several miles (outside the scope of this plan).	Cs-137	Airborne particulate to exposed surfaces, soil	Approximately 0.33 Ci particulate gross beta radioactivity released. Offsite- 44 pCi/g localized; 21pCi/g averaged over 2,500 m^2 (26,900 ft^2). Offsite data from Luckett.(13)	Several events contributed to the deposits. A DOG filter failure in March, and a main plant filter failure in September appear to have been the main sources of the observed depositions. Reported by NFS (12),(8)

LEGEND: ARPR = Acid Recovery Pump Room, DOG = dissolver off-gas, FRS = Fuel Receiving and Storage, OGA = Off-Gas Aisle,

NOTES:
(1) From Hemann and Steiner 1999.
(2) From Westcott 1998.
(3) From Carpenter and Hemann 1995.
(4) From Duckworth 1977.
(5) From Lewis 1967.
(6) From Taylor 1967.
(7) From Wischow 1967.
(8) From Marchetti 1982.
(9) From Taylor 1972.
(10) From WVNSCO 1983b.
(11) From WVNSCO 1995.
(12) From Urbon 1968a.
(13) From Luckett 1995.

2.3.5 WVDP Spills

Incidents occurring outside facility containment, and having the potential for residual environmental contamination are detailed as spills or unplanned releases. Spills that were confined inside facilities are not discussed because such spills did not lead to releases into the environment. For example, although the discovery of contaminant migration within the NDA in 1983 required action, the effects were contained within the facility (WVNSCO 1985a). Any residual contamination has been characterized along with the facility and is included in the respective facility radiological inventory.

Based on a review of event reports for the WVDP (1985 through 2008), one 1985 spill and one 1987 spill involving release of radioactive water were documented by unusual occurrence reports as identified below. These events are mentioned because they were considered to be serious enough to be reportable under DOE requirements. They are listed below in Table 2-18, along with three other unplanned releases of less significance.

Table 2-18. WVDP Spills Impacting Environmental Media (1982 – 2007)

Release Event and Origin Location	Principal Radionuclides			
	Type	Form	Activity or Concentration	Documentation Notes
1985 spill of radioactive water at the Waste Tank Farm. WMA3: from valve pit northwest of 8D-2, between 8D-2 and 8D-1.	Cs-137, H-3	Liquid to groundwater, soil	~400 gal at 4.6 E-02 µCi/mL gross beta, ~4E-03 µCi/mL H-3.	Spill of radioactive water March 1985 at the Waste Tank Farm from a condensate line running from Tank 8D-1 to Tank 8D-2 due to failure of flanged valve bolts. Some water (4.6E-02 µCi/mL gross beta) flowed out of valve pit. Contaminated soil was removed. Documented by Unusual Occurrence Report(1)
In 1987, condensate from a ventilation unit spilled on top of Tank 8D-2. WMA3: upon disassembling the unit, condensate leaked out onto the gravel surface.	Radionuclide mix typical of feed and waste contamination	Liquid to groundwater, soil	Less than 10 gallons spilled, water probably ~2E-5 µCi/mL gross beta.	A portable ventilation unit was disassembled after operations on March 2, 1987 near Tank 8D-2. Condensate from the housing spilled onto the gravel surface of Tank 8D-2 top. No soil or water contamination noted in samples collected.(2)
In 1987, the Neutralizer Pit overflowed during transfer of liquid waste to the interceptor. WMA2: the overflow went to the ground near the interceptors and Lagoon 1.	Radionuclide mix typical of feed and waste contamination	Liquid to groundwater, soil	Approximately 5,000 gallons of waste water was spilled, ~5E-05 uCi/mL gross beta.	The neutralizer pit overflowed on February 25, 1987 due to a malfunctioning drain valve. The overflow went to the ground near the interceptors and Lagoon 1. The flow was stopped when noted by an operator. Documented by Unusual Occurrence Report (3)

Table 2-18. WVDP Spills Impacting Environmental Media (1982 – 2007)

