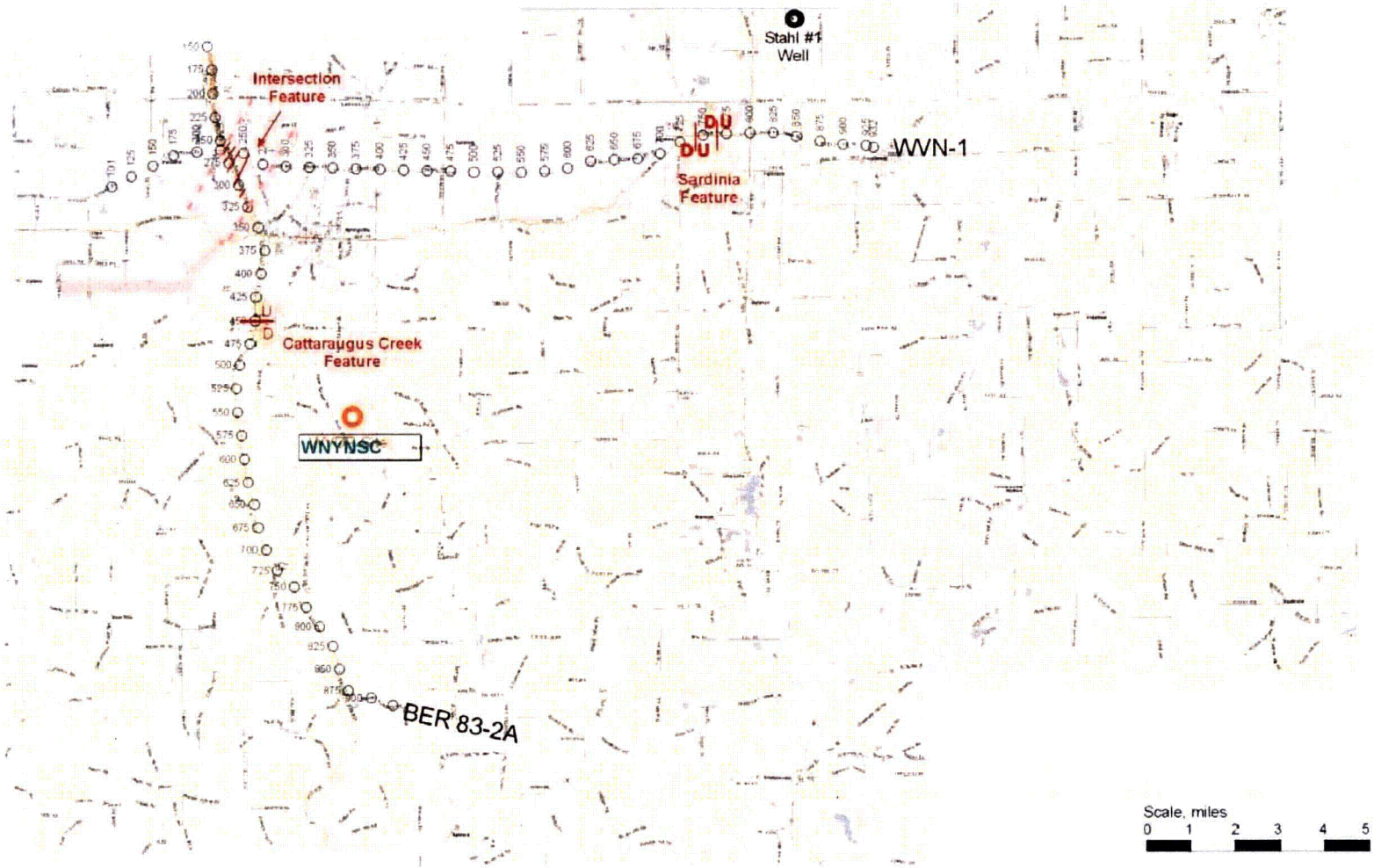

Figure 3-57. Location of Seismic Lines WVN1 and BER 83-2A

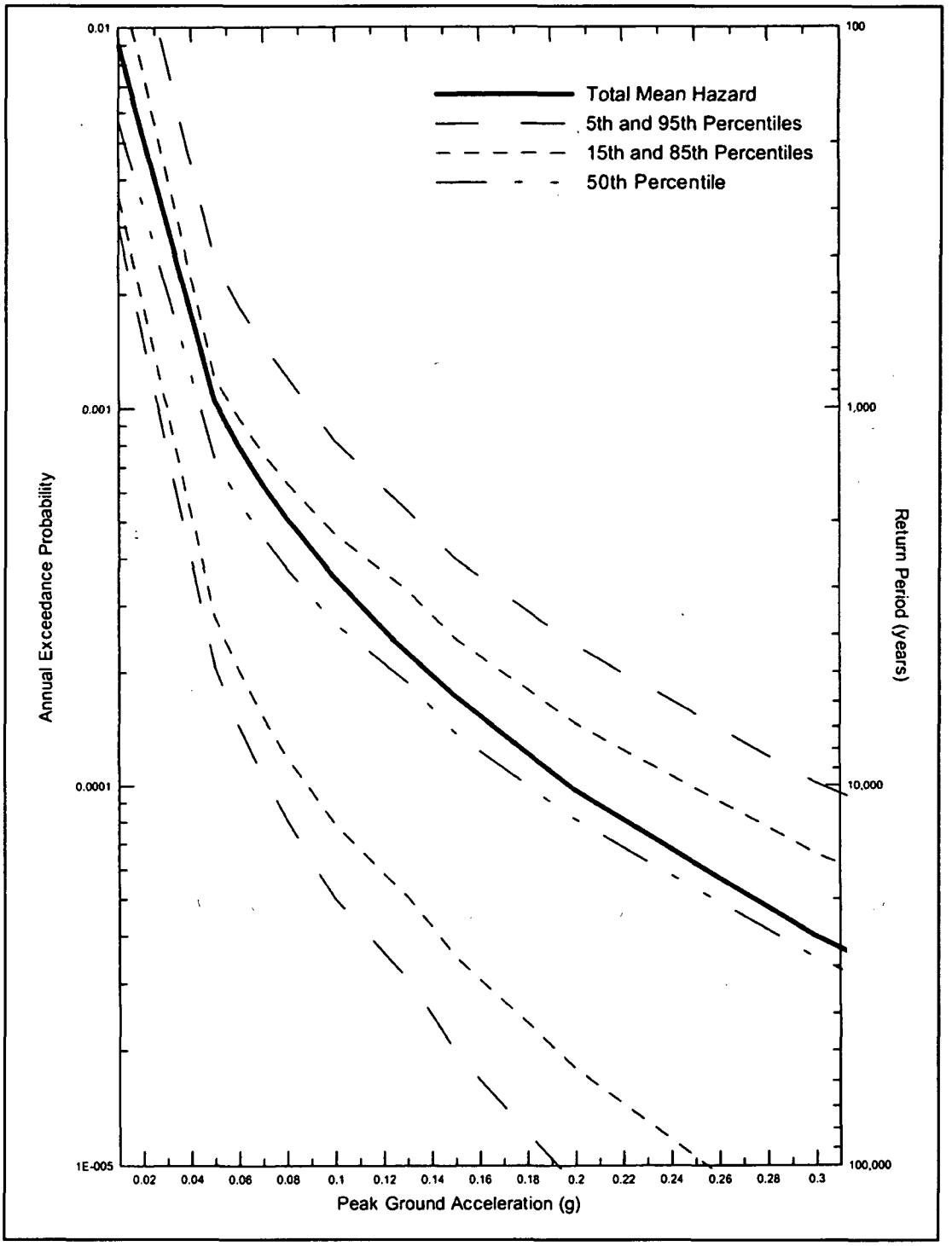

Figure 3-58. Seismic Hazard Curves for Peak Horizontal Acceleration

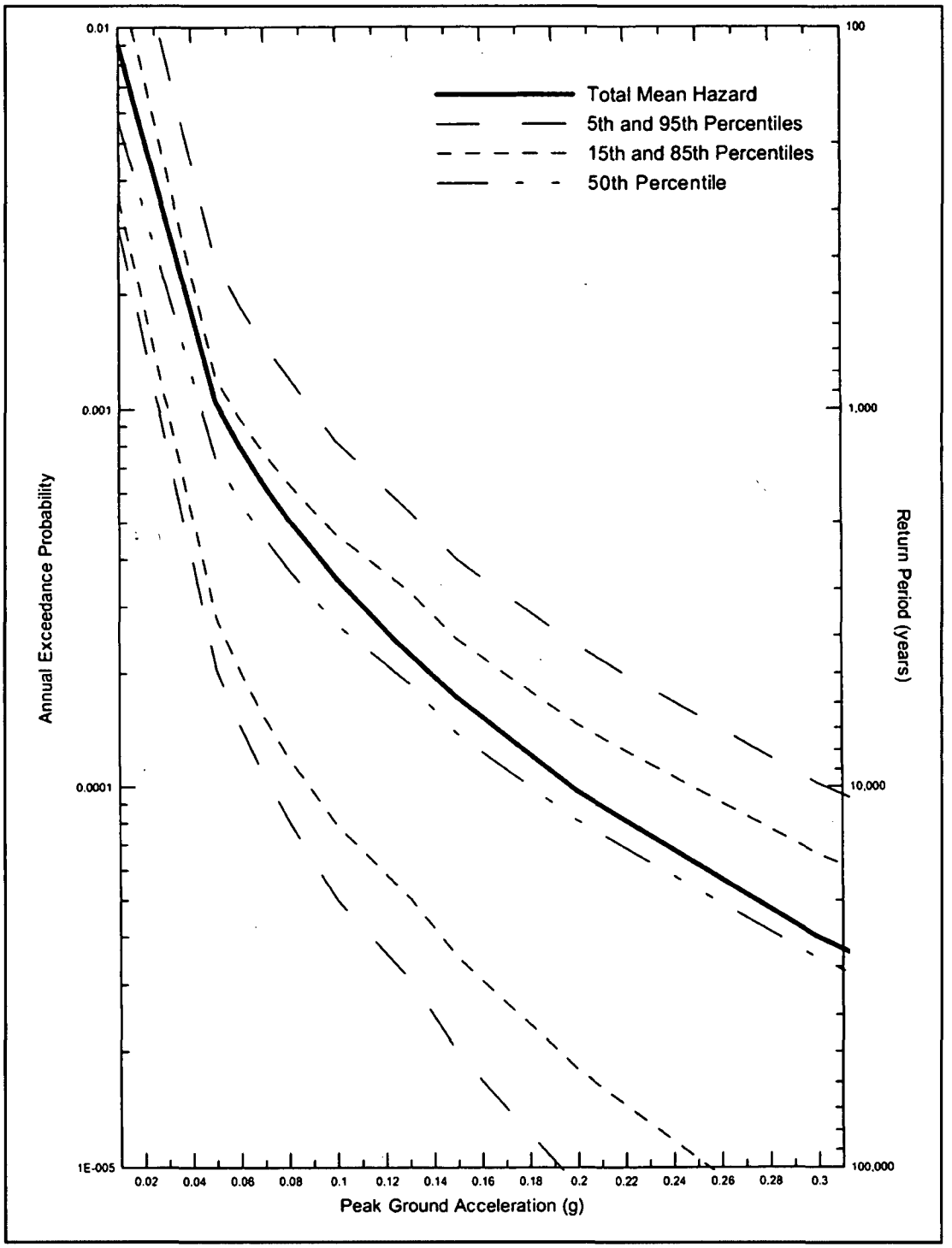

Figure 3-59. Seismic Hazard Curves for 1.0 Second Horizontal Spectral Acceleration

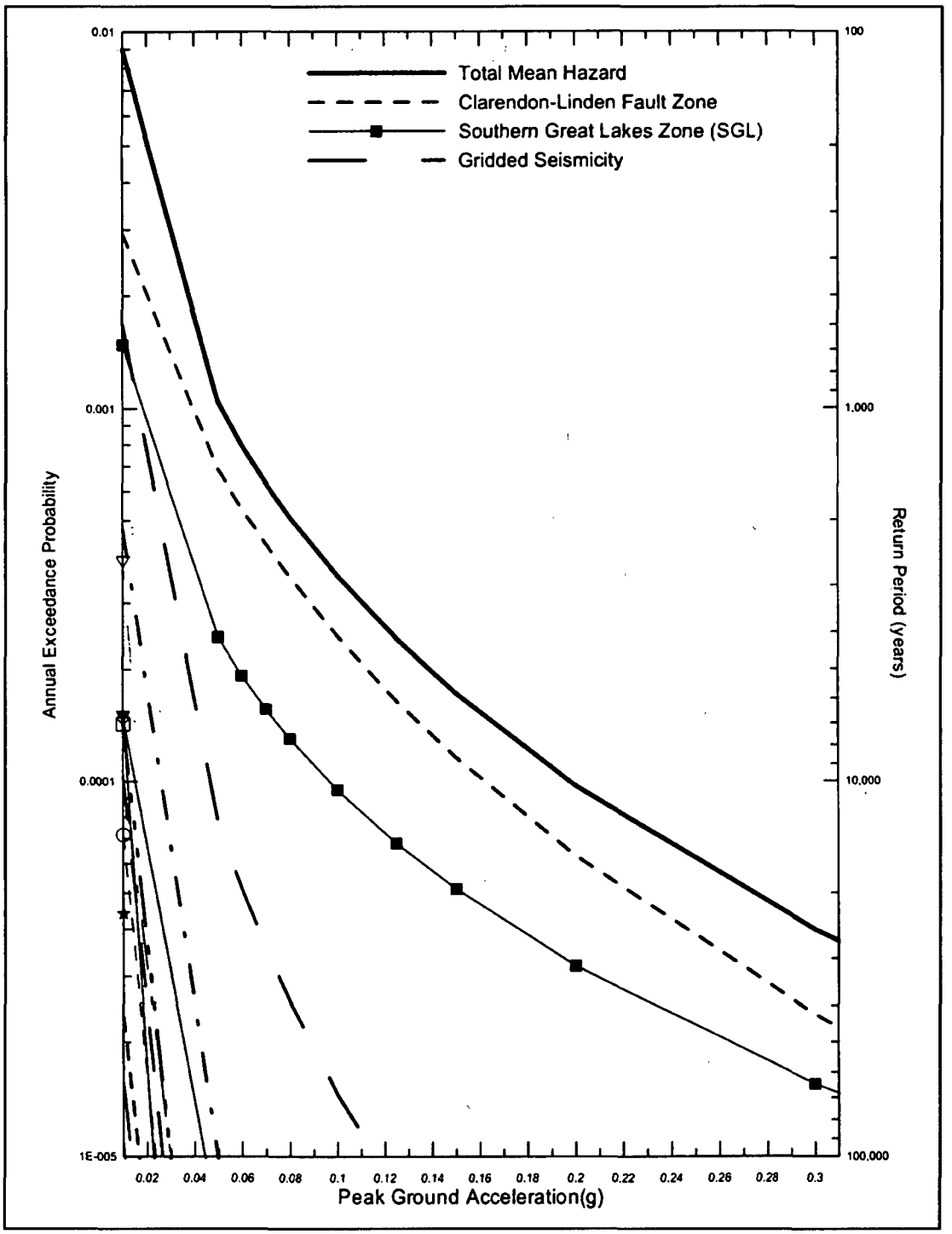

Figure 3-60. Seismic Source Contributions to Mean Peak Horizontal Acceleration Hazard

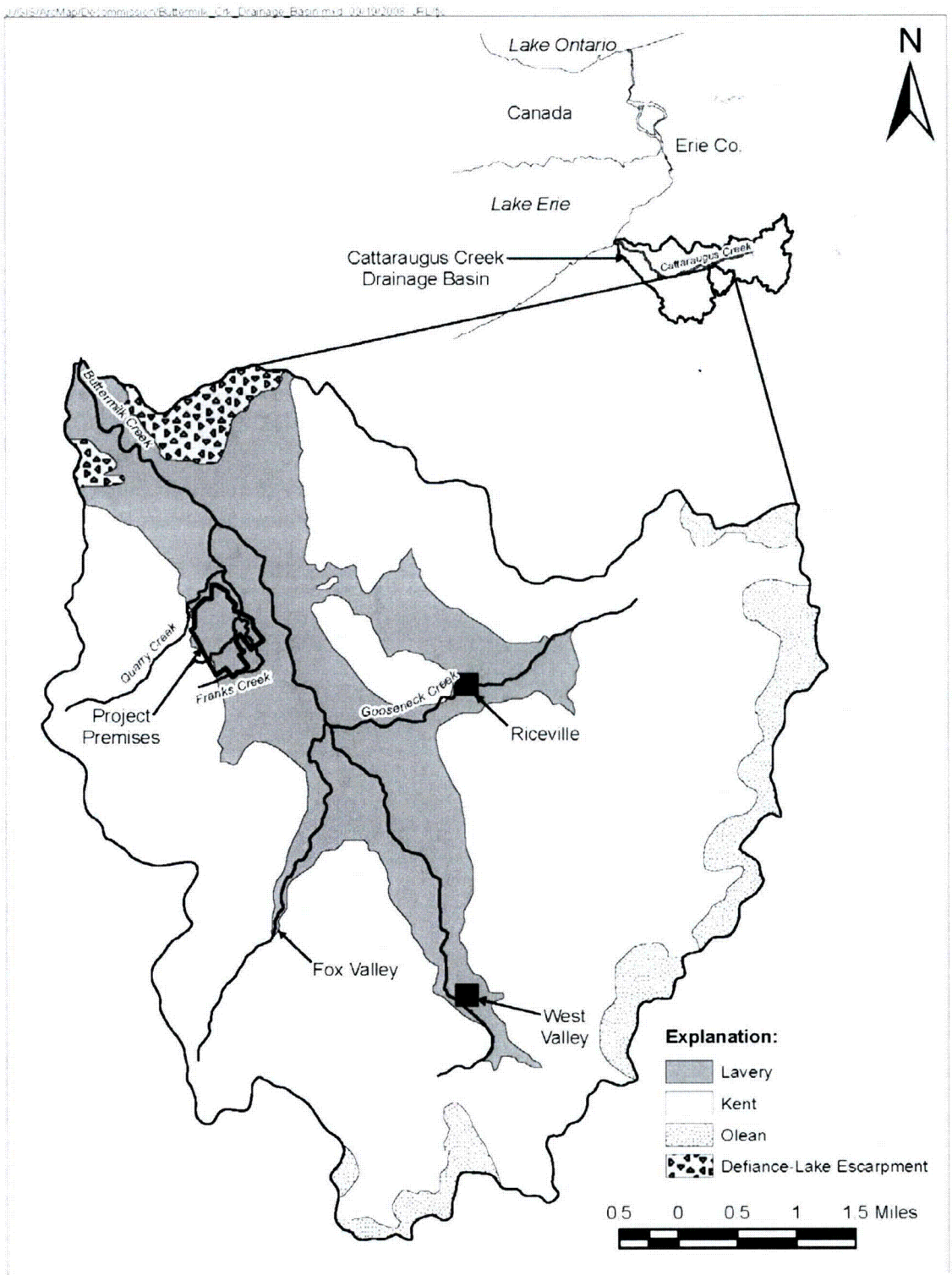

Figure 3-61. Buttermilk Creek Drainage Basin

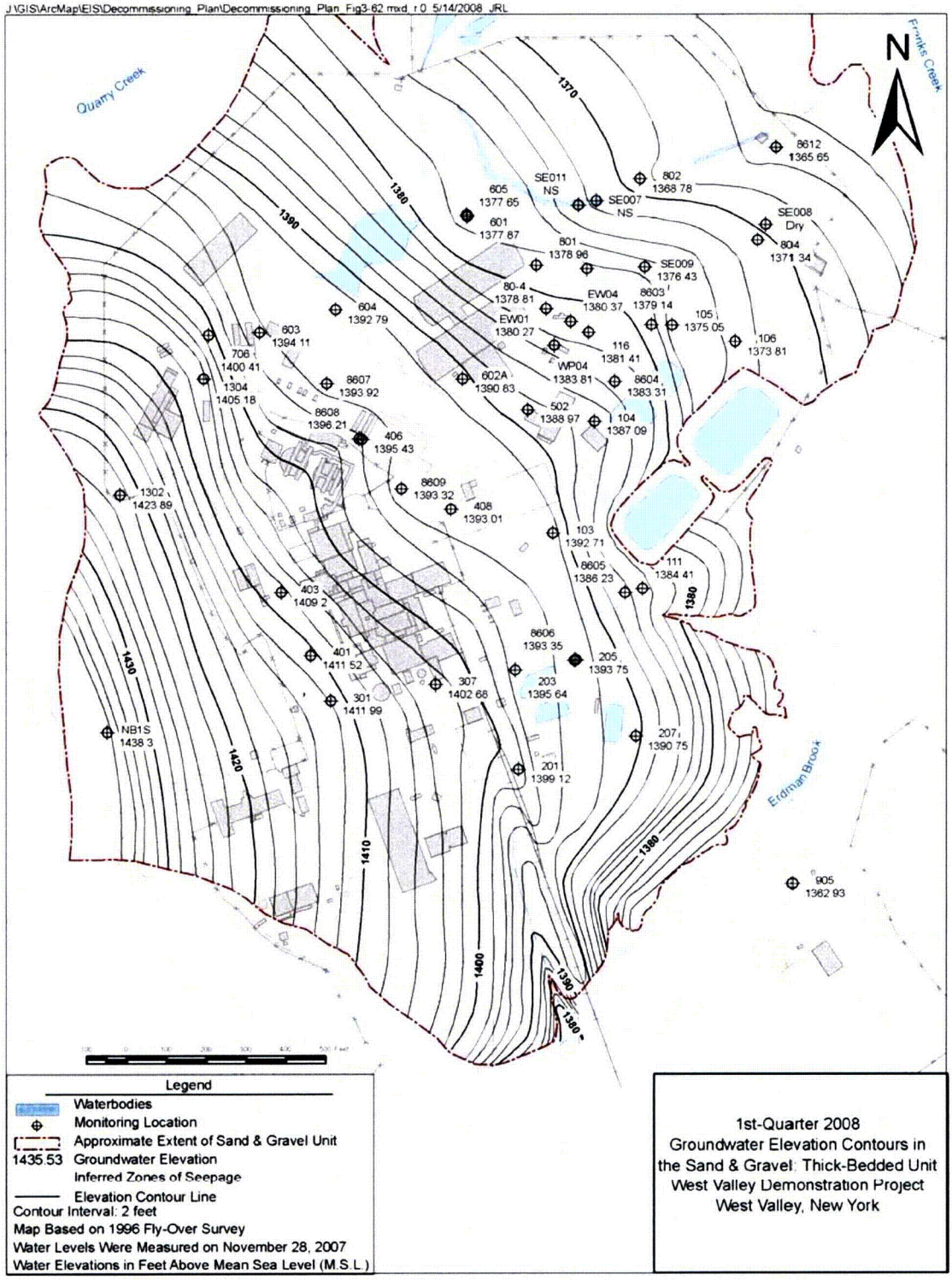

Figure 3-62. Groundwater Elevation Contours of the Sand and Gravel Unit, First Quarter 2008

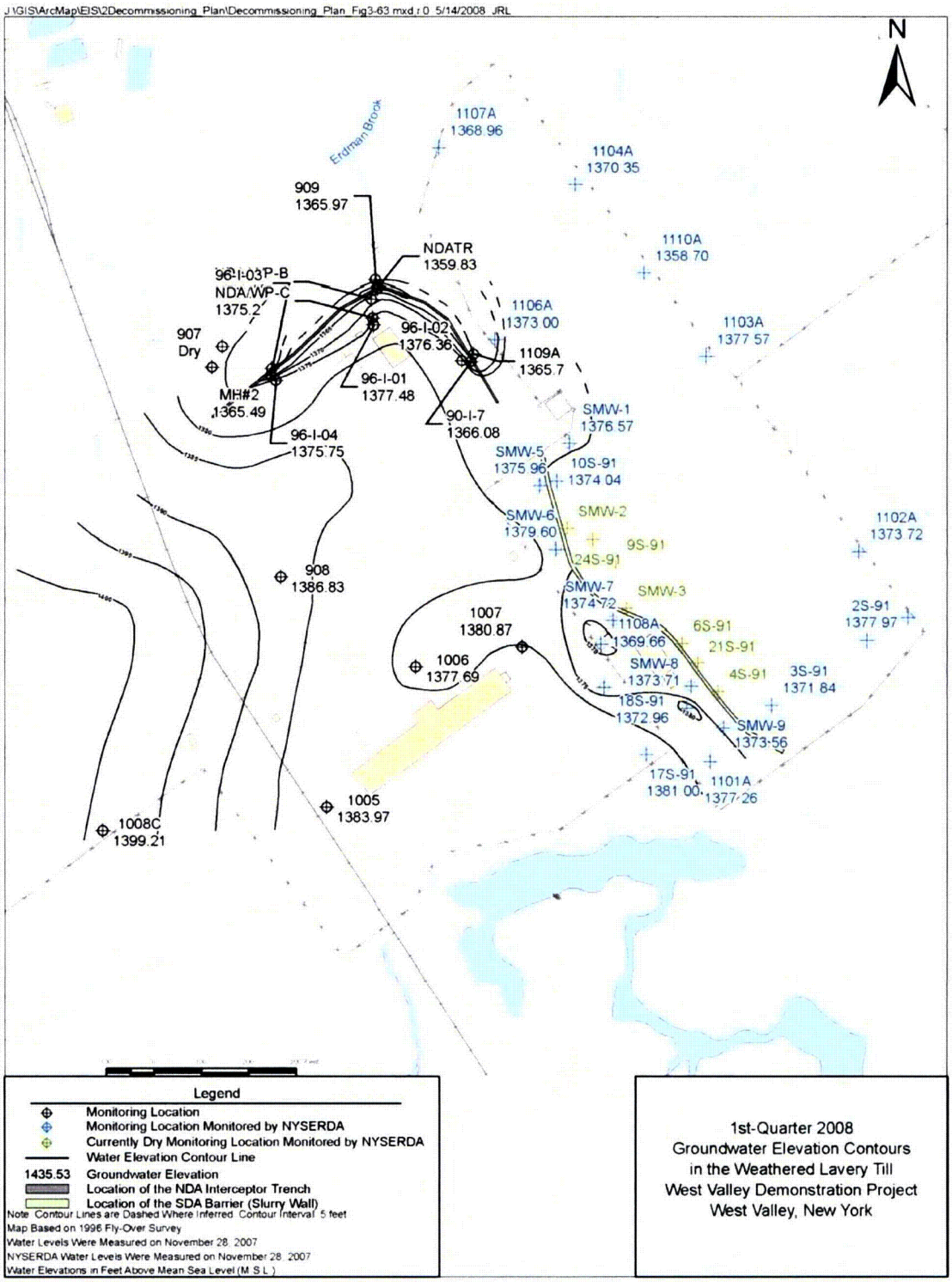

Figure 3-63. Groundwater Elevation Contours of the Weathered Lavery Till, First Quarter 2008

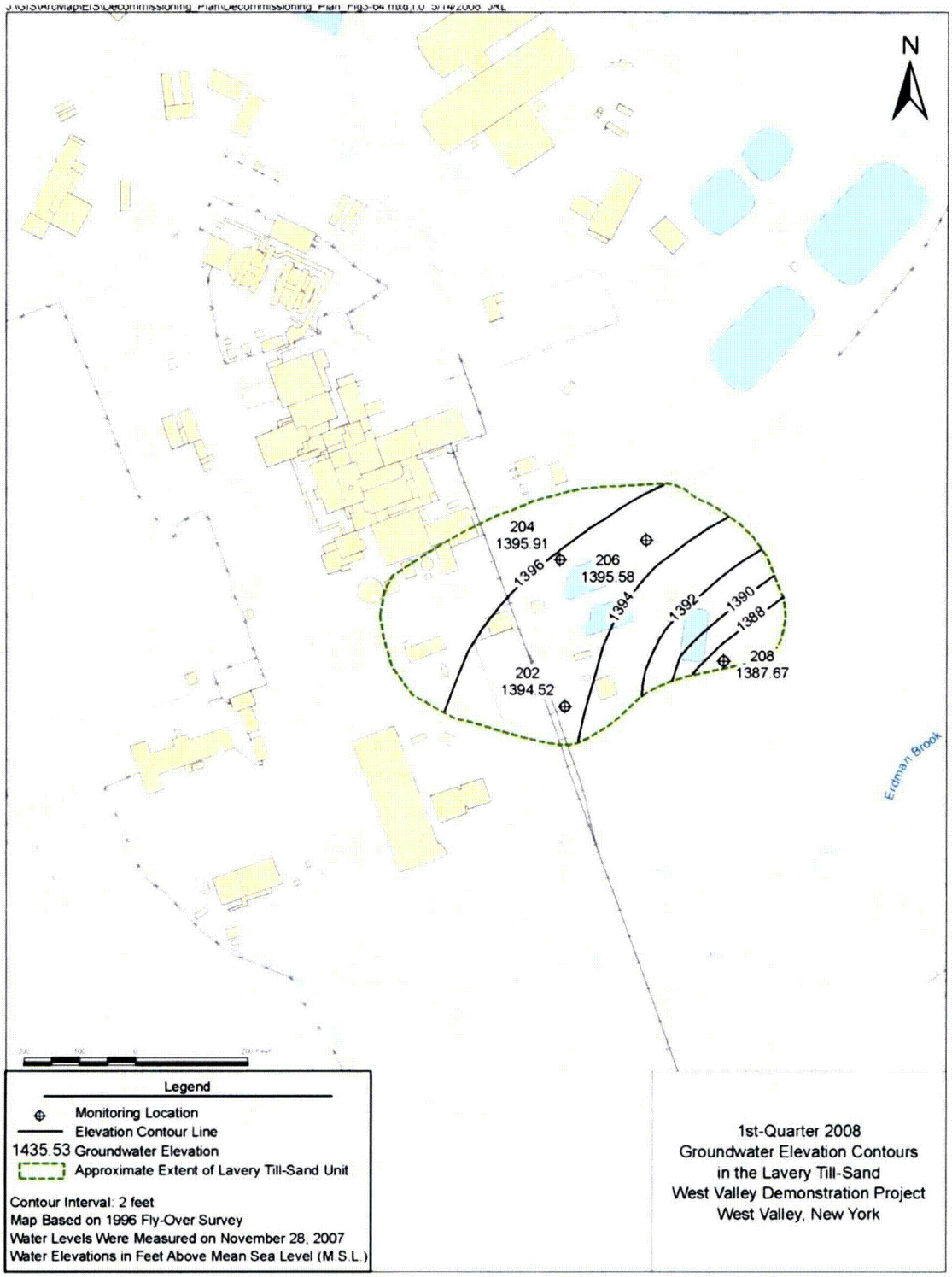

Figure 3-64. Groundwater Elevation Contours of the Lavery Till Sand, First Quarter 2008

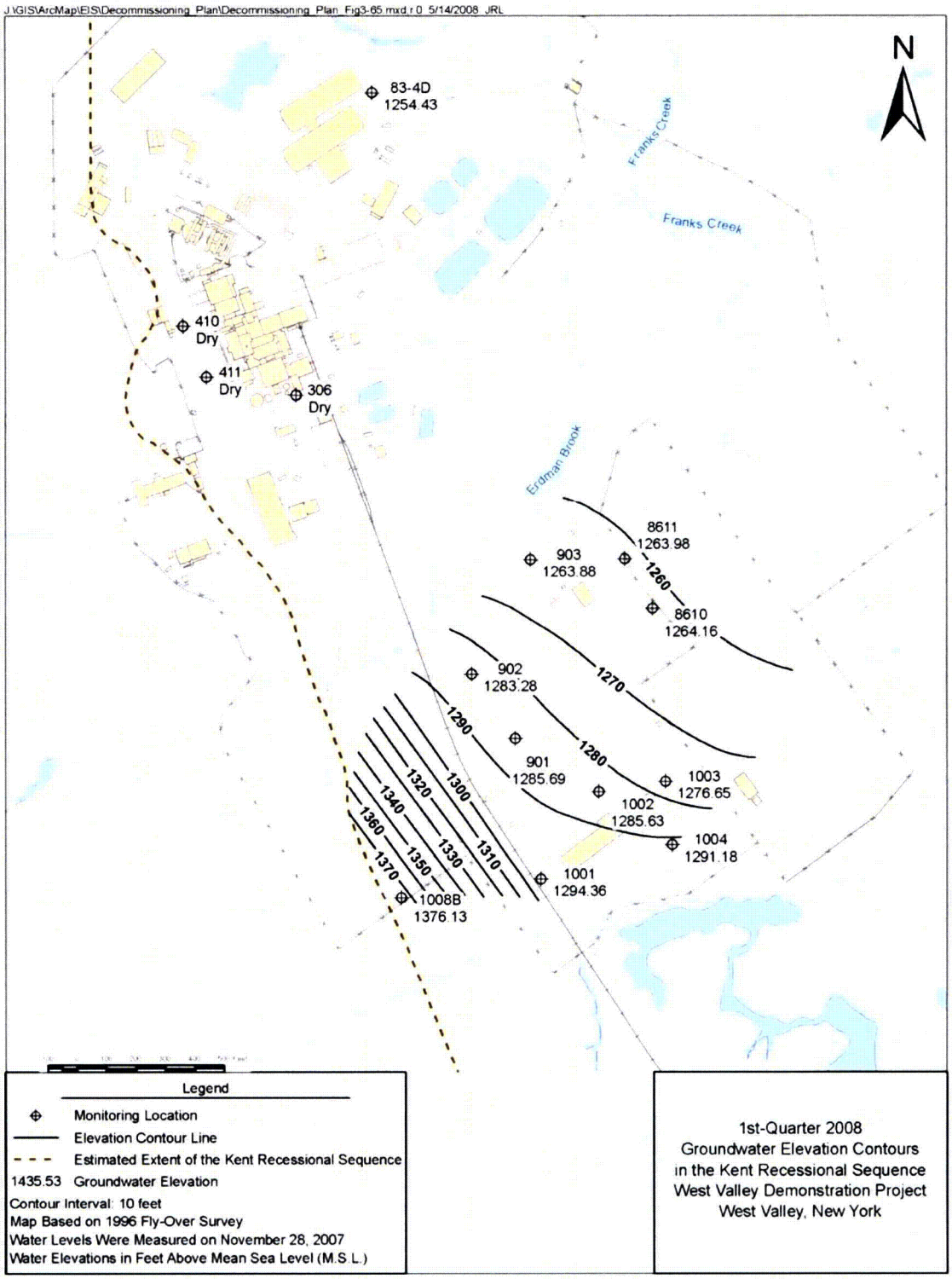

Figure 3-65. Groundwater Elevation Contours of the Kent Recessional Sequence, First Quarter 2008

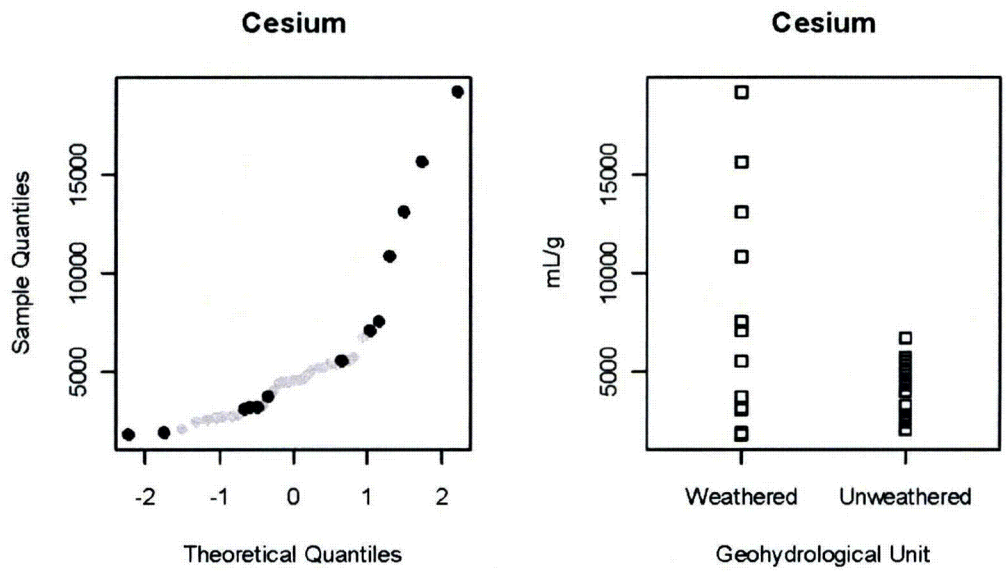

Figure 3-66. Vertical Distribution of Cesium K_d in the Weathered and Unweathered Tills (WVNSCO 1993a)

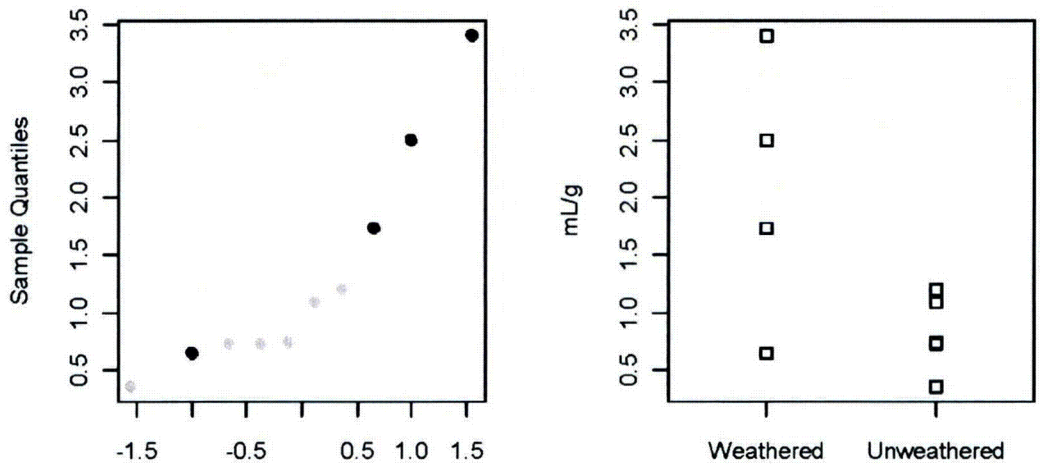

Figure 3-67. Vertical Distribution of Iodine K_d in the Weathered and Unweathered Tills (WVNSCO 1993a)

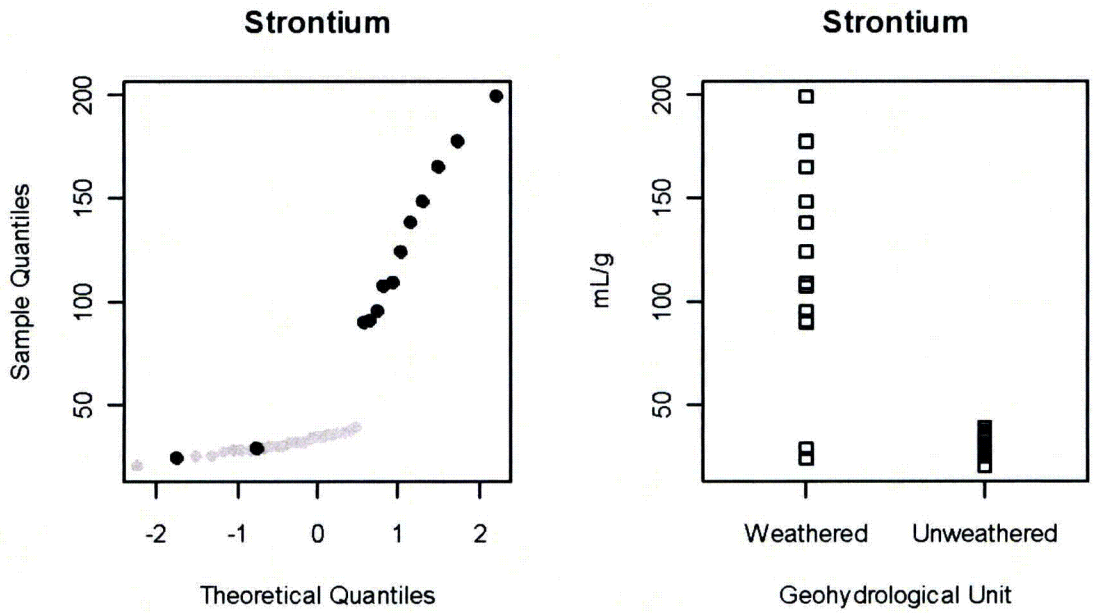

Figure 3-68. Vertical Distribution of Strontium K_d in the Weathered and Unweathered Tills (WVNSCO 1993a)

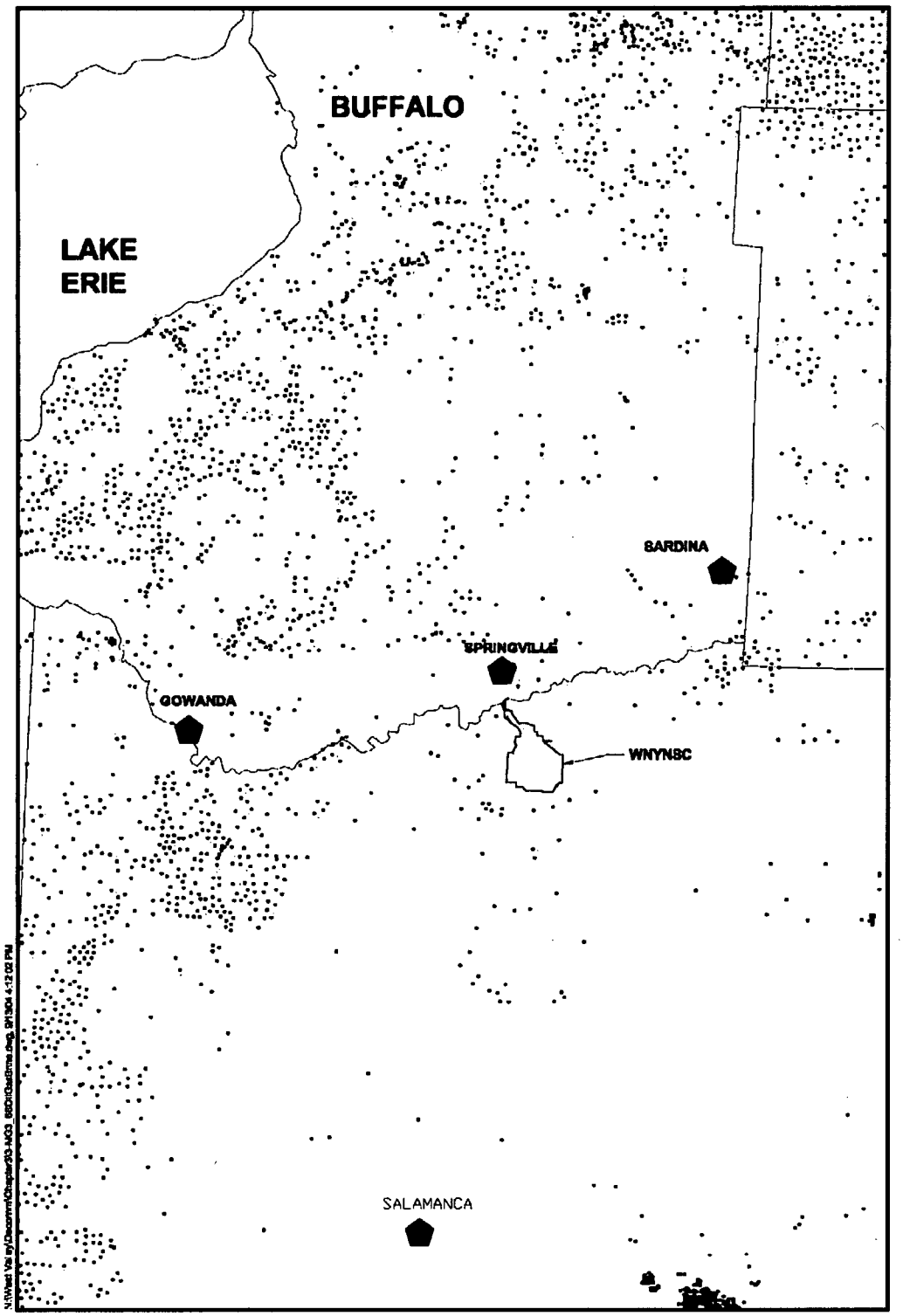

Figure 3-69. Locations of Natural Gas and Oil Wells in Western New York

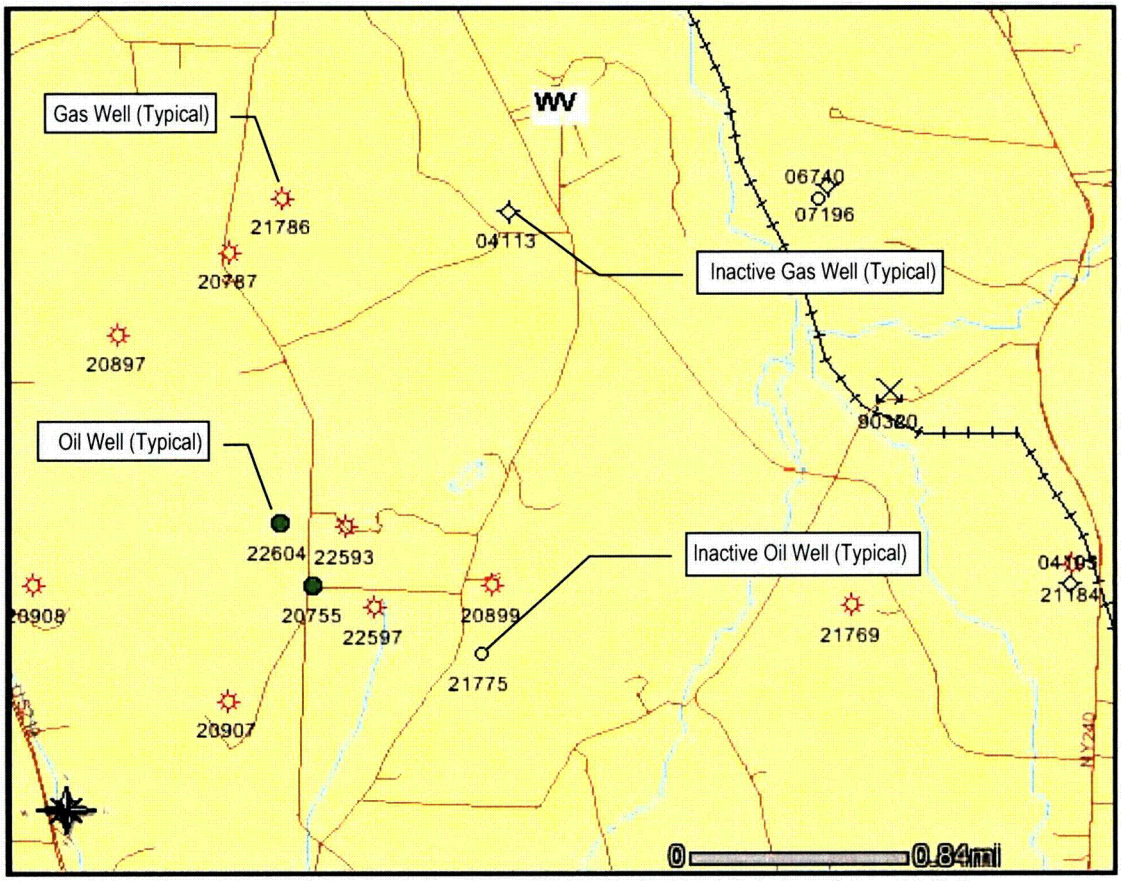

Figure 3-70. Locations of Natural Gas and Oil Wells in the Vicinity of the WVDP

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4.0 RADIOLOGICAL STATUS OF FACILITY

PURPOSE OF THIS SECTION

The purpose of this section and the related Appendix B is to provide summary information on the radiological status of the facilities and environmental media within the scope of the plan. This information is intended to enable readers to understand the types, levels, and general extent of radioactive contamination in the WVDP facilities and in soil, sediment, groundwater, and surface water on the project premises.

INFORMATION IN THIS SECTION

This section focuses mainly on facilities and areas within the scope of the plan.

- Section 4.1.1 discusses sources of available radiological data, background radioactivity, the origin of site radioactivity, and the mode of contamination in facilities.
- Section 4.1.2 identifies facilities impacted by radioactivity.
- Section 4.1.3 identifies facilities not impacted by radioactivity as of 2008.
- Section 4.1.4 provides information on radionuclide distributions in facilities.
- Section 4.1.5 summarizes the radiological status of the facilities of interest.
- Section 4.2 addresses the radiological status of surface soil, sediment, sub-surface soil, surface water, and groundwater and identifies impacted and non-impacted areas of the project premises. It also provides data on environmental radiation levels.

Additional radiological characterization would be performed where appropriate as described in Section 7 and Section 9.

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 plan;
- The information in Section 2 on site history, processes, previous decommissioning activities, and spills; and
- The facility descriptions, photographs, and illustrations in Section 3.

The radiological status information in this section provides the context for information provided in later sections, such as the dose modeling described in Section 5, the decommissioning activities in Section 7, and facility radiation surveys in Section 9.

4.1 Radiological Status of Facilities, Systems, and Equipment

This section summarizes existing data on radiological conditions in WVDP facilities, systems, and equipment. To fully define the radiological status of facilities and equipment within the scope of this plan, additional characterization would be performed in connection with proposed decommissioning activities as described in Sections 7 and 9.

4.1.1 Sources of Available Data

Radiological data on facilities, systems, and equipment are available from the Facility Characterization Project, which focused on the Process Building and the Vitrification Facility, and from several other sources.

Facility Characterization Project

The Facility Characterization Project, as described in the *Characterization Management Plan for the Facility Characterization Project* (Michalczak 2004a), produced conservative estimates of radionuclide inventories in various areas of the Process Building and in the 01-14 Building and the Vitrification Facility. These estimates are documented in a series of radioisotope inventory reports issued between 2002 and 2005.[1]

The Facility Characterization Project focused on the following radionuclides of interest:

Am-241	Cs-137	Pu-239	Tc-99	U-235
C-14	I-129	Pu-240	U-232	U-238
Cm-243	Np-237	Pu-241	U-233	
Cm-244	Pu-238	Sr-90	U-234	

Sixteen of these radionuclides (all except Sr-90 and Cs-137) were determined to be of interest because of their impacts in dose analyses associated with long-term performance assessment of the partially remediated site (Michalczak 2004a). Strontium-90 and Cs-137 were included because they are among the dominant radionuclides in site radioactive contamination and because they could have significant dose impacts in the near term.[2]

The process used to compile total activity estimates was inherently conservative for several reasons. These reasons include (1) assuming in dose rate-to-activity modeling that all measured gamma radiation was due to a single surrogate radionuclide (Cs-137 or Am-241), even though other gamma-emitting radionuclides may have also been present, and (2) use of the most conservative radionuclide distribution data for estimating scaling factors relating amounts of other radionuclides to Cs-137 in cases where multiple sets of radionuclide distribution data were available (Michalczak 2004a).

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[1]The Facility Characterization Project focused on source term estimates because when it was initiated the decommissioning approach was expected to entail in-place closure of a portion of the upper structure of the Process Building, as well as the underground portions of the structure and the Vitrification Facility.

[2] Additional information about selection of the radionuclides of primary interest for the Facility Characterization Project and in developing DCGLs for soil and sediment contamination appears in Section 5.2.

In addition to the source term estimates, the radioisotope inventory reports contain information on radiological history, radionuclide distributions, contamination levels, and radiation levels.

Characterization of the Underground Waste Storage Tanks

The four waste storage tanks have undergone detailed characterization. Data collection and analysis for Tanks 8D-1 and 8D-2 were performed in accordance with an approved data collection and analysis plan (Fazio 2001). The characterization results appear in three radioisotope inventory reports (Fazio 2002a, Fazio 2002b, and Fazio 2004c). These reports were provided to NRC in connection with preparation of the Decommissioning EIS.

In response to comments on the radioisotope inventory reports from NRC and other agencies, DOE prepared a supplemental report (WVNSCO and Gemini 2005) to clarify information on radionuclides of significance, address uncertainty in the inventory estimates, and provide additional information on the technical basis for scaling factors and on the mobile inventory estimate for Tank 8D-4.

Other Facility Residual Radioactivity Estimates

In 2008, the site contractor, West Valley Environmental Services (WVES), developed additional estimates for residual radioactivity in the Process Building, the Vitrification Facility, and underground waste storage Tanks 8D-3 and 8D-4 in the interim end state, i.e., at the beginning of the Phase 1 proposed decommissioning activities (WVES 2008a, WVES 2008b, and WVES 2008c, respectively). These estimates utilized the previous characterization results combined with projections based on additional decontamination to be performed in certain areas in connection with work to achieve the interim end state.

Analytical Data

The results of analyses of numerous liquid and solid samples performed by both onsite and offsite laboratories are available. These data, most of which are summarized in the radioisotope inventory reports, have been used to define radionuclide distributions in various areas of the Process Building and in the Vitrification Facility, the underground waste tanks, and other WVDP areas.

Routine Radiological Survey Data for Facilities

Routine radiological status surveys are performed in WVDP facilities in support of the WVDP radiation protection program. Data from these surveys, which typically include general area gamma radiation levels and removable beta contamination levels, reflect the current radiological status in accessible areas of most WVDP facilities.

Scoping Data

Available radiological data on facilities, systems, and equipment are generally considered to be scoping data, with the exception of data on the underground waste tanks, which have been appropriately characterized. As defined in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000), scoping survey data

identify radionuclide contaminants, relative radionuclide ratios, general levels, and the extent of contamination, yet may not comprise definitive characterization data. In some areas, available data are insufficient to meet the definition of scoping data, especially in cases where radionuclide ratios are not available or where the extent of contamination is not defined. (As noted previously, additional characterization would be performed in connection with proposed decommissioning activities as described in Sections 7 and 9.)

Background Radioactivity

Limited data are available on background radioactivity in facilities, although there are data from areas with a low potential for contamination. For example, typical routine surveys show gamma radiation levels <0.1 mR/h in the Solvent Storage Terrace and Acid Handling Area of the Process Building (Michalczak 2004b) and measurements taken with sodium-iodide detectors recorded in μR/h are available in some low-potential areas. During the characterization program outlined in Section 9, sufficient data would be acquired to establish background levels in facilities within the scope of the Phase 1 proposed decommissioning activities.

Origin of Site Radioactivity

Radioactivity associated with the project premises originated in irradiated nuclear fuel reprocessed in the Process Building. Analytical data on radioactivity in the fuel are available as described below. With the exception of one batch of thorium-uranium fuel, all fuel reprocessed was uranium based, as noted in Section 2.

Information on how the facilities became contaminated is contained in Section 2.

Mode of Contamination in Facilities

In many cases, radioactive contamination associated with facilities is located only on facility surfaces, and does not penetrate into the surfaces, and inside contaminated systems and equipment. In some cases contamination is also located on the outside of systems and equipment.

Exceptions primarily involve contamination of Process Building facility surfaces in depth from spills of radioactive acid on painted concrete surfaces and where radioactive water stood in the fuel pools. This conclusion is generally based on radiation level measurements on decontaminated surfaces that have minimal removable contamination. Quantitative information on the depth of penetration is available only in a single case: one sample from a wall of the Chemical Process Cell that showed contamination had penetrated approximately two inches into the concrete (URS 2001).

Data Provided in this Section

Section 4.1 provides estimates of residual radioactivity for the Process Building and the Vitrification Facility, which are within the scope of this plan, and for information and perspective, the underground waste storage tanks, and the NRC-Licensed Disposal Area (NDA). Data on radiation levels in representative areas of the Process Building, in the

Vitrification Facility, and in other areas are provided. Residual radioactivity in other areas is also discussed.

4.1.2 Impacted Facilities

The following facilities where licensed activities and/or WVDP activities have taken place are known or suspected to contain residual radioactive material in excess of background levels. Figures 4-1, 4-2, 4-3, 4-4, and 4-5 show the locations of these facilities. This list does not include facilities existing in 2008 that will be removed before the proposed decommissioning activities begin, which are addressed in Section 2.2.2. However, it does include for information and perspective some facilities that are not within the scope of Phase 1 of the proposed decommissioning.

WMA 1, Process Building and Vitrification Facility Area

- Process Building
- Utility Room and Utility Room Expansion
- Plant Office Building
- 01-14 Building
- Load-In/Load-Out Facility
- Vitrification Facility
- Vitrification off-gas trench lines
- Underground wastewater Tanks 35104, 7D-13, and 15D-6
- Underground lines

WMA 2, Low-Level Waste Treatment Facility Area

- LLW2 Building
- Old Interceptor
- New Interceptors (2)
- Neutralization Pit
- Lagoon 1 (deactivated)
- Lagoon 2
- Lagoon 3
- Lagoon 4
- Lagoon 5
- Solvent Dike

- Underground wastewater lines[3]
- French drain
- Maintenance Shop leach field
- North Plateau Groundwater Pump and Treat Facility (not in plan scope)
- Pilot permeable treatment wall (not in plan scope)
- Full-scale permeable treatment wall (to be installed, not in plan scope)

WMA 3, Waste Tank Farm Area

- Underground waste Tanks 8D-1 and 8D-2 and associated vaults[4]
- Underground waste Tanks 8D-3 and 8D-4 and their common vault[3]
- Con-Ed Building
- Equipment Shelter and Condensers
- HLW Transfer Trench piping
- Permanent Ventilation System Building (not in plan scope)
- Supernatant Treatment System Support Building (not in plan scope)
- Underground lines (not in plan scope)

WMA 4, Construction and Demolition Debris Landfill Area

- Construction and Demolition Debris Landfill (not in plan scope)
- Permeable reactive barrier (to be installed, not in plan scope)

WMA 5, Waste Storage Area

- Lag Storage Area 4 and Shipping Depot
- Remote Handled Waste Facility

WMA 6, Central Project Premises

- Demineralizer sludge ponds (2)
- Cooling Tower basin
- Rail Spur (because of nearby soil contamination, not within plan scope)

WMA 7, NDA and Associated Facilities

- Entire area (only the hardstand is within plan scope)

WMA 9, Radwaste Treatment System Drum Cell Area

- Radwaste Treatment System Drum Cell

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[3] Only those lines within planned excavations to remove facilities are within plan scope.
[4] Only the tank mobilization and transfer pumps and their support structures are with the scope of this plan.