Release Event and Origin Location	Principal Radionuclides			
	Type	Form	Activity or Concentration	Documentation Notes
In 1987, water from a 55-gallon drum containing spent resin leaked. WMA 5: water spilled on the ground before or during transfer of the drum to a processing station.	Radionuclide mix typical of feed and waste contamination	Liquid to soil, potentially to groundwater	<15 gallons likely spilled, wetted soil was <100 dpm/g gross beta.	Drum was being transferred from the Lag Storage Building hardstand to a waste solidification area in the Process Building when leakage was noted.(4)
In 2001, release of airborne particulate from Process Building stack in droplet form. WMA1 3: fallout was localized due to droplet size.	Radionuclide mix typical of Process Building stack particulate (Cs-137 & Sr-90)	Airborne particulate to exposed surfaces and soil	4.8E-04 µCi gross beta.	Over a period of two months, September-October 2001, excess moisture appears to have become entrained in the Main Plant Ventilation system, and was emitted from the stack as droplets containing radioactive particulates. The fallout was confined to the area several hundred feet from the Process Building. Radiological surveys were conducted and accessible above-background spots were decontaminated. Total releases were less than 0.5% of the administrative release limits.(5),(6)
In 2003, breach discovered in wastewater drain line allowing contaminated laundry water to leak into adjacent soils. WMA 1: during wastewater line inspection a breach was discovered, but no specific event was identified which would have caused the breach. The line was repaired.	Radionuclide mix typical of feed and waste contamination	Liquid to groundwater, soil	Amount unknown, water typically ~2E-07 µCi/mL gross beta.	Discovery of hole in riser to drain line 15-ww-569 from Laundry to Interceptors in October 2003: date of breach unknown. A sample of subsurface soil near the breach showed 3,300 pCi/g Cs-137 and 87 pCi/g Am-241 as shown in Table 4-12 in Section 4; the breached line may not have caused all of this contamination.(7), (8)

NOTES: (1) From WVNSCO 1985b. (5) From Nagel 2001.
(2) From WVNSCO 1987a. (6) From Nagel 2002.
(3) From WVNSCO 1987b. (7) From Maloney 2003.
(4) From WVNSCO 1987c. (8) From WVNSCO 2006.

2.4 Prior Onsite Burials

There are two prior burial sites within the NRC licensed property that contain radioactive material: Lagoon 1 and the NDA. A drainage area adjacent to the NDA is believed to contain contaminated soil below contouring fill. The location of these burial sites is shown in Figures 2-3 and 2-4.

2.4.1 Lagoon 1

In order to prevent further water infiltration, and to isolate contaminated fill removed in the 1980s from a hardstand north of the Process Building, radioactive wastes were stabilized and capped within Lagoon 1, one of five lagoons associated with the Low-Level Waste Treatment Facility. Lagoon 1 was an unlined basin in the system for treating liquid low-level waste. It was removed from service in 1984 because it was determined during initial WVDP environmental assessments to be a major source of tritium in nearby groundwater (Marchetti 1982).

After Lagoon 1 was taken out of service, liquid and sediment from it were transferred to Lagoon 2. Lagoon 1 was then filled with approximately 46,000 cubic feet of radioactively-contaminated debris removed during decontamination of the old/new hardstand area. Among this debris were asphalt, trees, stumps, roots, and weeds (WVNSCO 1995).

After being filled with debris, Lagoon 1 was then capped with clay, covered with topsoil, and re-vegetated. Table 2-19 provides an order-of-magnitude estimate for the residual radioactivity in Lagoon 1. Section 7 describes proposed decommissioning activities for Lagoon 1, which would include removal and offsite disposal of the buried waste.

Table 2-19. Estimated Residual Radioactivity in Lagoon 1(1)

Radionuclide	Activity (Ci)	Radionuclide	Activity (Ci)
C-14	0.053	U-234	0.012
Sr-90	19	U-235	0.0027
Tc-99	0.20	Np-237	0.0031
Cd-113m	0.065	U-238	0.025
Sb-125	0.0038	Pu-238	6.5
I-129	0.029	Pu-239	3.8
Cs-137	548	Pu-241	156
Eu-154	1.7	Am-241	11
U-233	0.22	Cm-244	0.22

NOTE: (1) From WVNSCO 1995, decay-corrected to January 2011. Most of the activity is estimated to be in the remaining sediment.

2.4.2 The NRC-Licensed Disposal Area

As explained in Section 3, the NDA is a 400-feet wide and 600-feet long shallow-land radioactive waste disposal site southeast of the Process Building. It includes three distinct areas: (1) the NFS waste disposal area, (2) the WVDP disposal trenches and caissons, and (3) the areas occupied by an interceptor trench and subsurface barrier wall (Figure 2-8).

Prior to 1972, the NDA was used exclusively for the disposal of highly radioactive solid wastes generated by the reprocessing plant. Wastes routinely buried in the area included spent fuel hulls, fuel assembly hardware, failed process vessels and large equipment,

degraded process solvent absorbed on suitable solid medium, and miscellaneous packaged trash including laboratory wastes, small equipment, ventilation filters, and other process-related debris.