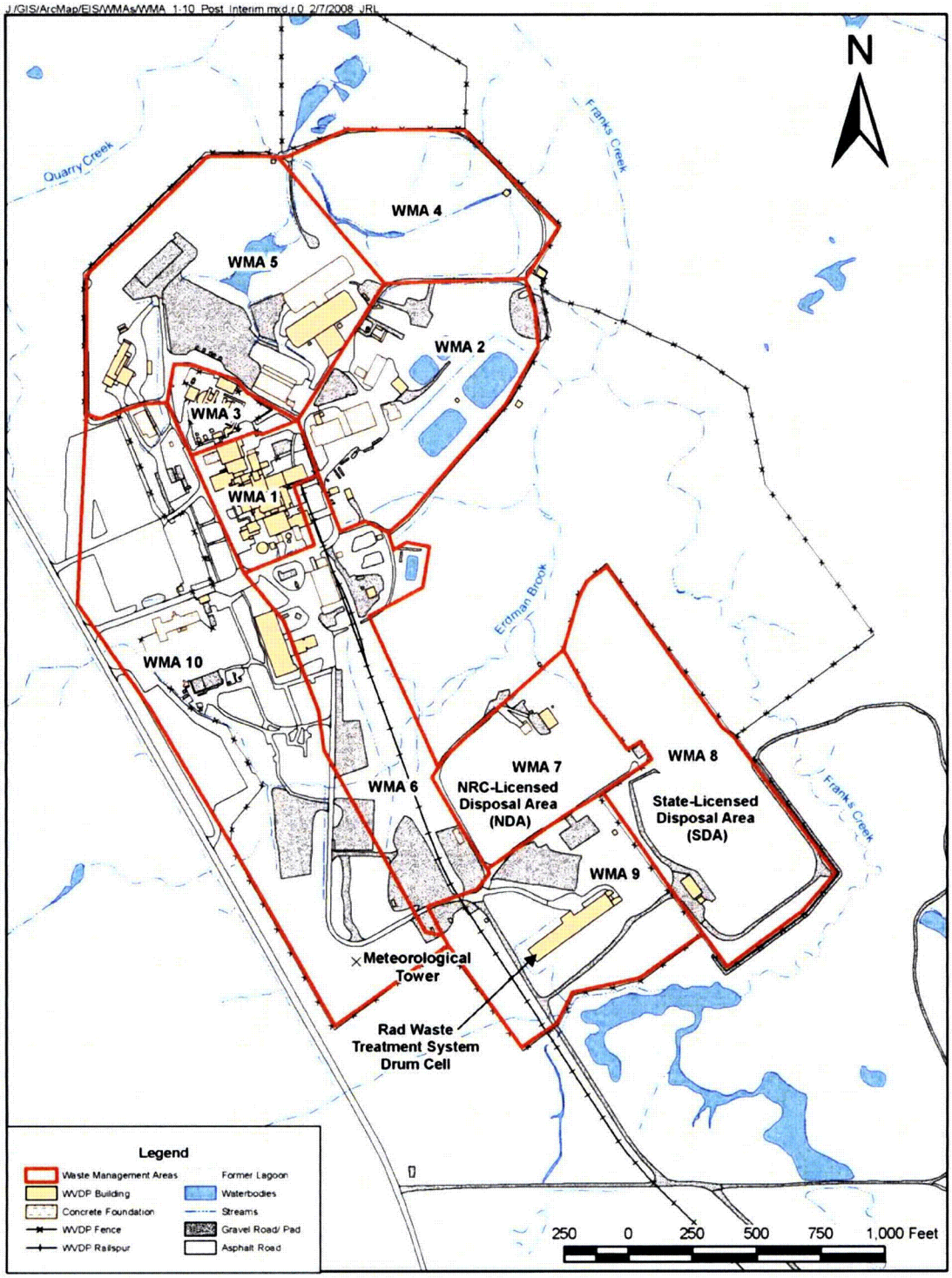

Figure 4-1. Location of Impacted and Non-Impacted Facilities

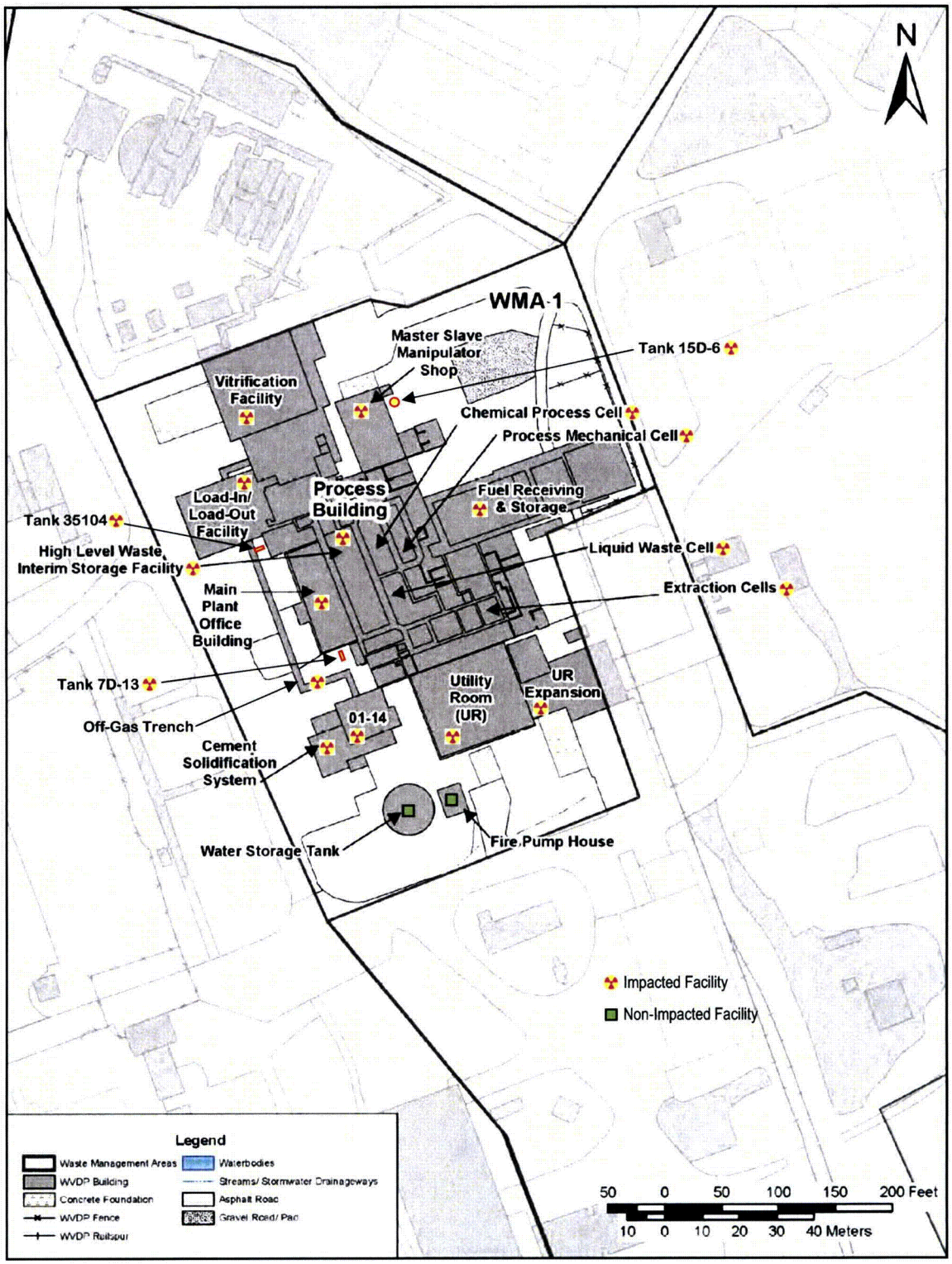

Figure 4-2. Impacted and Non-Impacted Facilities in WMA 1

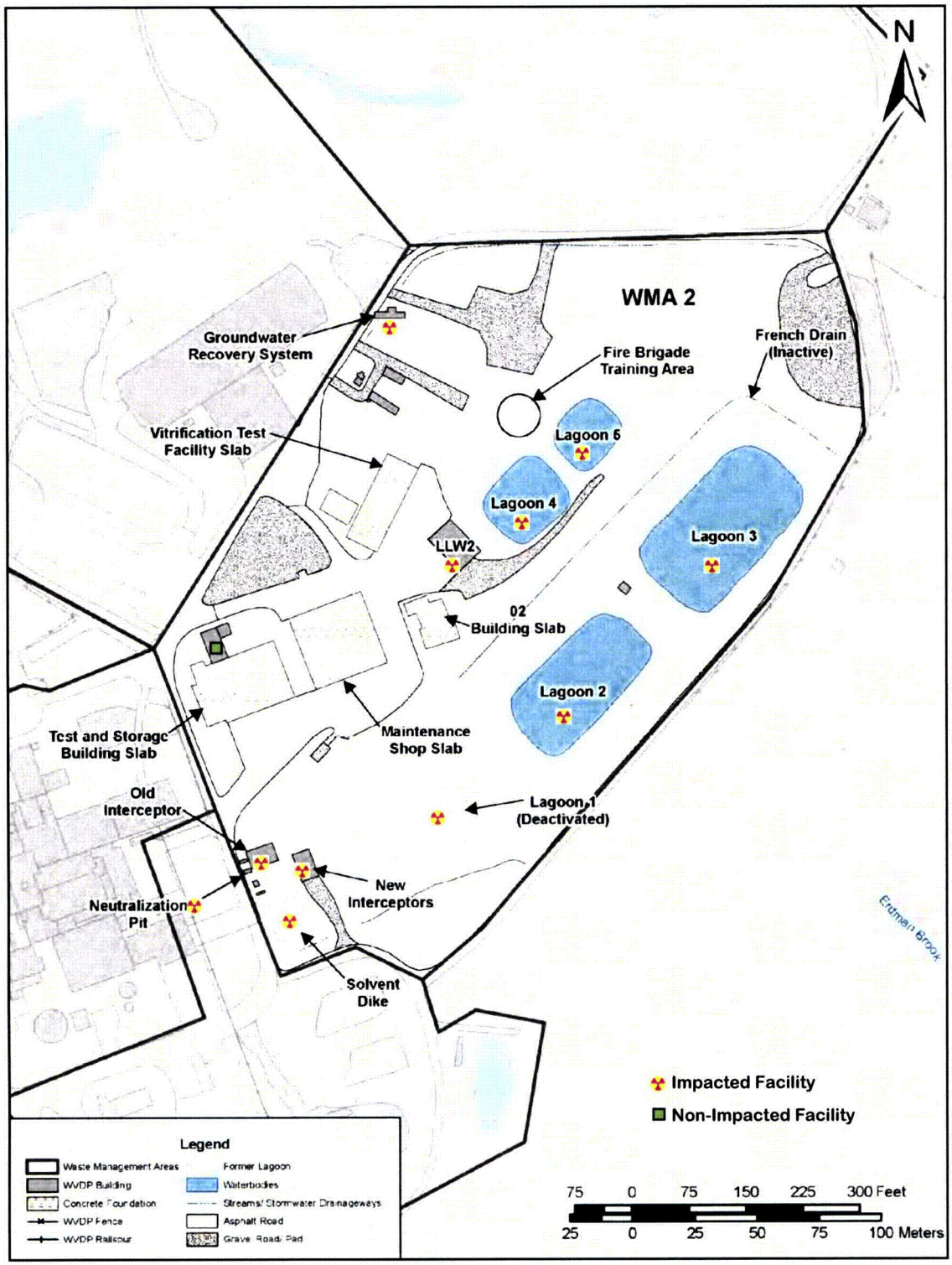

Figure 4-3. Impacted and Non-Impacted Facilities in WMA 2

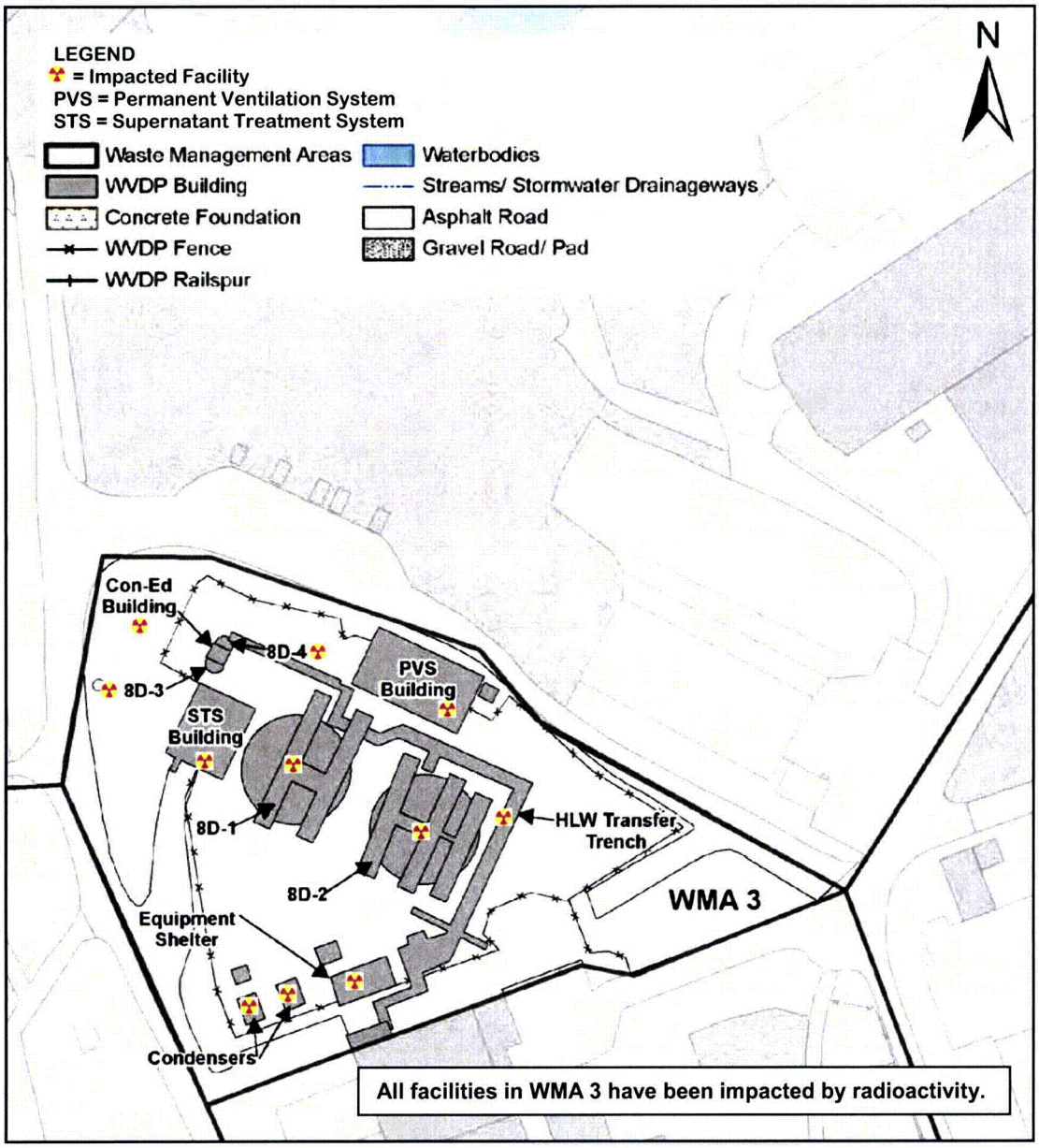

Figure 4-4. Impacted Facilities in WMA 3

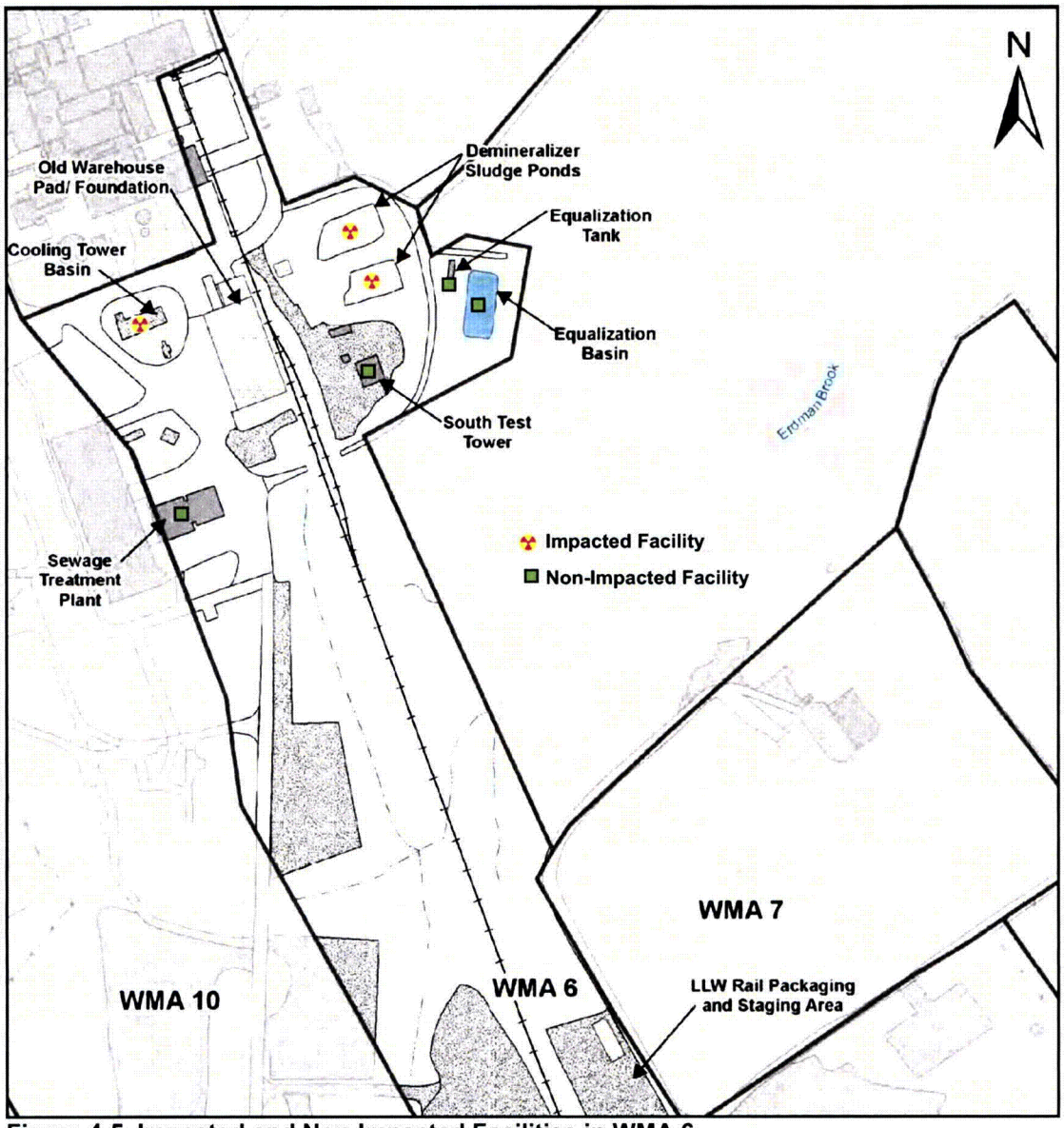

Figure 4-5. Impacted and Non-Impacted Facilities in WMA 6

4.1.3 Non-Impacted Facilities

The following structures and locations have not been impacted by radioactivity associated with licensed activities or WVDP activities as of 2008, based on process history, the results of routine radiological surveys, and the results of the WVDP environmental monitoring program (WVES and URS 2008). These facilities are shown in Figures 4-1, 4-2, or 4-5.

WMA 1, Process Building Area

- Fire Pump House
- Water Storage Tank
- Electrical Substation

WMA 6, Central Project Premises

- Sewage Treatment Plant
- South Waste Tank Farm Test Tower
- Equalization Basin
- Equalization Tank

WMA 10, Support and Services Area

- New Warehouse
- Meteorological Tower (not within plan scope)
- Security Gatehouse and Fences (not within plan scope)

Even though the Sewage Treatment Plant is considered not to have been impacted by radioactivity associated with licensed activities or the WVDP as of 2008, the excavation dug for its removal would be considered in Phase 1 final status surveys because of the potential buildup of naturally-occurring radioactivity in sewage sludge, as explained in Section 7.

Some WMAs also contain concrete floor slabs and foundations and gravel pads that would be removed during Phase 1. Some of the concrete slabs have been impacted by radioactivity as explained in Section 2 and may contain low levels of residual radioactivity.

Note that conditions in the non-impacted facilities are subject to change. DOE or its decommissioning contractor would reevaluate the conclusion that these facilities have not been impacted before decommissioning activities begin.

4.1.4 Radionuclide Distributions

Owing to the nature of spent fuel separation and purification processes, radionuclide distributions vary inside different areas of the Process Building and in other facilities of interest depending on the point in the reprocessing cycle where the contamination originated. Other factors discussed below also influenced radionuclide distributions inside the Process Building.

During the Facility Characterization Project, available analytical data and data from samples obtained and analyzed during this project were utilized to establish bounding radionuclide scaling factors. These scaling factors, which relate the concentrations of other radionuclides of interest to the concentration of Cs-137 or Am-241, were chosen to ensure that concentrations of radionuclides important to the dose evaluation were not underestimated[5].

The two principal radionuclide distributions that were available before the beginning of the Facility Characterization Project are known as the spent fuel distribution and the Batch 10 distribution. These distributions are discussed below.

Spent Fuel Distribution

Information on the radionuclide distribution associated with spent nuclear fuel has been derived primarily from the results of modeling of fuel processed by Nuclear Fuel Services (NFS) that was performed by Pacific Northwest National Laboratory using the ORIGEN2 computer code (Jenquin, et al. 1992). These data were used for all radionuclides of interest in spent fuel except U-235 and U-238, which were derived from NFS records for recovered and unaccounted for losses of uranium, and U-232, U-233, U-234, and U-236, which were established based on analytical results showing the U-232 to U-235/236 ratio from samples collected in the Acid Recovery Pump Room of the Process Building. The resulting scaling factors relating concentrations of other radionuclides of interest to the concentration of Cs-137 were determined to be conservative (Mahoney 2002). These scaling factors are shown in Table 4-1.

Table 4-1. Scaling Factors for Spent Fuel Reprocessed(1)

Nuclide	Ratio(2)	Nuclide	Ratio(2)	Nuclide	Ratio(2)
Am-241	8.58E-02	Np-237	4.5E-06	U-232	6.9E-01
C-14	1.3E-04	Pu-238	1.69E-02	U-233	1.40E+00
Cm-242	2.0E-04	Pu-239	2.84E-02	U-234	9.0E-02
Cm-243	5.9E-05	Pu-240	1.48E-02	U-235	1.5E-06
Cm-244	1.52E-03	Pu-241	9.10E-01	U-236	1.39E-01
I-129	6.3E-07	Tc-99	2.7E-04	U-238	2.6E-05

Notes: (1) From Mahoney 2002, Tables 1 and 2, reference date January 1, 1993

(2) All are scaled to Cs-137, except for U-232, U-233, U-234, and U-236, which are scaled to U-238. Sr-90 does not appear in the tables of calculated scaling factors in Mahoney, 2002. The Sr-90 to Cs-137 ratio was determined to be 9.5E-01 (WVNSCO 1989).

Note that in compiling estimates during the Facility Characterization Project, the reference date was adjusted to September 30, 2004 and the values for U-232, U-233, U-234, and U-236 were scaled to Cs-137 rather than U-238.

[5]Where multiple data sets were available, the highest values among radionuclide ratios from the different data sets were selected for each radionuclide for conservatism (Michalczak 2004a).

Batch 10 HLW Distribution

The vitrification Batch 10 distribution was used to establish bounding scaling factors related to Cs-137 for HLW. The Batch 10 sample analyzed was obtained from the first HLW transfer from underground waste Tank 8D-2 to the Vitrification Facility in 1996. It was representative of the waste in its most concentrated form when the highest ratios of alpha-emitting transuranic radionuclides to Cs-137 were present. Later batches contained relatively higher concentrations of Cs-137 (and lower ratios of alpha-emitting transuranics to Cs-137) because Cs-137 captured in zeolite resin was returned to Tank 8D-2 for subsequent transfer to the Vitrification Facility.

The Batch 10 sample was analyzed in May 1997 by the Radiological Processing Laboratory at Pacific Northwest National Laboratory. The analysis results are shown in Table 4-2.

Table 4-2. Batch 10 Sample Data(1)

Nuclide	µC/g	Nuclide	µC/g	Nuclide	µC/g
Am-241	3.21E+01	Np-237	2.00E-02	Tc-99	8.45E-02
C-14	4.90E-04	Pu-238	3.96E+00	U-232	(2)
Cm-243	2.58E-01	Pu-239	1.09E+00	U-233	3.60E-03
Cm-244	6.72E+00	Pu-240	7.70E-01	U-234	1.30E-03
Cs-137	2.85E+03	Pu-241	3.43E+01	U-235	3.80E-05
I-129	3.90E-07	Sr-90	2.75E+03	U-238	3.40E-04

Notes: (1) From Pacific Northwest National Laboratory results corrected for decay and ingrowth to May 15, 1997, included in Michalczak 2003b.
(2) No analysis was performed for U-232.

Process Building Distributions

During the Facility Characterization Project, the spent fuel distribution and the Batch 10 distribution were used in conjunction with sample analytical data to determine the appropriate radionuclide distribution for various representative areas of the Process Building.

Contamination in most areas of the building resulted primarily from spills and leaks of materials in the reprocessing feed and waste process streams. This feed and waste contamination is associated with reactor fuel before fission products have been separated or with the separated fission products. Until the point where the fuel was dissolved in the Chemical Process Cell, radionuclide ratios remained characteristic of the feed and waste process streams, typified by the Batch 10 distribution in Table 4-2.

Downstream of the dissolution process that took place in the Chemical Process Cell, radionuclide ratios began to change in the extraction cells, where the dissolved fuel underwent a solvent extraction process that separated uranium and plutonium from the

fission products. The uranium and plutonium products achieved their purest forms in the Product Purification Cell.

Contamination in other areas of the building came primarily from spills or leaks of the reprocessed products. These other areas are the Product Purification Cell, the Lower Warm Aisle, the Product Packaging and Handling Area, and the Extraction Sample Aisle.

There are substantial variations among distributions in different areas. One particular spill during reprocessing that affected radionuclide distributions in several areas was the release of highly radioactive nitric acid from an acid recovery line in the southwest corner of the building, as described in Section 2.

The dominant radionuclides in the Process Building contamination are typically Cs-137, Pu-241, Sr-90, Am-241, and Pu-238. The relative fractions of dominant radionuclides in the two basic distributions can be calculated based on the geometric means of the distributions in the various Process Building areas. Table 4-3 shows the results of these calculations. However, there are significant variations from these relative fractions in the different areas for which data were compiled.

Table 4-3. Relative Fractions of Process Building Dominant Radionuclides(1)

Relative Fractions of Dominant Radionuclides in Feed and Waste Contamination					
Radionuclide	**Pu-241**	**Cs-137**	**Sr-90**	**Am-241**	**Pu-238**
Fraction	0.404	0.281	0.216	0.065	0.035
Relative Fractions of Dominant Radionuclides in Product Contamination					
Radionuclide	**Pu-241**	**Am-241**	**Pu-238**	**Pu-239**	**Pu-240**
Fraction	0.754	0.133	0.045	0.039	0.029

NOTE: (1) Based on geometric means of radionuclides in the differently impacted areas using data from the Facility Characterization Project radioisotope inventory reports . These were the ratios on September 30, 2004, the reference date for the data used.

The information on radionuclide distributions for different Process Building areas found in the radioisotope inventory reports produced by the Facility Characterization Project would be used for planning decommissioning activities in the building and for waste management purposes.

The relative fractions of the dominant radionuclides in the Vitrification Facility are shown in Table 4-4.

Table 4-4. Relative Fractions of Vitrification Facility Dominant Radionuclides(1)

Radionuclide	**Cs-137**	**Sr-90**	**Am-241**	**Pu-241**	**Cm-244**
Fraction	0.506	0.482	0.007	0.005	0.001

NOTE: (1) Based on data in Radioisotope Inventory Report RIR-403-010 (Lachapelle 2003) as of December 31, 2006 as given in WVES 2008b.

4.1.5 Radiological Status of Facilities

Most of the residual radioactivity in facilities within the scope of this plan resides in two areas: the Process Building and the Vitrification Facility. Significant amounts of radioactivity are also located in Lagoon 1, Lagoon 2, the piping in the HLW transfer trench, the vitrification off-gas line that runs to the 01-14 Building, and underground piping in the Process Building area.

Radioactivity in WMA 1, the Process Building

The Facility Characterization Project provided residual inventory estimates for 33 different areas of the Process Building, including a group of "low ranking" areas. However, additional decontamination work is being accomplished in the Off-Gas Cell, the General Purpose Cell, and the Process Mechanical Cell.

Table 4-5 provides an estimate of the total amount of residual radioactivity that will be in the building when the interim end state is reached, that is, at the beginning of Phase 1 proposed decommissioning activities. The estimates account for the expected effectiveness of the planned decontamination work, which will include removal of certain equipment and two decontamination cycles for the floors and walls of the General Purpose Cell, the Process Mechanical Cell, and the Off-Gas Cell (WVES 2008a).

Table 4-5. Estimated Process Building Residual Activity at Start of Decommissioning(1)

Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)
Am-241	260	Np-237	0.57	Tc-99	4.9
C-14	13	Pu-238	200	U-232	0.75
Cm-243	0.27	Pu-239	63	U-233	0.41
Cm-244	6.3	Pu-240	47	U-234	0.19
Cs-137	2550	Pu-241	1100	U-235	0.03
I-129	0.63	Sr-90	1900	U-238	0.09

(1) From WVES, 2008a, not including the amounts for "yard" (i.e., the three underground wastewater tanks) and the 01-14 Building, with the estimates rounded to two significant figures or the nearest integer. These estimates were corrected for decay and ingrowth to 2011. They do not include activity associated with the HLW canisters or approximately 110 curies in embedded piping in the Process Building (McNeil 2005a).

Table 4-6 shows the total estimated residual radioactivity in different areas of the Process Building as of 2004.

Table 4-6. Estimated Total Activity in Representative Process Building Areas(1)

Area	Curies	Area	Curies
Analytical Decontamination Aisle	<1	Main Plant Stack	88
Acid Recovery Cell(1)	60	Miniature Cell	9
Acid Recovery Pump Room	31	Off-Gas/Acid Recovery Aisle	40

Table 4-6. Estimated Total Activity in Representative Process Building Areas(1)

Area	Curies	Area	Curies
Analytical Hot Cells	39	Off-Gas Blower Room	72
Building Roof	1	Off-Gas Cell(1)	250
Chemical Crane Room	6	Process Mechanical Cell(1)	1000
Chemical Process Cell	130	Process Sample Cells, 1C Sample Station	6
Equipment Decontamination Rm	36	Product Purification Cell	43
Extraction Cell 1(1)	47	Sample Storage Cell	17
Extraction Cell 2	2	Scrap Removal Room	<1
Extraction Cell 3(1)	11	Southwest Stairwell	5
Fuel Receiving and Storage	290	Upper Warm Aisle	18
General Purpose Cell(1)	3000	Uranium Load-Out Area	<1
GPC Crane Room and Extension	7	Uranium Product Cell	45
Head-End Ventilation Cell	610	Ventilation Exhaust Cell	67
Hot Acid Cell	<1	Ventilation Wash Room	74
Liquid Waste Cell	1000	Low Ranking Areas (31 areas)	25
Lower Warm Aisle	84	Embedded Piping	110

(1) From WVES, 2008a, with estimates corrected for decay and ingrowth to September 30, 2004 and here rounded to two significant figures or the nearest whole number, with the exception of the embedded piping estimate, which is taken from McNeil 2005a. These estimates assume that the work to achieve the interim end state will include additional decontamination of the floors and walls in three areas: the General Purpose Cell, the Off-Gas Cell, and Process Mechanical Cell. The estimates also assume that the vessels in the Acid Recovery Cell, the Hot Acid Cell, Extraction Cell 1, and Extraction Cell 3 will be removed.

Despite decontamination efforts, radiation levels remain relatively high in some areas of the building. Table 4-7 shows the highest radiation levels measured in representative areas.

Table 4-7. Measured Maximum Gamma Radiation Levels in Process Building Areas

Area	mR/h	Remarks	Source
Chemical Process Cell	15,000	At south sump in 1994	Michalczak 2003a
Equipment Decontamination Room	50	On floor in 1997	Michalczak 2003b
Fuel Receiving and Storage Area	8.5	Fuel Storage Pool, 2002	Fazio 2004a
	500	Cask Unloading Pool, 2002	Fazio 2004a
General Purpose Cell	200,000	3 feet above floor(1)	Choroser 2005a
	32,000	9 feet above floor(1)	Choroser 2005a
Head-End Ventilation Cell	50,000	On pre-filters in 2002	Michalczak 2003c
Liquid Waste Cell	1,800	In 2002	Choroser 2004
Miniature Cell	80	In 1998	Michalczak 2002a
Off-Gas Blower Room	700	In 2003	Michalczak 2002b

Table 4-7. Measured Maximum Gamma Radiation Levels in Process Building Areas

Area	mR/h	Remarks	Source
Process Mechanical Cell	40,000	In 2004, 3 feet above floor(1)	Choroser 2005b
Product Purification Cell	53	Hot spot on wall in 2003	Choroser 2003
Sample Storage Cell	1,950	On floor in 2001	Drobot 2003
Ventilation Wash Room	1,500	On ventilation duct	URS 2001

(1) Before planned additional decontamination described in report WVES 2008a.

Radiation levels on the vitrified HLW canisters measured in the 1996 to 2002 period during vitrification ranged from 1,770 to 7,460 R/h (Michalczak 2003a). The total activity in the average canister is approximately 37,000 curies, including approximately 13,600 curies of Sr-90 and approximately 23,400 curies of Cs-137, based on data in the waste form qualification report (WVNSCO 2007). The canisters remain stored in the HLW Interim Storage Facility in the former Chemical Process Cell, as noted previously.

Radioactivity in WMA 1, the Vitrification Facility

Table 4-8 shows the estimated residual radioactivity in the Vitrification Facility at the beginning of Phase 1 proposed decommissioning activities. Essentially all of this radioactivity is in the Vitrification Cell.

Table 4-8. Estimated Total Activity in the Vitrification Facility(1)

Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)
Am-241	14	Np-237	0.01	Tc-99	0.04
C-14	<0.01	Pu-238	1.6	U-232	<0.01
Cm-243	0.09	Pu-239	0.49	U-233	<0.01
Cm-244	1.9	Pu-240	0.35	U-234	<0.01
Cs-137	960	Pu-241	8.7	U-235	<0.01
I-129	<0.01	Sr-90	910	U-238	<0.01

(1) From WVES 2008b, corrected for decay and ingrowth to 2011 and rounded to two significant figures or the nearest integer.

Gamma radiation levels in the Vitrification Cell process pit in 2004 after equipment removal and decontamination ranged from 3.1 to 50.5 R/h, with levels in other parts of the cell in the 1.2 to 18.1 R/h range (WVNSCO 2004b).

Radioactivity in Other WMA 1 Facilities

The 01-14 Building together with the vitrification off-gas line that runs to the building from the Vitrification Facility is estimated to contain in 2011 approximately 340 curies, due principally to Sr-90 and Cs-137. Almost the entire amount is expected to be inside the off-gas line. The only place within the building itself where a significant amount of radioactivity is expected, besides the portion of the off-gas line in the building, is in the ventilation exhaust system filters (if these filter remain in place). (Michalczak 2004c)

While the Plant Office Building, the Utility Room, the Utility Room Expansion, and the Load-In Facility have been impacted, they are expected to contain insignificant amounts of radioactivity. Radiation levels in these structures are expected to be <1 mR/h with no removable surface contamination above the minimum detectable concentration (Michalczak 2004b).

Three underground wastewater tanks are located below grade outside of the Process Building: Tank 7D-13, Tank 15D-6, and Tank 35104 as shown in Figure 4-2. Tank 7D-13 has been estimated to contain 150 to 300 gallons of solids containing up to 84 curies in 2011, with the dominant radionuclides being Cs-137, Sr-90, Pu-241, Am-241, and Pu-239 (Michalczak 2004c). The other two tanks are not expected to contain significant amounts of radioactivity.

Most of the underground lines in WMA 1 are expected to be radioactively contaminated. A single line – HLW transfer line 7P120-3 – was estimated to contain more than 90 percent of the total activity. This line runs from under the Chemical Process Cell to Tanks 8D-3 and 8D-4 in WMA 3 and is expected to contain residual radioactivity of approximately 0.4 curie per linear foot in 2011, with almost all of this activity associated with Sr-90 and Cs-137. Several of the underground lines within WMA 1 are known to have leaked as discussed in Section 2. (Luckett, et. al 2004)

Radioactivity in WMA 2 Low-Level Waste Treatment Facility Area Facilities

Low levels of radioactivity are expected to be present in the LLW2 Building. Lagoon 1 is expected to contain a substantial amount of radioactivity, with more than 90 percent in the remaining sediment. Table 2-19 shows the estimated amounts in 2011.

Lagoon 2 is expected to contain residual radioactivity of the same order of magnitude as Lagoon 1 with a similar radionuclide distribution.[6] Lagoon 3 is expected to contain less radioactivity in its sediment than Lagoons 1 and 2. Lagoons 4 and 5 are expected to contain relatively low levels of radioactivity in sediment both above and below their liners. Table 4-14 shows the maximum measured concentrations of radioactivity in sediment samples obtained from each of the lagoons.

The Old Interceptor is expected to contain a significant amount of radioactivity based on available data, which include a gamma radiation level of 408 mR/h measured near the tank bottom in 2003 (WVNSCO 2003). As noted in Section 2, 12 inches of concrete was poured on the tank floor by NFS as radiation shielding. The New Interceptors and the Neutralization Pit are both expected to contain low levels of radioactive contamination.

The three septic tanks and other equipment in the Maintenance Shop leach field may have been impacted by the north plateau groundwater plume, but any resulting contamination levels are expected to be low.

[6] This conclusion is based on primarily on records showing that 22,400 cubic feet of sediment was pumped from Lagoon 1 to Lagoon 2 in 1984, with this sediment containing approximately 107 curies of total alpha activity and 1162 curies of total beta activity (Passuite and Monsalve-Jones 1993). Table 4-14 shows maximum measured radionuclide concentrations in the two lagoons, with Cs-137 concentrations being the same order of magnitude.

The contaminated underground wastewater lines within WMA 2 were estimated to contain a total of approximately 0.3 curies of residual radioactivity in 2004 (Luckett, et al. 2004). The French drain is expected to contain very low levels of residual radioactivity.

Radioactivity in the WMA 3 Waste Tank Farm Area Facilities

As explained in Section 1, only certain facilities and equipment within WMA 3 are within the scope of this plan. However, all WMA 3 facilities are briefly addressed here for perspective.

Table 2-5 in Section 2 provides estimates for the residual radioactivity in the underground waste tanks at the conclusion of reprocessing. Table 4-9 provides conservative estimates for residual radioactivity in the four underground waste tanks at the start of Phase 1 proposed decommissioning activities. These estimates were based on a comprehensive characterization program that made use of sample analytical data and radiation level measurements (WVNSCO and Gemini 2005)[7].

Table 4-9. Estimated Radioactivity in the Underground Waste Tanks(1)

Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)
Am-241	380	Np-237	0.55	Tc-99	12
C-14	0.036	Pu-238	170	U-232	0.90
Cm-243	3.6	Pu-239	39	U-233	0.34
Cm-244	80	Pu-240	28	U-234	0.14
Cs-137	310,000	Pu-241	630	U-235	0.005
I-129	0.018	Sr-90	33,000	U-238	0.039

NOTE: (1) From WVNSCO and Gemini 2005 and from WVES 2008c, corrected for decay and ingrowth to 2011 and rounded to two significant figures or a single integer.

The tank mobilization and transfer pumps are expected to contain significant amounts of radioactive contamination. Radiation levels near the bottom of Pump 55-G-003 exceeded 50 R/hr when this pump was removed in 1998 (WVNSCO 1998a). An order-of-magnitude estimate of the residual radioactivity in this removed pump was approximately 220 curies (WVNSCO 2001). The mobilization pumps remaining in the tanks will likely be similarly contaminated. The transfer pumps in Tanks 8D-1 and 8D-2 will likely have more contamination, since HLW passed through the entire length of the pump, rather than impacting only the lower portion as with the mobilization pumps. The other suction pumps in Tanks 8D-1 and 8D-2 that are described in Section 3 will likely have somewhat lower contamination levels than the mobilization and transfer pumps.

[7] These estimates addressed NRC comments provided on earlier characterization reports (NRC 2003). The characterization report (WVNSCO and Gemini 2005) included three different estimates: best case, conservative cases, and worst case. The conservative case on which Table 4-9 is based is considered to be conservative because it provides adequate safety margins, yet it is also considered to be realistic. The best and worst case estimates provide the lower and upper bounds on the realistic conservative case.

As explained in Section 3, the transfer pumps in Tanks 8D-3 and 8D-4 will be removed before Phase 1 of the proposed decommissioning and replaced with small submersible pumps. These submersible pumps are expected to contain much lower levels of contamination than the other transfer pumps.

The piping and equipment in the HLW transfer trench also contains significant amounts of residual radioactivity. Radiation levels measured in the trench in 2004 ranged from 0.6 to 9.6 mR/hr. Levels in the pump pits in 2003 ranged from background at the top of Pit 8Q-1 to 33.5 R/hr inside Pit 8Q-2. Conservative estimates indicated that the pump pits and the diversion pit contained approximately 440 curies and the transfer piping approximately 234 curies in 2004, with the dominant radionuclides being Cs-137, Sr-90, Am-241, Pu-241, and Cm-244, in that order. The transfer trench itself is not expected to be radiologically contaminated. (Fazio 2004b)

The equipment in the M-8 pump pit for Tank 8D-2 was estimated to contain approximately seven curies in 2004. Radiation levels up to 1.2 R/h were measured in the pit in 2000. (Fazio 2004b)

The Permanent Ventilation System Building is expected to contain a significant amount of activity inside the ventilation filter housing, but most other areas in the building typically show no removable contamination above minimum detectable concentrations.

In the Supernatant Treatment System Support Building, radiation levels as high as 8.2 R/hr were measured in the valve aisle in 2003. The valve aisle was conservatively estimated to contain 213 curies of residual radioactivity in 2004 (Fazio 2002c). Other areas of the building are not expected to contain significant radioactive contamination.

In the Equipment Shelter, most of the radiological inventory is expected to be located inside the ventilation system equipment. Radiation levels measured in 2003 ranged from 0.1 to 2.8 mR/hr. (Fazio, 2004b).

The Con-Ed Building is also radiologically contaminated, with the majority of the radiological inventory located inside the piping and equipment. Radiation levels measured in 2003 were typically 0.1 mR/hr. (Fazio, 2004b).

The total activity in the 40 underground lines in the immediate vicinity of the Waste Tank Farm has been estimated to be approximately 117 curies in 2004, with more than 99 percent of this activity associated with Cs-137 and Sr-90 (Luckett, et al. 2004).

Radioactivity in the Construction and Demolition Debris Landfill in WMA 4

Much of the buried waste in the landfill, which was not radioactive when it was emplaced, is now expected to have low-levels of radioactive contamination, mostly Sr-90, from the north plateau groundwater plume, which is addressed in Section 4.2.

Radioactivity in the Facilities in WMA 5, the Waste Storage Area

In WMA 5, Lag Storage Addition 4 and the attached shipping depot are expected to contain only low levels of radioactive contamination, if any. The Remote-Handled Waste

Facility is expected to contain only low levels of contamination after it is deactivated. Most of the residual radioactivity is expected to be in the Work Cell where high activity waste and equipment are being packaged for disposal.

Radioactivity in the Facilities in WMA 6, the Central Project Premises

The only facilities in WMA 6 that had been impacted by licensed radioactivity or the WVDP as of 2008 are the two demineralizer sludge ponds, which are addressed in Section 4.2, and the Cooling Tower basin. However, portions the Sewage Treatment Plant may contain radioactivity concentrations above background from sewage sludge which tends to concentrate naturally occurring radionuclides (ISCORS 2005).

Radioactivity in the NDA in WMA 7

The buried waste in the NDA is known to contain a large amount of radioactivity which has been estimated to total approximately 180,000 curies in 2011 as shown in Table 4-10.[8]

Table 4-10. Estimated Radioactivity in the NDA(1)

Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)	Nuclide	Estimate (Ci)
Am-241	2,000	Np-237	0.62	Tc-99	10
C-14	520	Pu-238	350	U-233	11
Co-60	7,000	Pu-239	580	U-234	0.71
Cs-137	29,000	Pu-240	400	U-235	0.13
H-3	35	Pu-241	9,100	U-238	1.5
I-129	0.022	Ra-226	0.039	-	-
Ni-63	110,000	Sr-90	22,000	-	-

NOTE: (1) From URS 2000, corrected for decay and ingrowth to 2011 and rounded to two significant figures.

Radioactivity in the Radwaste Treatment System Drum Cell in WMA 9

The Drum Cell – the only facility in WMA 9 and which is to be removed during Phase 1 – is expected to contain only low levels of residual radioactivity, if any.

WMA 10, the Support and Services Area

None of the facilities to remain within WMA 10 at the time the Phase 1 proposed decommissioning activities begin had been impacted by site radioactivity as of 2008.

[8] This table, which is the same as Table 2-21 in Section 2, is included here for completeness.

4.2 Radiological Status of Environmental Media

Section 4.2 describes the radiological status of surface soil, sediment, subsurface soil, surface water, and groundwater within the project premises as compared with background.

NOTE

Environmental media have not been fully characterized and, as a result, certain information normally included in decommissioning plans is not available. Additional characterization is planned in connection with the Phase 1 decommissioning work as described in Sections 7 and 9.

Additional characterization of subsurface soil is also being undertaken in 2008. This characterization is focusing on hazardous contaminants and radionuclides in the area of the north plateau groundwater plume (Michalczak 2007). DOE plans to provide a copy of the summary report of this characterization program to NRC and other involved agencies and to revise this plan to incorporate key data from this program.

The information provided below represents a compilation of environmental radiological data collected as part of the routine WVDP Environmental Monitoring and Groundwater Monitoring programs. It also includes data from nonroutine investigations designed to satisfy regulatory requirements (e.g., RCRA facility investigations) and other focused sampling activities.

Section 2.3 contains information on documented spills of radioactivity that have impacted environmental media on the project premises. These spills include the 1968 airborne radioactivity releases that produced the widespread area of surface contamination northeast of the Process Building known as the cesium prong and the release of radioactive acid under the southwest corner of the Process Building that resulted in the area of subsurface soil and groundwater contamination known today as the north plateau groundwater plume. This section focuses on environmental media conditions that exist today and duplicates information in Section 2.3 only where necessary for clarity.

Information in Section 4.2 is organized as follows:

- Section 4.2.1 identifies data sources used for this evaluation.
- Section 4.2.2 summarizes background levels of (1) radionuclide concentrations in surface soil, subsurface soil, stream sediment, surface water, and groundwater; and (2) environmental radiation.
- Section 4.2.3 summarizes radiological status of surface soil and sediment within the project premises.
- Section 4.2.4 provides the same information on subsurface soil.
- Section 4.2.5 summarizes maximum radionuclide concentrations at locations in each WMA where background levels were exceeded in soil, sediment, and subsurface soil.

- Section 4.2.6 provides information on environmental radiation levels on the project premises.
- Section 4.2.7 provides information on the radiological status of surface water on the project premises.
- Section 4.2.8 addresses the radiological status of groundwater on the project premises and, in particular, the north plateau groundwater plume.

Appendix B, *Environmental Radioactivity Data*, provides the following information:

- A description of how background radionuclide concentrations and environmental radiation levels were estimated;
- Maps showing locations where background data were taken;
- Summary statistics applicable to each medium;
- A description of how data from onsite sampling programs were evaluated to determine if radiological concentrations or environmental radiation levels were above background;
- Tables summarizing the ratios of above-background concentrations of radionuclides with Cs-137 in surface soil, sediment, and subsurface soil;
- Additional summary information about radiological concentrations from routine onsite sampling locations;
- Descriptions both impacted and non-impacted locations; and
- Tables that list the coordinates and descriptions of groundwater sampling locations, along with the depths and geologic units at which samples were collected.

4.2.1 Data Sources

Radiological data on surface soil, sediment, subsurface soil, surface water, groundwater, and environmental radiation levels were taken from the WVDP Laboratory Information Management System controlled database, which contains environmental data from 1991 through the present. This system is used to manage data from the WVDP Environmental Monitoring and Groundwater Monitoring Programs, as well as data from special sampling activities (e.g., RCRA facility investigations, north plateau groundwater plume investigations).

If necessary (i.e., if only pre-1991 data were available for an area), data were drawn from historical sources or summaries included in reports from previous evaluations.

Previous Evaluations

Radiological data from environmental media have been presented in formal reports, for example:

(1) WVDP Annual Site Environmental Reports (years 1982 through 2006 available on the Internet at www.wv.doe.gov);

(2) Groundwater trend analysis reports;

(3) Reports of RCRA facility investigations of various areas of the WVDP (WVNSCO 1995, WVNSCO 1996, WVNSCO 1997a, WVNSCO 1997b, WVNSCO and Dames & Moore [D&M] 1996a, WVNSCO and D&M 1996b, WVNSCO and D&M 1997a, WVNSCO and D&M 1997b, and WVNSCO and D&M 1997c); and

(4) Results from north plateau groundwater plume investigations (Carpenter and Hemann 1995, WVNSCO 1998, and URS 2002). The RCRA Facility Investigations and the north plateau investigations produced a substantial body of soil characterization data, most associated with nonradiological constituents.

Data Quality

WVDP environmental samples evaluated in this plan were collected in accordance with formal sampling plans. Samples were analyzed by onsite and offsite laboratories in accordance with controlled procedures as required by the WVDP quality assurance (QA) program. QA requirements applicable to the sampling programs include documented training of field personnel; controlled collection procedures; using appropriate containers, preservatives, and storage methods to protect samples from contamination and degradation; following appropriate field and analytical quality control guidelines; maintaining and documenting chain-of-custody; and conducting assessments and audits of field and analytical processes to verify compliance.

Data were validated by a separate data validation group, and validation and approval status of sample results were documented in the LIMS.

4.2.2 Background Levels

This subsection addresses background radioactivity in environmental media on the project premises and provides information on background radiation levels.

Background Radionuclide Concentrations in Environmental Media

Radionuclides for which backgrounds were estimated were selected with consideration of (1) radionuclides of interest from the Facility Characterization Project, listed in section 4.1.1, and (2) radionuclides that are routinely monitored in environmental media at the WVDP, for which sufficient data were available to develop a reliable estimate of background.

Background radionuclide concentrations were estimated for soil, sediment, subsurface soil, surface water, and groundwater for the following radionuclides:

Sr-90	U-232	U-235/236	Pu-238	Am-241
Cs-137	U-233/234	U-238	Pu-239/240	

Pu-241, Cm-243, Cm-244, and Np-237, which are radionuclides of interest in the Facility Characterization Project, are not routinely measured in environmental media at the WVDP so were not included in background estimates.

In addition, background concentrations were estimated for surface water and groundwater for the following nuclides that were not routinely analyzed in soil and sediment:

H-3 C-14 Tc-99 I-129

Although tritium (H-3) is not identified in Section 4.1.1 as a radionuclide of interest, it is commonly found in surface water and groundwater samples at the WVDP and so was included in the nuclide listing for environmental media. In addition, gross alpha and gross beta measurements are routinely used as screening (i.e., "surrogate" or "indicator") parameters for other nuclides, so background concentrations were estimated for gross alpha and gross beta activity. (For instance, gross beta measurements are used as a surrogate for Sr-90 measurements in the WVDP Groundwater Monitoring Program.)

Appendix B provides maps showing locations from which background data were taken and a description of how background concentrations were estimated. Appendix B also includes a table of summary statistics (e.g., number of samples, percentage of nondetect values, average concentrations, medians) for each constituent in each medium.[9] Median and maximum background concentrations are summarized in Table 4-11.

Table 4-11. Median and Maximum[(1)] Background Concentrations for Environmental Media at the WVDP

Constituent	Surface soil[(2)] (pCi/g dry)	Sediment (pCi/g dry)	Surface water (pCi/L)	Groundwater (pCi/L)
Gross alpha	1.3E+01 (2.7E+01)	9.2E+00 (2.2E+01)	<9.6E-01 (5.4E+00)	<2.6E+00 (2.2E+01)
Gross beta	2.0E+01 (4.0E+01)	1.6E+01 (2.7E+01)	2.3E+00 (2.0E+01)	4.6E+00 (2.8E+01)
H-3	NA	NA	<8.2E+01 (6.3E+02)	<8.6E+01 (9.4E+02)
C-14	NA	NA	<1.3E+01 (4.1E+02)	<2.7E+01 (7.4E+00)
Sr-90	9.5E-02 (3.1E+00)	<3.4E-02 (1.6E-01)	9.0E-01 (1.2E+01)	2.4E+00 (7.4E+00)
Tc-99	NA	NA	<1.8E+00 (7.3E+00)	<1.8E+00 (4.0E+00)
I-129	NA	NA	<7.9E-01	<6.0E-01

[9] Note that if a data set is symmetric, the average (i.e., mean) and the median will be the same. However, if the distribution is skewed to the right (i.e., contains a large number of low values and a few high values), the average will usually be higher than the median. For this reason, the median may be the more reliable estimator of central tendency. In this evaluation, both were estimated and are presented in Appendix B.

Table 4-11. Median and Maximum[1] Background Concentrations for Environmental Media at the WVDP

Constituent	Surface soil[2] (pCi/g dry)	Sediment (pCi/g dry)	Surface water (pCi/L)	Groundwater (pCi/L)
			(2.0E+00)	(1.6E+00)
Cs-137	4.2E-01 (1.2E+00)	3.8E-02 (7.8E-02)	<4.2E+00 (1.0E+01)	<2.2E+01 (1.9E+01)
U-232	<2.4E-02 (1.9E-02)	<3.1E-02 (3.9E-02)	<4.3E-02 (2.6E-01)	<4.9E-02 (3.8E-01)
U-233/234	7.9E-01 (9.4E-01)	6.6E-01 (8.6E-01)	9.9E-02 (3.0E-01)	1.6E-01 (8.2E+00)
U-235/236	5.2E-02 (2.2E-01)	4.6E-02 (2.8E-01)	<3.3E-02 (1.0E-01)	<5.0E-02 (1.9E-01)
U-238	7.9E-01 (9.3E-01)	6.5E-01 (9.0E-01)	5.7E-02 (4.0E-01)	1.2E-01 (5.3E+00)
Pu-238	<1.2E-02 (4.0E-02)	<1.4E-02 (1.3E-01)	<3.1E-02 (1.0E-01)	<4.6E-02 (2.2E-01)
Pu-239/240	1.6E-02 (2.3E-01)	<1.2E-02 (6.1E-02)	<2.7E-02 (2.0E-01)	<5.3E-02 (2.7E-01)
Am-241	<1.6E-02 (1.9E-01)	<1.4E-02 (8.6E-02)	<3.3E-02 (2.2E+00)	<3.8E-02 (1.8E-01)

NOTE: (1) Maxima are in parentheses. Maxima were selected from samples in which the radionuclide was detected (i.e., a "nondetect" result, indicated by a "<" sign, was used only if no detectable results were available).

(2) Data from only two subsurface locations sampled in 1993 were available for calculation of subsurface soil background concentrations. Therefore, surface soil backgrounds were used to evaluate subsurface soil data. (For comparability, data from the subsurface soil samples are summarized in Appendix B.)

LEGEND: NA = Not analyzed in this medium

Data on radionuclide concentrations in environmental media on the project premises were evaluated to determine the locations where radionuclide concentrations in excess of site background levels were found. Methods for evaluating sample data with respect to background were dependent on the type of data available for comparison (e.g., a single sample result, a data set encompassing several years). Methods for each are described in Appendix B.

Data evaluated in Section 4.2 were taken from samples collected over several years. While the majority of data points were from 1991 through the present, the earliest was from

a sample collected in 1967.[10] In Section 4.1, radionuclide activities in facilities on the project premises were decay-corrected to the year 2011. However, in Section 4.2 no attempt was made to decay-correct results from environmental samples because, unlike process cells or tanks, environmental media are not closed, static systems.

Media such as surface soil, sediment, subsurface soil, surface water, and groundwater are all subject to forces (aside from radioactive decay) with the potential to modify their radionuclide concentrations. Forces such as weathering, biological activity, atmospheric fallout, surface water runoff, wind erosion, and evaporation may act to deposit or remove radionuclides from a medium. Also, radionuclides are affected differentially by these mechanisms (e.g., Sr-90 is more mobile in water than Cs-137, which is more likely to bind to clay particles in soil and sediment).

Many of the radionuclides considered in this section are long-lived and it is unlikely that decay-correction would have affected the determination of whether or not background concentrations were exceeded. However, it is possible that estimates of radiological concentrations of the shorter-lived radionuclides (i.e., tritium [half-life of 12.3 years], Sr-90 [half-life of 28.9 years], and Cs-137 [half-life of 30 years]) are conservatively high, that is, overestimates.

NOTE

A soil characterization program will be undertaken in 2008. One of the goals will be to establish background soil concentrations. The interpretations in the following sections may be revised based on the results of this sampling program.

Background Environmental Radiation Levels

Radiation levels have been measured at the WVDP from 1986 through the present with a network of environmental thermoluminescent dosimeters (TLDs).[11] Average quarterly exposure measurements from four background locations over this time period was 19.3 mR per quarter (about 8.8E-03 mR/h). The maximum for any single quarter was 35 mR/quarter (about 1.6E-02 mR/h).

Background environmental radiation levels were used to evaluate measurements from onsite TLDs near process facilities, waste storage areas, and burial areas. (See Appendix B for a map showing the locations of background TLDs. See section 4.2.6 for a discussion of onsite exposure measurements.)

[10] Note that historical and current data, which were generated over more than 40 years of NFS and WVDP operations, may not be directly comparable because different sampling and analytical methodologies have been used over the years. Historical and current data were compared with background concentrations using different statistical methods, as described in Appendix B.

[11] While radiation levels were measured at the WVDP prior to 1986, the current methodology has been used only since 1986. Therefore, for comparability, only data generated from 1986 through the present were used in the background calculation.

4.2.3 Radiological Status of Surface Soil and Sediment

Since the facility has operated, numerous soil sampling studies have been conducted on site, not as part of a formal site-wide soil program, but rather as area-specific investigations in response to specific circumstances or events (WVNSCO 1994). In 1993, a site-wide soil sampling program was conducted to obtain additional data to support the EIS and RCRA processes. As part of this program, surface soil, sediment, and subsurface soil samples were collected. Results were summarized in WVNSCO 1994.

NUREG-1757 (NRC 2006) defines surface soil as the soil within the top 15 to 30 cm (six to 12 inches) of the soil column. That definition has been broadened in this plan to include soil within the top 60 cm (0 to two feet) of the soil column. This was done so that available data from the top interval (0 to two-foot depth) from onsite soil-borings collected as part of the 1993 program could be used to assess the radiological status of surface soil. Data from the subsurface portions of the boreholes (i.e., at depths greater than two feet) are discussed in section 4.2.5.

Areas With Radionuclide Concentrations in Excess of Site Background Levels

Figure 4-6 shows locations at which radiological concentrations exceeding background were noted in surface soil and sediment for (1) gross alpha or alpha-emitting radionuclides and (2) gross beta or beta-gamma emitting radionuclides.[12]

- The highest radionuclide concentrations were found in sediment from the lagoons in the WMA 2 Low-Level Waste Treatment Facility. See Table 4-14 for a listing of maximum radionuclide concentrations above background noted in the lagoon and drainage system. The highest radionuclide concentrations were noted in sediment from Lagoon 2. (Although higher concentrations are listed for Lagoon 1, the Lagoon 1 sediment was transferred to Lagoon 2 when Lagoon 1 was deactivated in 1984.)
- Cs-137 concentrations in excess of background were found in surface soil samples from all waste management areas at which samples had been collected. Although no surface soil data were available from WMA 1 (the Process Building and Vitrification Facility area), it is suspected that radionuclide concentrations in excess of background would be found here based on proximity to the Process Building and the elevated concentrations observed in adjoining WMAs. The highest levels noted in surface soil from other areas (i.e., 2.8E+02 pCi/g in WMA 2 near the Interceptors, 1.6E+02 pCi/g in WMA 6 near the Fuel Receiving and Storage Area and 2.3E+01 pCi/g in WMA 3 near the Waste Tank Farm) were all from areas in closest proximity to WMA 1. Elevated Cs-137 concentrations are thought to be largely attributable to historical releases and continuing low-level airborne releases from the main stack of the Process Building.

[12] WMA 12 is not labeled on the figures in this section because it extends to the boundaries of the Center. Areas on the project premises (i.e., within the security fence) that are considered to be part of WMA 12 include (1) the area between the north and south plateaus, which contains much of the drainage for Erdman Brook and Franks Creek, and (2) a small area north of WMA 4.

- Surface soil concentrations of Sr-90 exceeding background were noted in several areas, most notably in areas affected by the north plateau groundwater plume, such as WMA 2 (the Low-Level Waste Treatment Facility area) and WMA 4 (the area of the Construction and Demolition Debris Landfill).
- Radionuclide concentrations exceeding background, primarily from Sr-90 and Cs-137, were found in sediment samples from streams and drainage ditches in several waste management areas (WMAs 2, 4, 5, 6, 7, 10, and 12). Concentrations of alpha-emitting radionuclides (i.e., U-232, Pu-238, Pu-239/240, and/or Am-241) in excess of background were also noted in WMAs 2, 4, 5, 7, and 12 downgradient of liquid release points or waste burial areas.
- High radionuclide concentration levels were also associated with soil and sediment from the area of the Old Interceptors, the Solvent Dike, and inactive (filled-in) Lagoon 1 in WMA 2.
- South plateau areas with radionuclide concentrations exceeding background in surface soil include the two former shallow land burial disposal facilities, the NDA (WMA 7) and SDA (WMA 8). Elevated radiological concentrations in the surface and near-surface soils in the vicinities of those facilities is expected due to the nature of their operations. (As noted previously, WMA 8 is not within plan scope.)

Levels at which radionuclide concentrations in excess of background were found in surface soil and sediment are listed by WMA in the tables in section 4.2.5. As shown in Figure 4-6, only one surface soil sampling location (SS-11) had no concentrations exceeding background. All sediment sampling locations had at least one constituent exceeding background.

4.2.4 Radiological Status of Subsurface Soil

Figure 4-7 shows locations at which concentrations of radiological constituents above background were noted in subsurface soil for (1) gross alpha or alpha-emitting radionuclides and (2) gross beta or beta-gamma emitting radionuclides. All subsurface soil samples had at least one constituent that exceeded background concentrations.

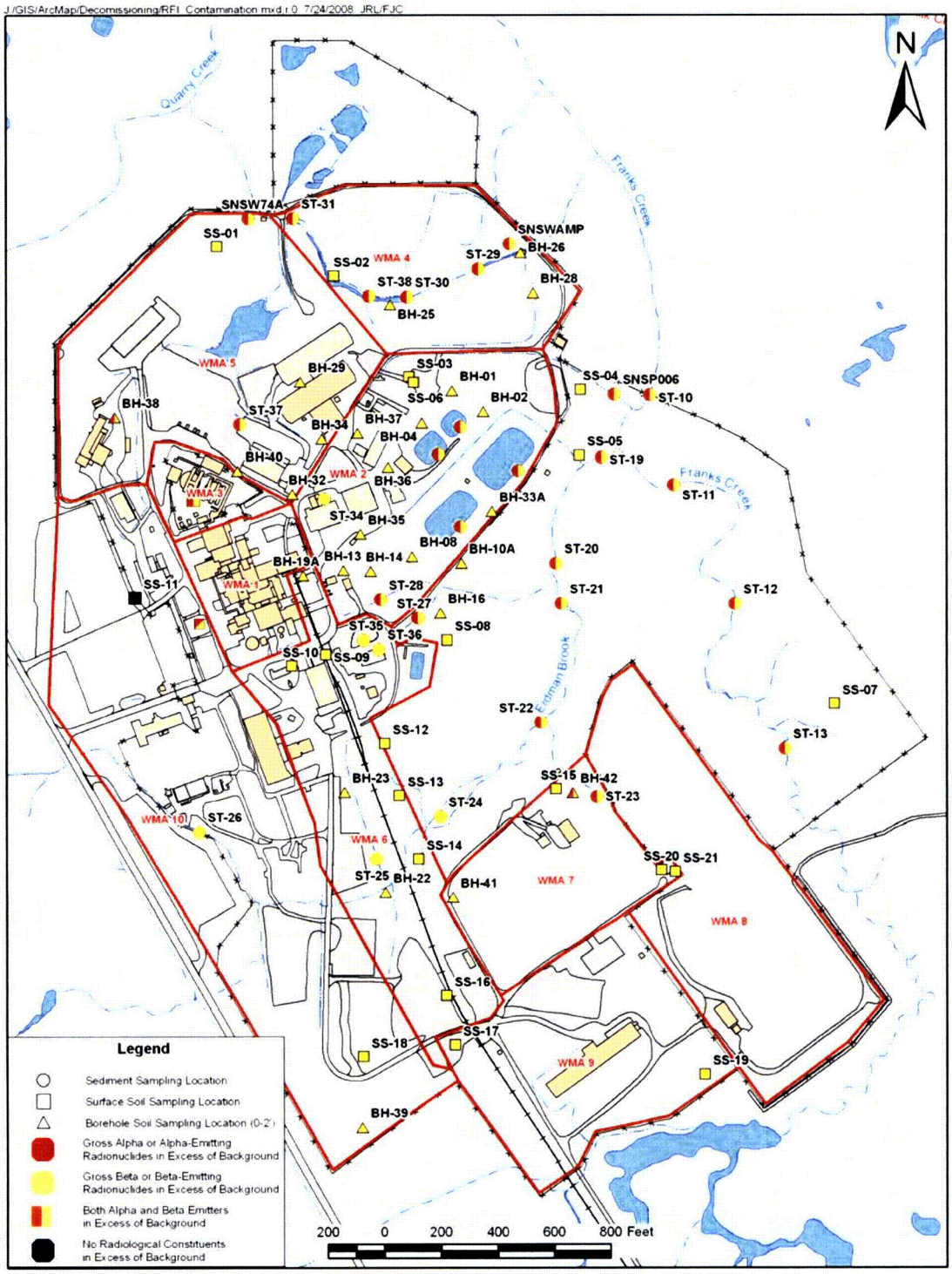

Figure 4-6. Surface Soil and Sediment Locations With Radionuclide Concentrations in Excess of Background

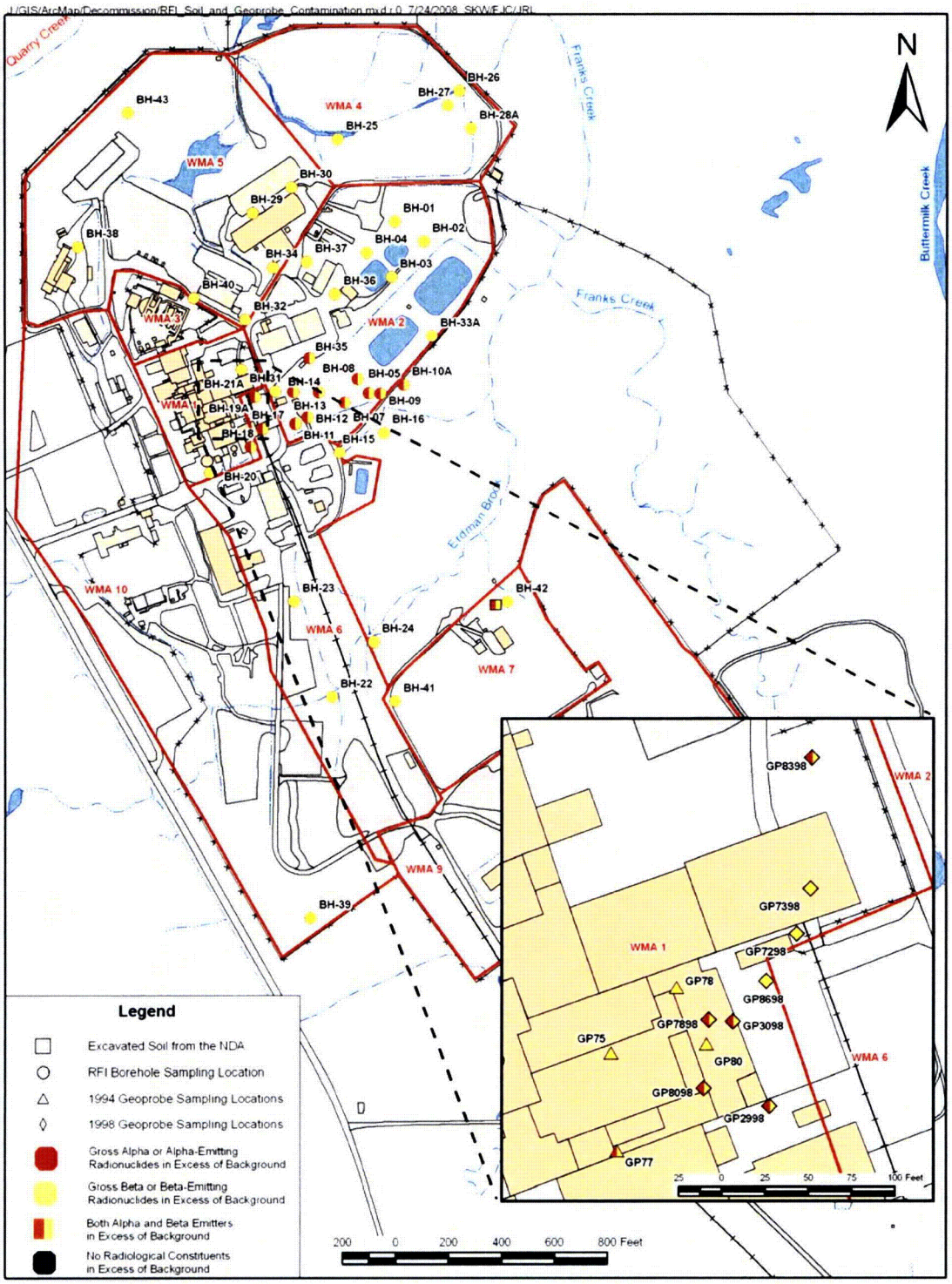

Figure 4-7. Subsurface Soil Locations With Radionuclide Concentrations in Excess of Background

Most subsurface soil data were taken from the 1993 RCRA Facility Investigation sampling program and two Geoprobe® sampling efforts (one in 1994 and one in 1998) to better define the origin and extent of the north plateau groundwater plume.

The highest subsurface radiological concentrations on the north plateau were observed in WMA 1 (the Process Building and Vitrification Facility area), WMA 2 (the Low-Level Waste Treatment Facility area), and WMA 6 (the Central Project Premises), downgradient of the Process Building. On the south plateau, highest concentrations were from WMA 7 (the NDA). Subsurface soil concentrations exceeding background were primarily associated with the north plateau groundwater plume (see Section 2) or with former waste processing or burial activities. Figure 4-8 presents a cross-section of Sr-90 concentrations in subsurface soil with depth in the north plateau below the Process Building. Data from this cross-section were taken from samples collected in 1993, 1994, and 1998 from WMAs 1, 2, and 6. The highest concentrations of Sr-90 were observed in the sand and gravel unit below the water table.

In WMA 1, high levels of Sr-90 were measured during the Geoprobe® investigations near the Process Building. In WMA 2, the highest levels of both beta-gamma and alpha-emitting radionuclides in subsurface soil were observed in sediments from borings taken near the Solvent Dike, the interceptors, and the Maintenance Shop leach field. In WMA 6, elevated subsurface soil concentrations were noted near the Utility Room and the Fuel Receiving and Storage Building. Data from WMA 7 were taken from rolloffs and boxes of excavated soil removed from "special holes" during NDA burial activities and from the Interceptor Trench, immediately downgradient of the NDA, when it was installed in 1990. Although the packaged soil has since been shipped offsite, it is likely that radionuclide concentrations in subsurface soil remaining in the NDA would be similar to those from the excavated soil.

Concentrations of radionuclides observed in excess of background levels in subsurface soils are summarized in Section 4.2.5.

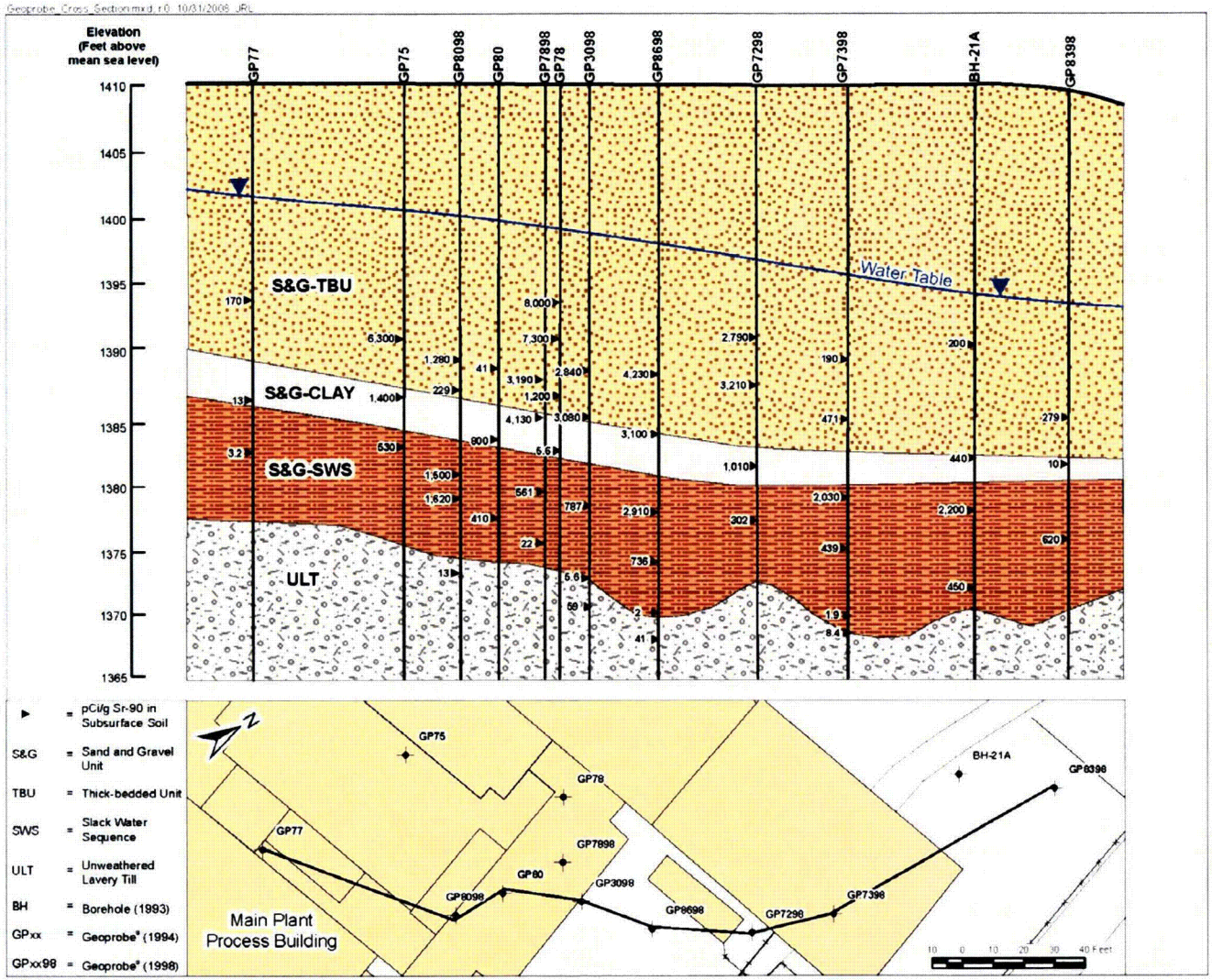

Figure 4-8. Cross-section of Sr-90 Concentrations Versus Depth in Subsurface Soil on the North Plateau

4.2.5 Radionuclide Concentrations Exceeding Background in Surface Soil, Sediment, and Subsurface Soil By WMA

The following tables summarize locations in each WMA where radionuclide concentrations were noted in excess of background. (See Table 4-11 and Appendix B for background radionuclide concentrations used to evaluate soil, sediment, and subsurface soil.) Data from surface soil, sediment, and subsurface soil are combined into one table for each WMA, except for WMA 2, where data are presented in three tables due to the large volume of information.

For each area, the maximum concentration at which the radionuclide was found is listed, together with source and location (i.e., reference or specific sample identifier). Identifiers from the 1993 RCRA Facility Investigation sampling program are specified as boreholes ("BH-"), surface soil ("SS-") or stream sediment ("ST-"). For subsurface soil, the depth at which the maximum was noted (if available) is also provided. Gross alpha and gross beta measurements are not presented because the measurements represent a mix of radionuclides (including those naturally occurring), and because data for specific alpha- and beta-emitting radionuclides were available. Ratios of above-background radionuclide concentrations to Cs-137 are presented in Appendix B in Tables B-9 (Surface Soil), B-10 (Sediment), and B-11 (Subsurface Soil).

WMA 1, Process Building and Vitrification Facility Area

Limited data are available for WMA 1, none for surface soil or sediment. Most subsurface soil data were taken from the 1994 and 1998 Geoprobe® investigations. Data from the Fuel Receiving and Storage Building subsurface soil and from near the Laundry were taken from the 1993 RCRA Facility Investigation sampling. Additional data were taken from one sample collected in 2004 near a breach in an underground wastewater line near the Laundry. The maximum Sr-90 value at the Fuel Receiving and Storage Building borehole (BH-21A) was observed in saturated soil from a depth of 30 to 32 feet. All maxima from the Geoprobe® samples were found at depths between 19 and 29 feet. Records from the 1998 sampling noted that this depth was in the saturated area. The ratio of the maximum Sr-90 to Cs-137 concentrations noted from the Geoprobe® samples was about 1600 to 1, reflecting the influence of the north plateau groundwater plume.

Table 4-12. Above-Background Concentrations of Radionuclides in Subsurface Soil at WMA 1[(1)]

Location	Maximum Concentration (pCi/g dry)			
	Cs-137	Sr-90	Pu-238	Am-241
Subsurface soil near Laundry (BH-18, Laundry line breach)	3.3E+03	≤Bkg	7.1E-02	8.7E+01
Subsurface soil north of FRS (BH-21A at a depth of 30-32')	≤Bkg	2.2E+03	≤Bkg	≤Bkg

Table 4-12. Above-Background Concentrations of Radionuclides in Subsurface Soil at WMA 1[(1)]

Location	Maximum Concentration (pCi/g dry)			
	Cs-137	Sr-90	Pu-238	Am-241
Subsurface soil from Geoprobe® sampling near Process Building (GP-72 [Cs-137 at 27-29' depth in 1998] ; GP-78 [Sr-90 at 19-23' depth in 1994]; GP-77 [Am-241 at 19-23' depth in 1994])	3.1E+01	8.0E+03	≤Bkg	9.8E-02

LEGEND: FRS = Fuel Receiving and Storage Building. "≤Bkg" = Background was not exceeded.

NOTE: (1) See Figure 4-2 for a map of facilities in WMA 1. The Laundry (not labeled in Fig. 4-2), is located adjacent to the Utility Room Expansion.

WMA 2, Low-Level Waste Treatment Facility Area

Extensive data, available both electronically and from historical reports, were available for WMA 2. The maximum concentrations observed at each location within WMA 2 are listed below. Due to the large volume, data are presented in three tables: Table 4-13 (surface soil), Table 4-14 (sediment), and Table 4-15 (subsurface soil).

The radionuclides observed above background in surface soil (Table 4-13) were Cs-137 and Sr-90. The maximum ratio of Sr-90 to Cs-137 (about 1.4 to 1) was observed in surface soil north of Lagoons 4 and 5, which is affected by the north plateau groundwater plume. No gross alpha concentrations or concentrations of alpha-emitting radionuclides were observed at concentrations above background in surface soil from WMA 2.

Table 4-13. Above-Background Concentrations of Radionuclides in Surface Soil From WMA 2[(1)]

Location	Maximum Concentration (pCi/g dry)	
	Cs-137	Sr-90
Surface soil near the Old and New Interceptors (BH-13)	2.8E+02	4.1E+00
Surface soil between the Interceptors and inactive Lagoon 1 (WVNSCO 1994 [Table 3-2] and BH-14)	1.4E+01	1.4E+00
Surface soil between inactive Lagoon 1 and active Lagoon 2 (BH-08)	4.8E+00	1.1E+00
Surface soil from Maintenance Shop Leach Field (WVNSCO 1994 [Table 3-2] and BH-35)	2.1E+01	1.3E+00
Surface soil near the LLW2 Facility (BH-36)	≤Bkg	3.2E-01
Surface soil near the Vitrification Test Facility (BH-37)	6.6E-01	≤Bkg

Table 4-13. Above-Background Concentrations of Radionuclides in Surface Soil From WMA 2[1]

Location	Maximum Concentration (pCi/g dry)	
	Cs-137	Sr-90
Surface soil north of Lagoons 4 and 5 (BH-04)	8.5E-01	1.2E+00
Surface soil between the lagoons and WMA 4 (SS-03, SS-06)	3.6E+00	3.6E-01
Surface soil between the road and Lagoon 2 (BH-33A)	8.9E-01	≤Bkg

LEGEND: "≤Bkg" = Background was not exceeded.

NOTE: (1) See Figure 4-3 for a map of facilities in WMA 2. Facilities not labeled in Fig. 4-3 include the former Maintenance Shop (which was located southwest of the LLW2 Facility), and the Vitrification Test Facility (located northwest of the LLW2 Facility). See Figure 4-6 for a map with the above sampling locations.

Radionuclides observed above background in sediment (Table 4-14) were Cs-137, Sr-90, U-232, U-233/234, U-235/236, U-238, Pu-238, Pu-239/240, and Am-241. Maximum ratios to Cs-137 for each were: Sr-90 (144 to 1), U-232 (0.0054 to 1), U-233/234 (0.056 to 1), U-235/236 (0.011 to 1), U-238 (0.057 to 1), Pu-238 (0.018 to 1), Pu-239/240 (0.019 to 1), and Am-241 (4.2 to 1). (See Appendix B, Table B-10, for a summary of radionuclide ratios in sediment from WMA 2.)

Maximum ratios to Cs-137 were found in sediment from (or downgradient of) the Solvent Dike (Sr-90, U-233/234, U-235/236, Pu-239/240, and Am-241), sediment from Lagoon 3 (U-232 and U-238), and sediment from the Lagoon 2 shoreline (Pu-238). The highest Am-241 to Cs-137 ratio (4.2 to 1) was from one Solvent Dike sediment sample collected in 1986. For comparison, the median Am-241 to Cs-137 ratio in WMA 2 was 0.0019 to 1.

Table 4-14. Above-Background Concentrations of Radionuclides in Sediment From WMA 2

Location	Maximum Concentration (pCi/g dry)								
	Cs-137	Sr-90	U-232	U-233/234	U-235/236	U-238	Pu-238	Pu-239/240	Am-241
Sediment from drainage north of Test and Storage Building (ST-34)	2.0E+00	3.5E-01	NA	NA	NA	NA	NA	NA	NA
Sediment from Solvent Dike (WVNSCO 1994, Table 3-12, 1986 samples)	3.1E+02	1.6E+03	NA	NA	NA	NA	NA	NA	1.1E+03
Sediment from drainage downgradient of Solvent Dike (ST-28)	1.7E+01	2.9E+00	≤Bkg	9.5E-01	≤Bkg	≤Bkg	2.9E-01	3.2E-01	7.1E-01

Table 4-14. Above-Background Concentrations of Radionuclides in Sediment From WMA 2

Location	Maximum Concentration (pCi/g dry)								
	Cs-137	Sr-90	U-232	U-233/234	U-235/236	U-238	Pu-238	Pu-239/240	Am-241
Sediment from Lagoon 1 (Passuite and Monsalve-Jones 1993, Tables 3-2 [1982 data] and 3-3 [1984 data])	4.7E+05	1.5E+05	NA	NA	NA	NA	3.9E+04	1.8E+04	1.9E+04
Sediment from Lagoon 2(1) (WVNSCO 1994, Tables 3-5 [1982 data] and 3-8 [1990 data])	2.7E+05	3.6E+04	NA	NA	6.5E-01	6.2E+00	8.0E+02	6.4E+02	8.3E+02
Sediment from Lagoon 3 (WVNSCO 1994, Tables 3-11 [1990 data], 3-9 [1967 data]; and 1994 Lagoon 3 sampling)	1.1E+04	7.7E+02	7.6E+00	4.5E+00	1.3E+00	8.8E+00	3.1E+00	1.4E+00	5.1E+00
Sediment from Lagoon 4 (1994 sampling)	3.2E+01	7.3E+00	NA	NA	NA	NA	NA	NA	NA
Sediment from Lagoon 5 (1994 sampling)	5.2E+01	4.1E+01	NA	NA	NA	NA	NA	NA	NA

NOTE: (1) In 1984, an estimated 22,400 cubic feet of sediment were pumped from Lagoon 1 to Lagoon 2 (Passuite and Monsalve-Jones 1993) so the 1982 sample results are not necessarily representative of the of the activity in Lagoon 2 sediment.

(2) See Figure 4-3 for a map of facilities in WMA 2. The Test and Storage Building (which was located near the southwestern boundary of WMA 2) is not labeled in Fig. 4-3. See Figure 4-6 for a map with the above sampling locations.

LEGEND: NA = No analysis. "≤Bkg" = Background was not exceeded.

Radionuclides observed above background in subsurface soil (Table 4-15) were Cs-137, Sr-90, U-232, U-233/234, U-235/236, U-238, Pu-238, Pu-239/240, and Am-241. The highest concentrations were noted downgradient of inactive Lagoon 1, near the Old and New Interceptors, near the Solvent Dike, and near the Maintenance Shop Leach Field.

All maxima downgradient of inactive Lagoon 1 were found in saturated soil from the six to eight foot depth. Near the Solvent Dike, the Cs-137 maximum was found in moist soil at the four to six foot depth, while maxima of the other radionuclides were found in saturated soil from the eight to ten foot depth. Near the Old and New Interceptors, maxima were located in saturated soil from the eight to ten foot depth.

The maximum ratios to Cs-137 for each were: Sr-90 (78 to 1), U-232 (0.081 to 1), U-233/234 (5.0 to 1), U-235/236 (0.74 to 1), U-238 (3.1 to 1), Pu-238 (0.089 to 1), Pu-239/240 (0.10 to 1), and Am-241 (0.15 to 1).

The maximum ratios to Cs-137 were found in subsurface soil near the Solvent Dike at the eight to 10 feet depth (Sr-90, U-233/234, U-235/236, U-238, Pu-239/240, and Am-241), from the Maintenance Shop leach field at the six to eight feet depth (U-232), and from between the interceptors and inactive Lagoon 1 at the 14-16' depth (Pu-238).

Table 4-15. Above-Background Concentrations of Radionuclides in Subsurface Soil From WMA 2[1]

Location	Maximum Concentration (pCi/g dry)								
	Cs-137	Sr-90	U-232	U-233/ 234	U-235/ 236	U-238	Pu-238	Pu-239/ 240	Am-241
Downgradient of inactive Lagoon 1 (BH-08 at 6-8' depth)	3.6E+04	1.5E+04	5.8E+02	2.7E+02	4.2E+00	6.8E+01	6.8E+02	1.2E+03	1.7E+03
Near Solvent Dike (BH-11 at 8-10' depth, Cs-137 max at 2-4' depth)	1.8E+02	5.6E+01	≤Bkg	3.6E+00	5.3E-01	2.2E+00	≤Bkg	7.5E-02	1.1E-01
Near the Old and New Interceptors (BH-13, 8-10' depth)	5.2E+03	1.9E+02	5.1E+01	2.4E+01	2.0E-01	3.7E+00	6.6E+01	5.1E+01	5.3E+01
Between the Interceptors and inactive Lagoon 1 (4-6' depth, Pu-238 at 14-16' depth)	6.1E+00	2.8E+01	1.0E-01	≤Bkg	≤Bkg	≤Bkg	1.7E-01	1.9E-01	2.8E-01
Maintenance Shop Leach Field (BH-35, 6-8' depth)	1.6E+01	3.9E+02	1.3E+00	≤Bkg	≤Bkg	≤Bkg	4.6E-01	7.4E-02	1.3E+00
Near Lagoon 4 (BH-03, 12-14' depth)	≤Bkg	2.3E+00	NA	NA	NA	NA	NA	NA	NA
North of Lagoons 4 and 5 (BH-04, 16-18' depth)	≤Bkg	3.5E+00	NA	NA	NA	NA	NA	NA	NA
Near Old Interceptor (BH-31, 10-12' depth)	≤Bkg	1.0E+01	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg
Between the Vitrification Test Facility and LSA 4 (BH-37, 8-10' depth)	≤Bkg	1.1E+02	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg
Near LLW2 facility (BH-36, 20-22' depth)	≤Bkg	2.9E+01	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg	≤Bkg

LEGEND: NA = No analysis. "≤Bkg" = Background was not exceeded.

NOTE: (1) See Figure 4-3 for a map of facilities in WMA 2. Facilities not labeled in Figure 4-2 include the former Maintenance Shop (which was located southwest of the LLW2 Facility), and the Vitrification Test Facility (located northwest of the LLW2 Facility). See Figure 4-7 for a map with the above sampling locations.

WMA 3. High-level Waste Tank Farm

Minimal data were available for the Waste Tank Farm. Table 4-16 lists maximum concentrations of radionuclides found in surface soil at levels above background. Data were from a 1990 sampling, as summarized in Table 3-2 of WVNSCO 1994. Concentrations in excess of background levels were noted for Cs-137, U-238, and Am-241. The ratios of U-238 and Am-241 to Cs-137 in surface soil from the Waste Tank Farm were 0.047, and 0.011, respectively. No sediment or subsurface soil data were available, although subsurface soil concentrations exceeding background are expected because of leaks or breaches in transfer lines (see Section 2) and because of elevated radionuclide concentrations found in groundwater as discussed below.

Table 4-16. Above-Background Concentrations of Radionuclides in Surface Soil at WMA 3(1)

Location	Maximum Concentration (pCi/g dry)		
	Cs-137	U-238	Am-241
Surface soil at the Waste Tank Farm (WVNSCO 1994, Table 3-2 [1990 data])	2.3E+01	1.1E+00	2.5E-01

NOTE: (1) See Figure 4-4 for a map of facilities in WMA 3 and Figure 4-6 for a map showing areas with above-background levels of radionuclides in surface soil.

WMA 4, Construction and Demolition Debris Landfill Area

Concentrations of radiological constituents measured at levels in excess of background in surface soil, sediment, and subsurface soil from WMA 4 are listed in Table 4-17. Surface soil from WMA 4, a portion of which includes the landfill, was found to contain concentrations of Cs-137 and Sr-90 in excess of background. The maximum ratio of Sr-90 to Cs-137 in surface soil was about 9.5 to 1.

Table 4-17. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil From WMA 4(1)

Location	Maximum Concentration (pCi/g dry)						
	Cs-137	Sr-90	U-233/ 234	U-238	Pu-238	Pu-239/ 240	Am-241
Surface soil along drainage though CDDL (SS-02 and WVNSCO 1994, Table 3-2 [1990 data])	9.1E+00	1.2E+01	NA	NA	NA	NA	NA
Sediment from drainage through CDDL (ST-31, ST-38)	7.0E+00	8.4E+01	NA	NA	7.3E-02	7.4E-02	1.3E-01
Sediment from Northeast Swamp drainage (SNSWAMP)	3.1E+01	3.0E+01	1.1E+00	1.1E+00	4.3E-01	6.4E-01	1.3E+00
Subsurface soil in CDDL (BH-27 [Cs-137 max at 2-4'], BH-25 [Sr-90 max at 12-14'])	7.3E-01	4.1E+00	NA	NA	NA	NA	NA

LEGEND: CDDL = Construction and Demolition Debris Landfill; NA = No analysis.

NOTE: (1) See Figures 4-6 and 4-7 for maps showing locations with radionuclide concentrations in excess of background.

Sediment from drainage locations on WMA 4 also contained Sr-90 and Cs-137 at levels exceeding background. However, it also contained above-background levels of the alpha-emitting radionuclides U-233/234, U-238, Pu-238, Pu-239/240, and Am-241. Maximum radionuclide ratios to Cs-137 were: Sr-90 (16 to 1), U-233/234 (1.4 to 1), U-238 (1.3 to 1) Pu-238 (0.057 to 1), Pu-239/240 (0.21 to 1), and Am-241 (0.22 to 1).

The maximum Sr-90 to Cs-137 ratio in sediment was noted from drainage through WMA 4 north of the landfill. The north plateau groundwater plume surfaces near ST-38 where this sample was taken (see Figure 4-6). Maximum ratios for the remaining nuclides were noted at the routine monitoring point SNSWAMP, which is located where drainage from WMA 4 leaves the site. Sediment (or soil, depending upon annual rainfall and drainage flow patterns) is collected at this location as part of the WVDP Environmental Monitoring Program. (See Appendix B for average and median radionuclide concentrations at the SNSWAMP location from 1995 through 2007.)

The comparatively high Sr-90 to Cs-137 ratios observed for surface soil and sediment in WMA 4 reflect the presence of Sr-90 in the north plateau groundwater plume.

Both Cs-137 and Sr-90 concentrations exceeding background were noted in subsurface soil from WMA 4. Because the landfill located on WMA 4 was not used for radioactive waste disposal, it was not thought to be the origin of the radionuclides. Cs-137 in subsurface soil is most likely leached from the overlying surface soil (the concentration of Cs-137 at the two to four feet depth was roughly one-tenth of the concentration at the surface). As seen in other areas, elevated levels of Cs-137 in surface soil were most likely attributable to airborne deposition (see Section 2). The maximum ratio of Sr-90 to Cs-137 for subsurface soil was about 0.73 to 1. As with the surface soil and sediment media, the north plateau groundwater plume is thought to be the origin of Sr-90 in subsurface soil in WMA 4.

WMA 5, Waste Storage Area

Concentrations of radiological constituents measured at levels in excess of background in surface soil, sediment, and subsurface soil from WMA 5 are listed in Table 4-18. Cs-137 and Sr-90 concentrations exceeding background were found in surface soil and sediment. Concentrations of the alpha-emitting radionuclides Pu-238, Pu-239/240, and Am-241 exceeding background were also found, possibly attributable to residual activity from the old/new hardstand, on which contaminated vessels and equipment from the Process Building had been stored when NFS was operating. Historical site surveys have noted elevated gamma radiation readings and soil contamination in the area of the old/new hardstand (Marchetti, 1982). Material from the hardstand was excavated and used to fill Lagoon 1 when it was closed in 1984. (See Section 2.)

Maximum ratios to Cs-137 in soil and/or sediment were: Sr-90 (3.3 to 1), Pu-238 (0.015 to 1), Pu-239/240 (0.096 to 1), and Am-241 (0.087 to 1). The maximum ratios were all found in sediment from the North Swamp drainage point SNSW74A.

No concentrations exceeding background of Cs-137 or alpha-emitting radionuclides were noted in subsurface soil samples from WMA 5. However, Sr-90 concentrations above background were found six to eight feet below-ground at a point between Lag Storage Addition 3 and Lag Storage Addition 4 and 22 to 24 feet below the surface at the southernmost point of WMA 5 near the Lag Storage Building.

Table 4-18. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 5(1)

Location	Maximum Concentration (pCi/g dry)				
	Cs-137	Sr-90	Pu-238	Pu-239/ 240	Am-241
Surface soil on north plateau near security fence (SS-01)	2.0E+01	3.7E-01	NA	NA	NA
Surface soil near Remote-Handled Waste Facility location (BH-38)	1.1E+01	8.2E-01	3.6E-02	1.6E-01	3.7E-01
Surface soil from footers for LSA 3 and LSA 4 (WVNSCO 1994, Table 3-15 [1990 data])	2.8E+01	NA	NA	NA	9.1E-01
Surface soil from the Lag Storage Building (BH-32)	7.8E-01	≤Bkg	≤Bkg	≤Bkg	≤Bkg
Sediment near old LSA 2 (ST-37)	6.1E+01	8.3E+00	≤Bkg	≤Bkg	6.5E-02
Sediment from north swamp drainage (SNSW74A)	8.8E+00	2.1E+00	≤Bkg	1.9E-01	2.6E-01
Subsurface soil between LSA 3 and 4 (BH-29, 6-8' depth)	≤Bkg	2.8E+00	NA	NA	NA
Subsurface soil by the lag storage building (BH-32, 22-24' depth)	≤Bkg	5.8E-01	≤Bkg	≤Bkg	≤Bkg

LEGEND: LSA = Lag Storage Addition. NA = No analysis. "≤Bkg" = Background was not exceeded.

NOTE: (1) See Figures 4-6 and 4-7 for maps showing locations with radionuclide concentrations in excess of background.

WMA 6, Central Project Premises

Concentrations of radionuclides measured at levels in excess of background in surface soil, sediment, and subsurface soil from WMA 6 are listed in Table 4-19. Cs-137 and Sr-90 were the only radionuclides found in concentrations exceeding background in surface soil and sediment from WMA 6. The highest concentrations of both Cs-137 and Sr-90 were found in surface soil collected near the Fuel Receiving and Storage Building.

The highest Sr-90 to Cs-137 ratio in surface soil (1.7 to 1) was also found in soil near the rail spur by the Fuel Receiving and Storage Building. The highest Sr-90 to Cs-137 ratio in sediment (0.59 to 1) was found in sediment from the south Demineralizer Sludge Pond.

The highest radionuclide concentrations in surface soil and sediment were from the northern portion of WMA 6, closest to the Process Building. However, elevated concentrations were also found along the rail spur south of the Sewage Treatment Plant. These elevated concentrations may be attributable to events in the 1960s and 1970s (e.g., increased radioactivity in treated effluents or possible line leaks [see further detail in Section 2.3.2]).

Subsurface soil samples – one from near the Utility Room and one from near the Fuel Receiving and Storage Building – contained Cs-137, Sr-90, Pu-238, Pu-239/240, and Am-241 concentrations exceeding background. The highest concentrations were found near the Fuel Receiving and Storage Building at a depth of 22 to 24 feet in the sand and gravel unit below the water table. (See Figure 4-8.) The maximum concentrations near the Utility Room were from 16 to 18 feet below the surface.

Ratios to Cs-137 for Pu-238, Pu-239/240, and Am-241 were similar for subsurface soil samples taken near the Utility Room and the Fuel Receiving and Storage Building (about 0.03 to 1, 0.04 to 1, and 0.2 to 1, respectively). However, the Sr-90 to Cs-137 ratios for each were strikingly different. Near the Utility Room, the ratio was about 1 to 1, but near the Fuel Receiving and Storage Building the ratio was 133 to 1, suggesting that the Fuel Receiving and Storage Building subsurface location was more central to the north plateau groundwater plume.

Table 4-19. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 6[1]

Location	Maximum Concentration (pCi/g dry)				
	Cs-137	Sr-90	Pu-238	Pu-239/240	Am-241
Surface soil along rail spur south of STP (BH-23, SS-13)	1.8E+00	3.2E-01	NA	NA	NA
Sediment along drainage by rail spur south of STP (ST-25)	2.1E+00	1.3E-01	NA	NA	NA
Surface soil by FRS (1994 sampling near rail spur)	1.6E+02	1.2E+01	NA	NA	NA
Surface soil by Cooling Tower (SS-10)	1.3E+01	1.4E+00	NA	NA	NA
Surface soil by Old Incinerator (WVNSCO 1994, Table 3-2 [1990 data])	1.9E+01	2.3E+00	NA	NA	NA
Surface soil by Old Warehouse (SS-09)	1.3E+01	9.3E-01	NA	NA	NA
Sediment from North Demineralizer Sludge Pond (WVNSCO 1994 Table 3-18 [1988 data], ST-35)	1.3E+01	7.7E-01	NA	NA	NA

Table 4-19. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 6[1]

Location	Maximum Concentration (pCi/g dry)				
	Cs-137	Sr-90	Pu-238	Pu-239/240	Am-241
Sediment from South Demineralizer Sludge Pond (WVNSCO 1994 Table 3-19 [1988 data], ST-36)	3.8E+01	3.5E-01	NA	NA	NA
Subsurface soil near the Utility Room (BH-17, 14-16' depth)	2.4E+00	2.7E+00	6.1E-02	9.7E-02	4.9E-01
Subsurface soil near the FRS (BH-19A, 22-24' depth)	4.3E+00	5.7E+02	1.5E-01	2.0E-01	8.0E-01

LEGEND: FRS = Fuel Receiving and Storage Building, STP = Sewage Treatment Plant

NOTE: (1) See Figure 4-5 for a map showing facilities in the northern portion of WMA 6. See Figures 4-6 and 4-7 for maps showing locations with radionuclide concentrations in excess of background.

WMA 7, NDA and Associated Facilities

Concentrations of radiological constituents measured at levels in excess of background in surface soil and sediment from WMA 7 are listed in Table 4-20. Cs-137, Sr-90, and Am-241 were found in concentrations exceeding background in surface soil. Sediment samples collected near the Interceptor Trench contained concentrations of Cs-137, Sr-90, Pu-238, and Am-241 in excess of background. Ratios of Sr-90 to Cs-137 in surface soil ranged from 0.11 to 1 to 8.2 to 1. The Sr-90 to Cs-137 ratio for sediment was about 3.7 to 1. Maximum ratios to Cs-137 for Pu-238, Pu-239/240, and Am-241 in surface soil and sediment were, respectively: 0.096 (sediment), 0.022 (surface soil), and 0.046 (sediment). All were found near the Interceptor Trench.

No concentrations above background were found in boreholes of subsurface soil taken in 1993 at WMA 7. (Note that the two subsurface soil borings done at this location in 1993 were taken from the edges of the burial area, one upgradient of the buried waste and the other on the opposite side of the Interceptor Trench downgradient of the area.) However, analytical results from boxes and rolloffs filled with subsurface soil excavated from "special holes" during burial activities on the NDA or during construction of the Interceptor Trench contained Am-241 concentrations well in excess of background. Ratios of Am-241 to Cs-137 ranged from 0.024 to 0.077 to 1. The excavated soil has been shipped offsite, however, results suggest that subsurface soil remaining in the NDA contains radionuclide concentrations exceeding background.

Table 4-20. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 7[1]

Location	Maximum Concentration (pCi/g dry)				
	Cs-137	Sr-90	Pu-238	Pu-239/ 240	Am-241
Surface soil by the NDA Interceptor Trench (SS-15, BH-42)	4.7E+00	3.3E+00	8.5E-02	9.2E-02	1.5E-01
Surface soil by the NDA Hardstand (SS-20)	6.8E+01	7.7E+00	NA	NA	NA
Surface soil at remainder of NDA (1994 data from special sampling)	3.2E+00	2.1E+01	NA	NA	NA
Sediment from drainage near Interceptor Trench (ST-23)	9.0E-01	3.3E+00	8.6E-02	≤Bkg	4.1E-02
Subsurface soil excavated from "special holes" or Interceptor Trench (1997 sampling of excavated soil in boxes and rolloffs)	3.5E+01	NA	NA	NA	1.8E+00

NOTE: (1) See Figures 4-6 and 4-7 for maps showing locations with radionuclide concentrations in excess of background. Not shown on the map, the Interceptor Trench borders the northeast and northwest boundaries of the NDA. The Trench was installed in 1990 to intercept and collect leaching from the NDA. The NDA Hardstand (not shown on the map) was located at the easternmost point of WMA 7.

WMA 9, Radwaste Treatment Drum Cell Area

Data from only two surface soil samples were available for WMA 9. Although gross beta concentrations exceeded background for both, data for specific beta-emitting radionuclides did not. (See Figure 4-6.) No subsurface soil or sediment data were available for WMA 9.

WMA 10, Support and Services Area

Concentrations of radiological constituents measured at levels in excess of background in surface soil and sediment from WMA 10, the Support and Services Area, are listed in Table 4-21. This area includes support facilities (e.g., administrative buildings, offices, parking lots, the Environmental Laboratory) that are not known to be radiologically contaminated. Note that only one surface soil sample shown on Figure 4-6 did not have concentrations exceeding background: SS-11 on the north plateau, located on the western side of the project premises in WMA 10.

Low-level concentrations of Cs-137 exceeding background were found in surface soil near support trailers close to the Process Building and in sediment from a drainage ditch south of the Environmental Laboratory. Elevated Cs-137 in surface soil is thought to be attributable to airborne releases. Elevated Cs-137 in the drainage ditch could be

attributable to runoff from WMA 6 (i.e., possibly related to historical releases or leaks from the old Sewage Treatment Plant that released radionuclides to drainage by the railroad bed, as discussed in Section 2). Although gross alpha and gross beta concentrations slightly above background were noted for certain surface soil samples from WMA 10 (as shown on Figure 4-6), no other concentrations of specific radionuclides above background have been reported.

Table 4-21. Above-Background Concentrations of Radionuclides in Surface Soil and Sediment at WMA 10(1)

Location	Maximum Concentration (pCi/g dry)
	Cs-137
Surface soil by former Trailer City (1998 special soil sampling)	1.0E+00
Sediment samples by drainage south of Environmental Laboratory (ST-26)	1.7E-01

NOTE: (1) See Figure 4-6 for a map showing locations with radionuclide concentrations in excess of background. Not shown on maps, the former Trailer City was located directly opposite the western entrance to the Process Building. The Environmental Laboratory (shown, but not labeled, on Figure 4-6) is located immediately north of sampling point ST-26.

WMA 12, Remainder of the Site

Concentrations of radiological constituents measured at levels in excess of background in surface soil and sediment from WMA 12 are listed in Table 4-22. Only the portion of WMA 12 within the project premises, which includes the onsite segments of Franks Creek and Erdman Brook, is addressed in this evaluation.

Surface soil concentrations of both Cs-137 and Sr-90 were noted in excess of background in WMA 12 (see Figure 4-6). Cs-137 and Sr-90 exceeding background concentrations were also found in sediment samples from both Franks Creek and Erdman Brook, as well as in drainage downgradient of the demineralizer sludge ponds. Sediment samples collected along the lengths of both Franks Creek and Erdman Brook also contained alpha-emitting radionuclides at concentrations in excess of background, although the radionuclides varied in relationship to the stream segment.

In Erdman Brook downstream of drainage from the NDA (locations ST-22 and ST-21), Am-241 and Pu-238 were observed in concentrations greater than background. Further downstream, at point ST-20, after the stream receives inflow from via a drainage from WMA 2, Am-241, Pu-238, and Pu-239/240 concentrations were all above background. At point ST-19, located downstream where the stream receives effluent from Lagoon 3, U-232 (in addition to the other nuclides) was also found above background.

Similarly, sediment at the southernmost segments of Franks Creek (points ST-13, ST-12, and ST-11) contained gross alpha concentrations in excess of background. However, at point ST-10, located downstream of its junction with Erdman Brook, concentrations of Am-241, Pu-238, and Pu-239/240 were found in its sediment in excess of background.

Table 4-22. Above-Background Concentrations of Radionuclides in Surface Soil and Sediment at WMA 12[(1)]

Location	Maximum Concentration (pCi/g)					
	Cs-137	Sr-90	U-232	Pu-238	Pu-239/240	Am-241
Surface soil near borders with WMA 2 and WMA 6 (SS-08 [Cs-137], BH-16 [Sr-90])	8.1E+00	1.3E+00	NA	NA	NA	NA
Surface soil near eastern fence line (SS-07)	1.6E+00	4.4E+00	≤Bkg	≤Bkg	≤Bkg	≤Bkg
Sediment from drainage downgradient of Demineralizer Sludge Ponds (ST-27)	6.0E+00	8.5E-01	≤Bkg	≤Bkg	7.3E-02	1.4E-01
Sediment from Erdman Brook (ST-19 [Cs-137, Sr-90, U-232], ST-20 [Pu-238, Pu-239/240], ST-22 [Am-241])	3.5E+01	1.6E+00	1.1E-01	2.5E-01	7.3E-02	1.4E-01
Sediment from Franks Creek (ST-10 [Cs-137 only], SNSP006)	1.0E+02	1.0E+01	1.4E-01	1.4E-01	1.1E-01	2.4E-01

NOTES: (1) See Figure 4-6 for a map showing locations with radionuclide concentrations in excess of background. The location of the Demineralizer Sludge Ponds is shown in Figure 4-5.

LEGEND: NA = No analysis. "≤Bkg" = Concentrations did not exceed background.

The highest concentrations of all radionuclides (except Pu-238, for which the maximum was found at point ST-20 on Erdman Brook) were observed in sediment from Franks Creek at location SNSP006, where it flows off site at the security fence.[13] As was found with sediment from Erdman Brook, sediment from Franks Creek collected downgradient of the controlled effluent water release point WNSP001 contained U-232 at concentrations exceeding background. (Effluent water discharged from lagoon 3 through WNSP001 often contains measureable quantities of U-232.) Summary statistics for radionuclide concentrations at SNSP006 are presented in Appendix B.

The highest ratio of Sr-90 to Cs-137 (about 3 to 1) in surface soil from WMA 12 was noted for one sample collected near the eastern edge of the fenced area. In sediment, the maximum ratios to Cs-137 for Sr-90 (0.1 to 1), Pu-239/240 (0.012 to 1), and Am-241 (0.023

[13] In 1990, a sample from a hot spot in Erdman Brook that measured 3000 μR/h during the ground-level survey showed 0.01 μCi/g (10,000 pCi/g) Cs-137. (This was a screening analysis that may have been performed on a wet sample; it was not validated.) This area of localized contamination was described as about six inches by six inches located one meter from the edge of the water. Limited investigation indicated that the contamination extended more than seven inches below the streambed surface. (Passuite and Monsalve-Jones 1993, Appendix C)

to 1) were all found downgradient of the Demineralizer Sludge Ponds. The highest ratios to Cs-137 of U-232 (0.003 to 1) and U-238 (0.007 to 1) were found in sediment from Erdman Brook, immediately after the point where it receives Lagoon 3 effluent.

4.2.6 Environmental Radiation Levels

As part of the WVDP Environmental Monitoring Program, since 1986 thermoluminescent dosimeters (TLDs) have been placed in the field to measure levels of integrated gamma radiation exposure. TLDs are placed:

(1) At background locations far from the Center,

(2) At communities near the Center,

(3) At a ring of perimeter locations around the Center, and

(4) At onsite locations near process areas, waste storage areas, and waste burial locations.

Figure 4-9 shows the locations of onsite TLDs.

Note that not all areas on the project premises have environmental TLD monitoring locations, therefore, data are not available for these areas. Average results over the last ten years, in mR/quarter and in mR/h, are summarized in Table 4-23. Onsite results are presented by waste management area. For comparison, measurements from background are included.

Exposure measurements from the ring of TLDs around the perimeter of the Center and at the community locations are evaluated each year as part of preparing the Annual Site Environmental Report. Values from offsite TLDs have consistently been indistinguishable from background.

Results from all onsite TLDs, with the single exception of DNTLD27 located on the eastern border of the security-fenced area, were in excess of background levels. Note that exposure levels in the above table may not be indicative of radionuclides in soil, but of radiation from the wastes being processed and/or stored nearby.

The on-site monitoring point with the highest dose readings was location DNTLD24 on the north plateau (Figure 4-9). Sealed containers of radioactive components and debris from the plant decontamination work are stored nearby in the Chemical Process Cell Waste Storage Area. Exposure rates at this location have been generally decreasing over time because the radioactivity in the materials stored nearby is decaying. This storage area is well within the Center boundary, just inside the WVDP fenced area, and is not accessible by the public.

The maximum quarterly exposure level (1298 mR/qtr [0.59 mR/hr]) was noted at DNTLD35, near the rail spur by the Drum Cell in the second quarter of 2007. This high reading was associated with waste storage and with staging and shipping drums of cement-stabilized waste from the Drum Cell. All remaining drums were shipped from the Drum Cell in 2007, and in the fourth quarter of 2007 the exposure level at DNTLD35 had dropped to 23 mR/qtr (0.011 mR/hr).

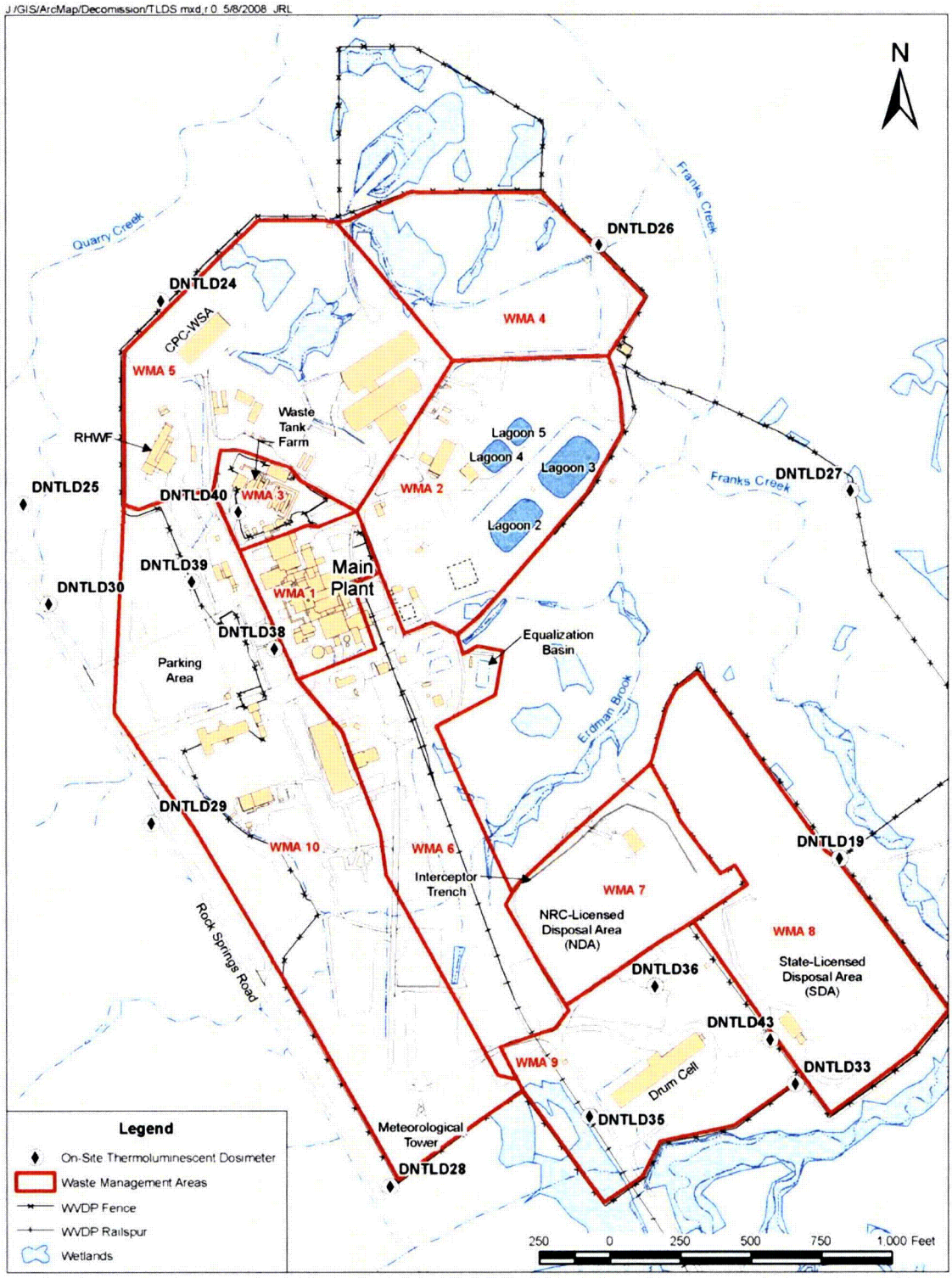

Figure 4-9. Onsite Environmental TLD Locations

Table 4-23. Environmental Radiation Levels on the WVDP Site (1998-2007 data)

TLD (s)	Location	Average mR/qtr	Average mR/h	Maximum mR/qtr	Maximum mR/h	(1)Exceeds Background?
DNTLD40	Waste Tank Farm (WMA 3)	119	0.054	268	0.122	Yes
DNTLD26	Construction and Demolition Debris Landfill fence line (WMA 4)	23	0.011	30	0.014	Yes
DNTLD24	Chemical Process Cell Waste Storage Area fence line (WMA 5)	523	0.239	717	0.327	Yes
DNTLD25	Quarry Creek, between security fence and public road (WMA 5)	23	0.011	31	0.014	Yes
DNTLD30	Northwest parking lot, near public road (WMA 10)	23	0.010	32	0.015	Yes
DNTLD39	On fence between parking lot and Process Building (WMA 10)	49	0.022	70	0.032	Yes
DNTLD38	Nurse's office across Process Building (WMA 10)	34	0.015	55	0.025	Yes
DNTLD29	On fence near Environmental Laboratory (WMA 10)	22	0.010	29	0.013	Yes
DNTLD28	Southwestern corner of Project Premises (WMA 10)	22	0.010	38	0.018	Yes
DNTLD35	(2)Near rail spur by Drum Cell (WMA 9)	109	0.050	1298	0.592	Yes
DNTLD36	(2)Drum Cell north fence (WMA 9)	61	0.028	458	0.209	Yes
DNTLD43	Drum Cell northeastern fence (WMA 9)	31	0.014	69	0.031	Yes
DNTLD33	Drum Cell southeastern corner (WMA 9)	32	0.014	54	0.025	Yes
DNTLD19	Western fence line near waste burial areas (WMA 12)	22	0.010	39	0.018	Yes
DNTLD27	Eastern fence line farthest from process and waste storage areas (WMA 12)	20	0.009	27	0.012	No
Background	Four background locations (map in Appendix B)	19	0.009	35	0.016	NA

NOTE: (1) Data sets from each location were compared with background data sets using one-way analysis of variance (see Appendix B).

(2) Exposure measurements near the Drum Cell have been elevated in the last several years because the area is being used as a storage area for vessels removed from the Process Building and for staging waste for shipping. Waste drums formerly stored in the Drum Cell itself were removed in 2007.

As summarized in WVNSCO 1994, two aerial radiation surveys of the WNYNSC in 1969 and 1979 identified above-background gamma radiation extending from the reprocessing plant in a northwest direction along Buttermilk Creek (1969) and in a prong extending westward offsite across Rock Springs Road (1979). Cs-137 was determined to be the source of the gamma activity. (See Section 2.)

Soil sampling by NYSDEC in 1971 and by WVNSCO in 1982 determined that Cs-137 activity was greater in soil northwest of the plant and that activity was greatest at the soil surface and decreased with depth (WVNSCO 1994). Activity in the cesium prong is attributed to airborne releases from a filter blow-out in 1968, as indicated in Section 2. Elevated radionuclide concentrations in the Buttermilk Creek drainage are attributed to routine radioactive liquid releases.