Also buried in the NDA are 42 ruptured spent fuel elements from the Hanford N-Reactor. According to records, the total radioactive waste volume in the NDA is approximately 361,000 cubic feet. The estimated total activity present in 2000 was approximately 299,000 curies (Wild 2000). Table 2-20 is an abridged summary of the wastes buried in the NDA. Table 2-21 is a summary of radioactivity in wastes buried in the NDA, corrected to the estimated radioactivity present in 2011.

The swale between the SDA and the NDA has been historically contaminated, presumably from spills during waste burial operations by NFS, and after SDA closure, during leachate control activities (DOE 1978). During the NDA tank removal and subsurface control period in the 1980s and 1990s, the swale area was re-contoured to prevent erosion. An unknown amount of low-level radioactive contamination remains in that area, evidenced by continuing elevated radioactive contaminant indicators in surface water immediately downstream (WVNSCO and URS 2007). The swale area averages approximately 30 feet wide running 300 feet north along the drainage from the old NDA hardstand. Based upon observations during radiation surveys in 1982, the contamination appeared to have permeated porous fill in the swale channel. Gamma readings in that area were five to seven times background, not inconsistent with observed downstream gross beta contamination (Marchetti 1982). Surface soil contamination is still occasionally noted in that area (WVNSCO 1986, WVNSCO 2007b).

Table 2-20 Summary of Wastes in the NRC-licensed Disposal Area(1)

NDA Location	General Waste Types (typical)	Volume (ft^3)	Estimated 2011 Activity (Ci)
NFS Deep Holes	Air filters, pumps, pipe, scrap, hulls, resin, solvent, fuel casing, shear ram, concrete, wood.	65,145	169,161
NFS Special Holes	Air filters, pumps, pipe, scrap, birdcages, resin, solvent, dissolver, jumpers, saw, shield, cask, railcar, LLWT sludge, trash.	97,298	58,914
WVDP Trenches	Air filters, metal tanks, scrap, resin, LLWT sludge, trash, concrete, wood, asphalt, glove box, snow blower.	197,656	926
WVDP Caissons	General waste, LLWT sludge.	823	0.15
	Disposal Totals	360,922	229,000

NOTE: (1) Based on the estimates in Wild 2000, decay corrected to 2011. Activity in each location estimated by proportion of overall 2000 activity.

Table 2-21. Estimated Radioactivity in the NDA[1]

Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)
Am-241	2,000	Np-237	0.17	Tc-99	10
C-14	520	Pu-238	350	U-233	11
Co-60	7,000	Pu-239	580	U-234	0.59
Cs-137	29,000	Pu-240	400	U-235	0.12
H-3	35	Pu-241	9,100	U-238	1.5
I-129	0.022	Ra-226	<0.01	-	-
Ni-63	110,000	Sr-90	22,000	-	-

NOTE: (1) From Wild 2000, radionuclide totals corrected for decay and in-growth to 2011 and rounded to two significant figures.