Posted Radiation Areas

At the WVDP Site, radiation areas are posted if exposure can exceed 5 mrem/hr at 30 centimeters (WVNSCO 2006). Posted radiological control areas on the project premises are shown in Figure 4-10. Posted radiation levels are generally indicative of surface and/or near surface contamination, storage of radioactive waste, and proximity to radiological process areas. Posted areas are delineated in accordance with 10 CFR 835, *Occupational Radiation Protection.*

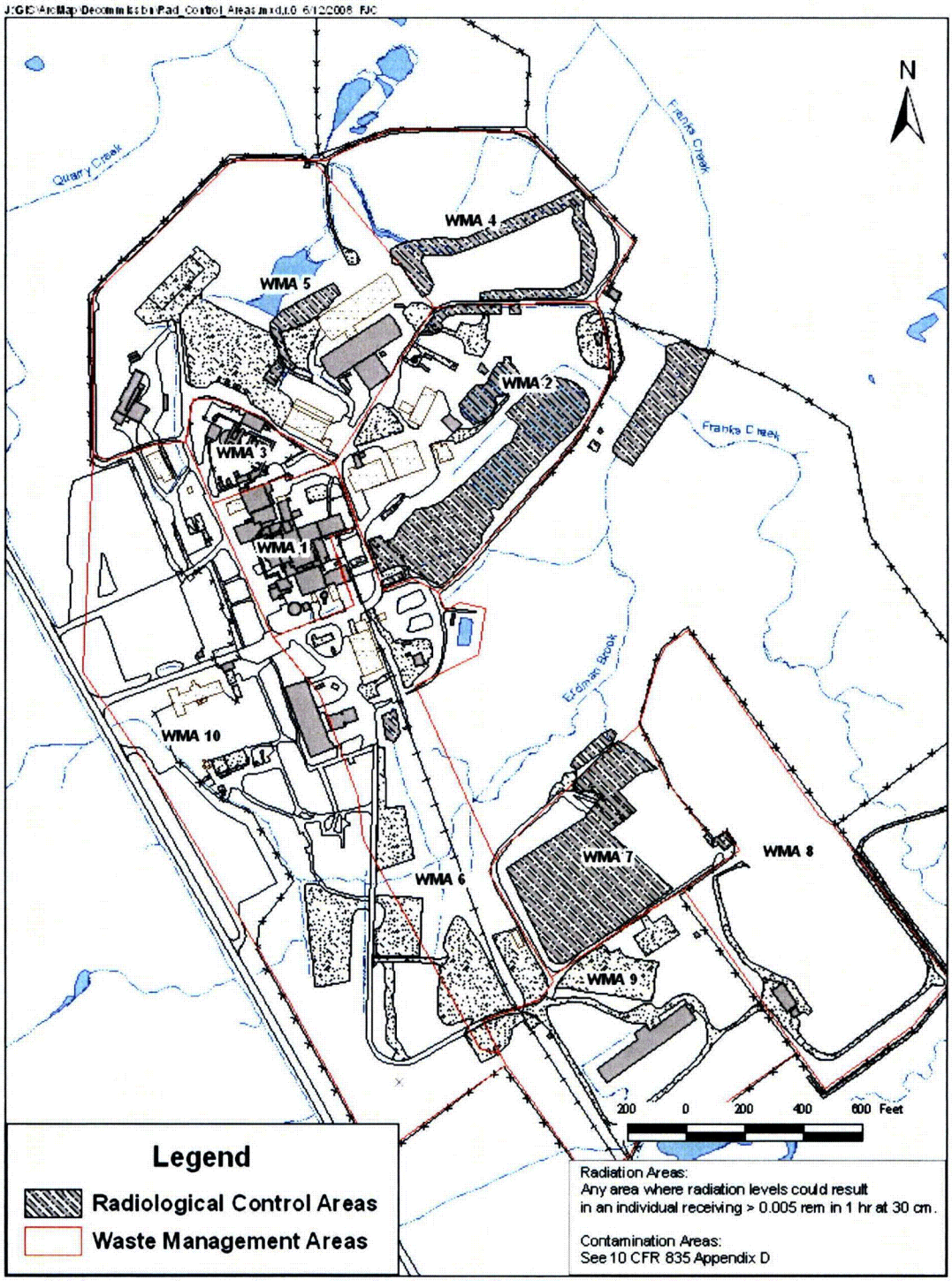

Figure 4-10. WVDP Radiological Control Areas. (Facilities with radiological controlled areas are outlined in black. Radiological Control Areas are current as of June 2008.)

4.2.7 Radiological Status of Onsite Surface Water

The WVDP Environmental Monitoring Program routinely collects surface water samples from the following locations on the project premises:

(1) Two controlled effluent discharges (releases from lagoon 3 through the weir at point WNSP001 and from the Sanitary Waste Treatment Facility at point WNSP007);

(2) Two drainages where water from the North Swamp and the Northeast Swamp leave the site (points WNSW74A and WNSWAMP, respectively);

(3) Facility cooling water from the Cooling Tower (WNCOOLW);

(4) Two drainage ditches (facility drainage [point WNSP005] and NDA surface drainage [point WNNDADR]); and

(5) Three locations on two streams (point WNERB53 on Erdman Brook, point WNFRC67 on Franks Creek, and point WNSP006 where Franks Creek leaves the project premises at the security fence).

Figure 4-11 shows the location of these routine surface water monitoring locations and indicates those with gross alpha (or alpha-emitting radionuclide) concentrations and gross beta (or beta/gamma-emitting radionuclide) concentrations in excess of background. All surface water locations had at least one constituent exceeding background (i.e., no non-impacted locations were noted).

Table 4-24 summarizes median, average, and maximum concentrations of those radionuclides observed to exceed background in surface water over the ten-year period 1998-2007. (For a complete summary of radionuclide concentrations in surface water, including those not detected above background, see Table B-13 of Appendix B.) Note that concentrations of the beta-emitting radionuclide Sr-90 exceeding background were observed in surface water throughout the project premises. (See Appendix B for comparable summary statistics for each radionuclide in surface water from background locations.) The highest Sr-90 concentrations were observed at location WNSWAMP, which is downstream of the point where the leading edge of the north plateau groundwater plume surfaces.

The full suite of radionuclides monitored in surface water was detected at above-background concentrations at the Lagoon 3 discharge point WNSP001. Tritium was detected downstream of the Low-Level Waste Treatment Facility (points WNSP001 and WNSP006), at the Northeast Swamp Discharge Point (WNSWAMP), at a point immediately downstream of the NDA on the south plateau (WNNDADR), and in Erdman Brook and Franks Creek on the south plateau (locations WNERB53 and WNFRC67, respectively).

Alpha-emitting radionuclides at concentrations exceeding background were noted only in surface water from the north plateau, primarily at locations downstream of the Low-Level Waste Treatment Facility discharge, but also at the North (WNSW74A) and Northeast Swamp (WNSWAMP) discharge points.

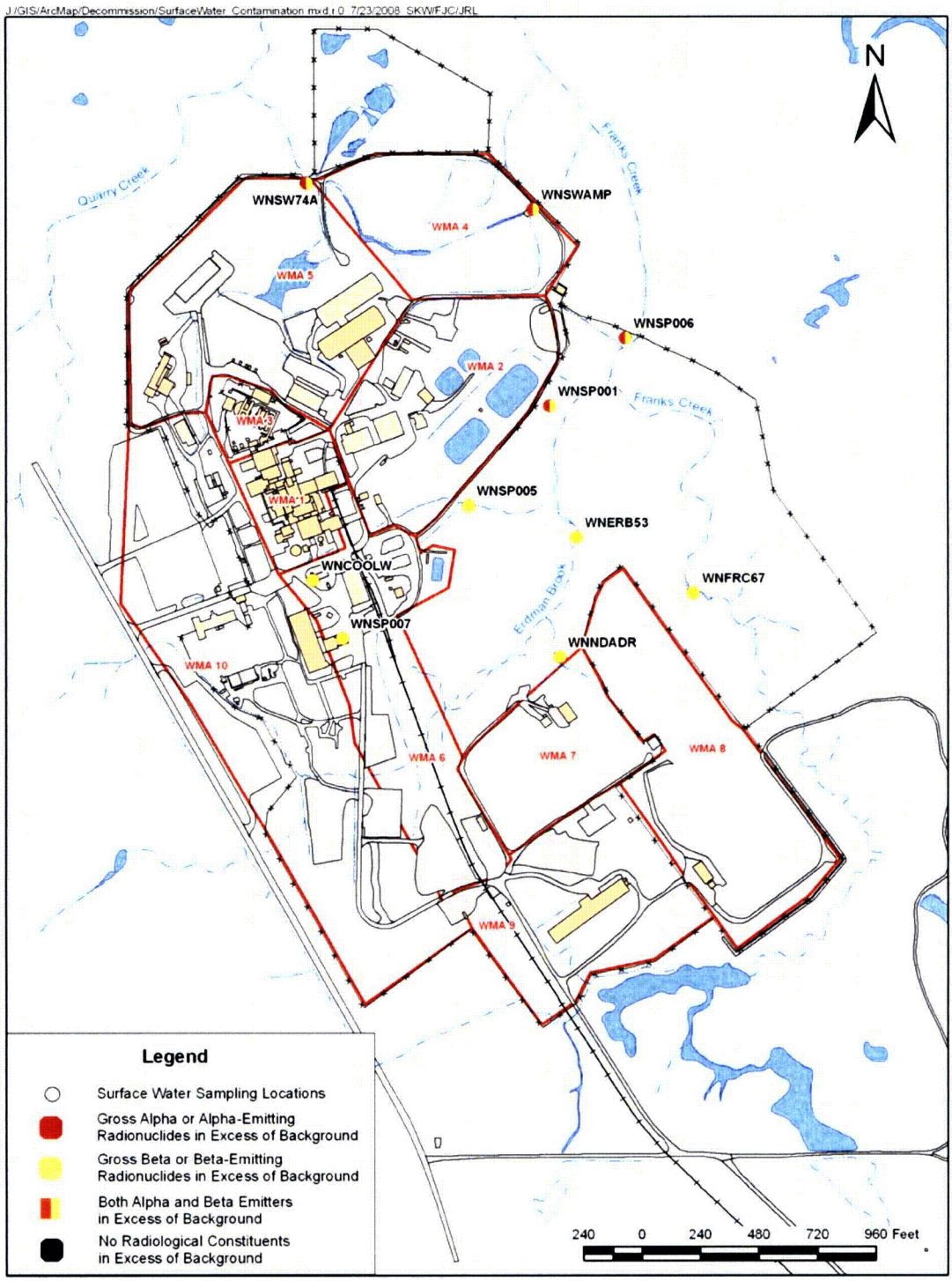

Figure 4-11. Surface Water Locations with Radionuclide Concentrations in Excess of Background

Table 4-24. Radionuclide Concentrations (pCi/L)[1] in Excess of Background in Surface Water[2]

Location	Median	Average			Maximum
		Result	±	Uncertainty	
Lagoon 3 discharge weir (WNSP001), WMA 2					
H-3	2.5E+03	2.8E+03	±	1.4E+02	7.2E+03
C-14	< 2.8E+01	1.4E+01	±	2.2E+01	4.8E+01
Sr-90	9.9E+01	1.2E+02	±	7.4E+00	3.2E+02
Tc-99	6.5E+01	7.9E+01	±	4.8E+01	3.4E+03
I-129	2.1E+00	2.4E+00	±	1.5E+00	1.0E+01
Cs-137	6.1E+01	7.6E+01	±	1.9E+01	3.3E+02
U-232	8.0E+00	9.0E+00	±	9.9E-01	2.1E+01
U-233/234	5.0E+00	5.5E+00	±	6.2E-01	1.4E+01
U-235/236	2.6E-01	2.8E-01	±	1.2E-01	5.8E-01
U-238	3.8E+00	3.8E+00	±	4.9E-01	7.6E+00
Pu-238	6.5E-02	1.5E-01	±	6.8E-02	1.6E+00
Pu-239/240	5.2E-02	1.3E-01	±	6.2E-02	1.4E+00
Am-241	6.8E-02	1.2E-01	±	6.0E-02	9.7E-01
Northeast swamp drainage (WNSWAMP), WMA 4					
H-3	1.1E+02	1.1E+02	±	8.2E+01	5.2E+02
Sr-90	1.5E+03	1.7E+03	±	3.1E+01	5.2E+03
U-233/234	1.7E-01	2.0E-01	±	1.4E-01	9.3E-01
U-238	1.0E-01	1.2E-01	±	1.1E-01	7.2E-01
North swamp drainage (WNSW74A), WMA 5					
Sr-90	5.5E+00	5.5E+00	±	1.8E+00	1.2E+01
U-233/234	1.5E-01	1.6E-01	±	8.4E-02	3.5E-01
U-238	1.0E-01	1.0E-01	±	6.6E-02	2.0E-01
Sanitary waste discharge (WNSP007), WMA 6					
Sr-90	3.1E+00	3.4E+00	±	1.9E+00	1.2E+01
Franks Creek at security fence (WNSP006), WMA 12					
H-3	< 8.5E+01	1.4E+02	±	8.3E+01	2.2E+03
Sr-90	1.9E+01	2.0E+01	±	3.0E+00	5.0E+01
Tc-99	< 2.1E+00	3.3E+00	±	2.1E+00	5.2E+01
Cs-137	< 8.0E+00	6.3E+00	±	9.5E+00	7.3E+01
U-232	3.2E-01	3.2E-01	±	1.3E-01	7.5E-01
U-233/234	3.7E-01	3.7E-01	±	1.3E-01	6.9E-01
U-238	2.5E-01	2.8E-01	±	1.1E-01	7.4E-01

Table 4-24. Radionuclide Concentrations (pCi/L)[1] in Excess of Background in Surface Water[2]

Location	Median	Average			Maximum
		Result	±	Uncertainty	
Pu-238	< 3.4E-02	2.1E-02	±	3.4E-02	1.4E-01
Facility yard drainage (WNSP005), WMA 12					
H-3	< 8.3E+01	3.8E+01	±	8.2E+01	1.2E+03
Sr-90	9.6E+01	1.0E+02	±	6.5E+00	2.0E+02
Drainage between NDA and SDA (WNNDADR), WMA 12					
H-3	1.0E+03	1.1E+03	±	1.0E+02	4.0E+03
Sr-90	8.5E+01	8.4E+01	±	5.4E+00	1.2E+02
Erdman Brook north of disposal areas (WNERB53), WMA 12					
H-3	< 8.3E+01	3.9E+01	±	8.0E+01	4.9E+02
Sr-90	8.2E+00	8.0E+00	±	2.0E+00	9.9E+00
Franks Creek East of SDA (WNFRC67), WMA 12					
H-3	< 8.3E-01	3.1E+01	±	8.1E+01	3.5E+02

NOTES: (1) 1 pCi/L = 3.7E-02 Bq/L

(2) Refer to Table 4-11 for median and maximum background values and to Appendix B for summary statistics of background radionuclide concentrations in surface water.

4.2.8 Radiological Status of Groundwater

Groundwater at the WVDP is routinely monitored in accordance with the WVDP Groundwater Monitoring Program. Although the primary focus of the program is on nonradiological constituents, all wells are monitored for radiological indicator parameters (gross alpha, gross beta, and H-3). Several wells, especially those impacted by the north plateau groundwater plume, are sampled for Sr-90. Select wells are monitored for a full suite of radionuclides. Table 4-25 lists routine groundwater monitoring locations at which radiological concentrations were found at levels exceeding background. Medians, averages, and maximum concentrations (in pCi/L) are presented for each.

For groundwater (unlike the other environmental media discussed in this section), gross alpha and gross beta concentrations exceeding background are presented. This is because limited radionuclide data are available for routinely monitored groundwater locations, and gross alpha and gross beta measurements, taken at all wells, may indicate the presence of other alpha- or beta-emitting radionuclides. For instance, gross beta measurements are used as a surrogate measurement for Sr-90 at monitoring points where the Sr-90-to-gross beta ratio has been determined to be approximately 0.5 to 1.

Locations at which gross alpha (or alpha-emitting radionuclide) concentrations and/or gross beta (or beta-emitting radionuclide, including H-3) concentrations exceeded background are shown on Figure 4-12. Locations at which no radiological constituents were found to exceed background are also shown. For a complete summary of

radionuclide data from both impacted and non-impacted routine groundwater monitoring locations, see Appendix B, Table B-14. A listing of supplementary information for each point (e.g., geographical coordinates, well construction, screened interval, geologic unit) is provided in Appendix B, Table B-15.

Table 4-25. Routine Groundwater Monitoring Locations With Radionuclide Concentrations (pCi/L)[1] in Excess of Background[2]

WMA	Monitoring Point	Constituent	Median	Average			Maximum
				Result	±	Uncertainty	
WMA 1	WP-A	Gross beta	2.4E+01	3.1E+01	±	4.6E+00	5.4E+01
		H-3	1.2E+04	1.1E+04	±	6.2E+02	1.3E+04
WMA 2	WP-C	Gross beta	2.4E+01	4.2E+01	±	5.5E+00	1.2E+02
		H-3	4.9E+04	4.7E+04	±	1.6E+03	6.6E+04
	WP-H	Gross alpha	6.1E+00	7.9E+01	±	2.3E+01	7.4E+02
		Gross beta	7.0E+03	7.2E+03	±	1.9E+02	1.2E+04
		H-3	3.0E+03	3.4E+03	±	5.0E+02	7.4E+03
	WNW0103	Gross beta	1.4E+02	1.8E+02	±	1.9E+01	5.5E+02
	WNW0104	Gross beta	5.9E+04	5.6E+04	±	1.6E+03	1.0E+05
		H-3	3.7E+02	3.9E+02	±	8.6E+01	7.5E+02
	WNW0105	Gross beta	3.9E+04	3.3E+04	±	1.5E+03	1.0E+05
		H-3	3.6E+02	3.7E+02	±	9.1E+01	7.1E+02
	WNW0106	Gross beta	1.6E+01	8.2E+01	±	8.0E+00	5.8E+02
		H-3	9.6E+02	1.0E+03	±	1.0E+02	1.8E+03
	WNW0107	Gross beta	7.0E+00	8.2E+00	±	2.6E+00	2.2E+01
		H-3	3.7E+02	4.8E+02	±	9.0E+01	9.9E+02
	WNW0108	Gross alpha	1.6E+00	1.5E+00	±	1.5E+00	4.3E+00
		H-3	1.2E+02	1.1E+02	±	8.4E+01	2.5E+02
	WNW0110	H-3	1.3E+03	1.3E+03	±	1.1E+02	1.7E+03
	WNW0111	Gross alpha	<4.4E+00	3.2E+00	±	5.1E+00	1.0E+01
		Gross beta	5.6E+03	5.9E+03	±	1.4E+02	1.2E+04
		H-3	2.0E+02	2.3E+02	±	8.4E+01	8.0E+02
	WNW0116	Gross beta	8.7E+02	2.0E+03	±	1.6E+02	9.5E+03
		H-3	1.7E+02	1.9E+02	±	8.2E+01	4.7E+02
	WNW0205	Gross beta	1.6E+01	1.7E+01	±	8.4E+00	4.1E+01
	WNW0408	Gross beta	4.0E+05	4.0E+05	±	3.0E+03	6.3E+05
		H-3	1.5E+02	1.9E+02	±	1.1E+02	2.2E+03
		Sr-90	1.5E+05	1.5E+05	±	1.7E+02	2.5E+05
		Tc-99	1.6E+01	1.7E+01	±	3.3E+00	2.5E+01

Table 4-25. Routine Groundwater Monitoring Locations With Radionuclide Concentrations (pCi/L)[1] in Excess of Background[2]

WMA	Monitoring Point	Constituent	Median	Average			Maximum
				Result	±	Uncertainty	
WMA 2		U-233/234	4.5E-01	5.3E-01	±	2.2E-01	1.3E+00
		U-238	2.9E-01	3.1E-01	±	1.6E-01	4.8E-01
	WNW0501	Gross beta	1.9E+05	1.9E+05	±	2.6E+03	3.2E+05
		H-3	1.4E+02	1.2E+02	±	8.4E+01	3.2E+02
		Sr-90	9.2E+04	9.3E+04	±	2.4E+02	1.5E+05
	WNW0502	Gross beta	1.7E+05	1.6E+05	±	2.8E+03	2.3E+05
		H-3	1.3E+02	1.4E+02	±	8.4E+01	5.0E+02
		Sr-90	8.4E+04	8.3E+04	±	2.1E+02	1.2E+05
	WNW8603	Gross beta	5.7E+04	4.8E+04	±	1.2E+03	9.0E+04
		H-3	3.4E+02	3.4E+02	±	8.8E+01	5.8E+02
	WNW8604	Gross beta	4.1E+04	4.6E+04	±	1.1E+03	1.0E+05
		H-3	3.5E+02	3.8E+02	±	8.4E+01	6.4E+02
	WNW8605	Gross alpha	9.1E+00	8.5E+00	±	7.7E+00	2.1E+01
		Gross beta	1.1E+04	1.1E+04	±	1.7E+02	1.6E+04
		H-3	3.7E+02	4.2E+02	±	8.7E+01	1.3E+03
WMA 3	WNW8609	Gross beta	1.5E+03	1.4E+03	±	4.2E+01	2.3E+03
		H-3	4.5E+02	4.7E+02	±	9.1E+01	7.9E+02
		Sr-90	8.0E+02	7.2E+02	±	2.1E+01	1.1E+03
WMA 4	WNW0801	Gross beta	8.0E+03	8.6E+03	±	2.7E+02	1.5E+04
		H-3	1.5E+02	1.6E+02	±	8.2E+01	3.8E+02
		Sr-90	4.1E+03	4.3E+03	±	4.7E+01	8.0E+03
	WNW0802	Gross beta	9.9E+00	3.5E+01	±	5.1E+00	2.8E+02
		H-3	<1.1E+02	9.0E+01	±	8.0E+01	4.2E+02
	WNW0803	Gross beta	1.5E+01	1.5E+01	±	4.7E+00	2.5E+01
		H-3	1.8E+02	1.6E+02	±	8.5E+01	3.4E+02
	WNW0804	Gross beta	2.6E+02	2.9E+02	±	1.1E+01	6.9E+02
		H-3	1.2E+02	1.1E+02	±	8.0E+01	3.6E+02
	WNW8612	H-3	4.2E+02	4.3E+02	±	8.9E+01	8.5E+02
WMA 5	WNW0406	Gross beta	7.4E+00	8.1E+00	±	3.5E+00	1.7E+01
		H-3	1.2E+02	1.1E+02	±	8.4E+01	4.4E+02
		Tc-99	2.2E+00	2.5E+00	±	1.9E+00	8.5E+00
	WNW0409	Gross alpha	<1.0E+00	9.4E-01	±	9.9E-01	2.3E+00
	WNW0602A	Gross beta	1.2E+01	1.3E+01	±	2.9E+00	3.5E+01

Table 4-25. Routine Groundwater Monitoring Locations With Radionuclide Concentrations (pCi/L)[1] in Excess of Background[2]

WMA	Monitoring Point	Constituent	Median	Average			Maximum
				Result	±	Uncertainty	
WMA 5		H-3	2.2E+02	2.2E+02	±	8.9E+01	4.9E+02
	WNW0604	Gross beta	6.1E+00	6.3E+00	±	3.0E+00	1.3E+01
	WNW0605	Gross beta	4.8E+01	5.1E+01	±	4.0E+00	8.8E+01
	WNW0704	Gross beta	8.0E+00	8.2E+00	±	3.0E+00	1.3E+01
	WNW8607	Gross beta	2.6E+01	2.7E+01	±	5.3E+00	7.6E+01
	WNW1304	U-233/234	2.7E-01	2.9E-01	±	1.3E-01	5.6E-01
		U-238	1.9E-01	2.2E-01	±	1.0E-01	5.8E-01
WMA 7	WNW0902	Gross alpha	1.5E+00	1.3E+00	±	1.3E+00	5.4E+00
	WNW0909	Gross beta	3.7E+02	3.7E+02	±	1.4E+01	6.4E+02
		H-3	8.2E+02	1.5E+03	±	1.2E+02	3.9E+03
		Sr-90	1.9E+02	1.8E+02	±	8.3E+00	2.2E+02
		Tc-99	<1.9E+00	1.3E+00	±	1.8E+00	5.0E+00
		I-129	6.2E+00	6.3E+00	±	1.9E+00	9.7E+00
		U-233/234	6.0E-01	7.4E-01	±	2.4E-01	1.3E+00
		U-238	4.7E-01	5.4E-01	±	2.0E-01	1.0E+00
	WNW0910	Gross alpha	<2.5E+00	1.9E+00	±	2.3E+00	3.4E+00
		Gross beta	3.8E+01	1.5E+02	±	8.5E+01	1.5E+03
	WNNDATR	Gross alpha	2.2E+00	2.1E+00	±	2.1E+00	1.1E+01
		Gross beta	1.5E+02	1.8E+02	±	8.4E+00	5.5E+02
		H-3	3.6E+03	5.0E+03	±	2.3E+02	2.0E+04
		Sr-90	5.8E+01	7.8E+01	±	5.5E+00	2.8E+02
		I-129	<9.1E-01	8.4E-01	±	9.4E-01	7.0E+00
		U-233/234	1.7E+00	1.5E+00	±	2.8E-01	2.1E+00
		U-235/236	1.1E-01	1.4E-01	±	9.5E-02	3.0E-01
		U-238	1.3E+00	1.2E+00	±	2.5E-01	1.7E+00
WMA 9	WNW1006	Gross alpha	<5.1E+00	4.2E+00	±	5.5E+00	1.0E+01

NOTES: (1) 1 pCi/L = 3.7E-02 Bq/L

(2) Refer to Table 4-11 for median and maximum background values and to Appendix B for summary statistics of background radionuclide concentrations in groundwater (Table B-7) and at non-impacted groundwater monitoring locations (Table B-14). Data sets from each location were compared with background data sets using the nonparametric Mann-Whitney "U" test, as described in Appendix B, section 4.3.

As shown in Figure 4-12, elevated gross beta concentrations are evident in groundwater northeast of the Process Building (WVNSCO and URS 2005). The beta activity is primarily found in the surficial sand and gravel unit, and the general direction of flow in this unit is to the northeast. Elevated gross beta concentrations are largely

attributed to Sr-90 in the north plateau plume. While concentrations of gross alpha or alpha-emitting radionuclides exceeding background were found at only a few locations, the locations were associated with (or downgradient of) historical waste processing or waste burial activities (i.e., WMAs 1, 2, and 7).

In December 1993, elevated gross beta concentrations were detected in surface water at a former sampling location near the edge of the north plateau. This discovery initiated a subsurface groundwater and soil Geoprobe® investigation in 1994 (Carpenter and Hemann 1995). Two additional Geoprobe® investigations were conducted in 1997 (Hemann and Fallon 1998) and 1998 (Hemann and Steiner 1999). A listing of the Geoprobe® locations, sample depths, and geologic units from which the groundwater was sampled is provided in Appendix B, Table B-16. (NOTE: For completeness, Appendix B, Table B-17, provides a listing of groundwater points — in addition to the routine groundwater monitoring and Geoprobe® locations included in this evaluation — that have been sampled over the years. Table B-17 presents information on the locations and depths of these points, and summarizes the reasons that the points were not included in the current evaluation [dry wells, wells dropped from program, unvalidated data, located in areas outside the scope of the Phase 1 DP, etc.].)

The principal source of the north plateau groundwater plume is believed to be a release of radioactively contaminated acid from the NFS acid recovery system in the 1960s when NFS was reprocessing fuel, during 10 CFR Part 50 licensed activities. A detailed description of the release is provided in Section 2, subsection 2.3.1. See also Table 2-15 for an estimate of radionuclide activity from this release expected to remain in the plume in 2011.

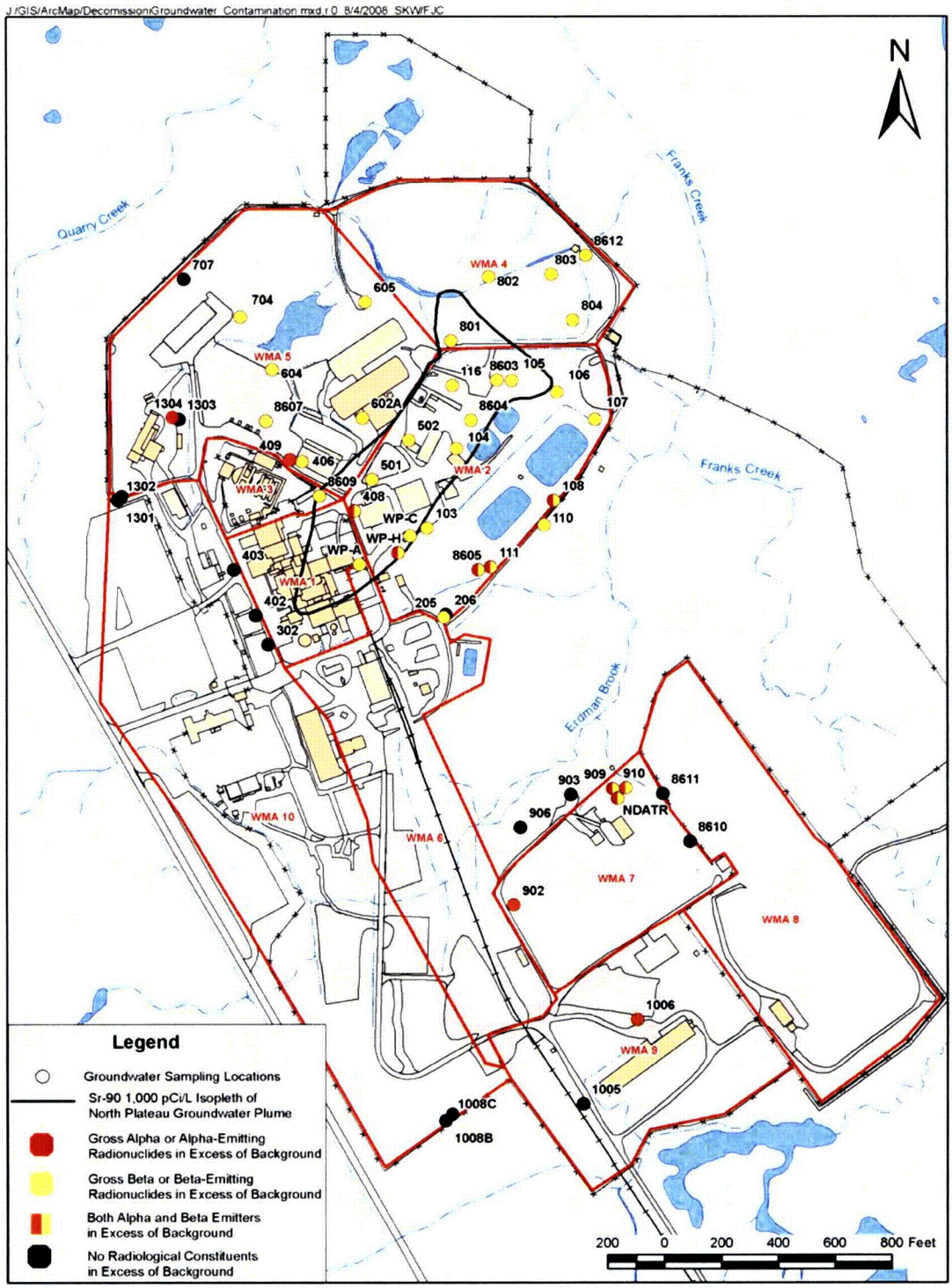

Figure 4-12. Routine Groundwater Monitoring Locations with Radionuclide Concentrations in Excess of Background

The Geoprobe® investigation results were used to estimate the extent of the north plateau groundwater plume beneath and downgradient of the Process Building. As part of the Geoprobe® investigations, a more extensive suite of radionuclides was analyzed in groundwater than was done for routine monitoring. Because the Geoprobe® groundwater samples differed from those taken from routine monitoring locations in that Geoprobe® samples may have been taken from several depths (and even from different geologic units) at a single location, the sample results were not directly comparable and have not been presented in the same table. However, results from the Geoprobe® investigations provide supplemental information about the presence of radionuclides in groundwater on the north plateau.

Geoprobe® locations at which concentrations of gross alpha (or alpha-emitting radionuclides) or gross beta (or beta/gamma-emitting radionuclides, including H-3) exceeded background are shown on Figure 4-13. The maximum measured radionuclide concentrations are summarized by WMA in Table 4-26. Since radionuclide data were available for these sampling locations, gross alpha and gross beta data were not included in this table.

As can be seen in Figure 4-13, concentrations of gross beta or beta/gamma-emitting radionuclides exceeding background are evident at all locations, with the exception of the four non-impacted points in WMA 5 northwest of the north plateau groundwater plume. Gross alpha or alpha-emitting radionuclide concentrations exceeding background were found immediately downgradient of the Process Building and downgradient of the Interceptors.

Table 4-26. Geoprobe® Groundwater Points with Radionuclide Concentrations (pCi/L) in Excess of Background[1]

WMA	Point	Constituent	Maximum	Point	Constituent	Maximum
WMA 1	GP8098	H-3	6.4E+04	GP76	Cs-137	8.5E+01
	GP29	C-14	2.3E+03	GP30	U-232	1.5E-01
	GP30	Sr-90	1.2E+06	GP73	U-233/234	1.2E+00
	GP72	Tc-99	1.2E+04	GP73	U-238	8.6E-01
	GP29	I-129	3.0E+01	GP76	Am-241	4.7E-01
WMA 2	GP47	H-3	3.4E+04	GP44	U-233/234	3.7E+01
	GP66	C-14	4.0E+02	GP44	U-235/236	6.2E-01
	GP8298	Sr-90	2.8E+05	GP60	U-238	1.5E+01
	GP68	Tc-99	5.8E+01	GP59	Pu-238	4.5E+00
	GP47	I-129	8.2E+01	GP59	Pu-239/240	7.9E+00
	GP46	Cs-137	1.5E+02	GP59	Am-241	5.9E+00
	GP44	U-232	7.8E+01	-	-	-
WMA 3	GP20	H-3	1.5E+03	GP20	I-129	2.5E+00
	GP2097	Sr-90	1.2E+04	-	-	-

Table 4-26. Geoprobe® Groundwater Points with Radionuclide Concentrations (pCi/L) in Excess of Background(1)

WMA	Point	Constituent	Maximum	Point	Constituent	Maximum
WMA 4	GP32A	H-3	1.3E+03	GP0397	Sr-90	8.6E+03
WMA 5	GP43	H-3	2.0E+04	GP53	Tc-99	8.0E+01
	GP40	Sr-90	3.8E+03	GP43	I-129	4.6E+00
WMA 6	GP70	H-3	6.8E+03	GP70	Tc-99	3.1E+01
	GP70	C-14	1.4E+02	GP70	I-129	1.1E+01
	GP70	Sr-90	2.8E+04	-	-	-
WMA 12	GP48	H-3	1.5E+03	GP50	U-238	7.2E-01
	GP50	Sr-90	1.3E+01	-	-	-

NOTE: (1) Points ending with "97" or "98" were collected in 1997 or 1998, respectively. The remaining points were collected in 1994. Sample results were compared with average background values as described in Appendix B, section 4.2.

The north plateau plume, as delineated by the 1,000 pCi/L isopleth, was approximately 300 feet wide and 800 feet long in 1994. By 2004, the plume area had expanded to approximately 350 feet by 1050 feet, and by 2007 to about 540 feet (at its widest point near the leading edge) by 1300 feet (WVES and URS 2008). (See Figure 4-14.)

The highest gross beta concentrations in groundwater and soil were found near the southeast corner of the Process Building. In the 1994 study, the maximum concentration in groundwater was 3.6E+06 pCi/L, and the maximum concentration in subsurface soil was 2.4E+04 pCi/g. Sr-90 and its progeny, Y-90, were determined to be the isotopes responsible for most of the elevated gross beta activity (WVNSCO and URS 2007).

As a result of recommendations from a 1997 external review of WVDP response actions on the north plateau, more attention was given in 1998 to the core area of the plume, determined to be beneath and immediately downgradient of the Process Building. Results from the 1998 investigation were presented in a summary report (Hemann and Steiner 1999) that compared groundwater and soil sampling data with the 1994 data. Concentrations detected in 1998 samples were generally lower than those in the 1994 samples due to radioactive decay and continuing migration and dispersion of the plume. The study also concluded that Lagoon 1 was a possible contributor of gross beta activity to groundwater downgradient of the Lagoon.

Figure 4-14 shows the 1E+03 pCi/L gross beta contour lines defining the extent of the plume in 1994, 2001, and 2007. (This figure, which duplicates Figure 2-6 in Section 2, is provided here for the sake of completeness.) Figure 4-14 also shows gross beta concentrations at the 12 routine groundwater monitoring locations that define the plume as of the fourth quarter of 2007. Contour lines show a gradual lengthening and expansion of the plume toward the northeast, with the highest concentration (i.e., well 408 at 3.9E+05

pCi/L) near the Process Building and lower concentrations near the leading edge. Further downgradient, the plume appears to be diverging – one prong moving to the north toward the surface drainage north of the CDDL and the other toward the east. Figure 4-14 also shows 1E+03 pCi/L contour lines of gross beta activity in groundwater over time near inactive Lagoon 1. This smaller area of elevated activity, likely associated with contamination remaining in Lagoon 1 sediment and backfill, appears to be migrating slightly eastward over time.

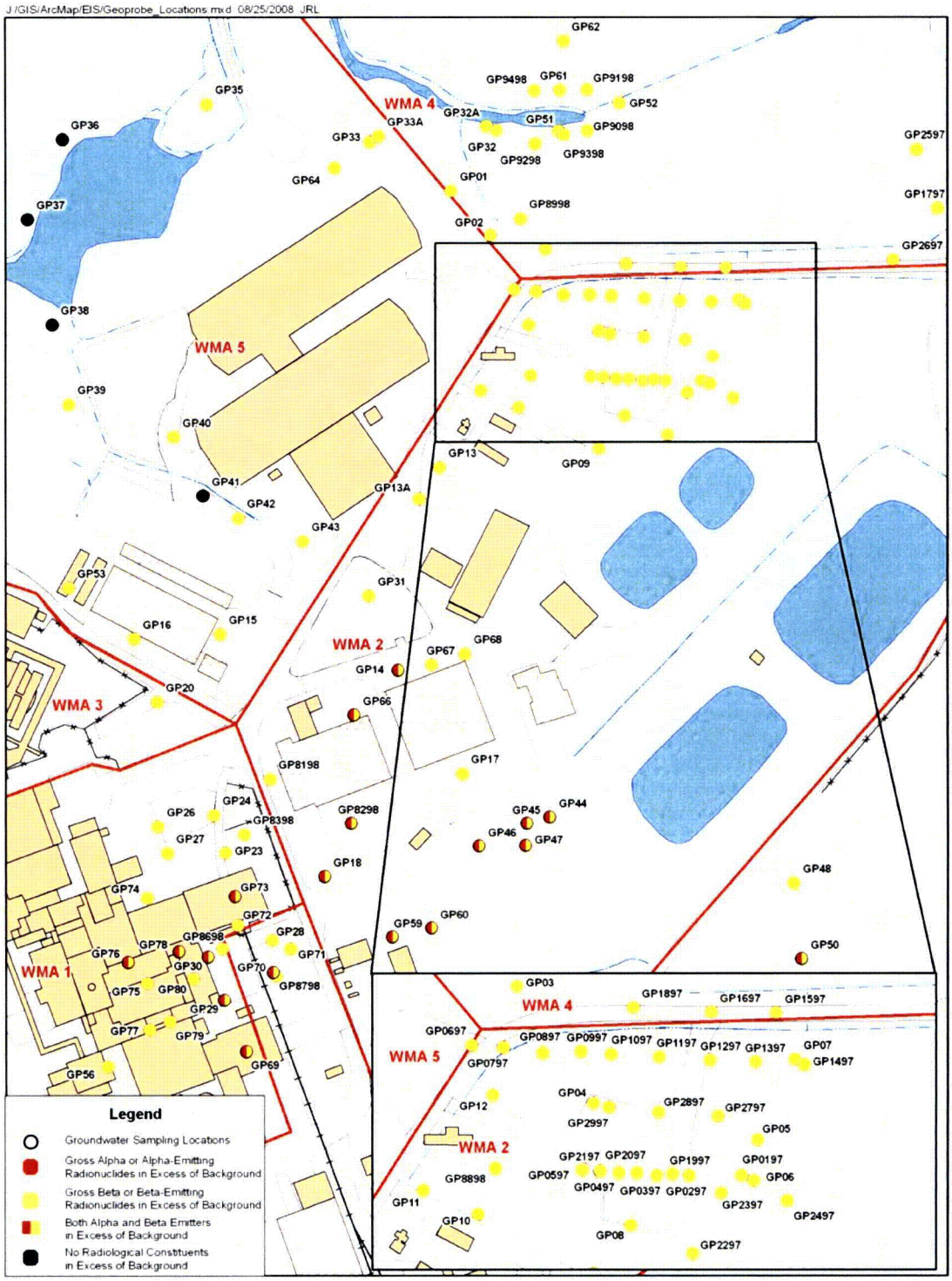

Figure 4-13. Geoprobe® Groundwater Locations with Radionuclide Concentrations in Excess of Background

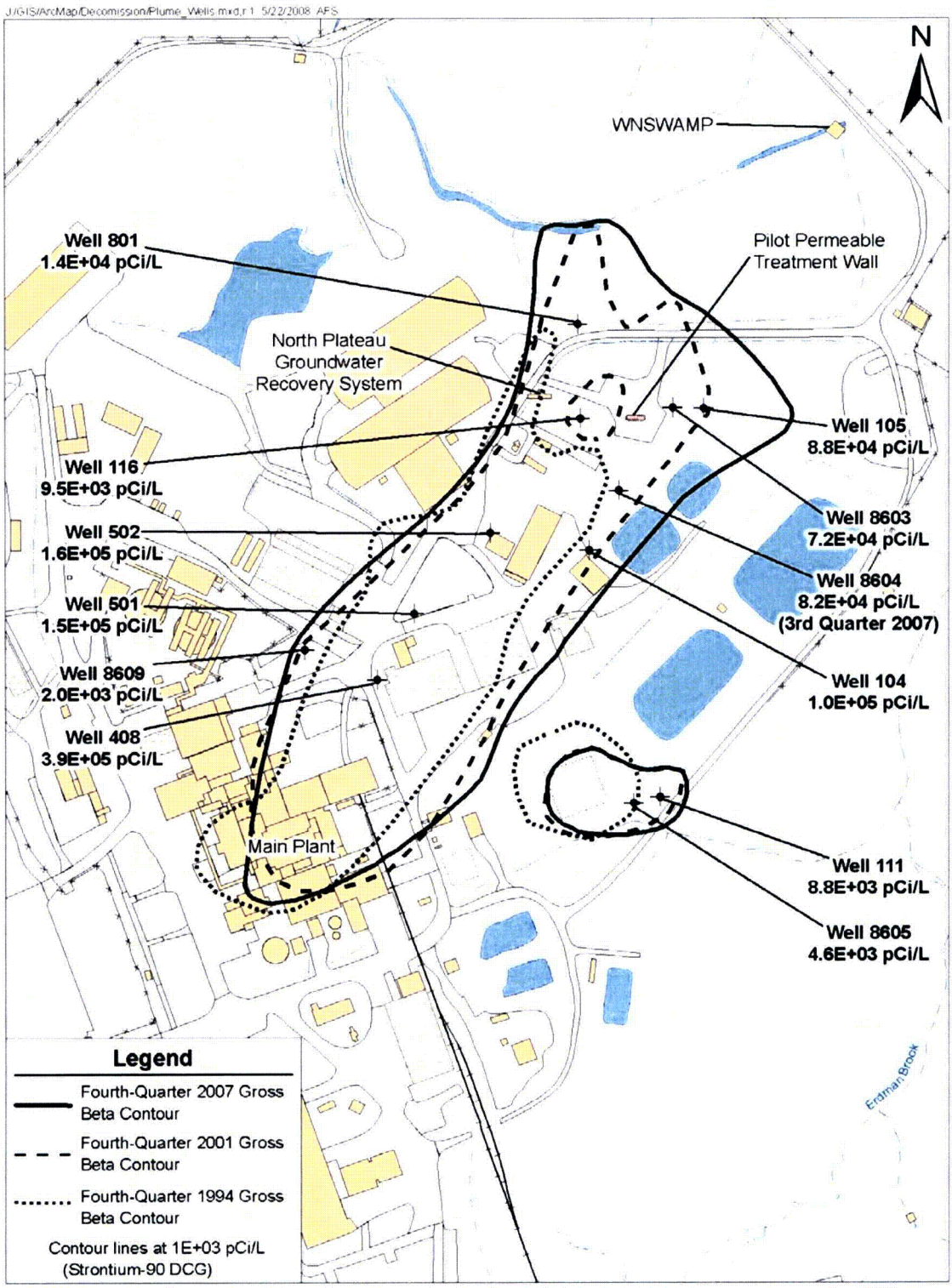

Figure 4-14. North Plateau Groundwater Plume

4.3 References

Code of Federal Regulations

10 CFR 50, *Domestic Licensing of Production and Utilization Facilities.*

10 CFR 835, *Occupational Radiation Protection.*

Other References

Carpenter and Hemann 1995, *Subsurface Probing Investigation on the North Plateau at the West Valley Demonstration Project*, WVDP-220, Rev. 0. Carpenter, B. and M. Hemann, West Valley Nuclear Services Company, West Valley, New York, April 26, 1995.

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5.0 DOSE MODELING

PURPOSE OF THIS SECTION

The purpose of this section is to describe dose modeling performed for Phase 1 of the proposed decommissioning to establish cleanup criteria that would not limit options for Phase 2 of the decommissioning.

INFORMATION IN THIS SECTION

This section provides the following information:

- Section 5.1 contains introductory material to place information in the following sections into context.
- Section 5.2 describes the three conceptual models and the mathematical model (RESRAD) used to develop derived concentration guideline levels (DCGLs) for 18 radionuclides of interest in surface soil, subsurface soil, and streambed sediment. It identifies the results in terms of $DCGL_W$ values and $DCGL_{EMC}$ values. It also discusses the results of deterministic sensitivity analyses of model input parameters.
- Section 5.3 discusses considerations related to dose integration and describes analyses performed to ensure that cleanup criteria used in Phase 1 would not limit Phase 2 decommissioning options.
- Section 5.4 provides cleanup goals; describes the process for refining the DCGLs and these cleanup goals; addresses use of a surrogate radionuclide in field measurements; provides a preliminary, order-of-magnitude dose assessment related to remediation of subsurface soil; and provides for a final such dose assessment after completion of the Phase 1 final status surveys.

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 this plan,
- The facility descriptions in Section 3,
- The information on site radioactivity in Section 4,
- The information in Section 6 on the as low as reasonably achievable (ALARA) analysis,
- The information in Section 9 on characterization surveys and the Phase 1 final status survey,
- The information in Appendix C that supplements the content of this section, and
- The information in Appendix D on engineered barriers and groundwater flow fields.

5.1 Introduction

To help place the dose modeling into context, it is useful to consider information about the applicable requirements and guidance, information on the environmental media of interest, and information relevant to consideration of doses from different parts of the project premises, along with information on matters that could impact dose modeling such as long-term erosion and potential changes in groundwater flow.

5.1.1 Applicable Requirements and Guidance

As explained in Section 1, certain areas of the project premises are being remediated in Phase 1 of the proposed decommissioning to NRC's unrestricted release criteria in 10 CFR 20.1402. These criteria state that a site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of the critical group that does not exceed 25 mrem per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are ALARA.

NRC provides guidance (NRC 2006) on two approaches that may be used to determine that these unrestricted release criteria have been achieved:

(1) The dose modeling approach, which involves characterizing the site – after remediation, if necessary – and performing a dose assessment; and

(2) The DCGL and final status survey approach, which involves developing or using DCGLs and performing a final status survey to demonstrate that the DCGLs have been met.

NRC observes that the second option is usually the more efficient or simpler method and that these two approaches are not mutually exclusive; they are just different approaches to show that the potential dose from a remediated site is acceptable (NRC 2006).

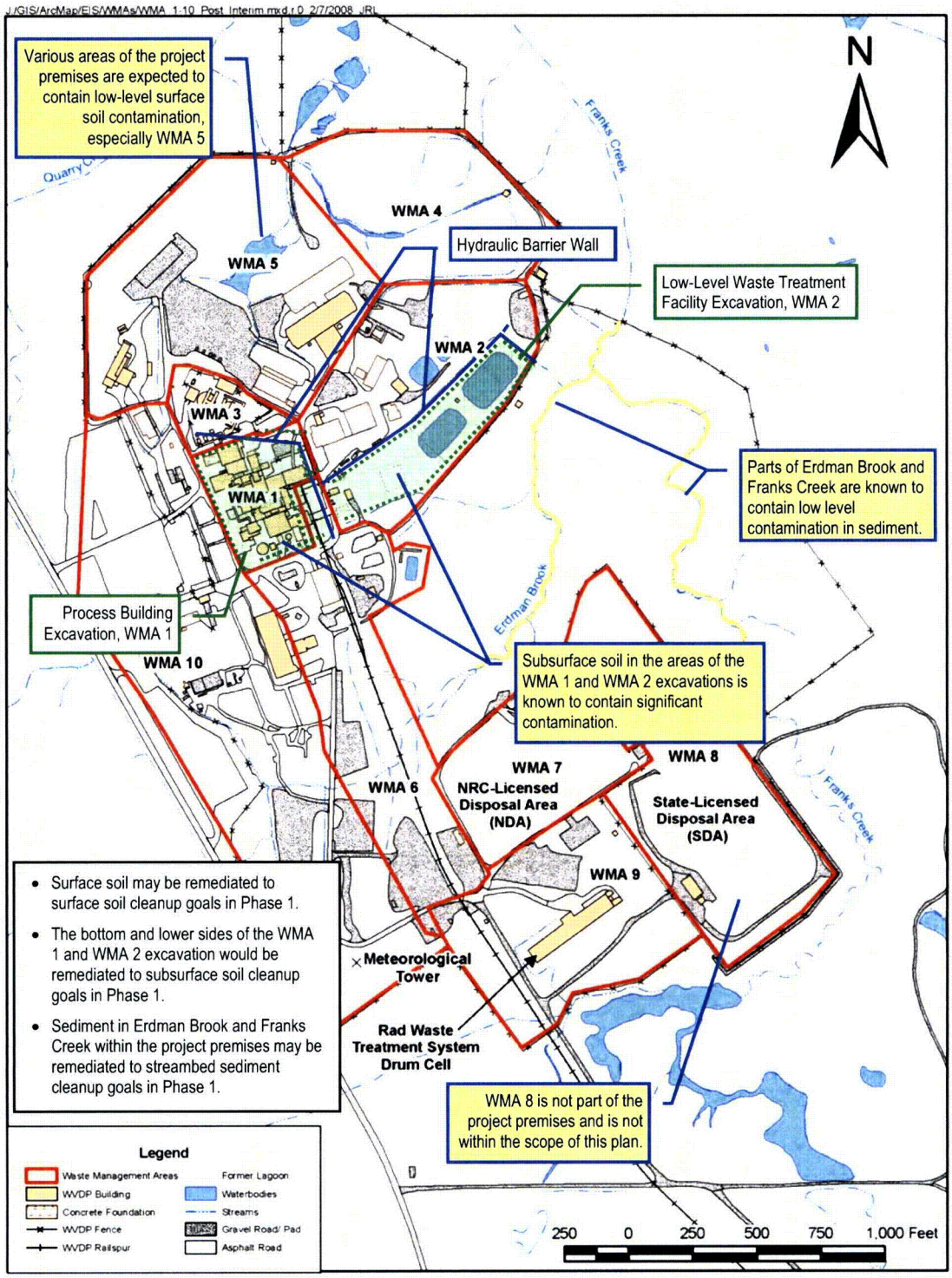
As explained below, DOE is using the DCGL approach in Phase 1 of the proposed decommissioning and then, after remediation of subsurface soil in the two areas of interest, would perform dose modeling using Phase 1 final status survey data to estimate potential future doses from these areas assuming the rest of the project premises were to also be cleaned up to the unrestricted release criteria in 10 CFR 20.1402.

DCGLs and Cleanup Goals

DCGLs are radionuclide-specific concentration limits used during decommissioning to achieve the regulatory dose standard that permit the release of the property and termination of the license. The DCGL applicable to the average concentration over a survey unit is called the $DCGL_W$ and the DCGL applicable to limited areas of elevated concentrations within a survey unit is called the $DCGL_{EMC}$ (NRC 2006). However, Phase 1 of the decommissioning would not result in the release of any property or in termination of the NRC license for the site. As explained below, cleanup goals below the DCGLs are used to ensure that Phase 1 criteria do not limit Phase 2 options.

5.1.2 Context for DCGL Development

Figure 5-1 shows the areas of interest for surface soil, subsurface soil, and streambed sediment for which separate DCGLs have been developed. Each of these areas is discussed below.



- Surface soil may be remediated to surface soil cleanup goals in Phase 1.
- The bottom and lower sides of the WMA 1 and WMA 2 excavation would be remediated to subsurface soil cleanup goals in Phase 1.
- Sediment in Erdman Brook and Franks Creek within the project premises may be remediated to streambed sediment cleanup goals in Phase 1.

Figure 5-1. Areas of Interest – Surface Soil, Subsurface Soil, and Streambed Sediment Within the Project Premises

Surface Soil

As explained in Section 1 of this plan, surface soil and sediment in drainage ditches on the project premises would be characterized for radioactivity early in Phase 1 to better define the nature and extent of radioactive contamination. Section 4.2 summarizes available data on radioactivity in these environmental media. Available data indicate that radioactive contamination is present in some areas but the magnitude and areal extent of this contamination have not been fully defined. Figure 4-6 shows locations where soil and sediment is known to have radioactivity concentrations in excess of background.

Cs-137 concentrations in excess of background have been measured in surface soil samples from all waste management areas (WMAs) where samples have been collected, with the highest measured concentration being 280 pCi/g. Sr-90 concentrations above background have been measured in surface soil samples from several WMAs, with a maximum of 12 pCi/g. Data on other radionuclides in surface soil are very limited, but above-background concentrations of Pu-238, Pu-239/240, and Am-241 have been identified as indicated in Section 4.2.

DCGLs for surface soil based on the unrestricted criteria in 10 CFR 20.1402 serve two purposes:

- They would support remediation of surface soil on selected portions of the project premises in Phase 1 of the proposed decommissioning if this plan were to be revised to provide for such remediation, and
- They would support decision-making for Phase 2 of the decommissioning.

Subsurface Soil

The subsurface soil DCGLs, which are also based on the unrestricted release criteria of 10 CFR 20.1402, apply only to the bottoms and lower sides of the two large excavations to be dug to remove facilities in WMA 1 and WMA 2.[1] Figure 5-2 shows a conceptual cross section view of the planned WMA 1 excavation with representative data on Sr-90 concentrations. Figure 5-3 shows a conceptual cross section view of the planned WMA 2 excavation with representative data. Both excavations would extend one foot or more into the Lavery till, as indicated in Section 7.

As explained in Section 1 and detailed in Section 7, the Process Building and the other facilities in WMA 1 would be completely removed during Phase 1 of the proposed decommissioning, along with the source area of the north plateau groundwater plume. The excavation for this purpose would be approximately 2.8 acres in size and extend more than 40 feet below the ground into the top surface of the unweathered Lavery till. Figure 5-1 shows the approximate location of this excavation.

[1] The subsurface soil DCGLs would be applied to the sides of these excavations at depths greater than three feet below the surface; the surface soil DCGLs would be applied to the portions of the excavation sides closer to the ground surface. Note that the sides of the excavations that are upgradient or cross-gradient (i.e., not hydraulically downgradient) of the contamination source are not expected to be contaminated.

These DCGLs may also be applicable to excavations made in Phase 2 of the decommissioning depending on the approach selected for Phase 2 and other factors if the conceptual model described in this section is representative of the Phase 2 conditions.

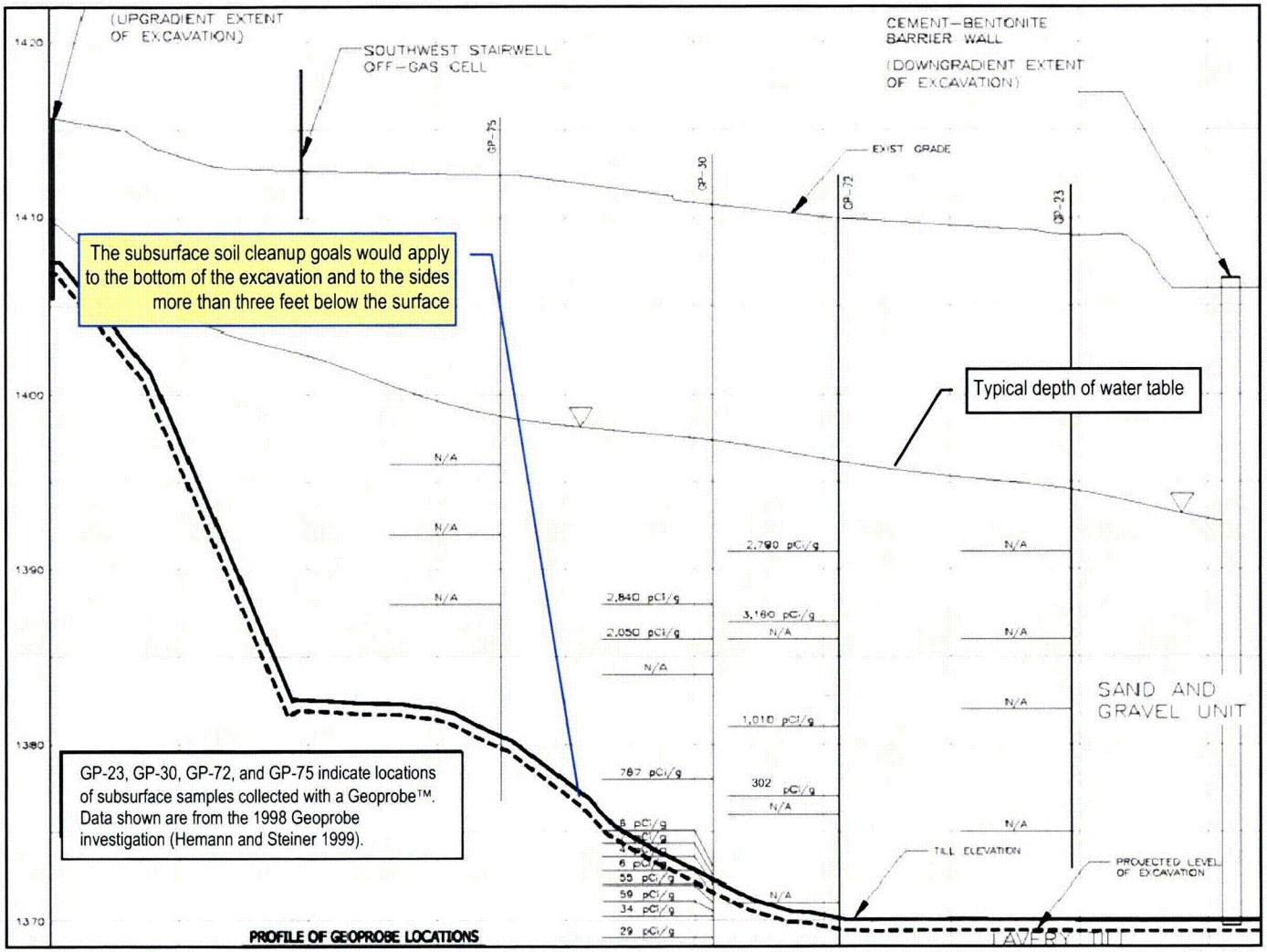


Figure 5-2. Conceptual Cross Section View of WMA 1 Excavation With Representative Data on Sr-90 Concentrations
(See Section 4.2 for more data and 7 for the excavation details.)

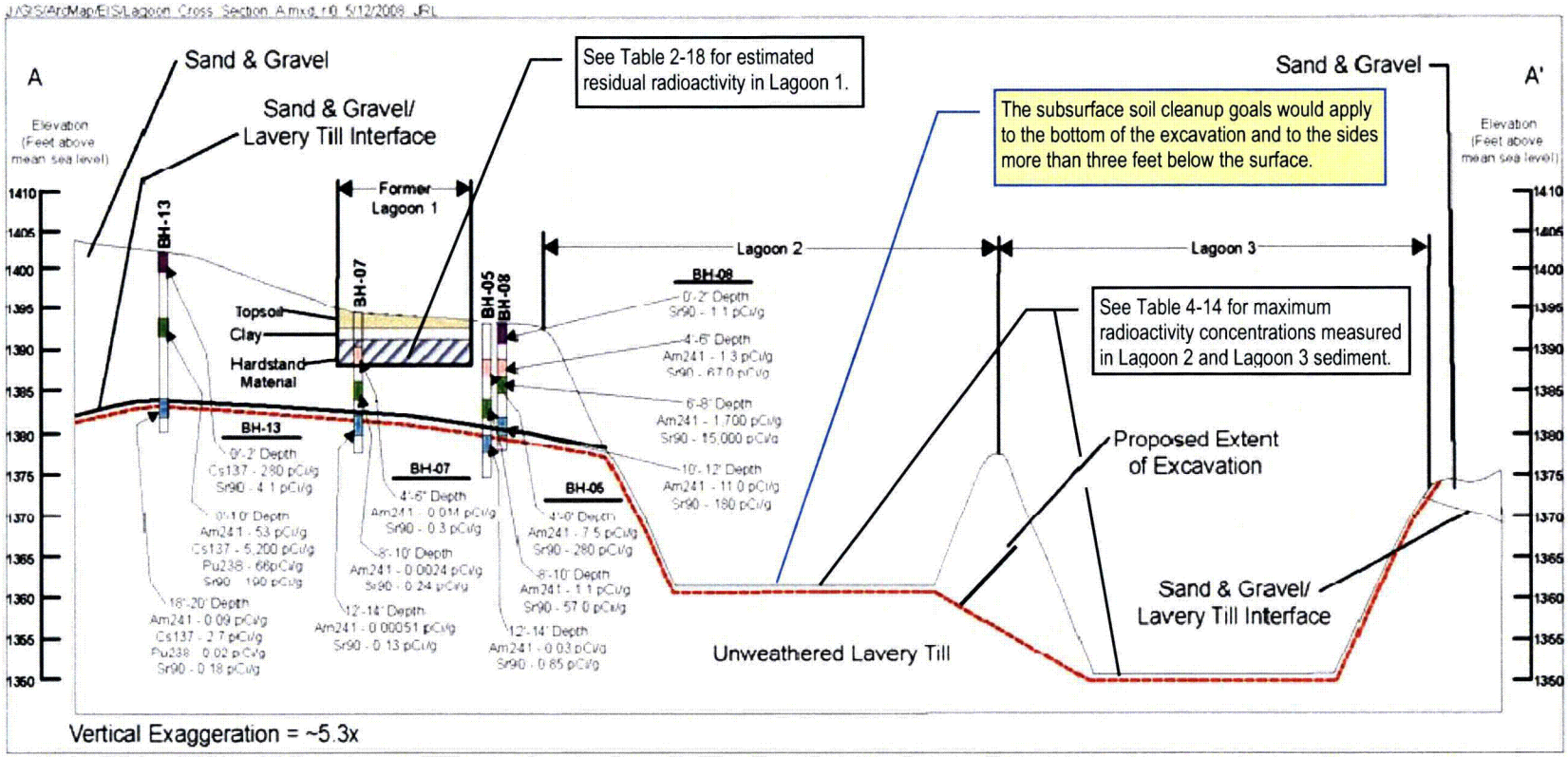


Figure 5-3. Conceptual Cross Section View of WMA 2 Excavation With Representative Data on Subsurface Soil Contamination
(See Section 4.2 for more data and 7 for excavation details. Analytical data shown are 1993 data from WVNSCO and D&M 1997.)

Available data on radioactive contamination in subsurface soil in WMA 1 described in Section 4.2 show Sr-90 to be the dominant radionuclide at depth. Figure 4-8 shows key data, which include three samples from several feet into the unweathered Lavery till that show Sr-90 concentrations of 13 pCi/g, 5.6 pCi/g, and 2.2 pCi/g at depths in the 35 to 40 feet range.

Other radionuclides with measured above-background concentrations in subsurface soil in WMA 1, with their maximum concentrations and the associated sample depth, include: Tc-99 (19 pCi/g at 19-23 feet), Cs-137 (31 pCi/g, at 27 to 29 feet), Pu-241 (15 pCi/g at 21 to 23 feet), and Am-241 (0.1 pCi/g, 19 to 23 feet). Table 5-1 shows the maximum measured radionuclide concentrations in the Lavery till in the areas of the large excavations in WMA 1 and WMA 2. Data in the Lavery till in these areas are limited – the complete set of data is provided in Table C-4 of Appendix C.

Table 5-1. Measured Maximum Lavery Till Radionuclide Concentrations(1)

Nuclide	WMA 1 Excavation Area		WMA 2 Excavation Area	
	Result (pCi/g)	Depth (ft)	Result (pCi/g)(3)	Depth (ft)
C-14	<8.6E-02	40-42	None	None
Sr-90	5.9E+01	38.5-39	8.5E-01(4)	12-14
Tc-99	<2.6E-01	40-42	None	None
I-129	<2.3E-01	40-42	None	None
Cs-137	2.6E-02	26-28	4.5E-01(4)	12-14
U-232	<7.4E-03(2)	36-38	1.2E-02(4)	12-14
U-233/234	1.6E-01	26-28	2.2E-01(5)	12-14
U-235	<5.8E-03	26-28	<6.6E-03(5)	12-14
U-238	1.1E-01	26-28	1.5E-01(5)	12-14
Pu-238	<4.8E-03(2)	36-38	1.0E-02(4)	12-14
Pu-239/240	<4.8E-03(2)	36-38	<6.2E-03(5)	12-14
Pu-241	1.3E+00	26-28	9.5E-01(5)	12-14
Am-241	<9.6E-03	26-28	3.0E-02(4)	12-14

NOTES: (1) Data are from the 1993 RCRA facility investigation and the Geoprobe® studies described in Section 4. Data for C-14, Tc-99, and I-129 were taken from only one sample at location GP80-98.

(2) From location BH-21A shown in Figure 4-8.

(3) Higher concentrations were measured at location BH-08, but the BH-08 sample contained material from the sand and gravel layer as well as from the Lavery till. The location of this sample and BH-5 are shown in Figure 5-3.

(4) From the lowest sample collected at location BH-05, just below the surface of the Lavery till, as shown in Figure 5-3.

(5) From location BH-07 shown in Figure 5-3.

Additional Characterization Planned

The characterization program to be undertaken early in Phase 1 of the decommissioning as described in Section 9 would provide additional data on radioactivity in subsurface soil in WMA 1 and WMA 2 and lagoon sediment in WMA 2. As noted in Section 4, additional characterization measurements being taken in 2008 are expected to somewhat better define subsurface contamination in both areas.

The actual depth of the WMA 1 excavation would be based on removal of soil exceeding the subsurface soil cleanup goals, as explained in Section 7. The excavation would extend at least one foot into the Lavery till, as noted previously, and this is the point where the cleanup goals would apply. The configuration of the residual source would therefore be similar to the bottom of the excavation shown in the representative cross section in Figure 5-2.

Figure 5-1 also shows the approximate location of the major excavation in WMA 2. As explained in Section 1 and detailed in Section 7, a single excavation would be made to remove Lagoons, 1, 2, and 3, the interceptors, the Neutralization Pit, and the Solvent Dike. The area of this excavation would be approximately 4.2 acres and its depth would vary from approximately 12 feet on the southwest end to approximately 26 feet on the northeast end.[2]

Figure 5-3 shows a conceptual cross section of the WMA 2 excavation. This figure also shows representative data on subsurface radioactivity. As indicated on the figure, Table 2-18 provides an estimate of residual radioactivity in Lagoon 1 and Table 4-14 shows maximum radionuclide concentrations measured in sediment in Lagoon 2 and Lagoon 3.

As indicated in order-of-magnitude estimates in Table 2-18, Cs-137 (at 510 curies) is expected to dominate the radioactivity in Lagoon 1. Other radionuclides expected to be present include Pu-241 (134 curies), Sr-90 (17 curies), and Pu-238 (6.4 curies). Table 4-14 shows significant concentrations of Sr-90, Cs-137, Pu-238, Pu-239/240, and Am-241 in Lagoon 2 sediment and lower concentrations of these radionuclides in Lagoon 3 sediment.

The actual depth of the WMA 2 excavation would be based on removal of soil exceeding the subsurface soil cleanup goals, as explained in Section 7. The excavation would extend at least one foot into the Lavery till or, in the cases of Lagoon 2 and Lagoon 3, approximately two feet below the bottom the lagoons, which extend into the Lavery till. The configuration of the residual source would therefore be similar to the bottom of the excavation shown in the representative cross section in Figure 5-3.

While the subsurface soil cleanup goals serve as the remediation criteria for the two excavations as specified in Section 7, actual residual contamination levels in the Lavery till are expected to be well below these criteria. The concentrations of Sr-90 and Cs-137 are expected to be of the same order of magnitude as the lower surface soil cleanup goals.

[2] The 26-foot estimate is based on using the ground surface adjacent to Lagoon 3 as a reference point. The excavation is expected to extend several feet below the bottoms of Lagoons 2 and 3 to remove sediment with radioactivity concentrations above DCGLs.

This conclusion is based on contamination data shown in Table 5-1 and the relative impermeability of the Lavery till to radionuclide migration compared to the sand and gravel layer above it.

Streambed Sediment

Streambed sediment refers only to sediment in Erdman Brook and the portion of Franks Creek running through the project premises. Surface soil DCGLs would be applied to sediment in ditches and in other parts of the project premises, with the subsurface DCGLs being applied to the bottom of Lagoons 2 and 3. Unique DCGLs are appropriate for Erdman Brook and Franks Creek because the areas of these streams would not support farming or grazing of livestock as would other areas of the project premises, owing to the steep stream banks.

Section 4.2 summarizes the limited available data on radioactivity in the sediment of Erdman Brook and the portion of Franks Creek on the project premises. Figure 4-6 shows sample locations, with five in Erdman Brook and four in Franks Creek. Table 4-22 shows the highest measured concentrations of Cs-137 and other radionuclides. The highest measured Cs-137 concentration was 100 pCi/g and the highest Sr-90 concentration was 10 pCi/g. Section 4.2 describes a hot spot found in Erdman Brook in 1990 with a gamma radiation level of 3000 μR/h; a sample collected at that location showed 10,000 pCi/g Cs-137. The characterization program to be undertaken early in Phase 1 would provide additional data in radioactivity in the sediment of the two streams.

DCGLs for streambed sediment based on the unrestricted use criteria in 10 CFR 20.1402, like the surface soil DCGLs, serve two purposes:

- They would support remediation of contaminated sediment in Erdman Brook and the portion of Franks Creek on the project premises in Phase 1 of the proposed decommissioning if this plan were to be revised to provide for such remediation, and
- They would support decision-making for Phase 2 of the decommissioning.

5.1.3 Context for the Integrated Dose Assessment

Three sets of DCGLs have been developed as described in Section 5.2 to be applied to the particular areas of interest, that is:

- Surface soil DCGLs for surface soil and sediment in drainage ditches on the project premises (except for the sediment in Erdman Brook and Franks Creek), and for the sides of the WMA 1 and WMA 2 excavations from the ground surface to three feet below the surface;
- Subsurface soil DCGLs for the bottoms of the WMA 1 and WMA 2 excavations and for the excavation sides more than three feet below the ground surface; and
- Streambed sediment DCGLs for sediment in Erdman Brook and the portion of Franks Creek on the project premises.

Each set of DCGLs was developed as if the area of interest remediated to the applicable DCGLs were the only area to which a hypothetical future resident or recreationist might be exposed. However, it is more likely that a variety of receptors would be exposed to multiple sources under a range of land use scenarios. Considering each source independently allows for flexibility in subsequent combined dose evaluations, as discussed further in Section 5.3.

Phase 1 and Phase 2 Sources

Inherent in the proposed phased decision-making approach is the concept of Phase 1 and Phase 2 sources. Figure 5-4 identifies these different sources.

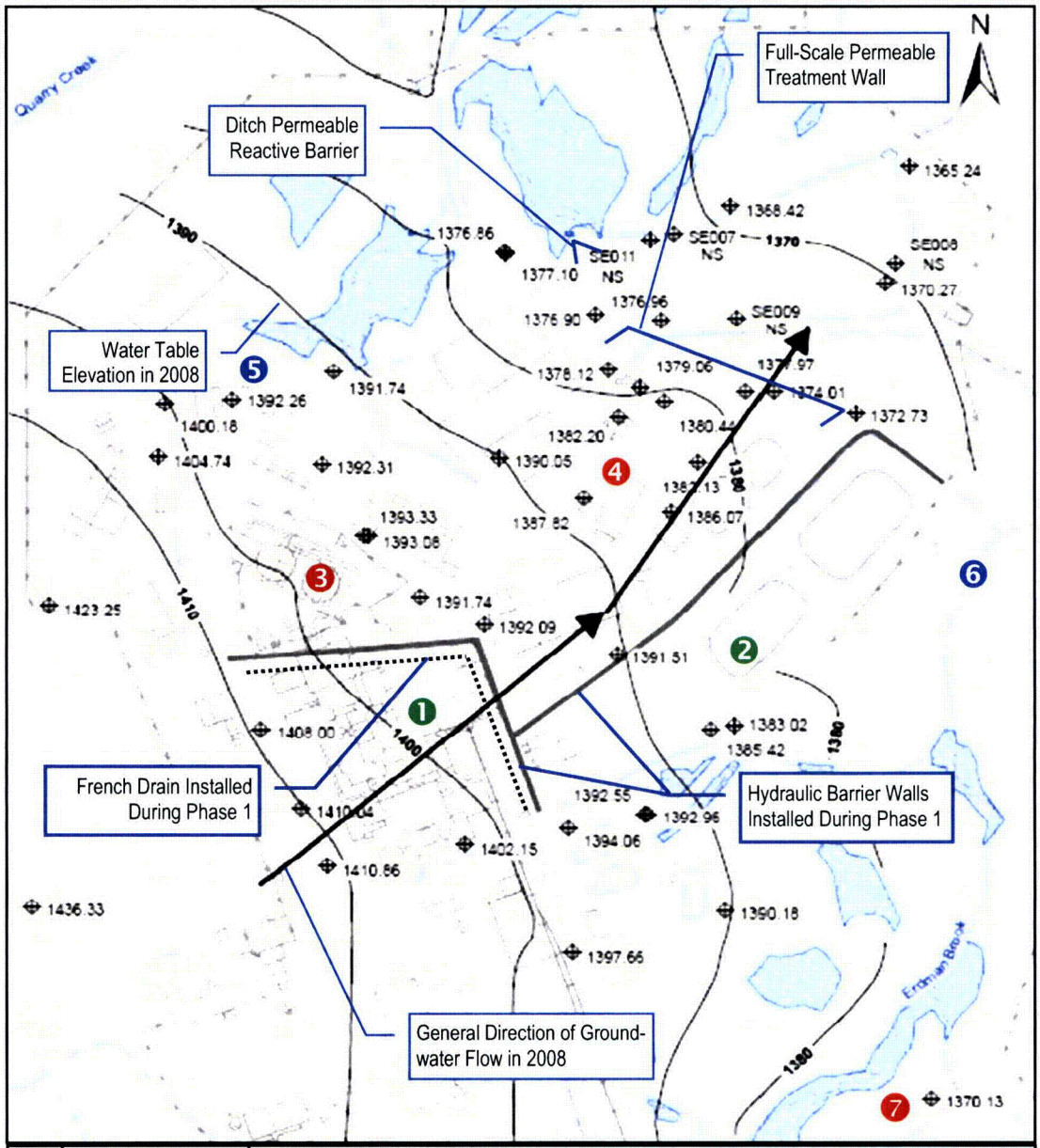
Phase 1 sources are those to be remediated during Phase 1 of the proposed decommissioning: mainly the WMA 1 area and the area in WMA 2 to be excavated. The surface soil and streambed sediment sources within the project premises may or may not be remediated in Phase 1[3]. Based on current characterization data, the main Phase 2 sources are the non-source area of the north plateau groundwater plume in WMA 2, WMA 4, and WMA 5; the Waste Tank Farm in WMA 3, and the NRC-Licensed Disposal Area (NDA) in WMA 7.

The table at the bottom of the Figure 5-4 shows the approximate amounts of total radioactivity in the different source areas based on estimates provided in Section 4. In this illustration, the remediated WMA 1 and WMA 2 excavated areas are the Phase 1 sources. The Waste Tank Farm, the non-source area of the north plateau groundwater plume, and the NDA are the Phase 2 sources. Low-level contamination in surface soil and streambed sediment – which may or may not be remediated during Phase 1 – could be either be a Phase 1 (remediated) or Phase 2 (remediated or not) source, with the potential impact from these sources much smaller than for the others.

Figure 5-4 shows other features of the project premises at the conclusion of the Phase 1 proposed decommissioning activities that could potentially influence future doses from residual radioactivity on the project premises:

- Groundwater flow, with the water table in the sand and gravel unit on the north plateau, with elevations expressed in feet above mean sea level, and the current pre-remediation general direction of groundwater illustrated on the figure;
- The two north plateau groundwater plume control measures to be installed before Phase 1 of the proposed decommissioning begins, the full-scale Permeable Treatment Wall and the Permeable Reactive Barrier; and
- The hydraulic barrier walls to be installed during Phase 1 of the proposed decommissioning as described in Section 7 and the French drain to be emplaced upgradient of the WMA 1 hydraulic barrier wall.

[3] As noted in Section 1, surface soil and sediment are to be remediated only in the Process Building-Vitrification Facility and Low-Level Waste Treatment Facility excavation areas during the proposed Phase 1 decommissioning activities. Soil and sediment in other areas may be remediated in Phase 1 by revision to this plan.



No.	Source	Nature
1	WMA 1 excavation	Area 30-45 ft. below grade remediated below subsurface DCGLs for unrestricted release
2	WMA 2 excavation	Area 12-26 ft. below grade remediated below subsurface DCGLs for unrestricted release
3	Waste Tank Farm	Underground tanks with ~345,000 curies in 2011
4	North plateau plume	Contaminated subsurface soil and groundwater, ~40 curies Sr-90 in 2041
5	Surface soil	Low-level contamination in some areas, may be remediated below DCGLs
6	Streambeds	Low-level contamination, especially Cs-137, may be remediated below DCGLs
7	NDA	NRC-Licensed Disposal Area buried waste containing ~180,000 curies in 2011

Figure 5-4. Sources at the Conclusion of Phase 1 of the Proposed Decommissioning

The effectiveness of these features impacts potential future doses to the receptor and overall contribution to the evaluation of combined dose from all sources.

Potential Conditions at the Conclusion of the WVDP Proposed Decommissioning

To determine whether criteria used in Phase 1 proposed remediation activities could potentially limit the decommissioning options for Phase 2 of the decommissioning, consideration must be given to potential approaches to Phase 2. The Decommissioning EIS evaluates a range of closure alternatives. Two of these alternatives would provide bounding conditions for assessment of whether the criteria used for Phase 1 remediation activities could limit Phase 2 options:

- The site-wide close-in place-alternative, where the major facilities would be closed in place, with residual radioactivity in the Waste Tank Farm and the NDA being isolated by engineered barriers and the non-source areas of the north plateau groundwater plume being allowed to decay in place; and
- The site-wide removal alternative, where the Phase 2 sources would be removed and the entire site remediated to the unrestricted release criteria of 10 CFR 20.1402.

Compatibility of Phase 1 Remediation With the Site-Wide Close-In-Place Alternative

With the site-wide close-in place-alternative, the Phase 2 source areas would remain under NRC license. With Phase 1 of the decommissioning being accomplished as proposed, the contamination remaining in the WMA 1 and WMA 2 excavations would be residual radioactivity at concentrations below the subsurface soil DCGLs located far below the surface and covered with uncontaminated earth.

Under a site-wide close-in-place approach, the remediated Phase 1 areas would be expected to fall within the controlled licensed area because of their close proximity to the Phase 2 source areas. In view of this situation, the proposed remediation of the Phase 1 areas to unrestricted release standards would clearly be compatible with the Phase 2 source areas remaining under license. That is, remediation of the Phase 1 source areas as planned would have no impact on the site-wide close-in place-alternative and would not limit its implementation in any way.

Compatibility of Phase 1 Remediation With the Site-Wide Removal Alternative

Under the site-wide removal alternative, the Phase 2 source areas would be remediated to unrestricted release standards like the Phase 1 source areas. All of the associated radioactive waste would be disposed of offsite. However, while the remediation standards would be the same, the critical group for potential future exposures would not be the same for all parts of the site. Because remediation to unrestricted release standards under Phase 1 of the proposed decommissioning does not preclude achievement of unrestricted release standards under Phase 2, all remedial options may be considered.

However, this situation requires consideration of potential exposures to members of the different critical groups, a matter which is addressed below.

Critical Group

Critical Group means the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances (10 CFR 20.1003).

Section 5.2 describes the critical groups for development of the different DCGLs. The average member of the critical group for development of the surface soil and subsurface soil DCGLs is a resident farmer. The average member of the critical group for development of the streambed sediment DCGLs is a recreationist, that is, a person who would spend time in the Erdman Brook and Franks Creek areas engaged in activities such as fishing and hiking.

One reasonably foreseeable set of circumstances would involve a person engaged in farming at some time in the future on one part of the remediated project premises who also spends time fishing and hiking at Erdman Brook and Franks Creek. This scenario would involve an individual being exposed to two different remediated source areas and being a member of the two different critical groups. Because this scenario is not considered in development of the DCGLs for the different areas of interest, it would be appropriate to consider whether it could result in such a hypothetical individual exceeding the unrestricted dose limit, that is, 25 mrem in one year, and whether the residual radioactivity has actually been reduced to levels that are ALARA in accordance with 10 CFR 20.1402.

Considering the foregoing discussion, Section 5.3 evaluates the potential impacts of this set of circumstance (combined sources of dose to receptor) on the DCGLs and the associated cleanup goals to be used to guide remediation during Phase 1 of the proposed decommissioning.

Two other factors that could potentially affect potential future doses from the remediated Phase 1 areas would be long-term erosion and potential changes in groundwater flow.

5.1.4 Potential Impact of Long-Term Erosion

The potential impact of long-term erosion is a consideration in development of DCGLs for Phase 1 of the proposed decommissioning and for estimating potential future doses from different parts of the project premises assuming that the entire site would be remediated for unrestricted use.

Section 3.5.3 of this plan describes the site geomorphology, including erosion processes such as channel incision, slope movement, and gully formation. Table 3-13 provides information on site erosion rates from various sources.

Detailed erosion studies performed in support of the Decommissioning EIS are described in Appendix F to that document. This appendix describes past studies and recent analyses that made use of two different landscape evolution models, SIBERIA and CHILD. The SIBERIA model is a physically based model that uses average precipitation over a specified timeframe and accounts for both fluvial and diffusional processes that move sediment through a drainage system (Willgoose 2000). The CHILD model performs

simulations like the SIBERIA model but incorporates additional features. Both models were calibrated for the site.

Analyses using these models were performed to predict erosion rates at the WVDP over a 10,000-year time period. The two models predicted a total erosion depth on the central portion of the north plateau generally no greater than 3.2 feet, with the assumption of no climate change over the evaluation period. This rate would amount to about four inches over a 1000-year period.

Limited field data showing actual sheet and rill erosion rates are available as indicated in Table 3-13. The maximum measured erosion among 19 measurements over an 11-year period ending in 2001 was 0.04 feet (approximately 0.5 inch) on the slope of a gully. One spot south of Lagoon 2 showed buildup of 0.04 feet (about 0.5 inch) during that period.

Conclusions that can be drawn from the available field data and the erosion studies detailed in Appendix F of the Decommissioning EIS include:

- The central portion of the north plateau is expected to be generally stable over the next 1000 years;
- The WMA 2 area, which is near the Erdman Brook stream valley, is more susceptible to erosion than the WMA 1 area;
- Existing gullies will propagate, becoming deeper and longer, and new gullies will form, mainly on the edges of the north plateau, if erosion is unchecked;
- Rim widening and channel downcutting could occur in Erdman Brook and Franks Creek;
- With unmitigated erosion, gullies could eventually extend into the areas of Lagoons 1, 2, and 3 during the 1000-year evaluation period; and
- With unmitigated erosion, rim widening and downcutting of Erdman Brook could possibly impact the eastern edge of the areas of these lagoons, especially Lagoon 3.

5.1.5 Potential Changes in Groundwater Flow Fields

Changes in the groundwater flow pattern that might result from installation of the hydraulic barriers shown in Figure 5-1 could increase the potential for recontamination of the areas remediated in Phase 1. Groundwater in the sand and gravel unit on the north plateau currently flows northeast as indicated on Figure 5-4. With this flow pattern, and with the WMA 1 and WMA 2 hydraulic barriers remaining in place, the potential for transport of contaminants by groundwater into the WMA 1 and WMA 2 areas remediated during Phase 1 of the proposed decommissioning from Phase 2 source areas is low.

Appendix D describes the results of an analysis performed to evaluate groundwater flow conditions near these engineered barriers. This analysis suggests that the potential for recontamination of the remediated WMA 1 and WMA 2 areas would not be significantly increased with the engineered barriers in place.

5.1.6 Seepage of Groundwater

Figure 5-5 shows the locations of groundwater seeps on the north plateau. As can be seen in the figure, any groundwater from the seeps located on the project premises runs into Erdman Brook or Franks Creek. (Dames and Moore 1994)

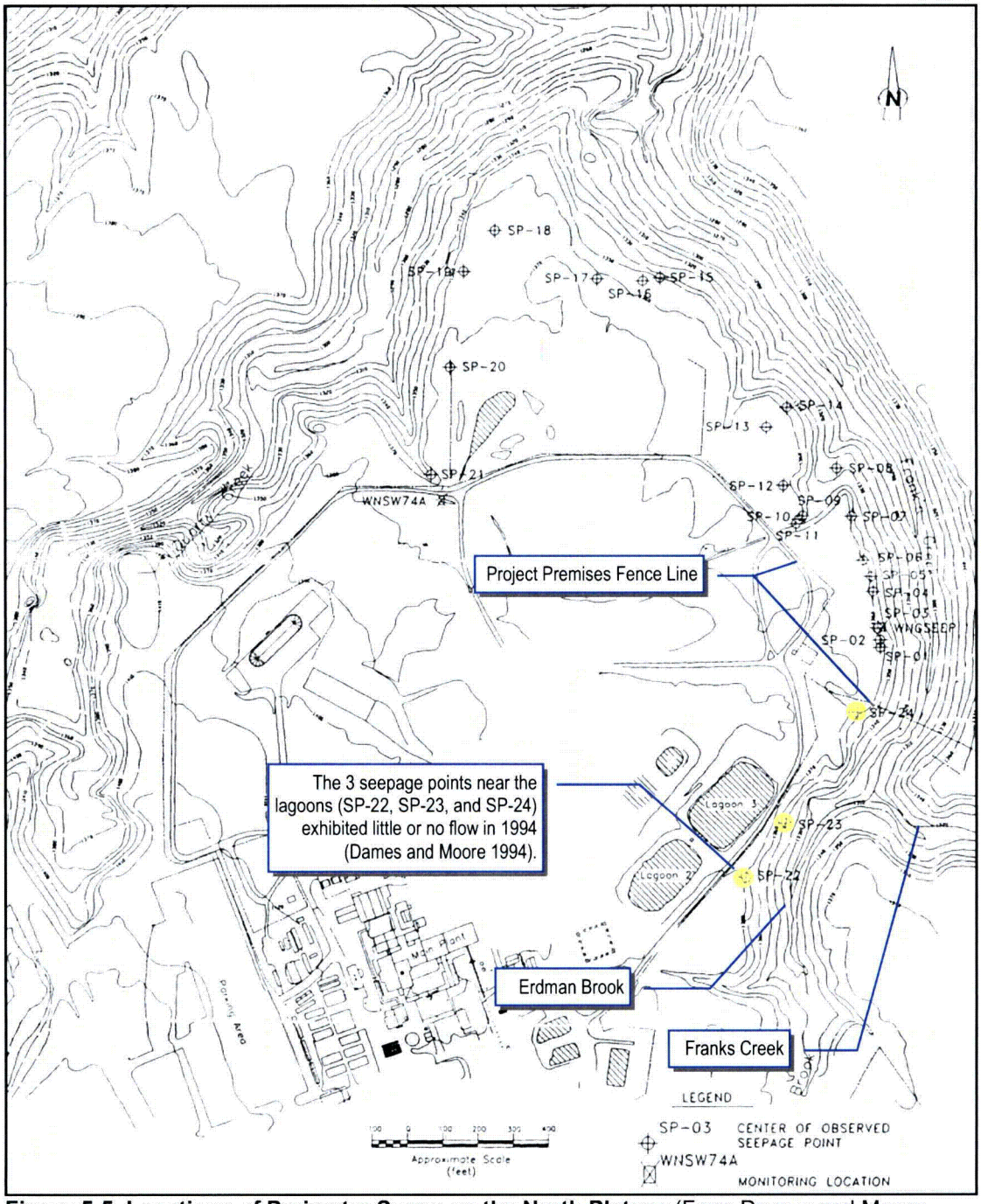


Figure 5-5. Locations of Perimeter Seeps on the North Plateau (From Dames and Moore 1994)

One other factor that could possibly affect conditions following Phase 1 of the proposed decommissioning is seepage of radioactively contaminated groundwater into Erdman Brook and Franks Creek.

As noted previously, surface soil and streambed sediment may be remediated during Phase 1 of the proposed decommissioning if this plan were to be revised to provide for these activities. The presence of groundwater seeps in the Erdman Brook area would be one factor taken into account in any decision to proceed with this remediation, since these seeps could possibly result in recontaminating the sediment in Erdman Brook.

However, the potential for significant radioactivity in seeps in this area following Phase 1 of the proposed decommissioning would be low due to the following factors:

- Any residual radioactivity that might remain in the Lavery till at the bottom of the remediated WMA 2 excavation would be at very low concentrations; and
- Groundwater flow changes with the Phase 1 vertical hydraulic barriers in place, as described in Appendix D, would be expected to substantially reduce the potential for contamination from the non-source area of the north plateau groundwater plume seeping into Erdman Brook.

Another factor that would be taken into account in any decision to proceed with remediation of sediment in Erdman Brook and in the portion of Franks Creek on the project premises during Phase 1 of the proposed decommissioning would be surface water runoff, especially runoff from the two radioactive waste disposal areas on the south plateau. Surface water runoff from both waste disposal sites is potentially contaminated due to surface soil contamination in these areas, although the potential impact on the streams is limited so long as the geomembrane covers for the waste disposal sites are intact.

5.1.7 Potential Impacts on the Kent Recessional Sequence

The potential for impacts on groundwater in the Kent Recessional Sequence from the any residual radioactivity that might remain in the bottom of the WMA 1 and WMA 2 excavated areas has been evaluated and found to be very low.

Groundwater in the sand and gravel unit generally flows to the northeast across the north plateau towards Franks Creek as shown in Figure 5-4. Water balance estimates (Yager 1987 and WVNSCO 1993a) suggest that approximately 60 percent of the groundwater from the sand and gravel unit discharges to Quarry Creek, Franks Creek, and Erdman Brook through surface water drainage discharge points and the groundwater seeps located along the margins of the north plateau that are shown in Figure 5-5.

Approximately two percent of the total discharge from the sand and gravel unit travels vertically downward to the underlying unweathered Lavery till, where groundwater flows vertically downward toward the underlying Kent Recessional Sequence at an average vertical groundwater velocity of 0.20 feet per year (WVNSCO 1993a). The unweathered Lavery till is approximately 30 to 45 feet thick below the planned WMA 1 excavation and 40 to 110 feet thick below the planned WMA 2 excavation (WVNSCO 1993b).

It would take approximately 200 years for groundwater to migrate through the unweathered Lavery till at WMA 1 and WMA 2 assuming a Lavery till thickness of 40 feet

and an average groundwater velocity of 0.20 feet per year. Mobilization and migration of the residual radionuclide inventory at the bottom of the WMA 1 and WMA 2 excavations through the Lavery till groundwater pathway would take even longer considering the sorptive properties of the Lavery till.

Short-lived radionuclides (Sr-90, Cs-137, and Pu-241) will have decayed away during these time frames. The long-lived radionuclide inventory is not an issue as the residual concentrations within the Lavery till are expected to be comparable to background concentrations for surface soil. The residual radionuclide concentrations in the Lavery till in the bottom of the WMA 1 and WMA 2 excavations are expected to be lower than those reported in Table 5-1 and would therefore not significantly impact the Kent Recessional Sequence. Groundwater reaching the Kent Recessional Sequence flows laterally to the northeast at an average velocity of 0.40 feet per year and eventually discharges to Buttermilk Creek.

The potential for impacts on groundwater in Lavery till sand has also been considered.

The Lavery till sand is located 30 to 40 feet below grade within the Lavery till and is recharged by downward groundwater flow from the Lavery till. The Lavery till sand is located south of the WMA 1 excavation (Figure 3-64) and would not be impacted by the Phase 1 excavation of WMA 1.

However, the Lavery till sand underlies approximately 15,000 square feet of the southwestern most portion of WMA 2 near the Solvent Dike (Figure 3-64). The Solvent Dike was originally excavated in 1986 and would be excavated down into the Lavery till during the excavation of WMA 2. Because any residual radionuclide concentrations are expected to be less than those reported in Table 5-1, groundwater flow from the Lavery till would not significantly impact the Lavery till sand.

5.1.8 General Dose Modeling Process

The general process for the dose modeling described in Section 5.2 and 5.3 is illustrated in Figure 5-6. As indicated in the figure, the process involves the following major steps:

- Calculating the DCGLs,
- Performing parameter sensitivity analyses and refining the conceptual models and the DCGLs as appropriate based on the results,
- Analyzing a combined source area exposure scenario,
- Factoring in the results of the ALARA analysis described in Section 6,
- Establishing cleanup goals (target levels below the DCGLs) to ensure that the degree of remediation in Phase 1 of the proposed decommissioning would not limit Phase 2 options,
- Characterizing surface soil, subsurface soil, and streambed sediment early in Phase 1,

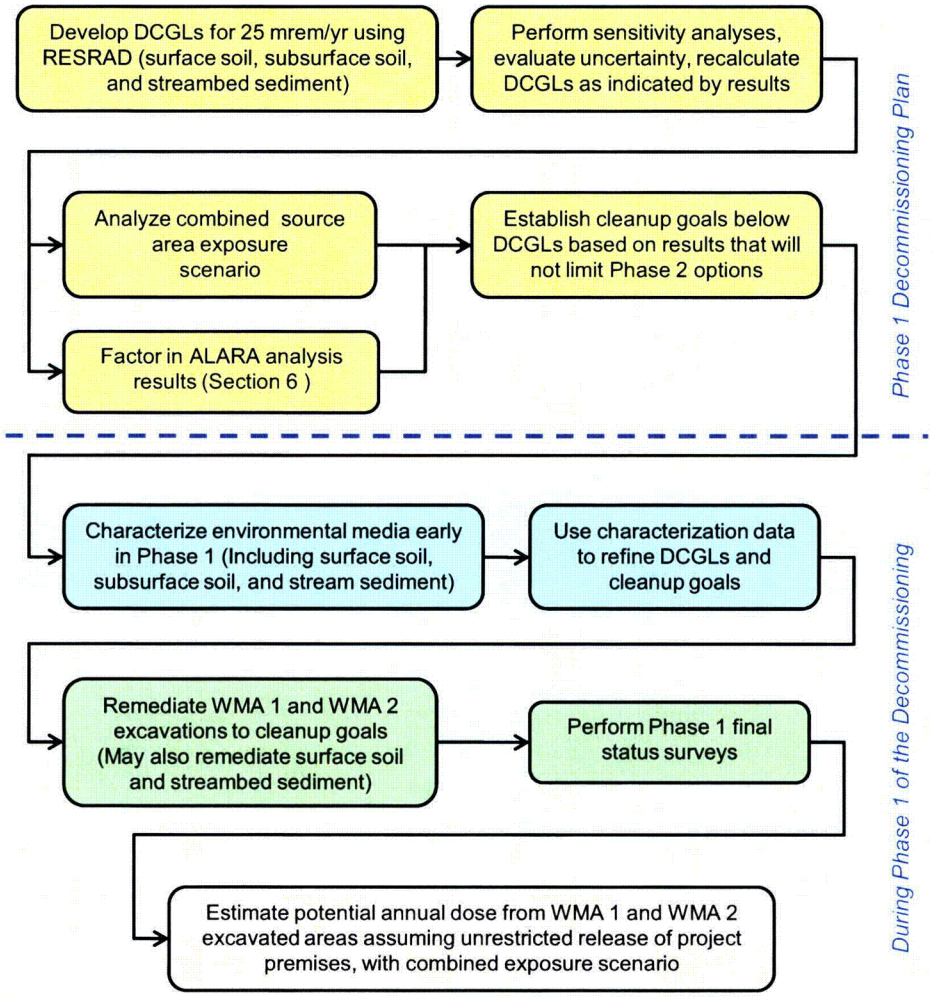


Figure 5-6. General Dose Modeling Process

- Refining the DCGLs and cleanup goals based on the resulting data[4],
- Completing remediation of the WMA 1 and WMA 2 excavations to the cleanup goals,
- Performing Phase 1 final status surveys in the remediated Phase 1 areas, and
- Making an estimate of the potential future doses for the remediated WMA 1 and WMA 2 areas using these data.

[4] The characterization to be performed early in Phase 1, which is described in Section 9, would provide data that may be useful in better defining source geometry in the conceptual model. For example, if the depth of surface soil contamination were to be found to typically be about six inches, rather than three feet (one meter) as used in the conceptual model, then the conceptual model thickness would be changed and the DCGLs recalculated. While DCGLs are developed for 18 radionuclides, characterization data may indicate that some radionuclides may be dropped from further consideration. This could be the case, for example, if one or more of the 18 radionuclides do not show up above the minimum detectable concentration in any of the soil or sediment samples.

Note that use of a surrogate radionuclide such as Cs-137 to represent all radionuclides in a mixture of radionuclides is not practical at this time because available data are not sufficient to establish radionuclide distributions in environmental media. This matter is discussed further in Section 5.4.3.

5.2 DCGL Development

This section describes the conceptual models used for developing DCGLs for surface soil, subsurface soil, and streambed sediment. It then describes the mathematical model (RESRAD) used to calculate these DCGLs and identifies the DCGLs. It concludes with a discussion of input parameter sensitivity and uncertainty.

The analyses simulate the behavior of residual radioactivity over 1000 years, a period during which peak annual doses from the radionuclides of primary interest would be expected to occur. DCGLs have been developed for residual radioactivity that would result in 25 mrem per year dose to the average member of the critical group for each of the following 18 radionuclides of interest:

Am-241	Cs-137	Pu-239	Tc-99	U-235
C-14	I-129	Pu-240	U-232	U-238
Cm-243	Np-237	Pu-241	U-233	
Cm-244	Pu-238	Sr-90	U-234	

Early studies related to the long-term performance assessment for residual radioactivity at the site included consideration of the initial inventory of radionuclides received on site and their progeny. This list was screened to eliminate short-lived radionuclides and those radionuclides present in insignificant quantities. Thirty radionuclides of interest remained after this screening process. These radionuclides were important to worker dose and/or long-term dose from residual radioactivity.

In characterization of radionuclides in the area of the Process Building, the north plateau groundwater plume, and the lagoons, it was determined that 18 of the 30 radionuclides were important for the development of Phase 1 DCGLs. These radionuclides were selected based on screening of simplified groundwater release and intrusion scenarios for north and south plateau facilities. The screening indicated that other radionuclides would in combination contribute less than one per cent of potential dose impacts at the individual facility.

The list of radionuclides for which DCGLs are initially developed would be expanded if necessary following completion of soil and sediment characterization early in Phase 1 of the proposed decommissioning. If other radionuclides show up in concentrations significantly above the minimum detectable concentrations, additional DCGLs would be developed for these radionuclides and their progeny, as appropriate. Conversely, if any of the 18 radionuclides of interest fail to show up in concentrations above the minimum detectable concentrations, then they may be omitted from the final DCGLs for the Phase 1 actions.

As explained in Section 1, the DCGLs for Sr-90 and Cs-137 were developed to incorporate a 30-year decay period from 2011. That is, achieving residual radioactivity

levels less than the DCGLs would ensure that dose criteria of 10 CFR 20.1402 would be met in 2041, around the time when the vitrified HLW canisters are expected to be shipped to the federal geologic repository.[5] Although a 30-year decay period could have been applied to all radionuclides, Sr-90 and Cs-137 were selected based on their prevalence in soil and sediment contamination, their expected peak doses at the onset of exposure, and the short half lives of these particular radionuclides.

5.2.1 Conceptual Models for DCGL Development

The conceptual model for development of surface soil DCGLs is described first.

Surface Soil Conceptual Model

Figure 5-7 illustrates the conceptual model for surface soil DCGL development. As is evident from this figure, which was adapted from the RESRAD Manual (Yu, et al. 2001), the basic RESRAD model is used.

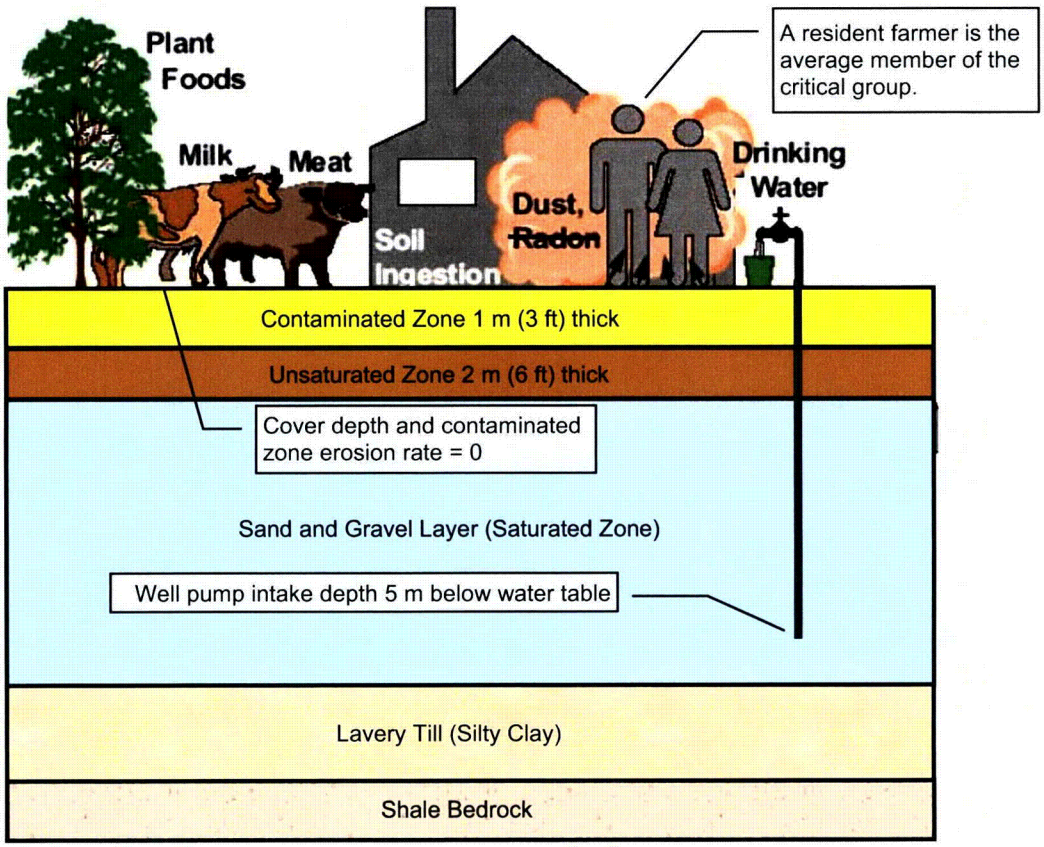


Figure 5-7. Conceptual Model for Surface Soil DCGL Development

[5] This approach would support any license termination actions that may take place in Phase 2 of the decommissioning, which could not be finalized before 2041 considering current expectations about shipment of the vitrified HLW canisters and the scope of effort necessary to achieve an unrestricted release of major portions of the project premises.

RESRAD is a computer model designed to estimate radiation doses and risks from RESidual RADioactive materials (Yu, et al. 2001). DOE Order 5400.5 designates RESRAD for the evaluation of radioactively contaminated sites, and NRC has approved the use of RESRAD for dose evaluation by licensees involved in decommissioning. RESRAD capabilities are discussed further in Section 5.2.2.

A resident farmer is the average member of the critical group for development of surface soil DCGLs. The hypothetical residence and farm are assumed to be located on a part of the project premises impacted solely by radioactivity in surface soil.

Other possible critical groups were considered. However, a resident farmer was determined to be most limiting because such an individual would be engaged in a wider range of activities that could result in greater exposure to residual radioactivity in surface soil than other critical groups considered.

The resident farmer would be impacted by a number of exposure pathways with long exposure durations. This hypothetical individual would utilize significant amounts of groundwater that involves consideration of secondary exposure pathways such as household water use, irrigation, and watering livestock. The resident farmer scenario also is consistent with current and projected future land uses for Cattaraugus County as discussed in Section 3.

Note that the geological units shown in Figure 5-7 are representative models of the north plateau as shown in Figure 3-6. Figure 3-7 shows that the geological units on the south plateau are different in that the sand and gravel unit does not extend to that area. However, DCGLs developed using the conceptual model illustrated in Figure 5-7 are appropriate for surface soil on the south plateau because the input parameters used in the modeling for the north plateau would generally be conservative for the south plateau. For example, site-specific distribution coefficients for the sand and gravel unit (where available) are typically lower than those for the Lavery till, and use of the lower values results in faster radionuclide movement though soil in the north plateau model, and less time for radioactive decay to take place.[6]

Table 5-2 shows the exposure pathways evaluated for development of the surface soil DCGLs.

Table 5-2. Exposure Pathways for Surface Soil DCGL Development

Exposure Pathways	Active
External gamma radiation from contaminated soil	Yes
Inhalation (airborne radioactivity from re-suspended contaminated soil)	Yes
Plant ingestion (produce impacted by contaminated soil and groundwater sources)	Yes
Meat ingestion (beef impacted by contaminated soil and groundwater sources)	Yes

[6] Table C-2 of Appendix C shows that site-specific K_d values for neptunium, plutonium, and strontium in the sand and gravel unit are used in the surface soil model. Table 3-20 of Section 3 shows the basis for these values.

Table 5-2. Exposure Pathways for Surface Soil DCGL Development

Exposure Pathways	Active
Milk ingestion (impacted by contaminated soil and groundwater sources)	Yes
Aquatic food ingestion	No(1)
Ingestion of drinking water (groundwater impacted by contaminated soil)	Yes
Ingestion of drinking water (from surface water)(2)	No
Soil ingestion (while farming and residing on contaminated soil)	Yes
Radon inhalation	No(3)

NOTES: (1) Fish ingestion is considered in development of the streambed sediment DCGLs and in the combined scenario discussed in Section 5.3.

(2) Groundwater was assumed to be the source of all drinking water because the low flow volumes in Erdman Brook and Franks Creek could not support the resident farmer. Also, use of surface water would not be as conservative as groundwater since surface water is diluted by runoff from the entire watershed area. Incidental ingestion of water from the streams is evaluated in development of the streambed sediment DCGLs as shown in Table 5-6.

(3) For the standard resident farmer scenario, the radon pathway is not considered (Appendix J, NRC 2006).

RESRAD requires a variety of input parameter values to completely describe the conceptual model. All of the input parameters for development of the surface soil DCGLs appear in Appendix C. Table 5-3 identifies selected key input parameters.

Table 5-3. Key Input Parameters for Surface Soil DCGL Development(1)

Parameter (Units)	Value	Basis
Area of contaminated zone (m^2)	1.0E+04	Necessary for subsistence farming.
Thickness of contaminated zone (m)	1.0E+00	Conservative assumption.(2)
Cover depth (m)	0	Contamination on surface.
Contaminated zone erosion rate (m/y)	0	Conservative assumption.(3)
Well pump intake depth below water table (m)	5.0E+00	Consistent with water table.
Well pumping rate (m^3/y)	5.72E+03	See Table C-2.
Unsaturated zone thickness (m)	2.0E+00	Typical for north plateau.
Distribution coefficient for strontium (mL/g)	6.16E+00	See Table C-2.
Distribution coefficient for cesium (mL/g)	2.8E+02	See Table C-2.
Distribution coefficient for americium (mL/g)	1.9E+03	See Table C-2.

NOTES: (1) See Appendix C for other input parameters. Metric units are used here because they are normally used in RESRAD.

(2) Available data discussed in Sections 2.3.2 and 4.2 suggest that most contamination will be found within a few inches of the surface except where the north plateau groundwater plume has impacted subsurface soil.

(3) This assumption is conservative because it results in no depletion of the source through erosion.

Key features of this conceptual model and key assumptions include:

- The areal extent of surface soil contamination, which has not been well defined, can be represented by a distributed source spread over a relatively large area (10,000 square meters or approximately 2.5 acres);
- The average depth of contamination (contamination zone thickness) is approximately 3.3 feet (one meter), a conservative assumption for the site;
- All water use (e.g., household, crop irrigation, and livestock watering) is from contaminated groundwater;
- Adequate productivity from a well pumping from the aquifer would be available in the future to support a subsistence farm;
- Soil erosion (i.e., source depletion) does not occur over the 1,000-year modeling period;
- The non-dispersion groundwater model is used because of the large contaminated area consistent with applicable guidance (Yu, et al. 2001, Appendix E);
- The groundwater flow regime under the post-remedial conditions is unchanged from the current configuration (e.g. flow direction, aquifer productivity); and
- DCGLs that reflect 30 years of decay (i.e., apply to the year 2041) are appropriate for Sr-90 and Cs-137. Although a 30-year decay period could have been applied to all radionuclides, Sr-90 and Cs-137 were selected based on their prevalence in surface soil, their expected peak doses at the onset of exposure, and the short half lives of these particular radionuclides, as noted previously.

Subsurface Soil Conceptual Model

Figure 5-8 illustrates the conceptual model for subsurface soil DCGL development. The basic RESRAD model is used as with development of surface soil DCGLs, with a resident farmer being the average member of the critical group. The hypothetical residence and farm are assumed to be located in the remediated WMA 1 area. Exposure to the subsurface radioactivity occurs following intrusion and surface dispersal when installing a water collection cistern.

Other possible critical groups were considered as with the conceptual model for surface soil DCGLs. However, a resident farmer was determined to be most limiting because such an individual would be engaged in a wider range of activities that could result in greater exposure to residual radioactivity in subsurface soil than other critical groups considered.

Consideration was given to a home construction scenario with the basement in the hypothetical home extending 10 feet below the surface. However, this scenario was not considered to be plausible because any contaminated subsurface soil would be more than 10 feet below the surface in the remediated WMA 1 and WMA 2 areas (the bottoms of the excavations would be more than 10 feet below the surface and uncontaminated soil would be used to backfill the excavations).

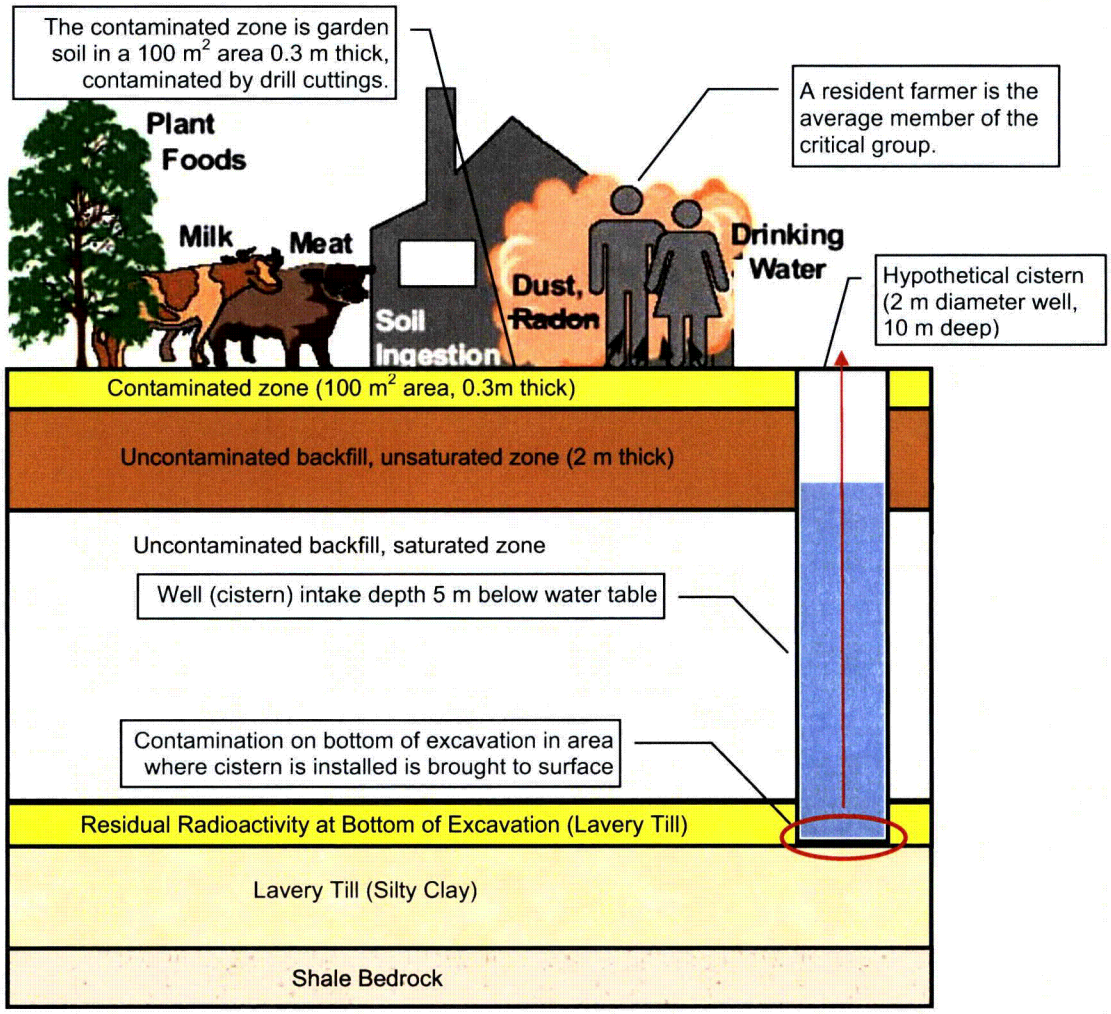


Figure 5-8. Conceptual Model for Subsurface Soil DCGL Development

Note that Section 7 specifies that the uncontaminated backfill as shown in the figure would be soil obtained from outside of the Center from an area that has not been impacted by site radioactivity. No soil removed during the excavation work would be used in filling the excavation, even if that soil were determined to be uncontaminated.

Consideration of NRC Guidance Related to Buried Radioactivity

Also considered in development of this conceptual model was NRC guidance related to assessment of buried radioactivity in Appendix J to NUREG-1757, Volume 2 (NRC 2006). This guidance applies to cases where radioactive material is buried deep enough that an external dose is not possible in its existing configuration; any radioactivity remaining at the bottom of the WMA 1 and WMA 2 excavations would meet this condition, and the WVDP situation is consistent with the intent of the guidance.

The NRC notes that a conservative analysis could be performed that assumes all of the material is spread on the surface. It describes two alternative exposure scenarios: (1) leaching of the radionuclides to groundwater, which is then used by a residential farmer, and (2) inadvertent intrusion into the buried radioactive material, with part of the radioactivity being spread across the surface where this fraction causes exposure to a resident farmer through various pathways. NRC further notes that

> "The second alternative exposure scenario encompasses all the exposure pathways and, although not all of the source term is in the original position, leaching will occur both from the remaining buried residual radioactivity (if there is any) and the surface soil. Unless differences in the thickness of the unsaturated zone will make a tremendous difference in travel time to the aquifer, the groundwater concentrations should be similar and, therefore, will generally result in higher doses than the first alternate scenario."

The surface soil DCGLs discussed previously represent the case where all of the radioactive material of interest is located on the surface; as explained in Section 6, possible application of these DCGLs to the subsurface soil of interest would be addressed in the ALARA analysis. DOE has selected the second alternative exposure scenario – inadvertent intrusion into the buried material, that is, into any residual radioactivity at the bottom of the WMA 1 and WMA 2 excavations – as the basis for development of the subsurface soil DCGLs. NRC discusses in Appendix J to NUREG-1757 (NRC 2006) the use of RESRAD in analysis of the inadvertent intrusion scenario, which DOE has implemented here.

This conceptual model has the following features, some of which are indicated on Figure 5-8:

- The initial modeled source of contamination brought to the surface consists of residual radioactivity in an area two meters (about six feet) in diameter and one meter (about three feet) thick, the top surface of which lies nine meters (about 30 feet) below the ground surface. The contamination assumed to be in this volume of subsurface soil represents the residual radioactivity of interest at the bottom of the WMA 1 or WMA 2 excavation. The exposure occurs when the subsurface radioactivity is deposited on the ground surface where it can result in exposure to members of the critical group through various pathways.

- For conservatism the hypothetical well is assumed to have a large diameter representative of a cistern, rather than the smaller diameter of a typical water supply well (eight inches). The larger diameter provides for a greater volume of contamination being brought to the surface, and is therefore conservative compared to the typical well diameter.
- The nine meters (about 30 feet) of uncontaminated backfill above the initial source of contamination comingles with the contaminated soil, and the mixture is assumed to uniformly cover a cultivated garden area of 100 square meters (about 1000 square feet), i.e., a small portion of the 10,000 square meter garden, to a depth of 0.3 meter (one foot).[7]
- The remainder of the contamination in the bottom of the excavation was not modeled as a continuing source to groundwater because this source is located below the assumed well pump intake depth and would not be expected to leach upward into the source of water available to the resident farmer. The potential dose contribution from this source has been determined to be small compared to the potential dose from contamination brought to the surface during installation of the hypothetical cistern. This matter is discussed further in Section 5.2.4.

Table 5-4 shows the exposure pathways for development of the subsurface soil DCGLs, which are the same as for the surface soil DCGLs.

Table 5-4. Exposure Pathways for Subsurface Soil DCGL Development

Exposure Pathways	Active
External gamma radiation from contaminated soil	Yes
Inhalation of airborne radioactivity from re-suspended contaminated soil	Yes
Plant ingestion (produce impacted by contaminated soil and groundwater contaminated by impacted soil)	Yes
Meat ingestion (beef impacted by contaminated soil and groundwater contaminated by impacted soil)	Yes
Milk ingestion (impacted by contaminated soil and groundwater contaminated by impacted soil)	Yes
Aquatic food ingestion	No(1)
Ingestion of drinking water (from groundwater contaminated by impacted soil)	Yes
Ingestion of drinking water (from surface water)(2)	No
Soil ingestion	Yes
Radon inhalation	No(3)

NOTES: (1) Fish ingestion is considered in development of the streambed sediment DCGLs and in the combined scenario discussed in Section 5.3.

(2) Groundwater was assumed to be the source of all drinking water because the low flow volumes in Erdman Brook and Franks Creek could not support the resident farmer. Use of surface water would also not be as conservative as groundwater since surface water is diluted by runoff from the entire

[7] Consideration was given to using a contaminated area larger than 100 square meters for the hypothetical garden. If the material brought to the surface during installation of the hypothetical cistern were spread over an area of 1000 square meters, for example, it would extend to an average depth of only about three centimeters (1.2 inches). If sufficient material were brought to the surface to cover 1,000 square meters to a depth of 0.3 meter (one foot), DCGLs would be reduced by a factor similar to that observed for surface soil DCGLs (reduction factors ranged from 1.3 for Cs-137 to 28 for C-14, see Appendix C).

watershed area. Incidental ingestion of water from the streams is evaluated in development of the streambed sediment DCGLs as shown in Table 5-6.

(3) In using the standard resident farmer scenario in modeling of buried radioactivity, the radon pathway is not considered (Appendix J, NRC 2006).

All of the input parameters for development of the subsurface soil DCGLs appear in Appendix C. Table 5-5 identifies selected key input parameters.

Table 5-5. Key Input Parameters for Subsurface Soil DCGL Development(1)

Parameter (Units)	Value	Basis
Initial source - cistern diameter (m)	2.0E+00	Conservative values used to estimate radioactivity brought to the surface to be mixed in garden soil.
Initial source – depth below surface (m)	9.0E+00	
Initial source – thickness (m)	1.0E+00	
Area of contaminated zone (m^2)	1.0E+02	Area drill cuttings from cistern installation spread on surface.
Thickness of contaminated zone (m)	3.0E-01	Contaminated soil depth in garden.
Cover depth (m)	0	Contamination on surface.
Contaminated zone erosion rate (m/y)	0	Conservative assumption.(2)
Well pumping rate (m^3/y)	5.72E+03	See Table C-2.
Unsaturated zone thickness (m)	2.0E+00	Reasonable for WMA 1 and WMA 2.
Distribution coefficient for strontium (mL/g)	1.5E+01	See Table C-2.
Distribution coefficient for cesium (mL/g)	4.8E+02	See Table C-2.
Distribution coefficient for americium (mL/g)	4.0E+03	See Table C-2.