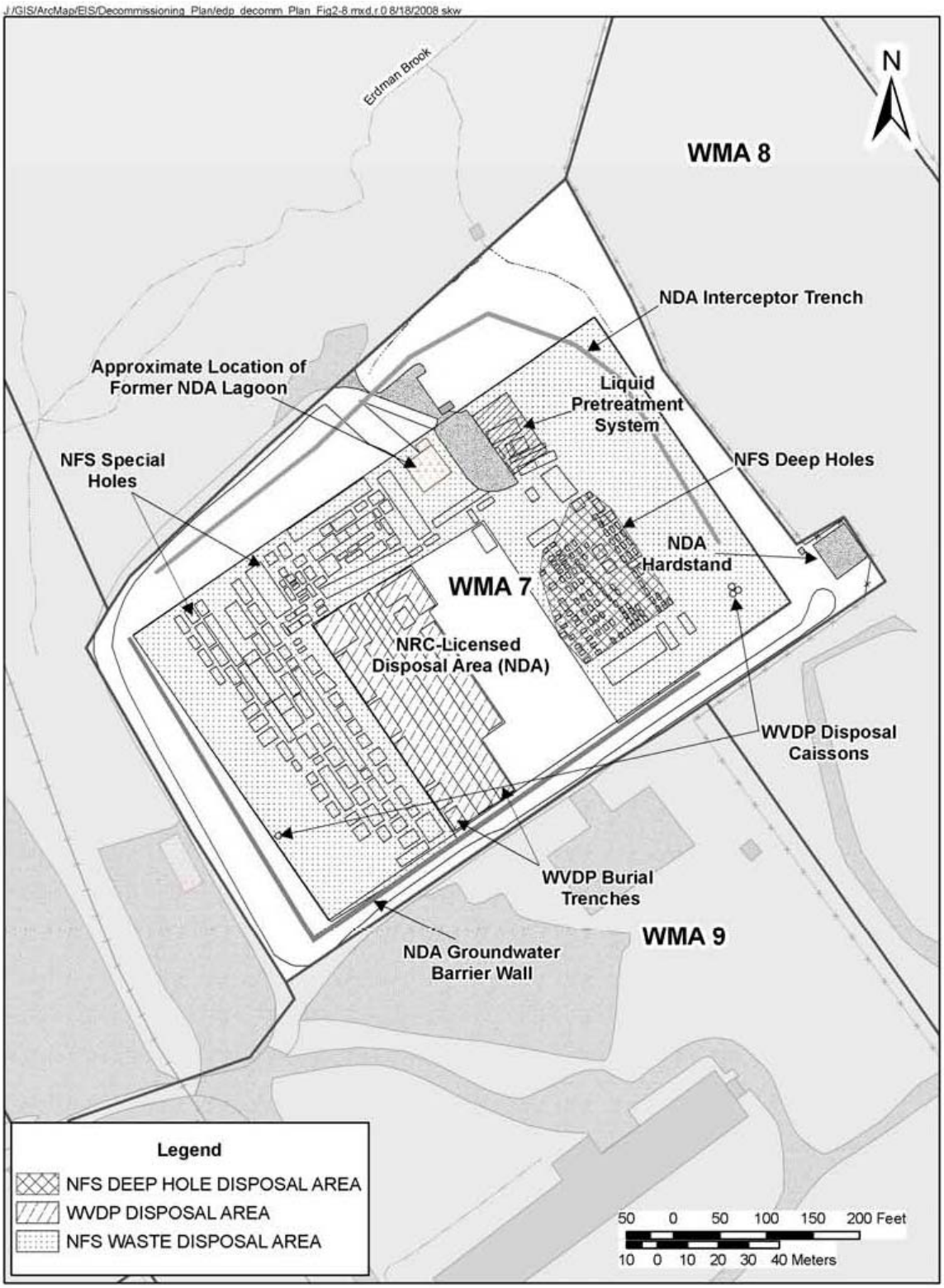

Figure 2-8. NDA Disposal Area Burials

2.4.3 Other Burial Locations

Two other areas on the Center contain buried radioactive material, although neither is within the scope of this plan[5]. One, the SDA, is not on the project premises. The other, the Construction and Demolition Debris Landfill in WMA 4, is briefly described here for completeness because it is located within the project premises.

Construction and Demolition Debris Landfill

The Construction and Demolition Debris Landfill in WMA 4 is located approximately 1,000 feet northeast of the Process Building. This landfill, the only facility within this WMA, covers approximately 1.5 acres in the southern part of the area. Nonradioactive waste material was typically placed in the landfill on existing grade in layers three to five feet thick, covered with soil, and compacted with bulldozers or trucks. The landfill is estimated to contain a total volume of 425,000 cubic feet of waste material and soil. It was initially used by Bechtel Engineering from 1963 to 1965 to dispose of nonradioactive waste generated during construction of the Process Building (WVNSCO1996).

NFS then used this landfill from 1965 to 1981 to dispose of nonradioactive construction, office, and facility generated debris, including ash from the NFS incinerator. The landfill was used from 1982 to 1984 to dispose of nonradioactive waste generated at the WVDP.

Disposal operations at the landfill were terminated in December 1984 and the DOE closed it in accordance with applicable New York State regulations. The final cover on the landfill was graded and grass planted to prevent erosion. In October 1986, the NYSDEC approved and certified the closure of the landfill (WVNSCO 1996).

Because this landfill is located in the path of the north plateau groundwater plume, radioactively contaminated groundwater in the plume is assumed to have come in contact with some of the waste buried in the landfill. Portions of the buried waste are therefore expected to be radioactive.

2.5 References

Federal Statutes

Atomic Energy Act of 1954

Energy Reorganization Act of 1974

West Valley Demonstration Project Act of 1980

Code of Federal Regulations

10 CFR 835, *Occupational Radiation Protection*.

[5] The condition of the old Sewage Plant drainage described in Section 2.3.2 could also considered to be buried radioactivity since the contaminated sediment is covered with soil.

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