NOTES: (1) See Appendix C for other input parameters. Metric units are used here because they are normally used in RESRAD.

(2) This assumption is conservative because it results in no depletion of the source.

Key assumptions associated with this conceptual model include:

- Contamination in the bottom one meter of the 10 meter deep excavation of the two meter diameter cistern would be brought to the surface, along with the overlying uncontaminated backfill, and blended into the soil over a 100 square meter area used by the resident farmer.
- All water used by the resident farmer (e.g., household, crop irrigation, and livestock watering) is groundwater which has been impacted by leaching of contaminants from surface soil (distributed excavated material) via infiltration of precipitation and irrigation water;
- Surface soil erosion (i.e., source depletion) does not occur over the 1,000 year-modeling period;
- The groundwater flow regime under the post-remedial conditions is unchanged from the current configuration (e.g. flow direction, aquifer productivity); and

- DCGLs that reflect 30 years of decay (i.e., apply to the year 2041) are appropriate for Sr-90 and Cs-137. Although a 30-year decay period could have been applied to all radionuclides, Sr-90 ad Cs-137 were selected based on expected peak doses at the onset of exposure and the short half lives of these particular radionuclides.

Other Possible Conceptual Models for Subsurface Soil DCGL Development

Other possible conceptual models were considered, such as a drilling worker. A drilling worker scenario would evaluate dose to a hypothetical individual installing the cistern, such as from contamination brought to the surface in the form of drill cuttings that could be set aside near the cistern.

A well driller scenario was evaluated in the Decommissioning EIS. The exposure pathways considered included inadvertent ingestion of contaminated soil, inhalation of contaminated dust, and direct exposure to contaminated water in a cuttings pond. The results, shown in Table H-44, indicate that dose to the hypothetical well driller in a representative area – the unremediated north plateau groundwater plume area after 100 years – would be insignificant (less than 1E-08 mrem per year).

Even considering the larger volume of removed contaminated soil in the two meter diameter cistern scenario, the potential dose to the drilling worker would be much smaller than the dose to a hypothetical resident farmer (see Section 5.4.4). Additionally, exposure to the drilling worker from the excavated Lavery till material would only occur in the final stages of the excavation because the majority of the material removed would be clean overlying soil. This factor would further reduce any potential exposure to the person constructing the hypothetical cistern.

Streambed Sediment Conceptual Model

Figure 5-9 illustrates the conceptual model for development of streambed sediment DCGLs. Table 5-6 identifies the exposure pathways considered.

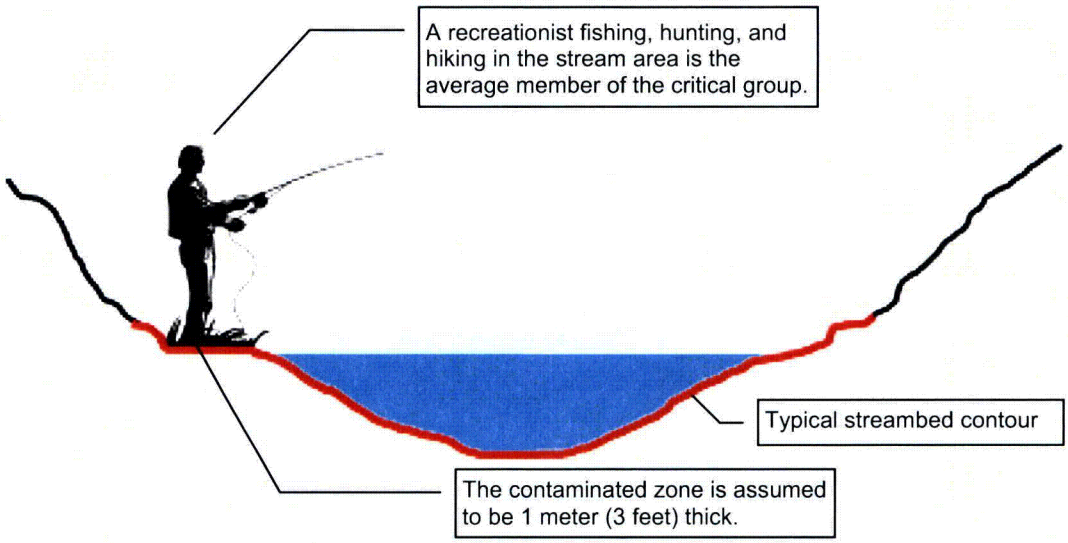


Figure 5-9. Conceptual Model for Streambed DCGLs Development

Table 5-6. Exposure Pathways for Streambed Sediment DCGL Development

Exposure Pathways	Active
External gamma radiation from contaminated sediment	Yes
Inhalation of airborne radioactivity from resuspended contaminated sediment	No(1)
Plant ingestion (produce impacted by soil and water sources)	No
Meat ingestion (venison impacted by soil and water sources)	Yes
Milk ingestion (impacted by soil and water sources)	No
Aquatic food ingestion (fish)	Yes
Ingestion of drinking water (from groundwater well)	No
Ingestion of drinking water (incidental from surface water)	Yes
Sediment ingestion (incidental during recreation)	Yes
Radon inhalation	No(2)

NOTES: (1) Sediments adjacent to streambed have significant moisture content that inhibits their resuspension potential and contradicts the consideration of inhalation exposure. Additionally, vegetation along the streambed would likely preclude significant wind scour and subsequent inhalation.

(2) The radon pathway is not considered because radon is primarily naturally occurring and neither radon nor its progeny are among the radionuclides of significant interest in dose modeling.

Key features of this conceptual model include the following:

- A person spending time in the area of the streams for recreation purposes was determined to be the appropriate member of the critical group; the area is not suitable for farming, livestock grazing, or residential use because of the steep stream banks, especially considering further erosion that is likely to occur as discussed previously.
- In this exposure scenario the primary radiation source is considered as the sediment deposited on the stream bank. The ability of sediment to adsorb and absorb radionuclides would be expected to concentrate otherwise dilute species of ions from the water (NRC 1977). The water in the stream provides some shielding and separation from radionuclides in sediments on the stream bottom, thus reducing direct exposure and incidental ingestion pathways from those sources.[8]
- The hypothetical recreationist is assumed to be located on the contaminated stream bank for 104 hours per year, which could involve spending two hours per day, two days per week for 26 weeks a year, reasonable assumptions considering the local climate.

[8] Note that modeling of transport, deposition, and concentrations of radionuclides in the stream itself would require assumptions on potential releases after Phase 1 of the decommissioning, and involve consideration of the Phase 2 end-state, which are not appropriate at this time.

- The contaminated zone of interest is located on the stream bank and is assumed to be three meters (10 feet) wide and 333 meters (1093 feet) long, with a total area of 1000 square meters (approximately ¼ acre).
- Having the contaminated zone on the stream bank takes into account a situation where the stream level might rise significantly then fall again to a lower level.
- The hypothetical recreationist is assumed to eat venison from deer whose flesh is contaminated with radioactivity from contaminated stream banks, such as from grazing on grass, and ingesting stream water.

Consideration was given to both receptor location and stream bank geometry.

Potential doses to a recreationist from impacted stream water would be less significant than potential doses from the stream bank for the following reasons:

- It would be plausible for the hypothetical recreationist to spend more time on the stream bank than immersed in stream water;
- The water would provide radiation shielding for radioactivity in the streambed sediment, which would decrease potential dose from direct radiation;
- While on the stream bank, the external dose from surface water would be negligible compared with the dose from the stream bank source; and
- Neglecting erosion of the stream bank source leads to greater doses than considering erosion of the source from the stream bank to the streambed, where significant shielding from surface water would reduce the dose.

The stream bank geometry was assumed to be represented by a plane source of contamination along the stream bank. Potential doses from alternative source configurations were not included in this evaluation for the following reasons:

- Any dose variation due to a sloped stream bank would likely result in doses similar to level sources due to movement of the receptor and exposure to an equivalent uniform dose (e.g. receptor is assumed to spend time moving throughout the source area and facing all directions for equal amounts of time);
- Although exposure to a source area wider than several meters is unlikely considering the steep terrain, the receptor is assumed to be externally exposed to a circular infinite plane source for conservatism; and
- Because the mass balance model was used for the sediment calculations, the source width parameter is not used in the calculations for water dependent pathways.

All of the input parameters for development of the streambed sediment DCGLs appear in Appendix C. Table 5-7 identifies selected key input parameters.

Table 5-7. Key Input Parameters for Streambed Sediment DCGL Development(1)

Parameter (Units)	Value	Basis
Area of contaminated zone (m^2)	1.0E+03	Area on stream bank.
Thickness of contaminated zone (m)	1.0E+00	Conservative assumption.
Fraction of year spent outdoors	1.2E-02	104 hours (out of a total of 8760 hours per year) in area.
Cover depth (m)	0	Contamination on surface.
Contaminated zone erosion rate (m/y)	0	Conservative assumption.(2)
Well pump intake depth (m below water table)	0	Only applicable to farming.
Well pumping rate (m^3/y)	0	Only applicable to farming.
Unsaturated zone thickness (m)	0	Contamination on stream bank surface.
Contaminated zone distribution coefficient for strontium (mL/g)	1.5E+01	See Table C-2.
Contaminated zone distribution coefficient for cesium (mL/g)	4.8E+02	See Table C-2.
Contaminated zone distribution coefficient for americium (mL/g)	4.0E+03	See Table C-2.

NOTES: (1) See Appendix C for other input parameters. Metric units are used here because they are normally used in RESRAD.
(2) This assumption is conservative because it results in no erosion of the source.

In development of the conceptual model, consideration was given to protection of environmental and ecological resources, as well as human health. It was determined that no changes to the model or the radioactivity cleanup criteria would be necessary for this purpose.[9]

5.2.2 Mathematical Model

As noted previously, RESRAD (Yu, et al. 2001) is used as the mathematical model for DCGL development. Version 6.4 was used to calculate the unit dose factors (in mrem/y per pCi/g) for each of the 18 radionuclides in each of the three exposure scenarios. Unit dose

[9] DOE Order 450.1, *Environmental Protection Program*, requires that DOE Environmental Management facilities such as the WVDP have an environmental management system to ensure protection of the air, water, land, and other natural and cultural resources in compliance with applicable environmental; public health; and resource protection laws, regulations, and DOE requirements. Implementing guidance includes DOE Standard 1153-2002, *A Graded Approach for Evaluating Radiation Doses to Aquatic and Terrestrial Biota*. This guidance includes the use of biota concentration guides to evaluate potential adverse ecological effects from exposure to radionuclides.

The WVDP routinely evaluates potential annual doses to aquatic and riparian animals and plants in relation to the biota concentration guides using the RESRAD-BIOTA computer code (DOE 2004) and radionuclide concentrations measured in water and streambed sediment. These evaluations show compliance with the guides (WVES and URS 2008). The environmental monitoring and control program for Phase 1 of the decommissioning described in Section 1.8 would ensure compliance with DOE Order 450.1 during the decommissioning activities.

factors were then scaled in Microsoft Excel to calculate individual radionuclide DCGLs corresponding to 25 mrem per year.

RESRAD was selected as the mathematical model for DCGL development due to the extensive use by DOE and by NRC licensees in evaluating doses from residual radioactivity at decommissioned sites. The RESRAD model considers multiple exposure pathways for direct contact with radioactivity, indirect contact, and food uptake, which are the conditions being evaluated at the WVDP.

RESRAD was used with the post-Phase 1 conceptual models described previously to generate doses for unit radionuclide source concentrations (i.e., dose per pCi/g of source). The resulting doses were then scaled to the limiting acceptable dose (25 mrem in a year) to provide the radionuclide specific DCGLs (see Appendix C). For example, the maximum estimated annual dose from 1 pCi/g of Cs-137 in surface soil was determined to be 1.7 mrem, so the DCGL for 25 mrem per year is 25 divided by 1.7 or 14.8 pCi/g prior to accounting for decay (see Table C-5). The calculated DCGLs were then input into the model as the source concentration to verify that the dose limit of 25 mrem per year was not exceeded.

Among the general considerations for the application of RESRAD to the post-Phase 1 decommissioning conceptual models were:

- Use of the non-dispersion groundwater pathways model for surface soil due to the relatively large source area;
- Use of the mass balance model, instead of the less conservative non-dispersion model, for the subsurface and streambed sediment models due to the relatively small source areas; and
- The conservative assumption of no erosion for soil and sediment sources in the development of DCGLs, so there would be no source depletion from erosion.

RESRAD input parameters were selected from the following sources, generally in the order given based on availability:

- Site-specific values where available, (e.g. groundwater and vadose zone parameters such as the distribution coefficients listed in Table 3-20);
- Semi site-specific literature values, (e.g. physical values based on soil type from NUREG/CR-6697 (Yu, et al. 2000) and behavioral factors based on regional data in the U.S. Environmental Protection Agency's *Exposure Factors Handbook* (EPA 1997);
- Scenario-specific values using conservative industry defaults, (e.g., from the *Exposure Factors Handbook*, the *RESRAD Data Collection Handbook* (Yu, et al. 1993), NUREG/CR-6697 (Yu, et al. 2000), and NUREG/CR-5512, Volume 3 (Beyeler, et al. 1999);
- The most likely values among default RESRAD parameters defined by a distribution, when available, otherwise mean values from NUREG/CR-6697 (Yu, et al. 2000).

5.2.3 Summary of Results

Table 5-8 provides the calculated individual radionuclide DCGLs for surface soil, subsurface soil, and streambed sediment which assure that the dose to the average member of the critical group would not exceed 25 mrem per year when considering the dose contribution from each radionuclide individually.

Table 5-8. DCGLs For 25 mrem Per Year (pCi/g)

Nuclide	Surface Soil		Subsurface Soil		Streambed Sediment	
	$DCGL_W$	$DCGL_{EMC}$(1)	$DCGL_W$	$DCGL_{EMC}$(1)	$DCGL_W$	$DCGL_{EMC}$(1)
Am-241	5.4E+01	4.4E+03	6.4E+03	4.6E+04	1.6E+04	3.7E+05
C-14	3.5E+01	1.7E+06	4.3E+05	1.5E+08	3.4E+03	1.1E+07
Cm-243	4.7E+01	8.4E+02	1.1E+03	9.0E+03	3.6E+03	3.3E+04
Cm-244	1.0E+02	1.4E+04	2.0E+04	1.5E+05	4.7E+04	3.2E+07
Cs-137(2)	2.9E+01	3.4E+02	4.4E+02	3.7E+03	1.3E+03	1.2E+04
I-129	6.5E-01	2.1E+03	4.2E+02	4.3E+04	3.7E+03	9.3E+05
Np-237	1.1E-01	2.3E+02	3.7E+01	3.7E+03	5.4E+02	1.7E+04
Pu-238	6.4E+01	8.5E+03	1.2E+04	9.2E+04	2.0E+04	1.6E+07
Pu-239	5.8E+01	7.7E+03	1.1E+04	8.3E+04	1.8E+04	1.4E+07
Pu-240	5.8E+01	7.7E+03	1.1E+04	8.3E+04	1.8E+04	1.5E+07
Pu-241	1.8E+03	1.5E+05	2.2E+05	1.5E+06	5.2E+05	1.3E+07
Sr-90(2)	9.7E+00	8.9E+03	3.1E+03	2.0E+05	9.5E+03	1.5E+06
Tc-99	3.2E+01	5.4E+04	1.1E+04	1.1E+06	2.2E+06	1.4E+08
U-232	6.3E+00	6.7E+01	1.2E+02	1.0E+03	2.7E+02	2.5E+03
U-233	2.2E+01	1.6E+04	1.7E+03	1.6E+05	5.8E+04	1.6E+06
U-234	2.3E+01	2.6E+04	1.7E+03	1.7E+05	6.1E+04	1.5E+07
U-235	1.6E+01	6.7E+02	9.5E+02	7.5E+03	2.9E+03	2.5E+04
U-238	2.4E+01	3.3E+03	1.8E+03	3.7E+04	1.3E+04	1.3E+05

NOTES: (1) $DCGL_{EMC}$ values are for an area 1 m^2 in size.
(2) Sr-90 and Cs-137 DCGLs reflect 30 years of decay and apply to the year 2041 and later.

The $DCGL_{EMC}$ values were calculated using each RESRAD model with an area of one square meter for the contaminated zone, in place of the larger contaminated zone area assumed in the base case model. This calculation produced the maximum dose in mrem per year in the peak year for a one square meter contaminated zone, which was used to estimate the $DCGL_{EMC}$ value.

As noted previously, the sum-of-fractions rule would be applied if characterization data indicate that a mixture of radionuclides is present in an area.

Conclusions About Results

Detailed outputs of the RESRAD simulations are presented in Appendix C. For surface soil, the results show that:

- Am-241 doses are due primarily to ingestion of plants,
- Cs-137 doses are due primarily to external exposure, and
- Sr-90 doses are due primarily to ingestion of plants.

The modeling to develop the subsurface soil DCGLs indicated that:

- Am-241 doses are due primarily to external exposure and ingestion of impacted plants,
- Cs-137 doses are due primarily to external exposure,
- Sr-90 doses are due primarily to ingestion of impacted plants, and
- DCGLs for subsurface soil are greater than those for the surface soil.

The modeling to develop the streambed sediment DCGLs indicated that:

- Am-241 doses are due primarily to incidental ingestion of sediment and to external exposure,
- Cs-137 doses are due primarily to external exposure, as well as ingestion of soil and venison,
- Sr-90 doses are due primarily to ingestion of venison, and
- DCGLs for the sediment source are orders of magnitude greater than those for surface soil.

Conservatism in Calculations

A number of factors make the calculated DCGLs conservative. For the surface soil DCGLs, these factors include, for example:

- Based on limited available data, the typical thickness of the contaminated zone is likely smaller than the one meter (about 3.3 feet) value used in the analysis.
- Because of the relatively short local growing season, it is likely that crop and forage yields would be less than those assumed for the site.

For the subsurface soil DCGLs, conservative factors include:

- As discussed previously, the diameter of the hypothetical well (cistern) at two meters (about 6.6 feet) is much larger than the diameter of a typical water well (eight inches)[10].

[10] With the larger diameter, much more contaminated soil and residual radioactivity would be brought to the surface where it could cause exposure through various pathways. The difference in volume would vary with the square of the radius; 100 times as much contaminated soil would be brought to the surface in the conceptual model with the two meter diameter well than with a model that assumed a 20 centimeter (eight inch) diameter well. The larger diameter well assumed ensures that the pumping needs of the residential farm would be met, since a smaller diameter well could not do this on some parts of the project premises.

- Use of the mass balance model within RESRAD is conservative in that all radionuclide inventory in leachate reaches the intake well.
- Because of the relatively short local growing season, it is likely that crop/forage yields would be less than those assumed for the site.

For the streambed sediment DCGLs, conservative factors include:

- Based on limited available data, the typical thickness of the contaminated zone is likely smaller than the one meter (about 3.3 feet) value used in the analysis.
- Based on available data, most contamination will be found in the stream beds, not on the banks.
- It is unlikely that the incidental ingestion rate (50 mg/d) for sediment will be exclusively from the contaminated area.
- It is assumed that all fish ingested by the recreationist are impacted by the streambed sediment source; however, it is more likely that a recreationist may ingest fish from other locations as well.
- Similarly, it is unlikely that the venison ingested would be impacted by streambed sediment sources exclusively. It is more likely that exposure would be from both impacted and non-impacted areas.
- Assumptions regarding the availability of an adequate fish population to allow long term fish ingestion may also result in overestimation of doses related to the sediment source, as there are currently no fish in the streams of sufficient quality or quantity for sustained human consumption.

5.2.4 Discussion of Sensitivity Analyses and Uncertainty

Table 5-9 summarizes the sensitivity analyses performed for the surface soil DCGLs, which are detailed in Appendix C.

Table 5-9. Summary of Parameter Sensitivity Analyses – Surface Soil DCGLs(1)

Parameter (Base Case)	Run	Change Made	Minimum DCGL Change		Maximum DCGL Change	
			Change	Nuclide(s)	Change	Nuclide(s)
Indoor/Outdoor Fraction (0.66/0.25)	1	-32%	-23%	U-232	0%	I-129
	2	21%	0%	I-129 U-234	30%	U-232
Source Thickness (1 m)	3	-50%	9%	Cs-137	82%	Sr-90
	4	200%	-30%	U-235	-0.1%	Cs-137
Unsaturated Zone Thickness (2 m)	5	-50%	-2%	U-238	6%	U-235
	6	150%	-4%	U-235	1%	U-238
Irrigation/Pump Rate (0.5 m/y/ 5720 m^3/y)	7	-57%	-1%	U-232	52%	I-129
	8	70%	-31%	I-129	2%	U-232

Table 5-9. Summary of Parameter Sensitivity Analyses – Surface Soil DCGLs(1)

Parameter (Base Case)	Run	Change Made	Minimum DCGL Change		Maximum DCGL Change	
			Change	Nuclide(s)	Change	Nuclide(s)
Soil/Water Distribution Coefficients (K_d) (Table C-2)	9	lower	-67%	Sr-90	6%	U-232
	10	higher	-4%	U-232	1146%	U-234
Hydraulic Conductivity (140 m/y)	11	-99%	0%	Sr-90	1873%	I-129
	12	150%	0%	Cs-137, Sr-90, U-232	122%	U-235
Runoff/Evapotrans-poration Coefficient (0.6/0.55)	13	-69%	-28%	U-234	3%	U-232
	14	64%	-3%	U-232	121%	U-234
Depth of Well Intake (5 m)	15	-40%	-42%	I-129	0.1%	U-232
	16	100%	0%	Cs-137	92%	I-129
Length Parallel to Aquifer Flow (100 m)	17	-50%	0%	Cs-137	78%	U-235
	18	100%	-44%	U-235	0.1%	U-232
Plant Transfer Factors (RESRAD default)	19	-90%	-4%	I-129	387%	Sr-90
	20	900%	-90%	Sr-90	-6%	I-129
Mass Balance Model (non-dispersion model)	21	-69%	-81%	U-234	0.1%	U-232
Contaminated Layer Area (10,000 m^2)	-	Various smaller areas	-	-	-	See note (1)

NOTES: (1) Information from the $DCGL_{EMC}$ calculations was used for evaluation of the sensitivity of the contaminated layer area. DCGLs generally increased with smaller areas. Results presented here are for radionuclides considered likely to contribute significantly to the overall surface soil dose based on available characterization data.

Discussion of Surface Soil Results

The uncertainty results for the surface soil source model been evaluated considering those radionuclides that are the primary dose drivers, i.e., those that are likely to contribute significantly to predicted dose based on available characterization data. The radionuclides

are Sr-90 (due to water independent plant uptake), I-129 (due to water dependent pathways), Cs-137 (external radiation dose), and most uranium radionuclides (water dependent pathways).

The sensitivity analysis of the surface soil model, for these radionuclides, indicates the following:

- A lower indoor exposure fraction results in the largest DCGL decrease for U-232 and no change for I-129. Similarly, a higher indoor exposure fraction results in the largest increase for U-232 and no change for I-129 and U-234. However, it is unlikely that the indoor fraction is too low based on the local climate. The U-232 doses are mainly due to external exposure, which accounts for the relative sensitivity to this parameter.
- Decreasing the source thickness increased the DCGL for all radionuclides and increasing the source thickness resulted in the most significant DCGL decrease for U-235. The sensitivity to this parameter is due to increased/decreased dose from the water ingestion and plant pathways (both water dependent and independent).
- Decreasing the unsaturated zone thickness resulted in an increased DCGL for U-235 and a decrease for U-238. Similarly, increasing the unsaturated zone thickness decreased the U-235 DCGL and increased the U-238 DCGL. Sensitivity to this parameter is mainly due to increased/decreased travel time of contaminants to the saturated zone, resulting in water dependent doses occurring earlier/later with respect to doses from water independent pathways.
- Reducing the irrigation/well pump rate increased the DCGL for I-129 most significantly. Similarly, increasing the pump rate decreased the DCGL for I-129. This is because reducing the pumping rate results in a lower dilution factor, and increasing the pumping rate results in more radionuclide inventory available for exposure.
- The most significant effects of varying the K_d values were observed for Sr-90 and U-234.
- Decreasing the hydraulic conductivity significantly increased the DCGL for I-129 due to increasing the travel time to the well. Increasing the hydraulic conductivity significantly increased the DCGL for U-235 because dilution is greater.
- Variations in the runoff/evapotranspiration coefficients had the greatest effect on U-234 and the least impact on U-232. Radionuclides that are most sensitive to this parameter have doses mainly due to water dependent pathways.
- Decreasing the well intake depth most significantly decreased the DCGL for I-129, while increasing this parameter results in significantly increased the DCGL for I-129, due to increased/decreased dilution in the well water.

- Changes to the parameter for length of contamination parallel to the aquifer flow had the most significant effect on the U-235 DCGL, due to increased/decreased dilution in the aquifer.
- Decreasing/increasing the plant transfer factors significantly increased/decreased the DCGL for Sr-90, as dose is mainly due to ingestion via plant uptake from soil.
- Use of the mass balance groundwater model significantly decreases the DCGL for U-234 but had no effect on U-232. Radionuclides most sensitive to this parameter have doses mainly due to water dependent pathways.

Table 5-10 summarizes the sensitivity analyses performed for the subsurface soil DCGLs, which are detailed in Appendix C.

Table 5-10. Summary of Parameter Sensitivity Analyses – Subsurface Soil DCGLs[1]

Parameter (Base Case)	Run	Change Made	Minimum DCGL Change		Maximum DCGL Change	
			Change	Nuclide(s)	Change	Nuclide(s)
Indoor/Outdoor Fraction (0.66/0.26)	1	-32%	-25%	Cs-137	0.1%	U-234
	2	21%	-1%	U-238	35%	U-232
Source Thickness (1m)	3	-67%	10%	U-238	193%	Sr-90
	4	233%	-66%	Sr-90	-1%	Cs-137
Unsaturated Zone Thickness (2 m)	5	-50%	-1%	U-238	0%	Cs-137, Sr-90, U-232, U-235
	6	150%	0%	Cs-137 Sr-90 U-232 U-235	1%	U-238
Irrigation/Pump Rate (0.5 m/y/ 5720 m^3/y)	7	-57%	-36%	I-129	0%	Cs-137
	8	70%	0%	Cs-137	159%	U-238
Soil/Water Distribution Coefficients (K_d) (Table C-2)	9	lower	-85%	U-238	9%	U-232
	10	higher	-27%	U-232	3144%	U-234
Hydraulic Conductivity (140 m/y)	11	-99%	-1%	U-238	3%	I-129
	12	150%	0%	Cs-137 I-129 Sr-90 U-232 U-233 U-234 U-235 U-238	0%	Cs-137, I-129, Sr-90, U-232, U-233, U-234, U-235, U-238
Runoff/Evapotrans-poration Coefficient (0.6/0.55)	13	-69%	-38%	U-234	16%	U-232
	14	64%	-19%	U-232	188%	U-234

Table 5-10. Summary of Parameter Sensitivity Analyses – Subsurface Soil DCGLs(1)

Parameter (Base Case)	Run	Change Made	Minimum DCGL Change		Maximum DCGL Change	
			Change	Nuclide(s)	Change	Nuclide(s)
Plant Transfer Factors (RESRAD defaults)	**15**	-90%	-0.4%	U-238	574%	Sr-90
	16	900%	-89%	Sr-90	-1%	U-234
Contaminated Layer Area (100 m²)	-	Various smaller areas	-	-	-	See note (1).

NOTES: (1) Information from the $DCGL_{EMC}$ calculations was used for evaluation of the sensitivity of the contaminated layer area. DCGLs generally increased with smaller areas. Results presented here are for radionuclides considered likely to contribute significantly to the overall subsurface soil dose based on available characterization data.

Discussion of Subsurface Soil Results

The uncertainty results for the subsurface soil source models have been evaluated considering those radionuclides that are the primary dose drivers, i.e., those that are likely to contribute significantly to predicted dose based on available characterization data (see Table 5-1). The radionuclides are Sr-90 (due to water independent plant uptake), I-129 (due to water dependent pathways), Cs-137 (external radiation dose), and uranium radionuclides (water dependent pathways).

The sensitivity analysis of the subsurface soil model for these radionuclides indicates the following:

- A lower indoor exposure fraction results in a DCGL decrease for Cs-137 and no change for U-234. A higher indoor exposure results in a significant increased DCGL for U-232. However, it is unlikely that the indoor fraction is too low based on the local climate. Doses for these isotopes are mainly due to external exposure, which accounts for the relative sensitivity to this parameter.
- The source thickness parameter sensitivity was most significant for Sr-90 The sensitivity to this parameter is due to increased/decreased dose from the water ingestion and plant pathways (both water dependent and independent).
- Decreasing or increasing the unsaturated zone thickness resulted in little change to the DCGLs.
- The I-129 and U-238 DCGLs were sensitive to changes in the irrigation/well pump rate but the Cs-137 DCGL was not. This effect is because reducing the pumping rate results in a lower dilution factor, and increasing the pumping rate results in more dilution for water dependent pathways.
- The most significant effects of varying the K_d values were observed for U-232, U-234, and U-238.
- Decreasing or increasing the hydraulic conductivity resulted in no change to the DCGLs due to use of the mass balance model.

- The U-232 and U-234 DCGLs are sensitive to changes in the runoff/ evapotranspiration coefficient. Radionuclides that are most sensitive to this parameter have doses mainly due to water dependent pathways.
- The plant transfer factor is most sensitive for Sr-90, as the dose is mainly due to ingestion via plant uptake.

Table 5-11 summarizes the sensitivity analyses performed for the streambed sediment DCGLs, which are detailed in Appendix C:

Table 5-11. Summary of Parameter Sensitivity Analyses – Streambed Sediment DCGLs(1)

Parameter (Base Case)	Run	Change Made	Minimum DCGL Change		Maximum DCGL Change	
			Change	Nuclide(s)	Change	Nuclide(s)
Indoor/Outdoor Fraction (0.66/0.25)	1	-50%	3%	Sr-90	86%	Cs-137
	2	100%	-48%	Cs-137	-5%	Sr-90
Source Thickness (1 m)	3	-50%	1%	Cs-137	29%	Sr-90
	4	200%	-0.2%	Sr-90	0%	Cs-137
Unsaturated Zone Thickness (2 m)	5	0 m to 1m	0.3%	Cs-137	8%	Sr-90
	6	0 m to 3 m	0.3%	Cs-137	8%	Sr-90
Soil/Water Distribution Coefficients (K_d) (Table C-2)	7	lower	0.5%	Cs-137	12%	Sr-90
	8	higher	0.3%	Cs-137	7%	Sr-90
Runoff/Evaporation Coefficient (0.6/0.55)	9	-54%	0%	Cs-137	0.4%	Sr-90
	10	78%	-0.3%	Sr-90	0%	Cs-137
Plant Transfer Factors (RESRAD defaults)	11	-90%	1%	Cs-137	82%	Sr-90
	12	900%	-82%	Sr-90	-9%	Cs-137
Fish Transfer Factors (RESRAD defaults)	13	-90%	0.3%	Cs-137	7%	Sr-90
	14	900%	-39%	Sr-90	-3%	Cs-137
Contaminated Layer Area (1000 m^2)	-	Various smaller areas	-	-	-	See note (1).

NOTES: (1) Information from the $DCGL_{EMC}$ calculations was used for evaluation of the sensitivity of the contaminated layer area. DCGLs generally increased with smaller areas. Results presented here are for radionuclides considered likely to contribute significantly to the overall sediment dose based on available characterization data.

Discussion of Streambed Sediment Results

The streambed sediment model sensitivity simulations have been evaluated considering those radionuclides that are likely to significantly contribute to the overall doses in this media, which are Sr-90 (venison ingestion) and Cs-137 (external radiation dose).

The sensitivity analysis for the sediment model, for these radionuclides, indicates:

- The DCGLs for Sr-90 and Cs-137 are inversely related to changes in outdoor fraction, with Cs-137 being the most sensitive. Radionuclides with primary doses from water independent pathways are more sensitive to changes in this parameter.
- Decreasing the source thickness results in higher DCGLs for Sr-90 and Cs-137. While increasing the source thickness has little effect on these radionuclides. Sr-90 is most sensitive to this parameter.
- Increasing the unsaturated zone thickness increases DCGLs for Sr-90 but had no effect on Cs-137. Radionuclides with primary doses from water dependent pathways are more sensitive to changes in this parameter.
- Varying the K_d values had no effect on the Cs-137 DCGLs, but increased the Sr-90 DCGLs due to doses from water dependent pathways.
- Varying the runoff/evapotranspiration coefficient had little effect on Cs-137 or Sr-90 DCGLs. Radionuclides most sensitive to this parameter have doses mainly due to water dependent pathways.
- Decreasing both plant and fish transfer factors resulted in increased DCGLs for Sr-90, and increasing these parameters resulted in decreased DCGLs for both Cs-137 and Sr-90.

Other Uncertainties

The RESRAD model does not account for the fate and transport of eroded particles due to surface soil source erosion/overland transport, and the rate of erosion input for RESRAD is only used to deplete the source. The assumption of no sediment source erosion is considered an appropriate simplification since it provides a conservative estimate of dose based on no source depletion via erosion. Additionally, while overland erosion via runoff is not considered, neither is the receiving water body diluted by the runoff.

The assumption of no change to groundwater conditions in terms of flow direction and aquifer productivity is a source of potential uncertainty. However, DCGLs based on this assumption can be further refined if site specific information indicates different conditions are likely.

Leaching of Residual Subsurface Contamination to Groundwater

The evaluation of DCGL radioactivity concentrations in the Lavery till (that is, at the bottom of the WMA 1 and WMA 2 excavations) as a continuing source to groundwater could not be modeled using RESRAD, because the code does not provide for a site configuration with a source below the water table. Pore water concentrations estimated

from the soil partition coefficients indicate that even assuming minimal dilution, the resulting well concentration would be low compared with the contribution from well cuttings leaching from the surface (see Appendix C). The uncertainty in neglecting this contribution to the overall dose is considered to be acceptable when considering the large percentage of the dose from pathways associated with subsurface soil cuttings spread on the surface compared to the potential dose from leaching of residual radioactivity at the bottom of the WMA 1 and WMA 2 excavations.

The following conditions suggest that the dose associated with subsurface soil cuttings as a surface source does not warrant consideration in the overall combined dose assessment:

- Even with conservative assumptions of a large cistern diameter and well depth, combined with a small thickness over which the cuttings are spread, the result is a source area of approximately 1,000 square feet (100 square meters). When this source area is used in conjunction with the required area for a resident farmer of 100,000 square feet (10,000 square meters), the result is a large DCGL for subsurface soil when compared with surface soil DCGLs (except in the case of Cs-137).
- Dilution of contaminated well cuttings with overlying clean fill results in further reduction of overall dose from subsurface sources relative to surface sources.
- Doses from potential surface soil sources are orders of magnitude greater than those from subsurface sources based on the resident farmer scenario.

Changes to Base-Case Models Based on Sensitivity Analysis Results

Development of the conceptual model for surface soil DCGLs was an iterative process that used conservative assumptions for model parameters and took into account the results of early model runs and the related input parameter sensitivity analyses.

The initial model runs produced inordinately low DCGLs for uranium radionuclides in surface soil. The calculated $DCGL_W$ for U-238, for example, was 1.0 pCi/g, slightly above measured background concentrations in surface soil shown in Table 4-11 of this plan.

The next iteration involved changes to radionuclide distribution coefficients. Evaluation of the basis for the original distribution coefficients and sensitivity analysis results led to the conclusion that some distribution coefficients used were inappropriate. These distribution coefficients were changed. The resulting distribution coefficients are based either on site-specific data for the sand and gravel layer or, where site-specific data are not available, values for sand from Sheppard and Thibault 1990, as shown in Table C-2.

These model changes produced higher $DCGL_W$ values for uranium radionuclides, e.g., 4.8 pCi/g for U-238. However, these values were still low compared to uranium DCGLs for unrestricted release developed at other sites. Further evaluation showed that the main reason for the low uranium DCGLs was the conservative use of the RESRAD mass balance model. After considering the results of the sensitivity analysis that evaluated use of

the non-dispersion model, and RESRAD Manual guidance[11], it was determined to be more appropriate to use the non-dispersion model in the surface soil analysis and this was done.

No other conceptual model changes were considered to be necessary given the approach of selecting input parameters that are generally conservative and taking into account the built-in modeling conservatism from selecting peak doses from all years and neglecting the decay of long-lived radionuclides. For the subsurface soil DCGL model, because of the limited amount of material excavated and distributed on the surface, the contaminated layer thickness at the ground surface was not increased (this provides a larger area over which to spread subsurface cuttings).

Overall Conclusion

The DCGLs developed for Phase 1 of the proposed decommissioning as shown in Table 5-8 are protective of human health. Evaluation of the dose modeling results indicates that:

- Primary contributions to dose associated with surface soil sources are due to external exposure to Cs-137 in surface soil, and ingestion of Sr-90 in plants. Surface soil source results indicate that Cs-137 dose is most sensitive to changes in the indoor/outdoor fraction and plant transfer factors, while Sr-90 is sensitive to changes in the contaminated zone thickness, plant transfer factors, and the use of the mass balance groundwater model.
- Primary contributions to dose associated with subsurface sources are due to external exposure to Cs-137 in excavated material, and ingestion of Sr-90 in plants. Subsurface soil source results indicate that Cs-137 is most sensitive to changes in indoor/outdoor fraction and source thickness. Sr-90 is most sensitive to source thickness and plant transfer factors.
- Primary contributions to dose associated with sediment sources are due to external exposure to Cs-137 in sediment, and ingestion of Sr-90 in venison. Sediment source results indicate that Cs-137 dose is most sensitive to the indoor/outdoor fraction, while Sr-90 is sensitive to plant transfer factors.

The DCGLs developed as described in this section were based on exposure to a single radionuclide in a specific source media (e.g., Sr-90 in sediment). The next section discusses refinement of the DCGLs to account for exposure to multiple radionuclides and sources.

5.3 Limited Site-Wide Dose Assessment

This section describes the limited integrated dose assessment performed to ensure that criteria used in Phase 1 remediation activities would not limit options for Phase 1 of the proposed decommissioning.

[11] The RESRAD Manual (Yu, et al. 2001) notes in Appendix E that:"The user has the option of selecting which [groundwater] model to use. Usually, the MB [mass balance] model is used for smaller contaminated areas (e.g., 1,000 m^2 or less) and the ND [non-dispersion] model is used for larger areas."

5.3.1 Basis for this Assessment

Section 5.1.3 explains why such a dose assessment is appropriate, considering the Phase 1 and Phase 2 sources illustrated in Figure 5-4. Section 5.1.3 also explains that the appropriate dose assessment involves a hypothetical individual engaged in farming at some time in the future on one part of the remediated project premises who also spends time fishing and hiking at Erdman Brook and Franks Creek.

This scenario would involve an individual being exposed to two different remediated source areas and being a member of the two different critical groups. As described in Section 5.2, the exposure group for the resident farmer scenario used for development of DCGLs for surface and subsurface soil is significantly different from the exposure group for the development of the streambed sediment DCGLs, which involves a hypothetical individual spending a relatively small fraction of his or her time hiking, fishing, and hunting in the areas of Erdman Brook and Franks Creek.

In both of these cases, it was assumed that the hypothetical individual (the average member of the critical group) would be exposed only to the residual radioactivity of interest. That is, the resident farmer would not be exposed to residual radioactivity in the areas of the streams and the recreationist would not be exposed to residual radioactivity in surface soil or subsurface soil.

5.3.2 Assessment Approach

The approach used involves partitioning doses between two critical groups and two areas of interest: (1) the resident farmer who lives in an area of the project premises where surface soil or subsurface soil has been remediated to the respective DCGLs and (2) the person who spends time in the areas of the streams hiking, fishing, and hunting (the recreationist). This approach is analogous to addressing multiple radionuclides in contaminated media of interest using the sum-of-fractions approach or unity rule (NRC 2006).

Consideration of potential risks related to the different areas led assigning 90 percent of the total dose limit of 25 mrem per year to the resident farmer activities and 10 percent to the recreational activities. This arrangement involves assigning an acceptable dose of 22.5 mrem per year to resident farmer activities and 2.5 mrem per year to recreation in the area of the streams, values which total 25 mrem per year.[12] The assessment was then performed using the base case analysis results for the resident farmer and the recreationist at Erdman Brook and Franks Creek.

Two separate assessments were performed with the resident farmer located in: (1) the area of the remediated WMA 1 subsurface soil excavation, and (2) the resident farmer

[12] This 0.90/0.10 split is based on judgment related to relative risk. Consideration was given to using a split based on the relative time the hypothetical farmer would spend in the area of the farm compared to the area of the streams. However, because the assumed time in the area of the streams is relatively small at 104 hours per year, such as spilt could result in an allowable annual dose of 24.7 mrem for resident farmer activities and 0.3 mrem for recreation at the streams. This split would have a minimal impact on the soil DCGLs while driving the streambed sediment DCGLs to unrealistically low levels.

located in an area where surface soil was assumed to have been remediated. Details appear in Appendix C.

5.3.3 Results of the Assessments

Table 5-12 provides the assessment results for the WMA 1 subsurface soil case and Table 5-13 provides the results for the surface soil case. The streambed sediment $DCGL_W$ values are the same in both cases because the apportioned dose limit of 2.5 mrem per year is the same.

Table 5-12. Limited Site-Wide Dose Assessment 1 Results (DCGLs in pCi/g)

Nuclide	Subsurface Soil $DCGL_W$ Values		Streambed Sediment $DCGL_W$ Values	
	Base Case(1)	Assessment(2)	Base Case(1)	Assessment(2)
Am-241	6.4E+03	5.8E+03	1.6E+04	1.6E+03
C-14	4.3E+05	3.8E+05	3.4E+03	3.4E+02
Cm-243	1.1E+03	1.0E+03	3.6E+03	3.6E+02
Cm-244	2.0E+04	1.8E+04	4.7E+04	4.7E+03
Cs-137(3)	4.4E+02	3.9E+02	1.3E+03	1.3E+02
I-129	4.2E+02	3.8E+02	3.7E+03	3.7E+02
Np-237	3.7E+01	3.3E+01	5.4E+02	5.4E+01
Pu-238	1.2E+04	1.1E+04	2.0E+04	2.0E+03
Pu-239	1.1E+04	9.9E+03	1.8E+04	1.8E+03
Pu-240	1.1E+04	9.9E+03	1.8E+04	1.8E+03
Pu-241	2.2E+05	2.0E+05	5.2E+05	5.2E+04
Sr-90(3)	3.1E+03	2.8E+03	9.5E+03	9.5E+02
Tc-99	1.1E+04	9.9E+03	2.2E+06	2.2E+05
U-232	1.2E+02	1.1E+02	2.7E+02	2.7E+01
U-233	1.7E+03	1.5E+03	5.8E+04	5.8E+03
U-234	1.7E+03	1.5E+03	6.1E+04	6.1E+03
U-235	9.5E+02	8.6E+02	2.9E+03	2.9E+02
U-238	1.8E+03	1.6E+03	1.3E+04	1.3E+03

NOTE: (1) The base case values from Table 5-8.
(2) The results for the analysis of the combined resident farmed located in the area of remediated surface soil and the recreationist in the area of the streams.
(3) These DCGLs apply in the year 2041 and later.

As can be seen from Table 5-13, the dose partitioning approach reduced the $DCGL_W$ values for surface soil by 10 percent and reduced the $DCGL_W$ values for streambed sediment by an order of magnitude.

Table 5-13. Limited Site-Wide Dose Assessment 2 Results (DCGLs in pCi/g)

Nuclide	Surface Soil $DCGL_W$ Values		Streambed Sediment $DCGL_W$ Values	
	Base Case(1)	Assessment(2)	Base Case(1)	Assessment(2)
Am-241	5.4E+01	4.9E+01	1.6E+04	1.6E+03
C-14	3.5E+01	3.1E+01	3.4E+03	3.4E+02
Cm-243	4.7E+01	4.2E+01	3.6E+03	3.6E+02
Cm-244	1.0E+02	9.4E+01	4.7E+04	4.7E+03
Cs-137(3)	2.9E+01	2.7E+01	1.3E+03	1.3E+02
I-129	6.5E-01	5.8E-01	3.7E+03	3.7E+02
Np-237	1.1E-01	9.6E-02	5.4E+02	5.4E+01
Pu-238	6.4E+01	5.8E+01	2.0E+04	2.0E+03
Pu-239	5.8E+01	5.2E+01	1.8E+04	1.8E+03
Pu-240	5.8E+01	5.2E+01	1.8E+04	1.8E+03
Pu-241	1.8E+03	1.6E+03	5.2E+05	5.2E+04
Sr-90(3)	9.7E+00	8.7E+00	9.5E+03	9.5E+02
Tc-99	3.2E+01	2.9E+01	2.2E+06	2.2E+05
U-232	6.3E+00	5.6E+00	2.7E+02	2.7E+01
U-233	2.2E+01	2.0E+01	5.8E+04	5.8E+03
U-234	2.3E+01	2.1E+01	6.1E+04	6.1E+03
U-235	1.6E+01	1.4E+01	2.9E+03	2.9E+02
U-238	2.4E+01	2.2E+01	1.3E+04	1.3E+03

NOTE: (1) The base case values from Table 5-8.
(2) The results for the analysis of the combined resident farmed located in the area of remediated surface soil and the recreationist in the area of the streams.
(3) These DCGLs apply in the year 2041 and later.

5.4 Cleanup Goals and Additional Analyses

This section (1) identifies the cleanup goals to be used in remediation of surface soil, subsurface soil, and streambed sediment and the basis for these cleanup goals; (2) describes how the DCGLs and the cleanup goals would be later refined; (3) discusses use of surrogate radionuclides; and (4) identifies plans for the dose assessment of the remediated WMA 1 and WMA 2 areas.

5.4.1 Cleanup Goals

As explained in Section 5.1.6, the dose modeling process includes establishing cleanup goals below the DCGLs developed to meet the 25 mrem per year unrestricted dose limit that are to be used to guide remediation efforts, considering the results of the analysis of the combined source area exposure scenario described in Section 5.3 and the ALARA analysis described in Section 6.

Combined Source Area Analysis

As indicated in Section 5.3, analysis of the limiting scenario for dose integration – a resident farmer living on the remediated project premises who spends time in the vicinity of Erdman Brook and Franks Creek hiking, fishing, and hunting – produced lower $DCGL_W$ values for both critical groups, with the reduction for the recreationist in the area of the streams being a much greater percentage.

ALARA Analysis

Section 6 describes the process used to evaluate whether remediation of surface soil, subsurface soil, and streambed sediment below DCGLs based on 25 mrem/y would be cost-effective, following the standard NRC methodology for ALARA analyses. Section 6 provides the results of a preliminary analysis and provides for a final ALARA analysis to be performed during the Phase 1 proposed decommissioning work.

The preliminary ALARA analysis suggests that the costs of removing slightly contaminated soil or sediment at concentrations below the DCGLs for 25 mrem per year would outweigh the benefits. That is, areas where surface soil, subsurface soil, and sediment are remediated to radioactivity concentrations at the DCGLs satisfy the ALARA criteria. The evaluation process balances the cost of offsite disposal of additional radioactively contaminated soil (cost of $6.76 per cubic foot) and the benefits of reduced dose (benefit of $2000 per person-rem as set forth in NRC guidance).

The final ALARA analysis that would be performed during the Phase 1 proposed decommissioning activities would make use of updated information, such as actual rather than predicted waste disposal costs. However, the results would likely be similar to the preliminary analysis.

Section 6 explains that the methods to be used in remediation of contaminated soil and sediment, which involve excavation of the material in bulk quantities, would generally remove more material than necessary to meet the DCGLs. As noted in Section 6, NRC recognizes that soil excavation is a coarse removal process that is likely to remove large fractions of the remaining radioactivity (NRC 1997). The contaminated soil and sediment removal method is therefore expected to produce residual radioactivity concentrations well below the DCGLs.

Cleanup Goals

Demonstration that the proposed decommissioning activities have achieved the desired dose-based criteria is through a process described in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000). Samples of the post-decommissioning media are analyzed for the individual radionuclides of interest (or for a surrogate radionuclide in a mixture[13]), and the *average* concentration is compared to the DCGL using various statistical tests. Because the average concentration is compared to

[13] Section 4.3.2 of the MARSSIM (NRC 2000) describes how for sites with multiple radionuclides it may be practical to measure just one of the contaminants and still demonstrate compliance with cleanup criteria for all of the contaminants through the use of surrogate measurements. Section 9 of this plan discusses the use of surrogate radionuclides in Phase 1 of the decommissioning.

the DCGL, and due to the statistical fluctuations inherent in measuring low concentrations of radioactivity, it is likely that some post-remediation samples would exceed the DCGL. It is not necessary that all samples be below the DCGL, but to increase success in the statistical evaluation, the planned post-remediation average (in-process or cleanup goal) should be somewhat below the DCGL. How far below the DCGL is appropriate depends on the variation of the post-remediation concentration across the area and on the inherent costs in responding to a false positive decision (concluding that remediation was successful but finding that analysis of samples from the area fails the statistical evaluation).

For surface soils and sediments in the WVDP Phase 1 areas, the field cleanup goal need not be too far below the DCGL, if at all. As discussed previously, bulk excavation would generally remove more material than necessary to meet the DCGL, so it is likely that the post-remediation average concentration would be below whatever in-process goal is chosen. And the costs for additional remediation of a surface soil or sediment site, while extra, are not unusually high.

However, for subsurface soils a field cleanup goal should be well below the DCGL because of the large costs to be incurred if additional remediation were necessary to an area that failed the statistical testing. Re-excavating to depth with shoring, engineering controls, and management or disposal of extensive overburden would be expensive compared to excavating some additional material in the original remediation.

Consideration of such factors led to DOE establishing in this plan the cleanup goals shown in Table 5-14.

Table 5-14. Cleanup Goals to be Used in Remediation in pCi/g[1]

Nuclide	Surface Soil[2]		Subsurface Soil[3]		Streambed Sediment[2]	
	CG_w	CG_{EMC}	CG_w	CG_{EMC}	CG_w	CG_{EMC}
Am-241	4.9E+01	4.0E+03	2.9E+03	2.1E+04	1.6E+03	3.7E+04
C-14	3.1E+01	1.5E+06	1.9E+05	6.6E+07	3.4E+02	1.1E+06
Cm-243	4.2E+01	7.6E+02	5.1E+02	4.0E+03	3.6E+02	3.3E+03
Cm-244	9.4E+01	1.2E+04	8.8E+03	6.6E+04	4.7E+03	3.2E+06
Cs-137[4]	2.7E+01	3.0E+02	2.0E+02	1.7E+03	1.3E+02	1.2E+03
I-129	5.8E-01	1.9E+03	1.9E+02	1.9E+04	3.7E+02	9.3E+04
Np-237	9.6E-02	2.1E+02	1.7E+01	1.7E+03	5.4E+01	1.7E+03
Pu-238	5.8E+01	7.7E+03	5.5E+03	4.1E+04	2.0E+03	1.6E+06
Pu-239	5.2E+01	6.9E+03	5.0E+03	3.8E+04	1.8E+03	1.4E+06
Pu-240	5.2E+01	7.0E+03	5.0E+03	3.8E+04	1.8E+03	1.5E+06
Pu-241	1.6E+03	1.3E+05	9.8E+04	7.0E+05	5.2E+04	1.3E+06
Sr-90[4]	8.7E+00	8.0E+03	1.4E+03	9.1E+04	9.5E+02	1.5E+05
Tc-99	2.9E+01	4.9E+04	5.0E+03	4.9E+05	2.2E+05	1.4E+07
U-232	5.6E+00	6.0E+01	5.3E+01	4.7E+02	2.7E+01	2.5E+02

Table 5-14. Cleanup Goals to be Used in Remediation in pCi/g(1)

Nuclide	Surface Soil(2)		Subsurface Soil(3)		Streambed Sediment(2)	
	CG_W	CG_{EMC}	CG_W	CG_{EMC}	CG_W	CG_{EMC}
U-233	2.0E+01	1.4E+04	7.5E+02	7.2E+04	5.8E+03	1.6E+05
U-234	2.1E+01	2.3E+04	7.7E+02	7.9E+04	6.1E+03	1.5E+06
U-235	1.4E+01	6.1E+02	4.3E+02	3.4E+03	2.9E+02	2.5E+03
U-238	2.2E+01	3.0E+03	8.2E+02	1.7E+04	1.3E+03	1.3E+04

NOTE: (1) These cleanup goals (CGs) are to be used as the criteria for the remediation activities described in Section 7 of this plan.

(2) The CG_W values for surface soil and streambed sediment are the same as the limited dose assessment DCGL values in Table 5-11. The CG_{EMC} values were producing by scaling the values provided in Table 5-8 and apply to 1 m^2 areas of elevated contamination.

(3) These CG_W values and CG_{EMC} values are the DCGL values in Table 5-8 reduced by a factor of 0.50 as discussed below.

(4) These cleanup goals apply in the year 2041 and later.

The basis for these cleanup goals is as follows. Compliance with the cleanup goals used for remediation when mixtures of radionuclides are present would be determined by use of the sum-of-fractions approach.

Basis for Cleanup Goals for Surface Soil

The surface soil CG_W values are the values in the Surface Soil $DCGL_W$ Assessment column of Table 5-13. DOE considers these goals to be conservative and appropriate to provide assurance that any remediation of surface soil and sediment in drainage ditches on the project premises that may be accomplished during Phase 1 of the proposed decommissioning would support releasing the remediated areas under the criteria of 10 CFR 20.1402, should the licensee eventually determine that approach to be appropriate for Phase 2 of the decommissioning.[14]

Basis for Cleanup Goals for Subsurface Soil

DOE has established the subsurface soil cleanup goals at 50 percent of subsurface soil DCGLs calculated in the limited site-wide dose assessments for 22.5 mrem per year (Table 5-12). The cleanup goals for subsurface soil would therefore equate to 11.25 mrem per year. DOE is taking this approach to provide additional assurance that remediation of the WMA 1 and WMA 2 excavated areas would support all potential options for Phase 2 of the proposed decommissioning.

Basis for Cleanup Goals for Streambed Sediment

DOE has used the $DCGL_W$ values from the limited site-wide dose assessment (the last column in Table 5-12 and Table 5-13) as the cleanup goals for streambed sediment. These values are substantially less than those developed for the base-case recreationist scenario

[14] As noted previously, surface soil may or may not be remediated in Phase 1 of the decommissioning. However, it is possible that characterization performed early in Phase 1 could identify surface soil contamination that would warrant remediation to reduce radiation doses during the period between Phase 1 and Phase 2 of the decommissioning. In the unlikely event that this situation developed, the areas of concern would be remediated in Phase 1.

and are considered to be supportive of any approach that may be selected for Phase 2 of the proposed decommissioning.

As noted in the discussion on the ALARA analysis results, DOE expects that the actual levels of residual radioactivity would turn out to be less than the DCGLs used for remediation, i.e., these cleanup goals, owing to the characteristics of the remediation method to be used.

5.4.2 Refining DCGLs and Cleanup Goals

The calculated DCGLs for 25 mrem per year and the associated cleanup goals would be refined as appropriate after the data from the soil and sediment characterization program to be completed early in Phase 1 of the proposed decommissioning becomes available. These data are expected to provide additional insight into the radionuclides of interest in environmental media and the depth and areal distribution of the contamination. Such information could, for example, lead to deleting one or more radionuclides from further consideration in the Phase 1 cleanup or lead to more realistic source geometry for development of DCGLs for surface soil contamination. Analytical data from the subsurface soil characterization measurements being taken in 2008 could also provide information to help refine the subsurface soil DCGLs.

If evaluation of the new data leads to refinement of the DCGLs and cleanup goals, then this plan would be revised accordingly to reflect the new values. Since such a change could affect the project end conditions, the plan revision would be provided to NRC for review and input prior to issue following the change process described in Section 1.

5.4.3 Use of a Surrogate Radionuclide DCGL

A *surrogate radionuclide* is a radionuclide in a mixture of radionuclides whose concentration is easily measured and can be used to infer the concentrations of the other radionuclides in the mixture. If actual radioactive contamination levels of the surrogate radionuclide are below the specified concentration, then the sum of doses from all radionuclides in the mixture would fall below the dose limit.[15]

The tables in this section do not provide $DCGL_W$ values for a surrogate radionuclide because available data on radionuclide distributions in soil and sediment are not sufficient to support this. However, surrogate radionuclide $DCGL_W$ values for the cleanup goals would be developed and incorporated into this section if evaluation of additional characterization data shows that Cs-137 or another easy to measure radionuclide can be used effectively as a surrogate for all radionuclides in source soil, subsurface soil, and/or streambed sediment in an area.

5.4.4 Preliminary Dose Assessment

Preliminary dose assessments have been performed for the remediated WMA 1 and WMA 2 excavations. These assessments made use of the maximum measured

[15] Guidance on the use of surrogate measurements provided in Section 4.3.2 of NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000) would be followed.

radioactivity concentration in the Lavery till for each radionuclide as summarized in Table 5-1, and the results of modeling to develop DCGLs for 25 mrem per year as shown in Table 5-8. The results were as follow:

WMA 1, a maximum of 1.0 mrem a year

WMA 2, a maximum of 0.08 mrem a year

Given the limited data available, these results must be viewed as order-of-magnitude estimates. However, they do suggest that actual potential doses from the two remediated areas are likely to be substantially below 25 mrem per year.

5.4.5 Final Dose Assessment

As noted previously, DOE would perform a dose assessment for the residual radioactivity in the WMA 1 and WMA 2 excavated areas using Phase 1 final status survey data. This assessment would use the same methodology used in development of the subsurface soil DCGLs to estimate the potential radiation dose using the actual measured residual radioactivity concentrations. The results of the dose assessment would be made available to NRC and other stakeholders. Note that a more-comprehensive dose assessment that also takes into account the Phase 2 sources may be performed in connection with Phase 2 of the proposed decommissioning, depending on the approach selected for that phase.

5.5 References

Code of Federal Regulations

10 CFR 20, Subpart E, *Radiological Criteria For License Termination (LTR).*

10 CFR 20.1003, *Definitions.*

DOE Orders

DOE Order 450.1, *Environmental Protection Program,* including Changes 1 and 2. U.S, Department of Energy, Washington, D.C. January 15, 2003.

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

DOE Technical Standards

DOE Standard 1153-2002, *A Graded Approach for Evaluating Radiation Doses to Aquatic and Terrestrial Biota.* U.S, Department of Energy, Washington, D.C., July 2002.

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NRC 2006, *Consolidated NMSS Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria, Final Report,* NUREG 1757 Volume 2, Revision 1. NRC, Office of Nuclear Material Safety and Safeguards, Washington, DC, September, 2006.

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WVNSCO 1993b, *Environmental Information Document Volume I, Geology,* WVDP-EIS-004, Revision 0. West Valley Nuclear Services Company, West Valley, New York, April 1, 1993

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6.0 ALARA ANALYSIS

PURPOSE OF THIS SECTION

The purpose of this section is to describe how DOE would achieve a proposed decommissioning goal below the 25 mrem per year dose limit in those areas remediated during Phase 1 of the proposed decommissioning and describe quantitative cost-benefit analyses to demonstrate that potential future doses from residual radioactivity in surface soil, subsurface soil, and streambed sediment would be as low as reasonably achievable (ALARA).

INFORMATION IN THIS SECTION

This section provides the following information:

- In Section 6.1, brief summaries of relevant NRC requirements and guidance and the planned remediation approach, along with a discussion of the derived concentration guideline levels (DCGLs);
- In Section 6.2, a brief summary of how DOE would achieve a proposed decommissioning goal below the dose limit; and
- In Section 6.3, a description of the ALARA analysis process, which focuses on the DCGLs, and the results of preliminary ALARA analyses which indicate that remediation of contaminated surface soil, subsurface soil, and streambed sediment below DCGLs for 25 mrem per year would not be cost-effective.

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. Useful background information is also provided in Section 2 on site history, in Section 3 on the facilities of interest, and in Section 4 and Appendix B on the radiological status of the project premises.

Section 5 describes the DCGLs that are the primary focus of the analysis process described in this section and summarizes how they were developed. Section 7 describes the Phase 1 proposed decommissioning activities.

6.1 Introduction

To put into context the ALARA process described below, it is useful to consider the applicable requirements and guidance, the planned remediation activities, and the DCGLs on which the ALARA process focuses.

After an area has been remediated to meet the cleanup criteria, additional remediation actions could be taken to further reduce the level of residual radioactivity. An ALARA analysis compares the benefits and costs of those additional remediation actions to determine whether or not it would be cost effective to implement any of them.

6.1.1 Applicable Requirements and Guidance

The NRC's Final Policy Statement on Decommissioning Criteria for the WVDP (NRC 2002) prescribed the NRC's License Termination Rule (10 CFR 20, Subpart E) as the decommissioning criteria for the WVDP. As explained in Section 1, certain areas of the project premises are being remediated in Phase 1 of the proposed decommissioning to NRC's unrestricted release criteria of the License Termination Rule. These criteria, which appear in 10 CFR 20.1402, state that:

> "A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE [total effective dose equivalent] to an average member of the critical group that does not exceed 25 mrem per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal."[1]

Appendix N of NUREG-1757, Volume 2 (NRC 2006) "describes methods acceptable to NRC staff for determining when it is feasible to further reduce the concentrations of residual radioactivity to below the concentrations necessary to meet the dose criteria", i.e., methods for performance of an ALARA analysis. NUREG/BR-0058 (NRC 2004) recommends use of a value of $2,000 per person-rem for ALARA analyses.

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[1] In 10 CFR 20.1003, NRC defines ALARA as follows: *ALARA* (acronym for "as low as is reasonably achievable") means making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part [10 CFR 20] as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

DOE defines ALARA in DOE Order 5400.5 as follows: "an approach to radiation protection to control or manage exposures (both individual and collective to the work force and the general public) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ... ALARA is not a dose limit, but rather it is a process that has as its objective the attainment of dose levels as far below the applicable limits of the Order as practicable."

How the ALARA process is applied for the subject analysis is discussed in Section 6.3.1.

As explained in Section 1.7 of this plan, the ALARA process is an integral part of DOE radiation control procedures applicable to Phase 1 of the proposed decommissioning. The ALARA process has been incorporated into the remediation strategy for the Phase 1 proposed decommissioning work as explained below.

6.1.2 Remediation Activities of Interest

Section 1.10.2 of this plan identifies the facilities within the scope of Phase 1 proposed decommissioning activities and explains that a soil and sediment characterization program would be undertaken early in the proposed decommissioning to better define the nature and extent of radioactive contamination in surface soil, subsurface soil, and streambed sediment on the project premises. This section also explains that radioactively contaminated subsurface soil in excess of DCGLs would be removed from large areas to be excavated in WMA 1, the Process Building and Vitrification Facility area, and WMA 2, The Low-Level Waste Treatment Facility area. Figure 1-2 shows these areas.

Section 1.10.2 also explains that remediation of environmental media during Phase 1 of the proposed decommissioning would be limited to soil within these large excavations unless this plan is revised. This plan may be revised to provide for remediation of surface soil in other parts of the project premises and streambed sediment in Erdman Brook and Franks Creek (within the project premises only) during Phase 1 of the proposed decommissioning depending on factors such as the results of the characterization program and available funding.

Section 7 of this plan provides additional details of Phase 1 proposed decommissioning activities including conceptual drawings showing the two major excavations and the methods for contaminated soil removal.

6.1.3 The DCGLs Involved

As explained in Section 5, three sets of DCGLs have been developed for Phase 1 of the proposed decommissioning. These DCGLs apply to (1) surface soil, (2) subsurface soil in the large WMA 1 and WMA 2 excavations, and (3) streambed sediment in Erdman Brook and Franks Creek.

The DCGLs were based on the unrestricted release dose limit of 25 mrem per year to the average member of the critical group of interest. Section 5 identifies the DCGLs and describes the conceptual models and the mathematic model (RESRAD) used in their development. Section 5 also describes additional dose assessments performed to ensure that remediation criteria used in Phase 1 do not limit potential options for Phase 2 of the decommissioning and the resulting cleanup goals, which are provided in Table 5-13.

6.2 Achieving a Decommissioning Goal Below the Dose Limits

DOE's plans to ensure that doses from residual radioactivity at the conclusion of the WVDP Phase 1 proposed decommissioning are ALARA include:

- A Phase 1 proposed decommissioning strategy that promotes ALARA,

- Conservatism inherent in development of DCGLs and the lower cleanup goals that would guide the decontamination efforts, and
- Use of remediation processes that are conservative by nature.

Cost-benefit analyses would be performed during Phase 1 of the proposed decommissioning to determine whether residual radioactivity levels should be decreased to further reduce future potential doses. The cost-benefit analysis process is described in Section 6.3.

Upon completion of Phase 1 of the proposed decommissioning and in preparation for Phase 2, additional dose evaluations would be performed utilizing Phase 1 final status survey data as a further demonstration that potential future doses from residual radioactivity in those areas remediated in Phase 1 are ALARA.

6.2.1 Phase 1 Proposed Decommissioning Strategy Promotes ALARA

As summarized in Section 1.10.2 and detailed in Section 7, DOE's Phase 1 proposed decommissioning strategy for the WVDP has been designed to reduce risk from residual radioactivity consistent with the ALARA process. For example:

- A new Canister Interim Storage Facility would be built on the south plateau and the vitrified HLW canisters moved there to allow removal of the contaminated Process Building.
- Most other contaminated surface structures would also be completely removed, including the Vitrification Facility, a process that would significantly reduce risk by reducing residual radioactivity on the project premises.
- The source area of the north plateau groundwater plume beneath the Process Building would be completely removed, a process that would also significantly reduce risk from residual radioactivity on the project premises.
- Vertical hydraulic barrier walls installed to support the WMA 1 and WMA 2 excavations would be left in place after Phase 1 of the proposed decommissioning to minimize the potential for contaminant migration though groundwater among different parts of the project premises, including the potential for recontamination of the remediated WMA 1 and WMA 2 excavated areas.
- All radioactive waste generated in Phase 1 proposed decommissioning activities would be disposed of offsite.
- Potentially contaminated soil and sediments within the project premises would be characterized to better define potential risk from residual radioactivity in these media, and surface soil and streambed sediment exceeding DCGLs may be remediated in Phase 1, which would effectively eliminate the risk associated with this environmental media contamination.

- Essentially all radioactive material that would remain after the Phase 1 activities have been completed would be located underground, primarily in the underground waste tanks and in the NDA. Controlled access to the WVDP would continue during the Phase 1 institutional control period, which would prevent access to this underground radioactivity.

6.2.2 Conservatism in DCGL Development

- The process for developing DCGLs for Phase 1 of the proposed decommissioning as described in Section 5 was conservative in several respects. Section 5 provides examples of this conservatism.

6.2.3 Conservatism from the Decontamination and Final Status Survey Processes

As explained in Section 7, bulk soil removal techniques using equipment such as tracked excavators and backhoes would be used to remove contaminated soil. These techniques are not precision processes, but remove soil (and its associated contamination) in discrete increments. Typically, they remove more soil than necessary so that the remaining concentration falls well below the DCGL. This inherent characteristic would result in average residual contamination in decontaminated areas generally being well below the $DCGL_W$ value.

NRC recognizes in NUREG-1496 (NRC 1997) that the soil remediation process would result in residual contamination below the DCGLs by stating:

> "In actual situations, it is likely that even if no specific analysis of ALARA were required for soil removal that the actual dose will be reduced to below 25 mrem/y because of the nature of the removal process. For example, the process of soil excavation is a coarse removal process that is likely to remove large fractions of the remaining radioactivity."

Another factor that adds conservation is the final status survey process, which is described in Section 9. This process follows guidance in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000) and the MARSSIM statistical techniques require the average residual radioactivity concentrations to be less than the $DCGL_W$ values. (In the case of this plan, the average residual radioactivity concentrations would be less than the cleanup goals or CG_W values.)

6.3 DCGL ALARA Analysis

This section describes the ALARA analysis process as a cost-benefit process as recommended by NRC (NRC 2006) and then provides the results of preliminary ALARA analyses for DCGLs for surface soil, subsurface soil, and streambed sediment.

6.3.1 ALARA Analysis Guidance

NRC guidance on ALARA analysis for remediation actions is found in Appendix N to NUREG-1757, volume 2 (NRC 2006). The guidance discusses possible costs and benefits that may be considered as indicated in Table 6-1.

Table 6-1. Possible Benefits and Costs Related to Decommissioning[1]

Possible Benefits	Possible Costs
Collective dose averted[2]	Remediation costs
Regulatory costs avoided	Additional occupational/public dose
Changes in land values	Occupational nonradiological risks
Esthetics	Transportation direct costs and implied risks
Reduction in public opposition	Environmental impacts
	Loss of economic use of site/facility

NOTES: (1) From Table N-1 of NUREG-1757, Volume 2 (NRC 2006).
(2) Collective dose averted is the primary possible benefit as discussed below.

The NRC guidance includes additional discussion of monetary costs that may be considered in the analysis, explaining that the costs associated with remediation beyond the cleanup goals (the remediation action) "generally include the monetary costs of: (1) the remediation action being evaluated, (2) transportation and disposal of the waste generated by the action, (3) workplace accidents that occur because of the remediation action, (4) traffic fatalities resulting from transporting the waste generated by the action, (5) doses received by workers performing the remediation action, and (6) doses to the public from excavation, transport, and disposal of the waste." (NRC 2006)

The NRC guidance also includes the following guidance related to limiting the scope of a preliminary analysis:

- "The primary benefit from a remediation action is the collective dose averted in the future, i.e., the sum over time of the annual doses received by the exposed population."
- "In the simplest form of the [ALARA] analysis, the only benefit estimated from a reduction in the level of residual radioactivity is the monetary value of the collective averted dose to future occupants of the site."

Consistent with this guidance, the only benefit considered in the preliminary ALARA analysis for the DCGLs is the collective dose averted by the action. The primary quantifiable cost is the disposal of the waste generated by the action, and that is the cost considered in this preliminary ALARA analysis.

6.3.2 Calculating Benefits and Costs

As defined in Section N.1.3 of NUREG-1757, Volume 2 (NRC 2006), the "residual radioactivity level that is ALARA is the concentration, Conc, at which the benefit from removal equals the cost of removal." The benefit from removal, i.e., the present worth of a future collective averted dose, can be calculated via NUREG-1757, Volume 2 (NRC 2006), Equations N-1 and N-2, combined below:

$$B_{AD} = \$2000 \times P_D \times A \times 0.025 \times F \times \frac{Conc}{DCGL_W} \times \frac{1 - e^{-(r+\lambda)N}}{r + \lambda}$$

where:

B_{AD}	=	benefit from an averted dose for a remediation action ($),
$2000	=	value in dollars of a person-rem averted (NRC 2004) ($/person-rem),
P_D	=	population density for the critical group scenario (persons/m^2),
A	=	area being evaluated (m^2),
0.025	=	annual dose to an average member of the critical group from residual radioactivity at the $DCGL_W$ (rem/y),
F	=	effectiveness, or fraction of the residual radioactivity removed by the remediation action (unit-less),
Conc	=	average concentration of residual radioactivity in the area being evaluated (pCi/g),
$DCGL_W$	=	derived concentration guideline equivalent to the average concentration of residual radioactivity that would give an annual dose of 25 mrem to the average member of the critical group (pCi/g),[2]
r	=	monetary discount rate (per year),
λ	=	radiological decay constant (per year), and
N	=	number of years over which the collective dose was calculated (years).

Setting the benefit from removal, B_{AD}, equal to the cost of the remediation, $Cost_T$, and solving for the ratio of the concentration, Conc, to the $DCGL_W$ gives NUREG-1757, Equation N-8:

$$\frac{Conc}{DCGL_W} = \frac{Cost_T}{\$2000 \times P_D \times 0.025 \times F \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}}$$

Where all parameters are as previously defined.

For convenience in the following discussion, the ratio of the concentration, Conc, to the $DCGL_W$ is defined as R.

When R is 1 or greater, the residual concentration (Conc) that is ALARA is equal to or greater than the $DCGL_W$, and no further remediation is needed to reduce the concentration to below the $DCGL_W$ level. When R is less than 1, then the concentration that is ALARA is less than the $DCGL_W$, and further remediation should be undertaken to reduce the residual concentration. For example, if R is equal to 0.5 for a particular remediation action, and the measured surface concentration is below the $DCGL_W$ value, but above 0.5 times the $DCGL_W$

[2] The DCGL applicable to the average concentration over a survey unit is called the $DCGL_W$ (W = Wilcoxon Rank Sum), whereas the DCGL applicable to limited areas of elevated concentrations within a survey unit is called the $DCGL_{EMC}$ (EMC = Elevated Measurement Comparison). (NRC, 2006).

value, then in order to meet the ALARA criterion that particular remediation action should be implemented.

6.3.3 Surface Soil Preliminary ALARA Analysis

For surface soil, the NUREG-1757, Volume 2 (NRC 2006), Table N.2 generic parameters are P_D = 0.0004 person/m^2, r = 0.03/y, and N = 1000 y. Also since surface soil remediation usually involves total removal of the soil, the remediation action efficiency (F) has been conservatively set to 1.0. Using these values to calculate the soil Conc to $DCGL_w$ ratio (R) gives:

$$R = \frac{C_{Tu}}{\$2000 \times 0.0004 \times 0.025 \times 1.0} \times \frac{0.03 + \lambda}{1 - e^{-(0.03+\lambda)1000}}$$

In the above equation the total cost of remediation ($Cost_T$) divided by the total area to be remediated (A) has been replaced by the total unit cost of remediation (C_{Tu}, \$/m^2).

If the surface soil concentration is set equal to the $DCGL_W$ (i.e., R = 1) then the above equation can be solved to determine the maximum remediation unit cost that would be ALARA. This is shown in the equation below, which has conservatively removed the radiological decay term.[3]

$$C_{Tu} = \$2000 \times 0.0004 \times 0.025 \times 1.0 \times \frac{1 - e^{-(0.03)1000}}{0.03}$$

Solving the above equation for C_{Tu} gives the maximum ALARA unit cost of \$0.67/m^2. In other words, if surface soil can be removed and disposed of for \$0.67/m^2, or less, then it would be consistent with the ALARA process to do so, but if it costs more than \$0.67/m^2 to remove and dispose of surface soil, then no further remediation below the $DCGL_W$ is necessary.

Removing six inches of soil would result in waste volumes of 5.38 cubic feet per square meter remediated. With a LLW disposal cost of \$6.76 per cubic foot (URS 2008, Table 3-16), the soil disposal component of the total remediation cost alone is about \$36.38/m^2. Consequently, residual radioactivity in surface soil at the $DCGL_W$ at the WVDP is ALARA, and soil remediation below the surface soil $DCGL_W$ is not necessary.

This result is consistent with NUREG-1496 (NRC 1997, page 7-6), which states: "there appears to be a strong indication that removing and transporting soil to waste burial facilities to achieve exposure levels at the site at or below a 25 mrem/y unrestricted use dose criterion is generally not cost-effective". It is also consistent with the surface soil example given in NUREG-1757, Section N.1.4, which states: "the dose limit [25 mrem/y] would be limiting by a

[3] Omitting the decay constant is conservative for shorter-lived radionuclides. For example, including a 30-year decay constant for Cs-137 or Sr-90 would result in a maximum ALARA unit cost of approximately \$0.38/m^2 for those radionuclides. The value of \$0.67/m^2 for long-lived radionuclides is not changed by omission of the decay constant in the equation.

considerable margin. Based on these results, it would rarely be necessary to ship soil to a waste disposal facility to meet the ALARA requirement. The licensee could use this [NUREG-1757] evaluation to justify not removing soil." (NRC 2006, page N-12).

6.3.4 Subsurface Soil Preliminary ALARA Analysis

For subsurface soil, it is appropriate to use the same parameter values to determine the Conc to $DCGL_W$ ratio (R) as were used for surface soil. Therefore, if subsurface soil can be removed and disposed of for $0.67/m^2, or less, then it is consistent with the ALARA process to do so, but if it costs more than $0.67/m^2 to remove and dispose of subsurface soil, then no further remediation below the $DCGL_W$ is necessary.

While the disposal unit cost for surface soil and subsurface would be the same, the cost to remediate subsurface soil would likely be higher than the cost for surface soil removal because removal of soil from the bottom or sides of the excavation would likely be more difficult than removal of surface soil.

Therefore, since for subsurface soil: (1) the Conc to $DCGL_W$ ratio (R) would be the same as for surface soil, (2) the cost to remediate would likely be higher than for surface soil, and (3) surface soil at the $DCGL_W$ is ALARA, it is concluded that remediation below the sub-surface soil $DCGL_W$ is similarly not necessary, and that subsurface soil at the $DCGL_W$ satisfies the ALARA criteria.

6.3.5 Streambed Sediment Preliminary ALARA Analysis

Likewise, for streambed sediment it is appropriate to use the same parameter values to determine the Conc to $DCGL_W$ ratio (R) as were used for surface and subsurface soils.[4] Therefore, if streambed sediment can be removed and disposed of for $0.67/m^2, or less, then it is consistent with the ALARA process to do so, but if it costs more than $0.67/m^2 to remove and dispose of streambed sediment, then no further remediation below the $DCGL_W$ is necessary.

The cost to remediate and dispose of streambed sediment would be similar to the cost for surface soil removal, except that streambed sediments of interest are located in Erdman Brook and the portion of Franks Creek on the project premises and are likely to be wet. Both of these factors would complicate the removal process – that is, managing the wet contaminated soil and the difficultly in providing equipment access owing to the steep stream banks – with the result that the remediation of streambed sediments would likely be more costly than the remediation of an equivalent amount of surface soil.

[4] One parameter that would be appropriately different for streambed sediment is the population density. The steep slopes in the areas of Erdman Brook and Franks Creek would reasonably be expected to preclude building residences in the area of these streams. However, use of the 0.0004 persons/m^2 value (about 1040 persons per square mile) is conservative because a more realistic smaller value would produce a higher R value. The population density in Cattaraugus County in 2000 was 64 persons per square mile using the total population figure in Table 3-6.

Therefore, since for streambed sediments: (1) the Conc to $DCGL_W$ ratio (R) would be the same as for surface soil, (2) the cost to remediate would likely be higher than surface soil, and (3) surface soil at the $DCGL_W$ is ALARA, it is concluded that remediation below the streambed sediment $DCGL_W$ is similarly not necessary, and that streambed sediment at the $DCGL_W$ is ALARA.

6.4 Additional Analyses

Additional ALARA analyses would be performed in connection with remediation of the WMA 1 and WMA 2 excavations. These analyses would make use of updated values for parameters such as LLW disposal costs, as well as in-process survey results for radioactivity in soil at the base of the excavation during soil removal activities.

Factors not included in the simple preliminary analyses such as other societal and socioeconomic considerations, the costs related to occupational risks, and transportation of additional waste would be taken into account in the additional ALARA analyses. Consideration would also be given in these analyses as to whether remediation of the WMA 1 and WMA 2 excavations to DCGLs (actually to the cleanup goals) for surface soil, rather than for subsurface soil, would be cost-effective.

NOTE

As mentioned previously, DOE has already established cleanup goals below the DCGLs calculated for 25 mrem per year for surface soil, subsurface soil and streambed sediment as explained in Section 5, based on considerations such as the complexity of the site and its different source areas, to ensure that cleanup criteria used in Phase 1 of the proposed decommissioning would support all potential options for Phase 2.

Also, as described in Section 5, a final dose analysis would be performed using Phase 1 final status survey data for the WMA 1 and WMA 2 excavations to estimate potential doses from residual radioactivity from these areas assuming that the entire project premises were to be remediated to the License Termination Rule criteria for unrestricted release.

6.5 References

Code of Federal Regulations

10 CFR 20.1003, *Definitions.*

10 CFR 20, Subpart E, *Radiological Criteria For License Termination (LTR).*

DOE Orders

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

Other References

NRC 1997, *Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities; Final Policy Statement.* NUREG-1496, Vol. 1. U.S. Nuclear Regulatory Commission, Office of Regulatory Research, Division of Regulatory Applications, Washington, D.C., July 1997.

NRC 2000, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, NUREG-1575, Revision 1. NRC, Washington, DC, August, 2000. (Also EPA 4-2-R-97-016, Revision 1, U.S. Environmental Protection Agency and DOE-EH-0624, Revision 1, DOE)

NRC 2002, *Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement.* U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Vol. 67, No. 22, February 1, 2002.

NRC 2004, *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission*, NUREG/BR-0058, Rev. 4. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, D.C., September 2004.

NRC 2006, *Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria, Final Report*, NUREG-1757, Vol. 2, Rev. 1. U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Division of Waste Management and Environmental Protection, Washington, D.C., September 2006.

URS 2008, *Facility Description and Methodology Technical Report*, WSMS-WV-08-0001, Revision 0. URS Washington Division, West Valley, New York, August 2008.

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7.0 PLANNED DECOMMISSIONING ACTIVITIES

PURPOSE OF THIS SECTION

The purpose of this section is to describe the Phase 1 decommissioning activities.

INFORMATION IN THIS SECTION

This section provides the following information:

- In Section 7.1, a brief summary of site conditions expected at the beginning of the Phase 1 decommissioning activities;
- In Section 7.2, a summary of the general approach and the general requirements that apply to the decommissioning activities;
- In Sections 7.3 through 7.10, descriptions of the Phase 1 decommissioning activities;
- In Section 7.11, a summary of the types of remediation and demolition technologies to be employed; and
- In Section 7.12, a discussion of the conceptual project schedule.

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 plan, Section 2 on facility operating history, and Section 3 that describes the facilities at the WVDP. One should also consider the radiological status information presented in Section 4.

The activities described here would be accomplished in accordance with requirements in other sections, as follows:

- Section 1.6, project management and project organization,
- Section 1.7, radiation safety and monitoring of workers;
- Section 1.8, environmental monitoring and control;
- Section 1.9, radioactive waste management;
- Section 8, quality assurance for engineering design, data, and calculations; for characterization; for engineered barrier installation; and for final status surveys; and
- Section 9, characterization surveys, in-process surveys, and final status surveys.

7.1 Conditions at the Beginning of the Phase 1 Decommissioning Work

Section 1.10 of this plan describes the interim end state to be reached at the conclusion of WVDP facility deactivation work. Section 4 summarizes the radiological conditions of facilities and areas within the scope of this plan. Table 7-1 notes the expected conditions in each facility or area in the interim end state, i.e., at the beginning of the Phase 1 proposed decommissioning work, based on information provided in Section 2 and Section 4. This table does not address soil and groundwater except in WMA 1 and WMA 2 where large areas would be excavated.

Table 7-1. Facility and Area Conditions at the Beginning of Phase 1[(1)]

WMA	Facility/Area	Conditions (See legend at table's end for acronyms)
1	Process Building	Partially decontaminated, high radiation levels in some cells, vitrified HLW canisters in the HLW Interim Storage Facility, CSRF removed.
	Vitrification Facility	Partially decontaminated, high radiation levels in Vitrification Cell.
	01-14 Building	Significant contamination in filters, portion of off-gas line in building[(2)].
	Vitrification off-gas line	Significant residual radioactivity.
	Utility Room	No contamination above MDC in most areas.
	Utility Room Expansion	No contamination above MDC in most areas.
	Load-In/Load-Out Facility	No contamination above MDC in most areas.
	Plant Office Building	No contamination above MDC.
	Fire Pump House	Not impacted by radioactivity.
	Water Storage Tank	Not impacted by radioactivity.
	Electrical Substation	Not impacted by radioactivity.
	Underground tanks	Significant contamination in Tank 7D-13, little in others.
	Underground lines	Significant contamination in some lines, especially 7P120-3.
	Subsurface soil, groundwater	Significant contamination in plume source area under the Process Building
	Surface soil	Low-level contamination may be present in several areas.
2	Lagoon 1	Deactivated, significant radioactivity in sediment.
	Lagoon 2	In use, radioactive water, significant radioactivity in sediment.
	Lagoon 3	In use, radioactive water, low levels of radioactivity in sediment.
	Lagoon 4	In use, radioactive water, low levels of radioactivity in sediment.
	Lagoon 5	In use, radioactive water, low levels of radioactivity in sediment.
	Interceptors	In use, significant contamination in Old Interceptor, less in new ones.
	Neutralization Pit	In use, low-level contamination.
	LLW2 Building	In use, low level contamination, radioactive water in sump.
2	Underground lines	Most in use, low-level contamination.
	Solvent Dike	Low-level contamination in soil.
	Subsurface soil, groundwater	Contaminated with Sr-90 in plume area, other subsurface soil contamination.

Table 7-1. Facility and Area Conditions at the Beginning of Phase 1[1]

WMA	Facility/Area	Conditions (See legend at table's end for acronyms)
	Surface soil	Low-level contamination in much of area.
3	Tank 8D-1[3]	Laid up, one HLW transfer pump and five mobilization pumps in place.
	Tank 8D-2[3]	Laid up, one HLW transfer pump and four mobilization pumps in place.
	Tank 8D-3[3]	Laid up, one submersible pump in place.
	Tank 8D-4[3]	Laid up, one submersible pump in place.
	Con-Ed Building	Low levels of residual radioactivity, mostly inside equipment.
	Equipment Shelter	Low levels of residual radioactivity, mostly inside equipment.
	HLW transfer trench	High levels of residual radioactivity inside piping and equipment.
4	Construction and Demolition Debris Landfill	Low level Sr-90 contamination from the north plateau groundwater plume in some buried waste and in other parts of WMA 4. [WMA 4 and the landfill are not within the Phase 1 proposed decommissioning scope.]
5	Lag Storage Addition 4, Depot	No contamination above MDC.
	RHWF	Low levels of contamination, but may be significant in Work Cell.
6	Sewage Treatment Plant	Not impacted by radioactivity.
	South WTF Test Tower	Not impacted by radioactivity.
	Demineralizer sludge ponds	Low levels of radioactivity in soil.
	Equalization basin	Not impacted by radioactivity.
	Equalization tank	Not impacted by radioactivity.
7	NRC-Licensed Disposal Area (NDA)	Significant radioactivity in buried waste, low-level surface soil contamination. [The NDA is not within the Phase 1 proposed decommissioning scope.]
9	Drum Cell	No contamination above MDC.
10	New Warehouse	Not impacted by radioactivity.

NOTES: (1) See also Table 2-12 in Section 2, which contains information on the radiological status of remaining concrete floor slabs and foundations.
(2) The filters may be removed before Phase 1 begins.
(3) These tanks contain significant amounts of residual radioactivity and the mobilization and transfer pumps are expected to have high radiation levels as indicated in Section 4.1.

LEGEND: CSRF = Contact Size Reduction Facility (former Master-Slave Manipulator Repair Shop)
MDC = minimum detectable concentration
RHWF = Remote-Handled Waste Facility
WTF = Waste Tank Farm

7.2 General Approach and General Requirements

7.2.1 General Approach

As explained in Section 1, it is proposed that the WVDP decommissioning be accomplished in two phases. The following activities would take place in Phase 1.

Facility and Equipment Removal

The following facilities and equipment would be removed:

- All WMA 1 facilities, including the three underground wastewater tanks and the underground lines;
- In WMA 2, the five lagoons, the Interceptors, the Neutralization Pit, the LLW2 Building, the Solvent Dike, the Maintenance Shop leach field, the remaining concrete slabs and foundations, and the underground wastewater lines within the large excavation;
- In WMA 3, the waste tank mobilization and transfer pumps, the Con-Ed Building, the Equipment Shelter and condensers, and the piping and equipment in the HLW transfer trench;
- In WMA 5, the two remaining structures – Lag Storage Addition 4 and the Remote-Handled Waste Facility – and the remaining concrete floor slabs and foundations;
- In WMA 6, the Sewage Treatment Plant, the south Waste Tank Farm Test Tower, the two demineralizer sludge ponds, the equalization basin, the equalization tank, and the remaining concrete floor slabs and foundations;
- In WMA 7, the remaining gravel pads associated with the NDA hardstand;
- In WMA 9, the Integrated Radwaste Treatment System Drum Cell, the sub-contractor maintenance area, and the trench soil container area; and
- In WMA 10, the New Warehouse.

The following facilities and equipment on the project premises are not within the scope of the Phase 1 proposed decommissioning activities:

- In WMA 2, the North Plateau Pump and Treat System, the Pilot Scale Permeable Treatment Wall, the Full-Scale Permeable Treatment Wall, and underground lines not within the excavated areas;
- In WMA 3, the four underground waste tanks, the Permanent Ventilation System Building, the Supernatant Treatment System Support Building, the HLW transfer trench itself, and the underground lines;
- In WMA 4, the Construction and Demolition Debris Landfill and the new Permeable Reactive Barrier;
- In WMA 6, the rail spur;
- In WMA 7, the NDA and the associated interceptor trench; and
- In WMA 10, the Meteorological Tower and the Security Gatehouse.

Approach

Soil and sediment on the project premises would be characterized for radioactivity. Before the Process Building is removed, the new Canister Interim Storage Facility would be built on the south plateau, the Load-In Facility converted to a Load-Out Facility, and vitrified HLW canisters transported to the new Canister Interim Storage Facility.

One large excavation would be dug to remove the WMA 1 facilities and a second large excavation dug to remove key WMA 2 facilities. These excavations would extend down into the underlying Lavery till. Contaminated surface and subsurface soil in these excavations

would be removed to achieve derived concentration guideline levels (DCGLs) for unrestricted release specified in Section 5[1]. The source area of the north plateau groundwater plume in WMA 1 would be removed, but not the non-source area portion of the plume, except for those portions that fall within the large WMA 1 and WMA 2 excavations.

Activity Integration

The work would be sequenced for maximum efficiency. For example, the Low-Level Waste Treatment Facility would be kept in service until the Process Building is taken down so its wastewater treatment capabilities can be utilized during the Process Building decontamination and demolition work. The conceptual schedule in Figure 7-15 describes the general sequence. Section 1.6 describes the more-detailed schedules that would be used in management of the project.

More details would appear in one or more Decommissioning Work Plans, which would be completed before the Phase 1 proposed decommissioning activities begin and would address matters such as demolition of the Process Building and the Vitrification Facility.

7.2.2 General Requirements

The following general requirements would be adhered to during proposed decommissioning activities described in Sections 7.3 through 7.10.

Use of Approved Written Procedures

Following DOE policy, the proposed decommissioning activities would be accomplished in accordance with written procedures formally approved by the appropriate member(s) of the decommissioning team.

Remedial Technologies

The decommissioning contractor would utilize efficient, proven technologies in accomplishment of the work. Section 7.11 provides examples of these technologies. DOE has generally avoided being prescriptive in methods to be used to give the decommissioning contractor the flexibility to make use of improved methods that may become available. Exceptions include the conceptual designs for engineered barriers, which are more specifically described because of their importance in support of Phase 2 of the proposed decommissioning. The Decommissioning Work Plan(s) would provide more-detailed information on remedial technologies to be used.

Dealing With Unique Remediation Issues

Given the complexities of the site, some remediation issues would be faced during Phase 1 of the proposed WVDP decommissioning that are highly unusual, if not entirely unique. Two such issues are demolition of the Process Building and removal of the

[1] As explained in Section 5, cleanup goals have been established below the DCGLs for unrestricted release to account for combined exposure scenarios that could potentially be encountered if the entire project premises were to be cleaned up to unrestricted release standards in Phase 2 of the decommissioning. Where the term *DCGLs* is used in this section, it refers to the cleanup goals specified in Section 5. The surface soil cleanup goals would be applied from the ground surface to a depth of three feet; below that depth the subsurface soil cleanup goals would apply.

radioactive contamination in the source area of the north plateau groundwater plume that extends far below the building.

The Process Building is an unusually complex structure, much of which is built of heavily-reinforced concrete. Some cells and the spent fuel handling and storage areas extend far below the ground as explained in Section 3. Despite extensive decontamination efforts over a lengthy period, significant amounts of residual radioactivity and high radiation levels will remain in some parts of the structure at the beginning of the Phase 1 proposed decommissioning work as indicated in Tables 4-7 and 4-8 of Section 4. Equipment containing significant amounts of radioactive contamination will also remain in some areas, such as the Liquid Waste Cell.

The process to be followed in demolition of the Process Building is outlined in Sections 7.3.3 and 7.3.8 below. To assist the decommissioning contractor with demolition of the building, DOE is having a Decommissioning Work Plan prepared. This work plan, which would provide implementing details for the requirements in this plan, is being prepared by DOE's current WVDP contractor to take advantage of that contractor's experience with deactivation and partial decontamination of various parts of the building. Experience with demolition of large contaminated buildings at other DOE sites is also being considered in development of this work plan.

Remediation of the source area of the north plateau groundwater plume is being carefully planned. The process to be followed is outlined in Section 7.3.8. Conceptual engineering work performed in support of the Decommissioning EIS has been considered in design of the excavation. The excavation design makes use of an unusually thick (13 feet) vertical hydraulic barrier on the downgradient side to facilitate removal of as much contaminated soil as practical in that area. DOE has considered deep soil remediation experience at other DOE and commercial sites in developing plans to deal with this unusual remediation issue.

Mitigative Measures

Actions would be taken as necessary to eliminate or reduce potential impacts to human health and the environment during the proposed decommissioning work and to prevent recontamination of remediated areas. For example, the excavations for WMA 1 and WMA 2 would be planned to minimize the impacts associated with handling of removed contaminated soil, such as protecting laydown areas with a suitable covering material. Fixatives and water spray would be used as necessary to minimize airborne radioactivity during demolition of contaminated structures and equipment. Suitable covering material would be placed over removed contaminated soil and other loose radioactive waste to prevent the spread of contamination.

Confinement structures also would be used or other radiological control measures taken to minimize the release of airborne radioactivity associated with removal of soil containing significant concentrations of radioactivity. Appropriate dust suppression measures would be taken also during demolition of noncontaminated concrete and steel and during transportation of waste generated in such work.

Mitigative measures would include as low as reasonably achievable (ALARA) considerations, such as removal of contaminated soil to concentrations below the cleanup

goals in cases where this would be practical. Details would be provided in the Decommissioning Work Plan(s) or in a separate Mitigative Measures Plan.

Radiological Controls

Radiological controls and personnel monitoring during proposed decommissioning activities would be in accordance with the DOE radiological control procedures identified in Section 1.7.

Worker Safety

DOE would follow its internal requirements discussed in Section 1.7 and all other applicable requirements to ensure worker safety during the proposed decommissioning work. These requirements would be detailed in a project Health and Safety Plan.

Waste Management

Radioactive waste generated during proposed decommissioning activities would be managed in accordance with DOE procedures identified in Section 1.9, characterized, and disposed of offsite at appropriate government-owned or commercial disposal facilities. Hazardous and toxic waste would be managed and disposed of offsite in accordance with applicable requirements. Non-radioactive equipment and demolition debris would be disposed of offsite at a construction and demolition debris landfill.

Quality Assurance

The quality assurance requirements of Section 8 would be adhered to during engineering analysis and design, compilation of engineering data, characterization, and the Phase 1 final status surveys. Applicable DOE quality assurance requirements would be implemented in other proposed decommissioning activities.

Conceptual and Detailed Designs

This plan describes the processes to be utilized during remediation activities in general terms and designs for engineered barriers and supporting facilities in a conceptual fashion. Detailed procedures for the remediation processes would later be developed consistent with the DOE policy stated above. Likewise, more detailed designs would later be developed for engineered barriers and other engineered features of the proposed decommissioning.

Characterization

As indicated in Section 4, the WVDP facilities and areas had not been completely characterized for radioactivity as of 2008. Additional characterization would be performed as necessary in accordance with the Characterization Sample and Analysis Plan, as explained in Section 9. The soil and sediment characterization would include the portions of the streambeds of Erdman Brook and Franks Creek located on the project premises[2].

[2] It is not intended that the characterization extend outside of the project premises, even in cases where environmental media contamination has been previously identified outside of the project premises, i.e., in the cesium prong area to the northwest of the project premises and in stream sediment in Franks Creek downstream of the project premises.

Some specific cases where additional characterization surveys and sampling would be necessary are identified in this section.

Characterization of subsurface soil in the area of the large WMA 1 and WMA 2 excavations would include collecting samples in the top portion of the Lavery till. Samples of subsurface soil would also be collected along the upgradient and cross-gradient edges of the excavation footprint in WMA 1 and on the edges of the WMA 2 excavation footprint. Analytical data from these samples (1) would help determine the best location for the excavation boundaries, (2) may be useful in refining the conceptual model used in developing subsurface soil DCGLs as described in Section 5, and (3) would support planning Phase 1 final status surveys to be performed on the sides of the excavations.

Characterization measurements would include those necessary for waste management purposes. The decommissioning contractor would provide a procedure for characterizing materials for waste management purposes and obtain DOE approval of this procedure. This procedure would be consistent with applicable DOE requirements and guidance, as well as any applicable State-specified waste acceptance criteria for radioactivity in the offsite landfill(s) where uncontaminated material may be disposed of. This procedure would apply to, among other materials, surface and subsurface soil not known to have been impacted by radioactivity.

Note that the specific proposed decommissioning activities described below are based on assumptions about conditions that will be encountered during the course of the work. If characterization were to disclose unexpected conditions, the proposed decommissioning activities would be changed as necessary to ensure that conditions at the conclusion of the Phase 1 proposed decommissioning activities meet the DCGLs (i.e., the cleanup goals). This plan would be revised as appropriate under these circumstances with NRC involvement as described in Section 1.13.

DCGLs and Cleanup Goals

DCGLs for surface soil, subsurface soil, and stream sediment referred to in this section are the cleanup goals specified in Section 5. The DCGLs for Sr-90 and Cs-137 are based on a 30-year decay period, as discussed previously.

ALARA Analyses

The results of the preliminary ALARA analysis are described in Section 6. As specified in Section 6, additional ALARA analyses would be performed during the WMA 1 and WMA 2 excavations using in-process survey data. These analyses would determine whether remediation to residual radioactivity concentrations below the cleanup goals would be cost-effective. If this is determined to be the case, then additional subsurface soil would be removed as indicated by the results of the analyses.

In-Process Radiological Surveys

In-process surveys would be performed in connection with the proposed decommissioning activities for radiation protection and waste management purposes in accordance with the requirements of Section 9.

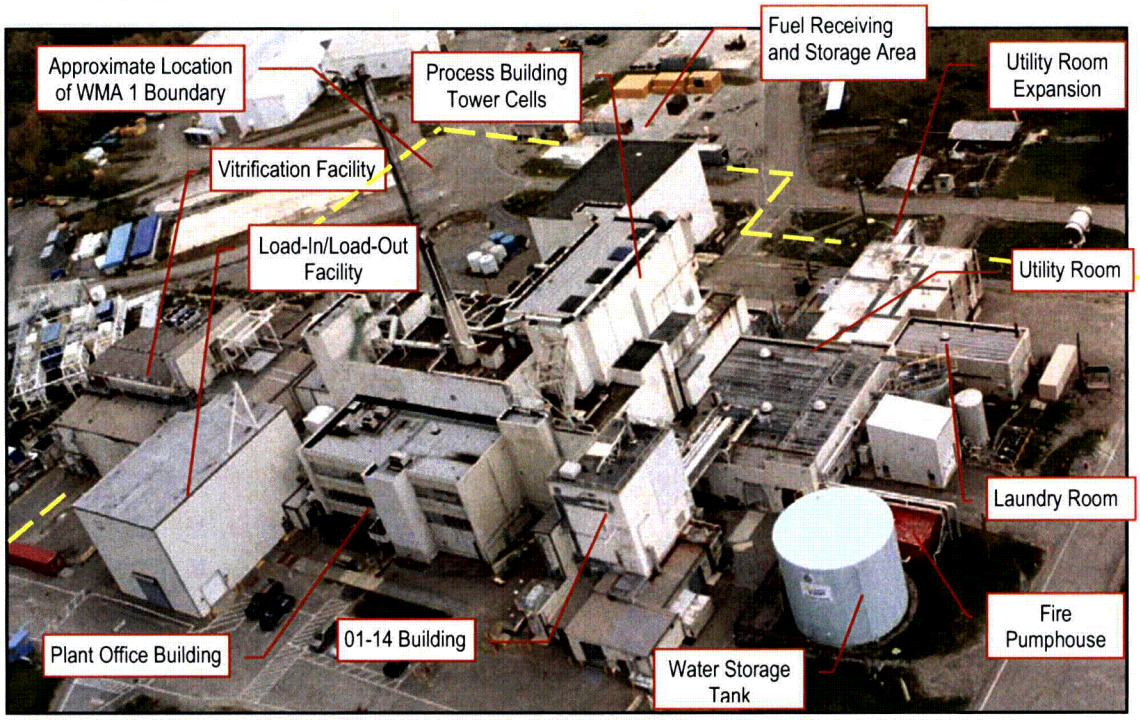
Final Status Surveys and Confirmatory Surveys

Phase 1 final status surveys would be accomplished in accordance with the Final Status Survey Plan as explained in Section 9 of this plan, which would also address confirmatory surveys to be performed by NRC or its contractor. When Phase 1 final status surveys are specified below, inherent in the survey process would be any additional remediation necessary to achieve the cleanup criteria and resurveys of areas remediated to ensure that the criteria were achieved.[3]

The Phase 1 final status surveys focus on areas to be made inaccessible by proposed decommissioning activities. Phase 1 final status surveys would be performed and confirmatory surveys coordinated with NRC or its contractor before these areas are made inaccessible. An example of such an area would be the lagoon excavation in WMA 2, which would be filled with earth only after the Phase 1 final status surveys and confirmatory surveys have been accomplished and the resulting data reviewed and accepted.

7.3 WMA 1 Proposed Decommissioning Activities

This section describes the proposed decommissioning activities in WMA 1, the Process Building and Vitrification Facility area, to be accomplished in Phase 1. Figure 7-1 shows WMA 1.



7-1. WMA 1 in 2007

[3] Section 9 uses the term *Phase 1 final status surveys* to describe these surveys of excavations, which would follow the final status survey protocols of the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000).

7.3.1 Characterizing Soil and Streambed Sediment

Soil and sediment in WMA 1 would be characterized for residual radioactivity in accordance with the Characterization Sample and Analysis Plan described in Section 9. The results of this effort would be used in planning the excavation work described below.

7.3.2 Relocating the Vitrified HLW Canisters

The 275 vitrified HLW canisters would be relocated to the new Canister Interim Storage Facility to permit demolition of the Process Building.

General Approach

The new Canister Interim Storage Facility (if the approach is selected by DOE) would be set up on the south plateau. The Equipment Decontamination Room would be modified to support handling the vitrified HLW canisters and the Load-In Facility would be converted to a Load-Out Facility. The vitrified HLW canisters would then be moved from the HLW Interim Storage Facility (the former Chemical Process Cell) and loaded into shielded dry storage canisters. Each storage canister would be placed in a shielded onsite transport cask and moved by truck to the new Canister Interim Storage Facility. The storage canisters would be maintained there in protective storage until they can be transported to the federal geologic repository.

This approach is among several approaches described in a preliminary conceptual engineering study (WVNSCO and Scientech 2000) which is currently under evaluation by DOE. If this approach is selected by DOE, detailed designs based on the preliminary conceptual designs would be developed. These designs would take into account the size of the canisters (two feet in diameter by 10 feet long), their weight (approximately 5,000 pounds each), their high radiation levels (about 1,750 to 7,500 R/h when they were moved into the HLW Interim Storage Facility in the former Chemical Process Cell), and the amounts of radioactivity they contain (an average of approximately 37,000 curies each in 2005) (WVNSCO 2006)[4]. The DOE is expected to make a decision on the preferred approach in the near future. A shielded dry interim storage system similar to those used at nuclear power plants for spent nuclear fuel is assumed for purposes of this plan.

Procurement of Interim Storage System for the Vitrified HLW Canisters

The interim storage system would include 69 shielded canisters and shielded modules made of reinforced concrete in which to store these shielded canisters. Each shielded canister would be capable of (1) holding four vitrified HLW canisters, (2) being loaded in a horizontal position, (3) being transported onsite within a shielded transport cask by truck, and (4) being transported within a shielded transport cask to the geologic repository by rail. The shielded canisters would be used for both onsite storage within the reinforced concrete storage modules and transport within a shielded transport cask.

The onsite shielded transport cask would be capable of (1) holding a single shielded canister, (2) loading and discharging the shielded canister in a horizontal position, and (3)

[4] Table 2-10 in Section 2 shows the activity estimate for a typical HLW canister.

being positioned on the onsite transport trailer so the open end can be partially inserted into a shielded area during both loading and discharge.

> NOTE
>
> The conceptual designs described below for the modifications to the Equipment Decontamination Room and the Load-In Facility and for the new Canister Interim Storage Facility for the vitrified HLW canisters depend on the characteristics described above. If DOE were to use an interim storage system with different characteristics, this plan would be revised to reflect the appropriate changes.

Modifications to the Equipment Decontamination Room

These modifications would involve setting up the Equipment Decontamination Room to remotely handle the vitrified HLW canisters and prepare them for insertion into the shielded canisters. The vitrified HLW canisters would be moved into the Equipment Decontamination Room from the HLW Interim Storage Facility using the existing transfer cart, which holds four canisters in a vertical position, or in a similar conveyance. New equipment would be installed to remove the canisters from the transfer cart, lower them into a horizontal position, and move them into a shielded transfer cell constructed in the Load-In/Load-Out Facility.

Conversion of the Load-In Facility

The shielded transfer cell would be constructed at the east wall of the facility between the shield door to the Equipment Decontamination Room and the air lock. This cell would be designed for operators to remotely perform the following activities: (1) verify canister dimensions as necessary, (2) weigh the canisters, (3) measure gamma radiation levels and removable surface radioactivity, (4) decontaminate the outside surfaces of the canisters, (5) load them in the shielded storage canisters, (6) weld the storage canister lids in place, and (7) load the shielded storage canisters into the onsite transport cask.

The transfer cell would be constructed of material such as steel plate to provide necessary radiation shielding and facilitate dismantlement after use. One or more viewing windows and remote manipulators would be provided, along with ventilation utilizing high-efficiency particulate air (HEPA) filters.

To avoid the need to remove the shielded transport cask from the trailer, the transfer cell would be designed so that trailer can be backed up to it to position the cask to receive a loaded shielded storage canister. With this arrangement, the trailer would be supported by jacks for stability, the open end of the onsite transport cask would be positioned within the outer part of the transfer cell to provide necessary radiation shielding, and the loaded shielded canister would be inserted into the cask and the cask shield plug installed. Figure 7-2 shows the conceptual arrangement.

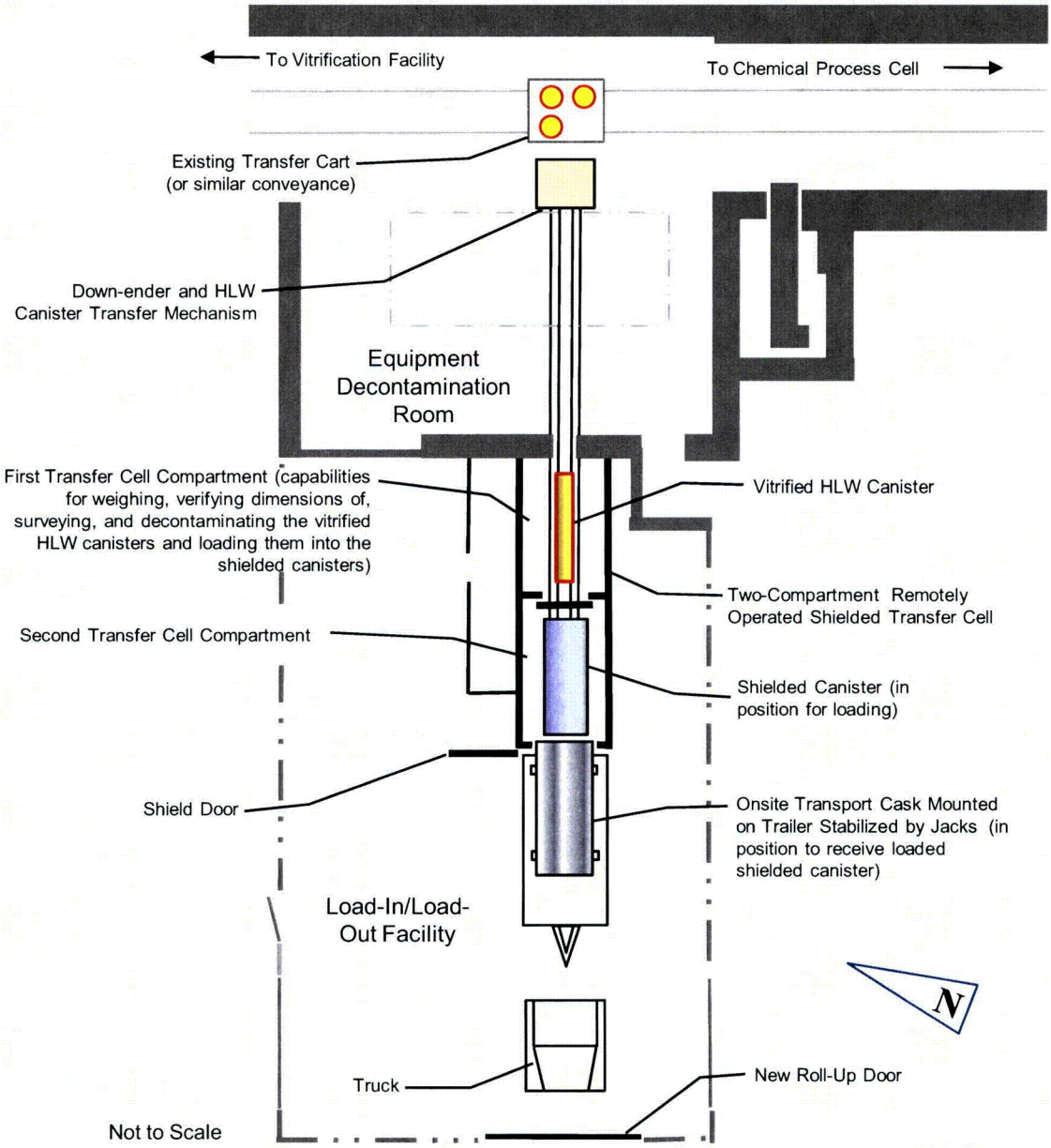


Figure 7-2. Conceptual Arrangement for Transferring Vitrified HLW Canisters

Construction of the New Canister Interim Storage Facility

The new Canister Interim Storage Facility would be constructed on the south plateau near the rail spur. The facility would consist of a reinforced concrete pad with reinforced concrete storage modules to provide radiation shielding and mechanical protection. The concrete pad would be sufficient in size and load capacity to accommodate reinforced concrete storage modules for the 69 loaded shielded canisters.

Figure 7-3 shows the conceptual design for a storage module, which is similar to the NUHOMS® standard horizontal storage module provided by AREVA (Transnuclear Incorporated) for dry storage of containerized spent nuclear fuel. (This design is provided as an example only and its inclusion here does not imply that DOE would necessarily select this interim storage system, which is among a variety of systems approved by NRC for general use that would be considered by DOE.)

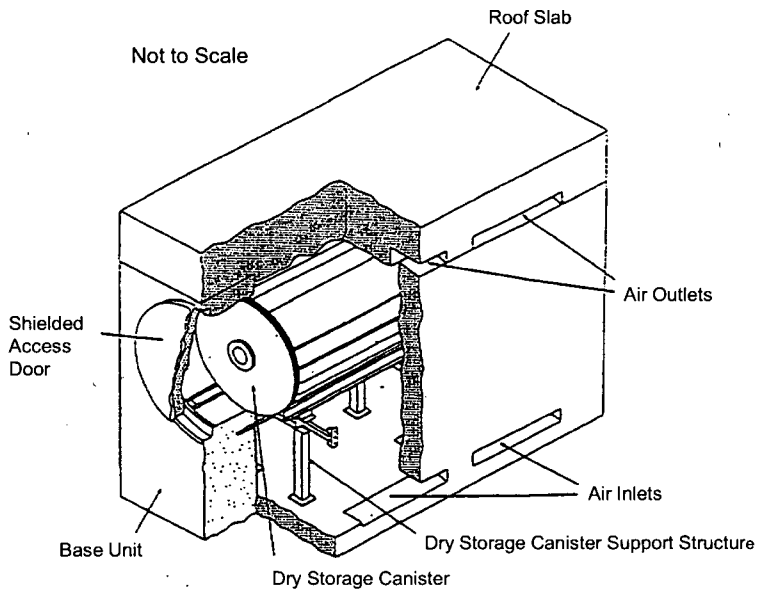


Figure 7-3. Storage Module Conceptual Design (from WVNSCO and Scientech 2000)

Appropriate fence(s), lighting, and remote monitoring equipment for security purposes would be provided. DOE would consider applicable NRC guidance in detailed design of the new Canister Interim Storage Facility, such as that found in NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1997). DOE would provide information on the detailed design of the facility to NRC and consult with NRC on preparation of the related safety analysis report.

Moving the Vitrified HLW Canisters to the New Canister Interim Storage Facility

A process such as the following would be used to transport the vitrified HLW canisters to the new Canister Interim Storage Facility:

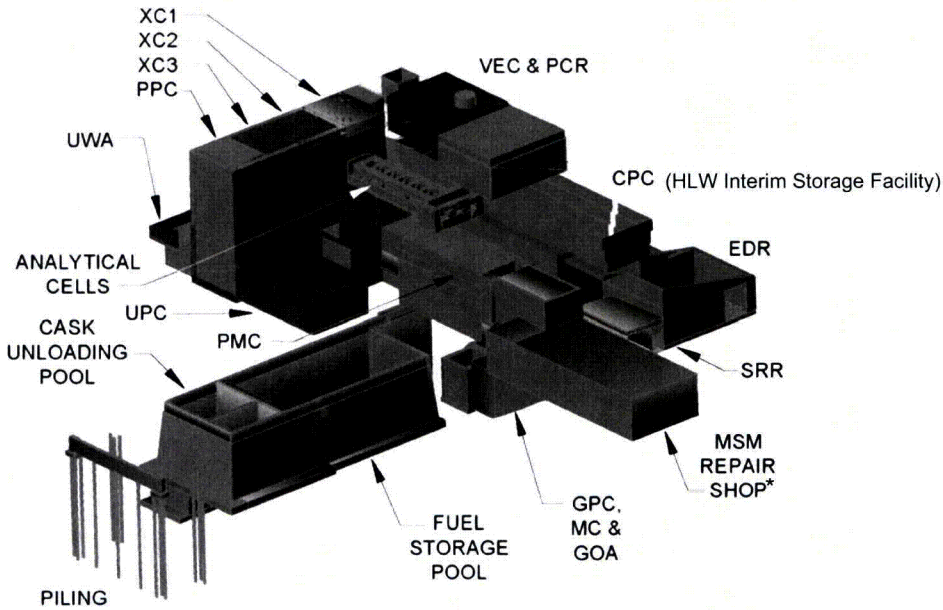
- Readiness reviews would be performed to ensure that all preparations for the move have been satisfactorily completed;
- The first shielded canister would be placed inside the shielded transfer cell;
- The onsite transport cask to receive the first shielded canister would be moved into the Load-In/Load-Out Facility and positioned next to the transfer cell;
- The first group of four vitrified HLW canisters would be moved into the Equipment Decontamination Room on the transfer cart or similar conveyance;

- The vitrified HLW canisters would be lifted from the cart one by one, lowered to a horizontal position, and moved into the transfer cell where appropriate measurements would be taken;
- After measurements and any necessary decontamination are completed, each of the four vitrified HLW canisters would be loaded into a shielded canister and the shielded canister would be loaded into the onsite transport cask; and
- The cask would be transported to the new Canister Interim Storage Facility where the shielded canister would be inserted into the designated reinforced concrete storage module and the module shielded access door installed.

This process would be repeated until all 275 vitrified HLW canisters have been relocated to the new Canister Interim Storage Facility.

7.3.3 Removing the Above-Grade Portion of the Process Building

As explained in Section 3, the Process Building is a complex structure comprised of various shielded cells, rooms, aisles, and supporting areas. It is approximately 270 feet long, 130 feet wide, and stands 79 feet above ground. Much of the structure is formed of heavily reinforced concrete. Figure 7-4 illustrates the Process Building and identifies key areas.



LEGEND:

CPC = Chemical Process Cell	PPC = Product Purification Cell
EDR = Equipment Decontamination Room	SRR = Scrap Removal Room
GOA = General Purpose Cell Operating Aisle	UPC = Uranium Product Cell
GPC = General Purpose Cell	UWA = Upper Warm Aisle
MC = Miniature Cell	VEC = Ventilation Exhaust Cell
MSM = Master-Slave Manipulator	XC1 = Extraction Cell 1
PCR = Process Chemical Room	XC2 = Extraction Cell 2
PMC = Process Mechanical Cell	XC3 = Extraction Cell 3

*The MSM Repair Shop and the Contact Size-Reduction Facility now located in this area will be removed before the decommissioning begins.

Figure 7-4. Process Building General Arrangement

Removal of the above-grade portion of the Process Building would be performed as specified below. The below-ground portion of the building would be removed as specified in Section 7.3.8. As indicated previously, this work would be performed in accordance with the Decommissioning Work Plan, which would provide more details on the activities described below.

Removing Equipment

Equipment would be removed during demolition of the building. Equipment to be removed from the areas that supported interim storage of the vitrified HLW canisters includes the canister storage racks and ventilation equipment in the HLW Interim Storage Facility, remote manipulators, the two cranes in the Chemical Crane Room, the vitrified HLW canister handling equipment in the Equipment Decontamination Room, and various pieces of ventilation equipment.

Other equipment remaining inside the Process Building after the interim end state is reached – such as the vessels in the Liquid Waste Cell, other vessels and equipment, the other cranes, and the master-slave manipulators – would also be removed. This equipment would be size reduced as necessary, characterized, packaged, and disposed of offsite. Size reduction would be accomplished either in the areas where the equipment is located or in another area set up for this purpose, such as the Vitrification Cell in the Vitrification Facility.

Removing Hazardous and Toxic Materials

Hazardous and toxic materials in the building would be removed to the extent practical before demolition. These materials would include:

- Any remaining temporary lead shielding and all permanently-installed lead shielding from areas such as the wall outside of the Off-Gas Blower Room and the shield doors and door frames in the Radiological Counting Room;
- The lead-glass viewing windows, whose frames contain lead;
- Any remaining bulk hazardous materials;
- Any electrical equipment known to contain polychlorinated biphenyls (PCBs); and
- Any remaining piping insulation known to contain asbestos.

These materials would be size reduced as necessary, characterized, packaged, and disposed of at an appropriate offsite disposal facility.

Completing Process Building Decontamination

Process Building areas known to have significant residual radioactivity would be evaluated and decontaminated as necessary to support unconfined demolition of the building, including the following areas used to support vitrified HLW canister storage:

- HLW Interim Storage Facility
- Chemical Crane Room
- Ventilation Exhaust Cell
- Head-End Ventilation Building

- Equipment Decontamination Room

The process used would involve activities such as the following:

- Removing remaining equipment from these areas, size reducing it as necessary, characterizing it, packaging it, and disposing of it at appropriate offsite disposal facilities;
- Performing radiological characterization surveys as specified in Section 9 to assess the extent of contamination on facility surfaces; and
- On the basis of characterization data results, verify that the process building can be demolished without exceeding National Emission Standards for Hazardous Air Pollutants (NESHAP) limits (40 CFR 61), making use of the CAP88-PC code (EPA 2007) and considering other sources of airborne radioactivity emissions during the calendar year in which the demolition would be accomplished.

Removing the Building to Grade Level

The Process Building would be demolished to grade level using conventional demolition methods such as those described in Section 7.11. Fixatives would be applied to building surfaces with significant radioactive contamination before this is accomplished to help avoid the need for radiological containment. The resulting debris would be sized reduced as necessary, packaged for disposal or managed as bulk waste, and disposed of offsite at an appropriate waste disposal facility.

Demolition of the building to grade level would be coordinated with demolition of other WMA 1 facilities and installation of the vertical hydraulic barrier wall for the WMA 1 excavation described in Section 7.3.8.

7.3.4 Removing the Above-Grade Portion of the Vitrification Facility

As explained in Section 3, this structural steel frame and sheet metal building houses the reinforced concrete Vitrification Cell, operating aisles, a control room, and other support areas. It is approximately 91 feet wide and 150 feet long. The peak of the roof stands approximately 50 feet high with the crane house extending another 26 feet above the roof. Figures 3-11 through 3-21 show the outside of the building and representative interior areas.

Removal of the above-grade portion of the Vitrification Facility would be performed as specified below. The below-grade portion of the building would be removed as specified in Section 7.3.8.

Preparing for Facility Removal

Preparations to remove the Vitrification Facility to grade would be similar to those for the Process Building. Installed equipment would be removed as necessary, along with the nine lead glass viewing windows in the Vitrification Cell and any remaining hazardous and toxic materials. Residual radioactivity levels inside the Vitrification Cell would be evaluated to ensure compliance with NESHAP emission limits during demolition. Fixatives would be applied to surfaces with significant radioactive contamination levels.

Removal of the Facility to Grade Level

After such preparations are completed, the Vitrification Facility would be removed to grade level using conventional demolition methods such as those described in Section 7.11. The thick reinforced concrete walls and roof structures would be segmented as necessary using a technique such as diamond wire cutting.

The resulting debris would be sized reduced as necessary, packaged for disposal or managed as bulk waste, and disposed of offsite at an appropriate waste disposal facility. The demolition work would be coordinated with demolition of the Process Building and the other WMA 1 facilities and with removal of piping in the HLW transfer trench in WMA 3, which connects to the north side of the building.

7.3.5 Removing the 01-14 Building and the Vitrification Off-Gas Line

As indicated in Section 3, the four-story 01-14 Building stands at the southwest corner of the Process Building. Figures 3-11 shows the building. The 10-inch vitrification off-gas line runs from the Vitrification Facility to the 01-14 Building in a 340 feet long subgrade concrete trench.

An approach such as the following would be used to remove this building to its floor slab and foundation:

- Performing characterization surveys;
- Removing any remaining equipment from the building, along with any hazardous or toxic materials and the lead-glass viewing window (the frame contains lead);
- Decontaminating the building structure and applying fixatives if necessary to allow demolition without the use of containment; and
- Demolishing the structure to its floor slab and foundation, as well as the cement silo and the entrance enclosure; and
- Characterizing the resulting debris, packaging it for disposal or managing it as bulk waste, and disposing of it at an offsite disposal facility.

The floor slab and foundation would remain in place temporarily and would be removed in connection with the excavation of the underground portions of the Process Building and Vitrification facility and the source area of the north plateau groundwater plume.

The off-gas line would be cut into segments, removed from the concrete trench, characterized, packaged for disposal, and disposed of at an offsite disposal facility. The trench itself would remain in place temporarily and would be removed in conjunction with removal of the WMA 1 subgrade structures and the plume source area.

7.3.6 Removing the Load-In/Load-Out Facility

As explained in Section 3, this 60 feet by 70 feet by 54 feet high steel building has a concrete floor. The process for removal of this building would be similar to the process used for the 01-14 Building and would include major steps such as the following:

- Performing characterization surveys;

- Removing equipment such as the vitrified HLW canister handling system, lead glass windows in the transfer cell, and the crane;
- Decontaminating the facility and applying fixatives to surfaces with significant radioactive contamination to facilitate demolition without containment;
- Demolishing the structure to its floor slab and foundation; and
- Characterizing the resulting debris, packaging it for disposal or managing it as bulk waste, and disposing of it at an offsite disposal facility.

The floor slab and foundation would remain in place temporarily and would be removed in conjunction with removal of the WMA 1 subgrade structures and the plume source area.

7.3.7 Removing Other WMA 1 Structures

The remaining WMA 1 structures would be removed to their concrete floor slabs and foundations, which would removed during excavation of the subgrade structures and the plume source area.

Utility Room and Utility Room Expansion

The Utility Room and the Utility Room Expansion are concrete block structures containing site utilities as explained in Section 3. The proposed decommissioning process for these facilities would include steps such as the following:

- Performing characterization surveys,
- Removing equipment from the building, along with any hazardous or toxic materials;
- Demolishing the building to its floor slab and foundation;
- Characterizing the resulting debris, managing it as bulk waste, and disposing of it at an offsite disposal facility.

Plant Office Building

The three-story concrete block Plant Office Building is shown in Figures 3-11 and 7-1. Decommissioning this structure would entail a process such as the following:

- Performing characterization surveys;
- Removing equipment from the building, along with any hazardous or toxic materials;
- Demolishing the building to its floor slab and foundation; and
- Characterizing the resulting debris, managing it as bulk waste, and disposing of it at an offsite disposal facility.

Fire Pump House

As of mid-2008, this 20 feet by 24 feet by 10 feet high steel building was not known to have been impacted by radioactivity. Decommissioning this structure would entail a process such as the following:

- Performing characterization surveys to confirm that the building is not impacted by radioactivity;
- Removing equipment only to the extent necessary to support building demolition; and
- Demolishing the building to its floor slab and foundation, disposing of the debris in an offsite landfill.

Water Storage Tank

This 475,800-gallon tank was not known to have been impacted by radioactivity as of late 2008. Decommissioning would entail emptying the tank, draining the water to the storm sewer system, and dismantling the tank.

Electrical Substation

This 34.5 kilovolt/480 volt transformer was not known to have been impacted by radioactivity as of late 2008. Decommissioning would entail de-energizing it and removing it, with the equipment containing PCBs managed in accordance with applicable State and U.S. Environmental Protection Agency requirements.

7.3.8 Removing the Underground Structures and Equipment and the Plume Source Area

Figure 7-5 shows the layout of the underground portions of the Process Building. The floor of the melter pit in the Vitrification facility, which is not shown on this figure, also extends approximately 14 feet below grade.

To facilitate removal of the underground structures of the Process Building and Vitrification Facility, along with the source area of the north plateau groundwater plume, an area larger than the footprint of both buildings would be excavated. Figure 7-6 shows this area.

Figure 7-6 provides information on Sr-90 contamination in groundwater that represents the upgradient portion of the north plateau groundwater plume based on measurements made in the 1998 investigation (Hemann and Steiner, 1999). This figure also shows the location of the main source of the plume, identified near the bottom of the drawing as "7P-240 Release," and key underground lines in the area.

Figure 7-7 shows a cross section view of the excavation. This figure also shows key soil contamination data from Geoprobe® samples collected in the 1998 investigation (Hemann and Steiner 1999).

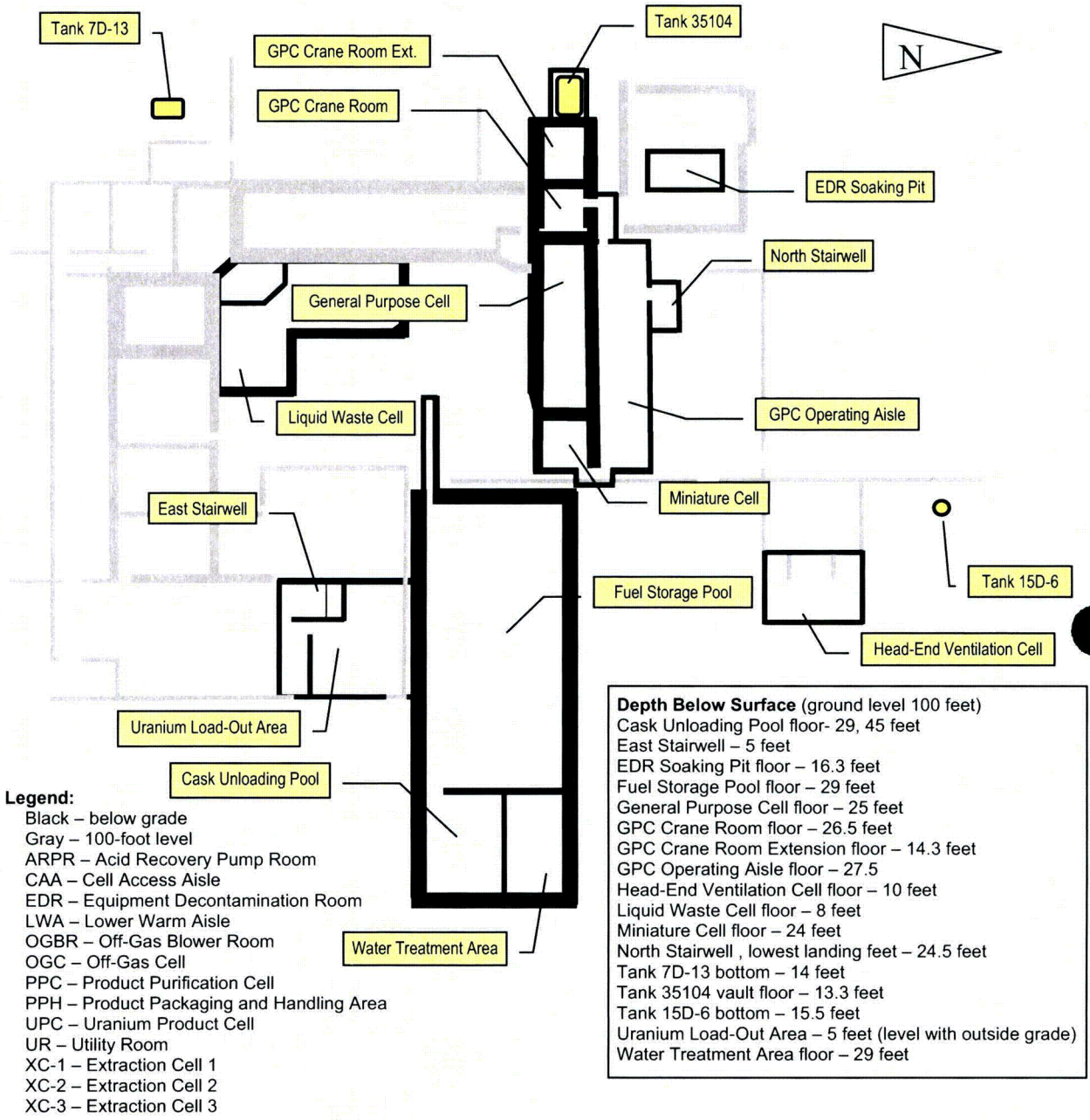
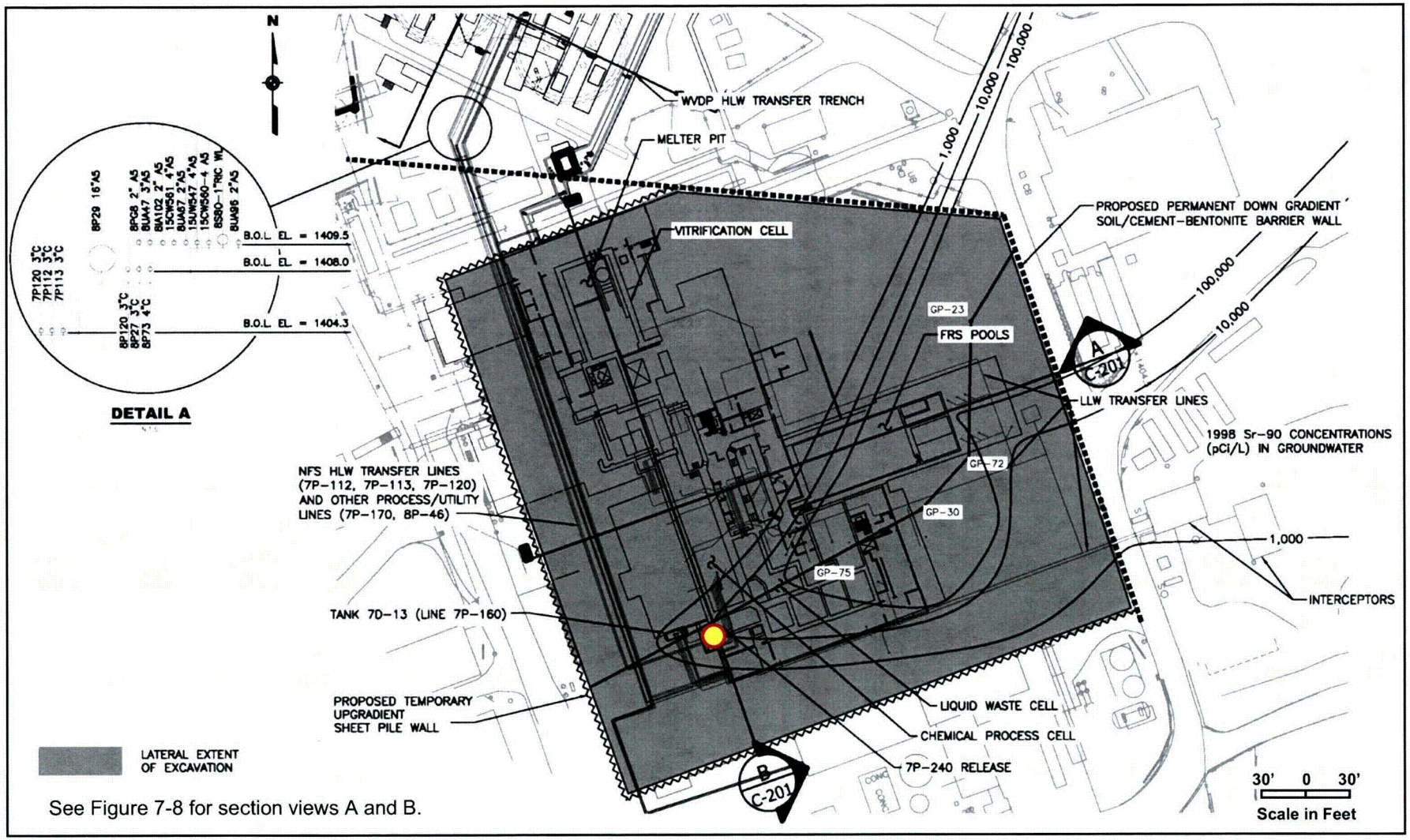


Figure 7-5. Layout of Process Building Underground Structures



See Figure 7-8 for section views A and B.

Figure 7-6. Conceptual Layout of WMA 1 Excavation

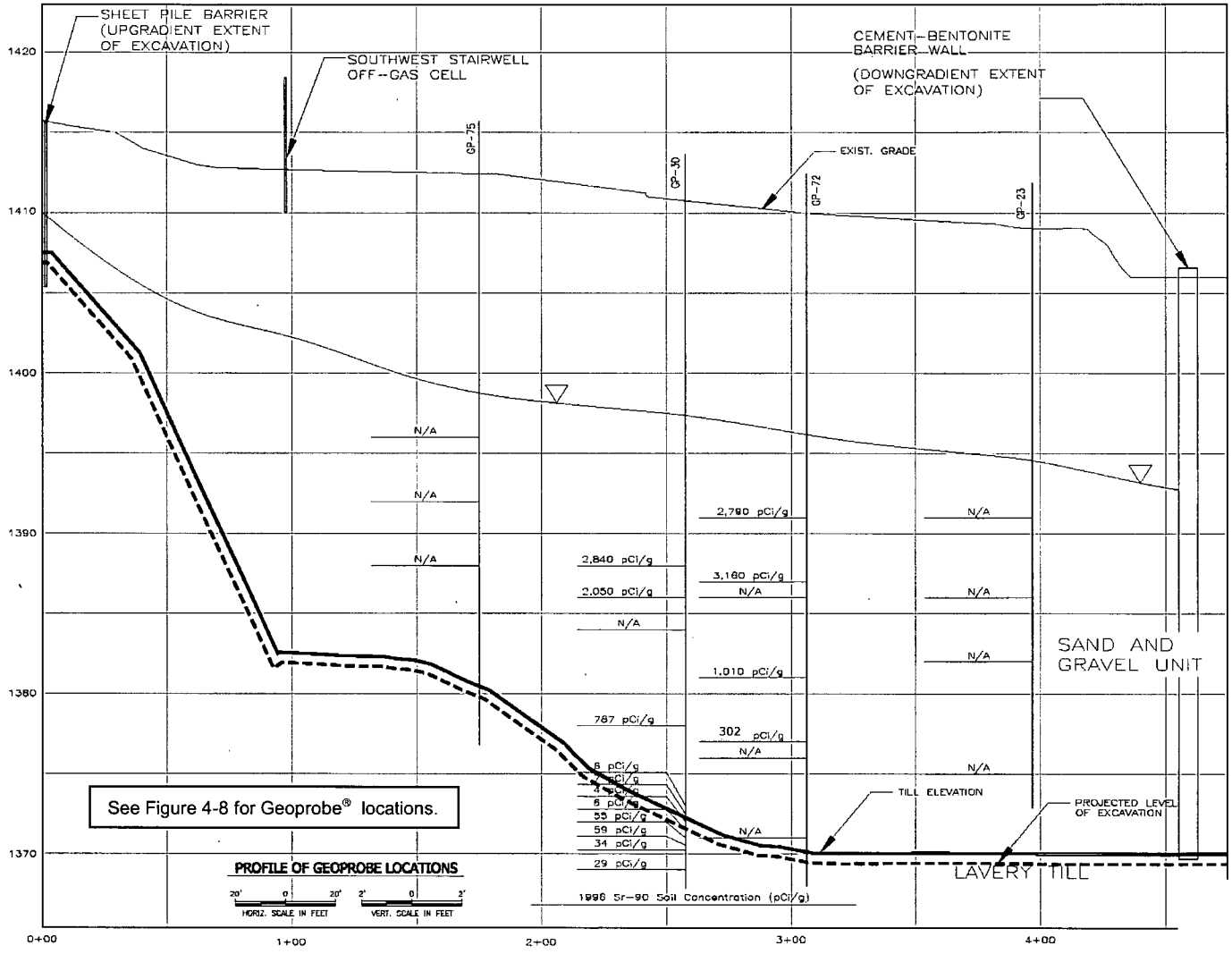
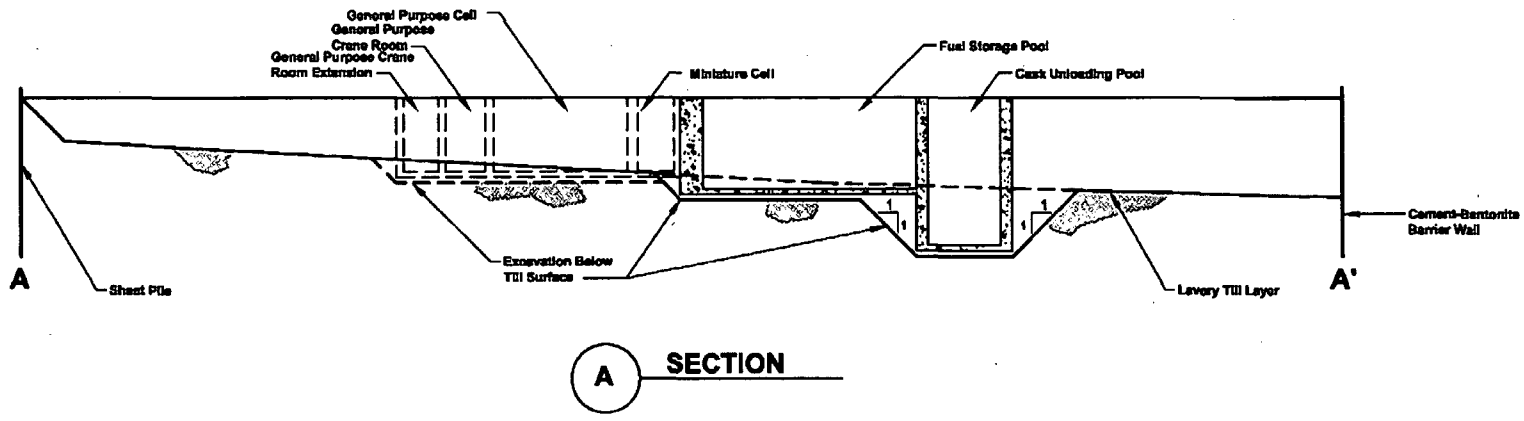


Figure 7-7. Conceptual WMA 1 Excavation Contour, With Selected Subsurface Soil Data



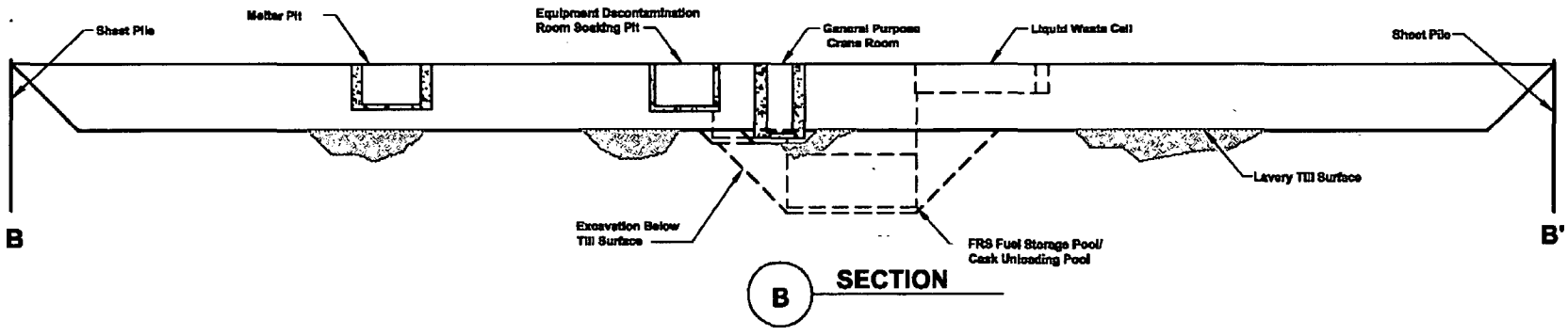


Figure 7-8. Excavation Cross Sections (From URS Drawing C-102)

Excavation Conceptual Design

The horizontal limits of the excavation would be based primarily on physical considerations, although consideration would also be given to analytical data on subsurface soil contamination at the planned excavation boundary acquired early during Phase 1. As can be seen in Figure 7-6, the western edge of the excavation would lie near the road in front of the Plant Office Building. The northern edge of the excavation would follow the walkway between the Vitrification Facility and the Waste Tank Farm. The eastern edge would follow the road between the Process Building area and the interceptors. The southern edge would correspond with a line running immediately south of the 01-14 Building, the Utility Room, and the Utility Room Expansion. The footprint of the excavation would comprise approximately three acres.

The depth of the excavation would vary depending on the subsurface structures. Figure 7-8 shows two representative cross sections (which are identified on Figure 7-6).

Hydraulic Barrier Wall Installation

To control groundwater, a vertical hydraulic barrier would be installed around the area to be excavated as shown in Figure 7-6 and Figure 7-7. The upgradient portion would be built of sheet pile. The downgradient portion would consist of a soil-cement-bentonite slurry wall. Both would extend approximately two feet into the Lavery till and the slurry wall would remain in place after the excavation is backfilled.

Before the hydraulic barrier wall is installed, underground lines in its footprint that carried radioactive liquid would be located. Sections of these lines in the area where the barrier walls would be constructed would be removed in a controlled manner to avoid unnecessary release of contamination. During this process, characterization measurements would be taken in the end of each line that would remain in place outside of the excavation and the line capped.

The total length of the slurry wall would be approximately 750 feet, with approximately 525 feet of this length directly adjacent to the WMA 1 area to be excavated. The 525-foot portion of the slurry wall adjacent to the area to be excavated would be sufficiently wide to provide the stability necessary to permit excavation up to the base of the wall, with the remainder a more typical two foot width. The extra width of the main portion of the slurry wall and the inclusion of cement in the mixture would provide the stability necessary to accommodate the nearby excavation.[5]

The sheet pile section of the hydraulic barrier wall would be installed using a conventional pile driver. Construction of the soil-cement-bentonite slurry wall would involve activities such as the following:

- Making preparations to handle the soil to be excavated, with characterization data, including data collected during the excavation process, used to determine the portion of the soil that is radioactively contaminated;
- Using a hydraulic excavator to dig the trench for installation of the slurry wall;
- Preparing the slurry and backfill mixtures in earthen containment berms that would be constructed near the slurry wall;
- Keeping the trench filled with slurry during the excavation process to help support the trench walls during the excavation;

[5] Consideration of industry experience in use of slurry walls at the boundaries of deep excavations indicates that the barrier planned for the WMA 1 excavation would perform as planned in controlling groundwater intrusion and supporting the excavation design. The extra thickness would accommodate some excavation into the upper portion of the barrier wall with sufficient thickness remaining to ensure satisfactory performance.

- Backfilling the trench with a mixture of clean soil, cement, and bentonite to displace the slurry, which would then be used to continue the trench excavation;
- Collecting the radioactively contaminated removed soil in lift liners, adding absorbent to the saturated soil, and transporting it offsite for disposal as low specific activity waste; and
- Disposing of the uncontaminated soil at an appropriate offsite disposal facility..

The resulting slurry wall would have a maximum in-place saturated hydraulic conductivity of 6.0E-06 cm/s. It would extend to within about three feet of grade and be topped with uncontaminated soil.

Preparations for Removal of Contaminated Soil and Groundwater

Removal of contaminated soil and groundwater is addressed first because of the issues in dealing with highly contaminated soil expected beneath the Process Building. However, removal of the underground structures and equipment would be coordinated with soil removal since the north plateau plume source area lies beneath the Process Building. Detailed planning for the excavation would take into account available information on radioactivity in the soil and groundwater as summarized in Section 4.2 and the results of the soil characterization program. The depth of the water table in the area – typically about 10 feet below the surface – would also be taken into account.

Preparations, in addition to installation of the hydraulic barrier wall, would include installation of extraction wells to dewater the excavation. The removed water would be sent to the Low-Level Waste Treatment Facility for treatment prior to discharge through a State Pollutant Discharge Elimination System (SPDES)-permitted outfall or, as an alternative, a portable wastewater treatment system using ion exchangers and filters provided for this purpose. Preparations would also include making provisions for appropriate radiological controls, such as design and erection of a pre-engineered confinement structure over the north plateau plume source area or over the entire excavation to provide for weather protection and airborne radioactivity control.

Removal of Contaminated Soil and Groundwater

Before excavation begins, the hydraulic barrier wall would be installed, the sheet piles installed, the dewatering wells installed and placed in operation, and appropriate radiological controls established. The excavation process would be accomplished in two phases using conventional excavation equipment.

The first phase would involve removal of soil in the vadose zone, except for the soil in the north plateau plume source area and soil immediately downgradient of this area. Excavation of soil in the saturated zone would begin after the dewatering wells have removed groundwater in the confined area to the extent practical. The groundwater would be treated as discussed previously and discharged to Erdman Brook through a SPDES-permitted outfall after confirmation that radioactivity concentrations are acceptably low. The groundwater extraction wells would be removed during the excavation.

Soil would be excavated to a depth of at least one foot into the Lavery till, with the extent of additional soil removal determined by the use of the cleanup goals specified in the Section 5. Remedial action surveys would be performed during the course of the work and soil on the bottom and sides of the excavation with radioactivity concentrations exceeding the cleanup goals would be removed and disposed of offsite as radioactive waste.[6] Contaminated soil with radioactivity

[6] It is unlikely that the sides of the excavation that are not hydraulically downgradient will be contaminated. In any case, the extent of soil remediation on the sides of the excavation would be limited by the excavation boundaries.

concentrations below cleanup goals would be removed where practical, consistent with the ALARA process as described in Section 6 and Section 7.2.2. Soil would be excavated as close to the hydraulic barrier wall as practical. The other sides of the excavation would have a slope of approximately 45 degrees as indicated on Figure 7-8.

Removal of Underground Structures, Floor Slabs, and Foundations

The demolition of below-grade cells and structures shown in Figure 7-5 would be coordinated with the removal of the three underground tanks, the underground piping, and contaminated soil associated with the source area of the north plateau groundwater plume. All remaining concrete floor slabs and foundations in the area, including those outside of the excavation, would be removed early in the process to facilitate the excavation work. After soil is excavated to expose their structures, the below-grade cells would be demolished with conventional demolition equipment such as diamond wire saws.

The foundation pilings supporting the Process Building would be cut off at the bottom of the excavation or slightly below the bottom and the cut-off portion removed as well. All demolition debris would be characterized and disposed of offsite. In connection with this work, samples of soil would be collected around representative pilings, including at points several feet below the surface. Analytical data from the samples would be used to evaluate the potential for preferential flow paths around the pilings and be considered in the Phase 1 final status surveys described in Section 9.

Removal of Underground Tanks and Piping

The three underground tanks and radioactively contaminated underground piping within the excavated area would be removed and disposed of as radioactive waste. Planning for underground line removal would take into account one line of particular interest: waste transfer line 7P120-3, which is expected to contain high levels of residual radioactivity as described in Section 4.1. The concrete off-gas trench would be removed. (Removal of the piping in the trench was provided for in Section 7.3.5.)

Duriron wastewater piping under the Process Building and east of the building, which contains lead in the piping joints, would be cut near the joints, with pieces containing the joints being disposed of as mixed waste. The remainder of this piping would be disposed of as LLW.

This process would apply to radioactive lines and also to nonradioactive sanitary lines and utility lines, which would be removed during the course of the work because it is unlikely that it would be practicable to leave them in place. Underground piping outside of the excavation would remain in place.

7.3.9 Site Restoration

Once the below-grade structures of the Process Building and Vitrification Facility, the three wastewater tanks, the underground piping, and the remaining concrete floor slabs and foundations have been removed, and the underlying contaminated soils associated with the source area of the north plateau groundwater plume have been removed, a Phase 1 final status survey would be performed in the excavation bottom and sides as specified in Section 9 to verify that residual radioactivity levels are below the cleanup goals. Special attention would be paid to areas around the remaining sections of the Process Building support pilings. Surveys performed around the support pilings would extend to sufficient depth to evaluate the extent, if any, of the downward

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That is, any soil found to exceed the cleanup goals would be removed only within the confines of the downgradient hydraulic barrier wall and the sheet piles installed on the other sides of the excavation.

migration of contamination along the pilings. Arrangements would also be made for an independent verification survey to be performed on behalf of the regulatory agencies.

After the verification survey is completed and regulatory approval is received, the area would be backfilled with uncontaminated earth and graded as necessary to restore to it a near natural appearance. The backfill material would be obtained from similar offsite geologic deposits. The properties of this material (especially the texture, hydraulic conductivity, and distribution coefficients) would be similar to those of the sand and gravel layer on the project premises as described in Section 3.

A French drain would be emplaced during backfilling of the excavation to prevent groundwater from mounding near the hydraulic barrier wall. Water from the French drain would be allowed to passively discharge into a small tributary of Erdman Brook. More detail on the French drain design appears in Appendix D.

The sheet pilings installed on the upgradient sides of the excavation would be removed after the excavation is backfilled. The piling and any confinement structure used would be disposed of offsite at appropriate waste disposal facilities.

Appendix D addresses monitoring and maintenance of the WMA 1 area between the completion of Phase 1 of the proposed decommissioning and the beginning of Phase 2. Appendix D also provides information on expected changes to the groundwater flow field that would occur with completion of the Phase 1 proposed decommissioning activities in WMA 1.

7.4 WMA 2 Proposed Decommissioning Activities

This section addresses proposed decommissioning of the Low-Level Waste Treatment Facility area, which is shown in Figure 7-9.

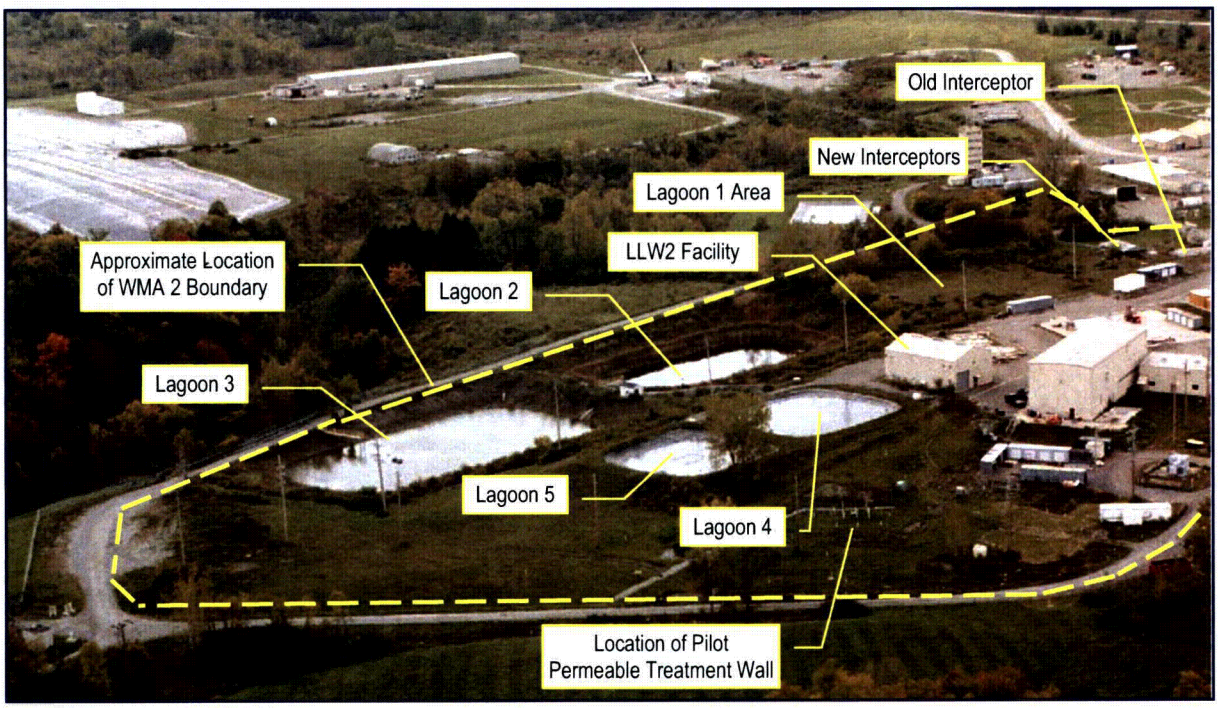

Figure 7-9. WMA 2 in 2007

The sequence for the Phase 1 proposed decommissioning activities in WMA 2 would be developed during detailed planning. The LLW2 facility would be kept in service until it is no longer needed to treat the water in the lagoons and contaminated groundwater removed from the excavation before it is discharged though an SPDES-permitted outfall into Erdman Brook.

Demolition debris, soil, sediment, and other material removed during this work would be characterized for waste management purposes and disposed of at appropriate offsite waste disposal facilities. Absorbents would be added as necessary to containers of wet contaminated soil to absorb moisture.

7.4.1 Characterizing Soil and Sediment

Soil and sediment in WMA 2 would be characterized for residual radioactivity in accordance with the Characterization Sample and Analysis Plan described in Section 9. The results of this effort would be used in planning the excavation work described below. (This characterization would not include subsurface soil in areas impacted by the north plateau groundwater plume except in the portion of WMA 2 where the excavation would be located.)

7.4.2 Removing Structures

The structures in WMA 2 would be removed with appropriate radiological controls, along with the remaining concrete floor slabs and foundations. Removal of the Neutralization Pit, the Old Interceptor, the New Interceptors, and Lagoons 1, 2, and 3 would be coordinated with digging the WMA 2 excavation addressed in Section 7.4.3, which would encompass the area of these facilities as well as the Solvent Dike. During this process, characterization measurements would be taken in the end of each underground line that would remain in place and the line capped.

LLW2 Facility

This metal-sided building with skid-mounted process equipment and a 900-gallon stainless steel lined sump is expected to contain low levels of radioactive contamination. Its demolition would involve activities such as the following:

- Removing the process equipment;
- Removing any water in the sump, stabilizing it in cement for disposal as LLW;
- Demolishing the structure to grade level;
- Removing the floor slab and foundation and the sump liner;
- Removing soil under the floor slab and foundation to a depth of approximately two feet[7];
- Performing Phase 1 final status surveys in the area excavated for these purposes;
- Making arrangements for any independent confirmatory surveys to be performed in the excavated area; and
- Filling in the excavated area with clean earthen backfill.

[7] The two-foot prescriptive excavation depth was selected to avoid unnecessary excavation into soil contaminated by the north plateau groundwater plume during Phase 1 of the decommissioning. As noted previously, the plume would be among the sources considered in Phase 2 of the decommissioning.

Neutralization Pit

The Neutralization Pit would be removed using a process similar to the following:

- Removing any residual water, treating it for disposal via an SPDES-permitted outfall or solidifying it for disposal as LLW; and
- Removing the liner, concrete walls, and floor of the pit.

The underground wastewater lines in the area of the Neutralization Pit would be removed in connection with digging the WMA 2 excavation described in Section 7.4.3. Phase 1 final status surveys, independent confirmatory surveys, and filling the excavation are also addressed in Section 7.4.3.

Old Interceptor

The Old Interceptor would be demolished using a process similar to that used for the Neutralization Pit, with additional radiological controls appropriate to the larger amount of residual radioactivity it contains.

New Interceptors

The New Interceptors would be demolished using a process similar to that used for the Neutralization Pit.

Concrete Floor Slabs and Foundations

The concrete floor slabs of the O2 Building, Test and Storage Building, Vitrification Test Facility, Maintenance Shop, Maintenance Storage Area, and the Vehicle Maintenance Shop would be removed and the building footprints excavated approximately two feet below grade. Phase 1 final status surveys would be performed in the excavated areas, and arrangements made for an independent verification survey if desired by the regulators. After the surveys have been completed, the excavations would be filled with earth.

7.4.3 Decommissioning the Lagoons

Decommissioning of Lagoons 1, 2, and 3 would involve constructing a vertical hydraulic barrier on the northwest side of the lagoons and digging a single large excavation. Lagoons 4 and 5 would be removed separately. Figure 7-10 shows the conceptual plan view of the large excavation and the location of the hydraulic barrier wall. Figure 7-11 shows the conceptual cross section.

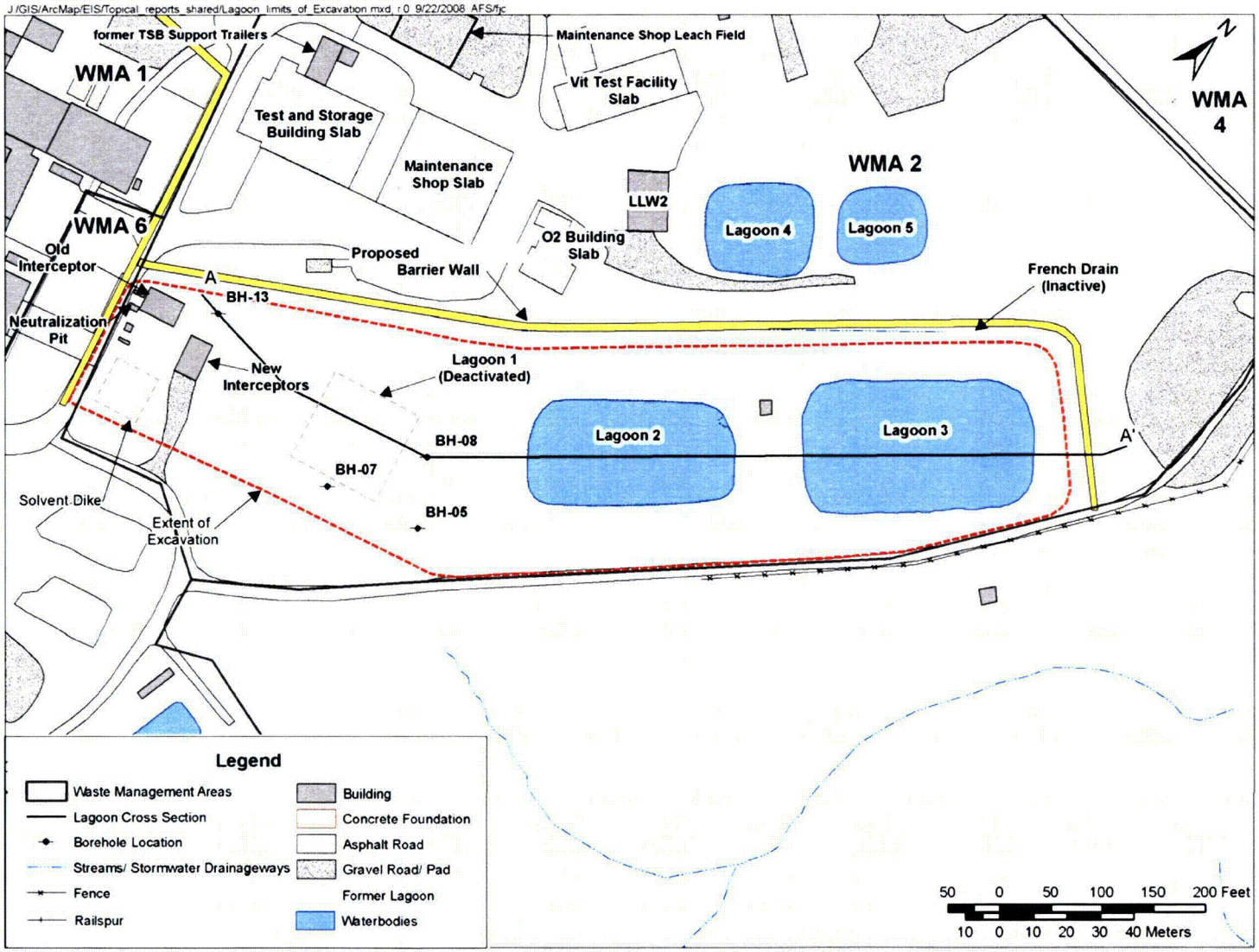

Figure 7-10. Conceptual Arrangement of WMA 2 Excavation, Plan View

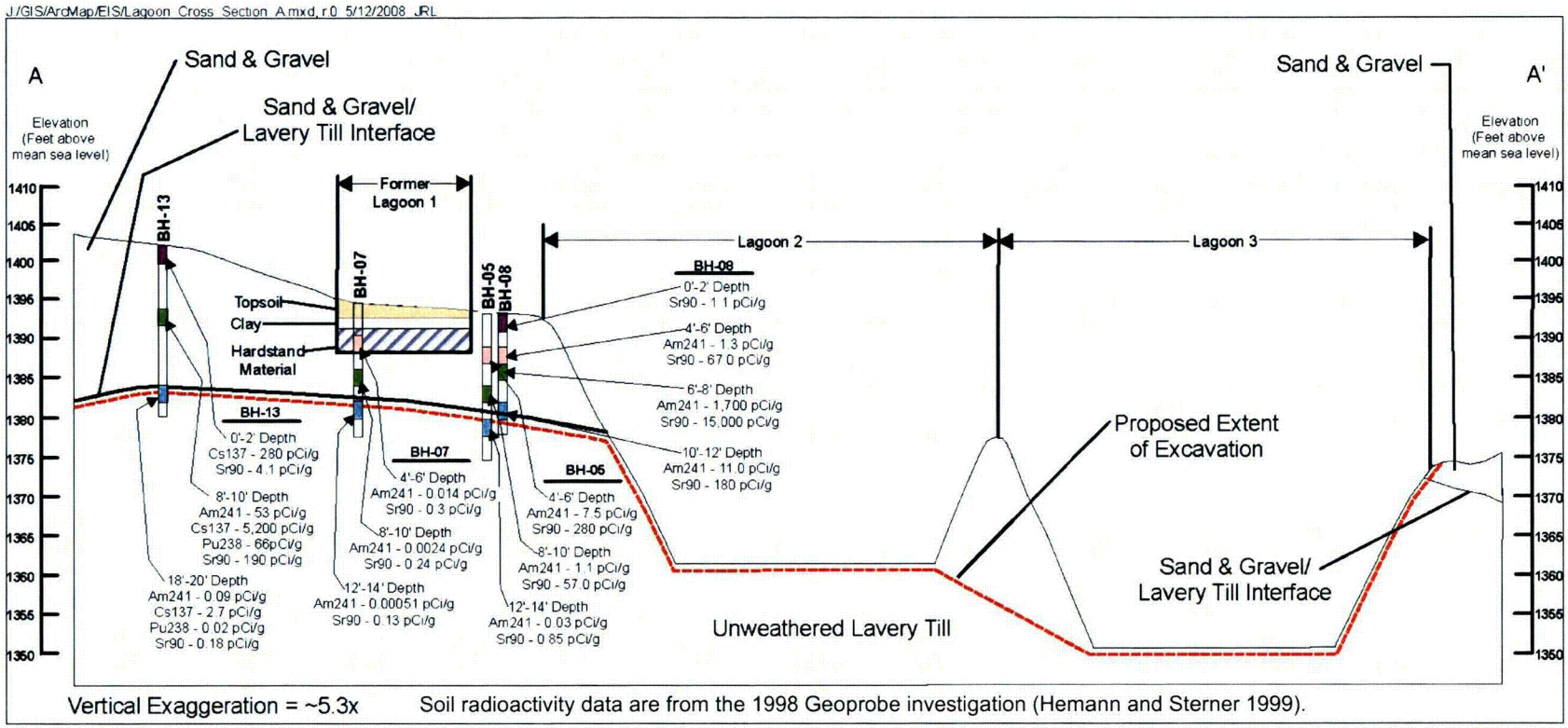

Figure 7-11. Conceptual Arrangement of WMA 2 Excavation, Cross Section

Hydraulic Barrier Wall Installation

To isolate the area of WMA 2 to be excavated from the north plateau groundwater plume, a vertical hydraulic barrier wall would be installed as shown in Figure 7-10. This hydraulic barrier would consist of a soil-cement-bentonite barrier wall that would extend approximately two feet into the Lavery till. It would remain in place after the excavation is backfilled.

Before the hydraulic barrier wall is installed, underground lines in its footprint that carried radioactive liquid would be located. Sections of these lines in the area where the wall would be constructed would be removed in a controlled manner to avoid unnecessary release of contamination. During this process, characterization measurements would be taken in the end of each line that would remain in place and the line capped.

The total length of the barrier wall would be approximate 1100 feet. It would be sufficiently wide to provide the stability necessary to permit excavation up to the base of the wall. This barrier wall would connect with the WMA 1 hydraulic barrier wall as shown in Figure 7-10. It would be constructed in the same manner as the WMA 1 slurry wall and have an in-place maximum saturated hydraulic conductivity of approximately 6E-06 cm/s. It would extend to within about three feet of grade and be topped with excavated material. Sheet piles on the southeastern side of the excavation are not expected to be necessary to control groundwater, except possibly in the Lagoon 1 area as indicated below.

Preparations for Removal of Contaminated Lagoon Sediment and Soil

Detailed planning for the excavation would take into account available information on radioactivity in the lagoon sediment, soil, and groundwater as summarized in Section 4, along with the results of the soil characterization program. The depth of the water table in the area – typically about seven feet below the surface – would also be taken into account.

Preparations, in addition to installation of the hydraulic barrier wall, would include provisions for appropriate radiological controls to minimize airborne radioactivity releases during the excavation work, such as a single-span confinement structure for the Lagoon 1 area.

Removal of Contaminated Soil and Underground Wastewater Lines

Removal of Lagoons 1, 2, and 3 and the facilities within the area to be excavated as described below would be coordinated with removal of soil in other parts of the excavation. Before excavation begins, the hydraulic barrier wall would be installed. The excavation process would be accomplished in two phases using conventional excavation equipment.

The first phase would involve removal of soil in the vadose zone. It is expected that approximately one-half of the total amount of soil to be removed would be unsaturated.

The second phase would involve removal of soil in the saturated zone. Wastewater piping within the excavated area would be removed. Groundwater accumulating in the excavation would be pumped out, treated using a portable treatment system containing ion exchangers and filters, and discharged to Erdman Brook through an SPDES-permitted outfall.

Figure 7-11 shows the planned depth of excavation. The excavation would extend at least one foot into the Lavery till and one foot below the sediment in the bottoms of Lagoons 2 and 3 as indicated in the figure, with the amount of additional soil removal determined by the use of cleanup

goals specified in Section 5.[8] Remedial action surveys would be performed during the course of the work and soil on the bottom and sides of the excavation with radioactivity concentrations exceeding the cleanup goals would be removed. Soil with radioactivity concentration exceeding cleanup goals would be excavated as close to the hydraulic barrier as practicable. However, the lateral extent of the remediation would not exceed the boundary shown in Figure 7-10 during Phase 1.

Lagoon 1

Lagoon 1 during operation was approximately 82 feet by 82 feet by five feet deep. It now contains radioactively contaminated sediment, asphalt, soil and vegetation and is capped with clay and covered with topsoil.

Sheet piles would be installed around Lagoon 1 as necessary to control groundwater flow into the area to be excavated. The excavation would be dug to encompass an area roughly 100 feet by 100 feet and extend approximately two feet into the Lavery till, with a total depth of approximately 14 feet. The clay cap, hardstand waste, and contaminated sand and gravel underlying Lagoon 1 would be excavated, along with the underlying sediment. The excavation would extend at least one foot into the underlying Lavery till, with the cleanup goals specified in Section 5 being used to determine the need for any additional soil removal. Phase 1 final status surveys would be performed in the excavated area and arrangements would be made for independent confirmatory surveys before the excavation is filled in, as described below. (These surveys would be performed when the entire WMA 2 excavation has been completed.)

Lagoon 2

As indicated previously, Lagoon 2 is an unlined basin approximately 280 feet long, 195 feet wide, and 17 feet deep with a significant amount of radioactive contamination in the bottom sediment.

Water in the lagoon would be treated in the LLW2 Facility and discharged though an SPDES-permitted outfall into Erdman Brook. Auxiliary equipment such as piping in the pump shed and the shed itself would be removed. Contaminated lagoon sediment would be removed along with at least one foot of underlying Lavery till, with the cleanup goals specified in Section 5 being used to determine the extent of any additional soil removal. As with Lagoon 1, Phase 1 final status surveys would be performed in the excavated area and arrangements would be made for independent confirmatory surveys before the excavation is filled in, as described below.

Lagoon 3

As indicated previously, Lagoon 3 is an unlined basin similar in design to Lagoon 2, but 24 feet deep rather than 17 feet deep, with low level radioactivity in the sediment. It would be decommissioned using the same process as Lagoon 2.

Solvent Dike

Radioactively contaminated soil in the Solvent Dike area would be removed before the large excavation is dug. This sequence would facilitate management of any unexpected wastes that might be present.

[8] Note that Figure 7-11 shows the interface between the sand and gravel unit and the Lavery till in the area of Lagoon 1; Lagoon 2 and Lagoon 3 extend well into the Lavery till.

Other Parts of the Excavation

Removal of soil in between the facilities in the area to be excavated would be coordinated with excavation of the facilities themselves so that the entire area is excavated as indicated in Figures 7-10 and 7-11, with the excavation extending at least one foot into the Lavery till. Any sheet piles installed to facilitate excavation of Lagoon 1 would be removed after that lagoon is excavated,

Surveying and Backfilling the Excavation

Phase 1 final status surveys would be performed in the bottom and sides of the excavation to verify that the cleanup goals have been achieved and arrangements made for independent confirmatory surveys. After these surveys are completed and any issues resolved, the excavation would be filled with uncontaminated earthen backfill and the surface leveled with the surrounding area. The backfill material would be obtained from similar offsite geologic deposits. The properties of this material would be similar to the backfill used in the WMA 1 excavation.

Lagoons 4 and 5

Lagoons 4 and 5 are similar above-grade lagoons that were constructed in 1971 from till material. Lagoon 4 has a capacity of 204,000 gallons and Lagoon 5 has a capacity of 166,000 gallons. Both are now lined with concrete grout and geomembranes. Low levels of radioactive contamination are expected in sediment both above and below the lagoon liners.

The geomembranes and the concrete and clay liners in Lagoons 4 and 5 would be removed and underlying soil excavated to a maximum depth of two feet. After completion of this work, a Phase 1 final status survey would be performed in the area, and arrangements made for any independent verification surveys described by the regulators. The excavated area would be filled with clean earth after the surveys.

Appendix D addresses monitoring and maintenance of the WMA 2 area between the completion of Phase 1 of the proposed decommissioning and the beginning of Phase 2. Appendix D also provides information on expected changes to the groundwater flow field that would occur with completion of the Phase 1 proposed decommissioning activities in WMA 2.

7.5 WMA 3 Proposed Decommissioning Activities

This section addresses proposed decommissioning activities in the Waste Tank Farm area, which include removal of two structures, piping and equipment in the HLW transfer trench, and the mobilization and transfer pumps in the underground waste tanks, along with requirements for continuing maintenance of the underground waste tanks. WMA 3 is shown in Figure 3-29.

7.5.1 Removing Structures

The Con-Ed Building and the Equipment Shelter and Condensers would be removed with appropriate radiological controls and the resulting demolition debris characterized and disposed of at an appropriate offsite disposal facility.

Con-Ed Building

This small concrete block building located over the Tank 8D-3/8D-4 vault would be removed by removing the installed equipment, demolishing the structure to grade level, and performing Phase 1 final status surveys in the area of the building footprint.

Equipment Shelter

This concrete-block building – which is approximately 40 feet long, 18 feet wide, and 12 feet high – would be removed using a process similar to that used for the Con-Ed Building. The condensers would also be removed and disposed of at an offsite waste disposal facility. Soil in the footprints of the building and condenser foundations would be removed to a maximum depth of two feet below grade. Phase 1 final status surveys would be performed in the excavated areas and arrangements made for any independent confirmatory surveys to be performed. Afterwards, the excavated areas would be filled with clean earthen backfill.

7.5.2 Removing Waste Tank Pumps and Pump Support Structures

As noted previously, Tank 8D-1 contains five HLW mobilization pumps and Tank 8D-2 contains four of these centrifugal pumps. Tanks 8D-1 and 8D-2 also each contain a HLW transfer pump. Each pump has an overall length of more than 50 feet and contains significant amounts of radioactive contamination. Figure 3-32 shows both pump designs. Figure 3-34 shows a typical pump pit. As noted in Section 3, Tanks 8D-1 and 8D-2 each contain another suction pump and Tanks 8D-3 and 8D-4 are each expected to contain a small submersible pump.

The HLW mobilization and transfer pumps have been impacted by liquid HLW. DOE would follow applicable provisions of DOE Manual 435.1-1, *Radioactive Waste Management Manual*, concerning these pumps.

The HLW mobilization pumps, transfer pumps, and suction pumps would be removed and disposed of offsite using a process such as the following:

- Preparations would be made for handling the removed pumps in a controlled manner consistent with their expected high radiation and contamination levels and the expected waste classification of different parts of the pump assembly;
- Each pump would be removed using appropriate radiological controls, decontaminated as necessary, cut into sections during removal, and packaged for disposal;
- The pump support structures would be removed in conjunction with removal of the pumps; and
- The pump segments and the support structures would be disposed of offsite at appropriate waste disposal facilities.

The submersible pumps in Tanks 8D-3 and 8D-4 would also be removed using appropriate radiological controls and disposed of offsite as radioactive waste.

7.5.3 Removing HLW Transfer Trench Piping and Equipment

As noted previously, the HLW transfer trench, which is shown in Figure 3-33, is approximately 500 feet long, extending from the Tank 8D-3/8D-4 vault to the Vitrification Facility. The trench contains lines comprising approximately 3000 linear feet of double-walled stainless steel pipe. Each pump pit contains a waste transfer pump (which would be removed as specified in Section 7.5.2), discharge piping, and flow monitoring equipment; Pump Pit 8Q-2 also contains grinding equipment that was used to size reduce contaminated zeolite. The inner piping, valves, and the other equipment are expected to contain significant radioactive contamination.

The piping that was actually used and some of the other equipment were wetted by liquid HLW. DOE would follow applicable provisions of DOE Manual 435.1-1, *Radioactive Waste Management Manual* concerning the piping and such other equipment.

The piping and other equipment would be removed and disposed of offsite using a process such as the following:

- Preparations would be made for handling the removed piping and other equipment in a controlled manner consistent with their expected high radiation and contamination levels;
- The piping would be cut into sections and packaged for disposal;
- The other equipment would be removed, segmented as necessary, and packaged for disposal, with this effort coordinated with removal of the piping and waste mobilization and transfer pumps; and
- The piping and other equipment would be disposed of offsite at an appropriate waste disposal facility.

After the piping has been removed, Phase 1 final status surveys would be performed in the empty transfer trench and the trench covers reinstalled.

7.5.4 Monitoring and Maintenance

Monitoring and maintenance of the Waste Tank Farm would continue during the Phase 1 proposed decommissioning and until such time that Phase 2 of the proposed decommissioning begins. The tank and vault drying system installed during the work to establish the interim end state described in Section 3 would remain in operation.

The existing dewatering well would continue to be used to artificially lower the water table to minimize in-leakage of groundwater into the tank vaults. After the Low-Level Waste Treatment Facility is taken out of operation, the water from this well would be collected, sampled, treated if necessary using a portable wastewater treatment system, and released to Erdman Brook through a SPDES-permitted outfall.

Appendix D provides additional information on these matters.

7.6 WMA 5 Proposed Decommissioning Activities

This section addresses removal of Lag Storage Addition 4, the Remote-Handled Waste Facility, and remaining concrete floor slabs and foundations and gravel pads in WMA 5, the Waste Storage Area. Figure 3-35 shows this area.

7.6.1 Removing Lag Storage Addition 4 and the Shipping Depot

Lag Storage Addition 4, a clear-span structure with a pre-engineered frame and steel sheathing, is approximately 291 feet long, 88 feet wide, and 40 feet high. The attached steel framed, steel sided structure houses the Shipping Depot and Container Sorting and Packaging Facility.

These structures would be removed and the demolition debris disposed of at an appropriate off-site waste disposal facility using a process such as the following:

- Demolishing the structure to grade level;
- Removing the floor slab and excavating the building footprint to approximately two feet below grade;
- Disposing of the demolition debris at appropriate offsite waste disposal facilities;
- Performing Phase 1 final status surveys in the area excavated;
- Making arrangements for any independent confirmatory surveys to be performed in the excavated area; and
- After completion of the surveys, filling in the excavated area with clean earthen backfill.

7.6.2 Removing the Remote-Handled Waste Facility

This metal-sided, steel-frame building, which became operational in 2004, includes a receiving area, a buffer cell, a work cell, a waste packaging area, an operating aisle, and a load-out/truck bay. It is shown in Figures 3-36 and 3-37.

This facility is used to remotely section and package high-activity equipment and waste and is permitted as a mixed waste treatment and storage containment building. The closure of the facility under an approved Resource Conservation and Recovery Act closure plan would be coordinated with the demolition under this plan.

The Remote-Handled Waste Facility would be removed using a process such as the following:

- Removing the installed equipment such as the cranes and tanks;
- Demolishing the structure to grade level;
- Removing the floor slab and foundation, removing the below-grade part of the structure, and excavating the rest of the building footprint to approximately two feet below grade;
- Disposing of the demolition debris at appropriate offsite waste disposal facilities;
- Performing Phase 1 final status surveys in the area excavated;
- Making arrangements for any independent confirmatory surveys to be performed in the excavated area; and
- After completion of the surveys, filling in the excavated area with clean earthen backfill.

The underground decontamination waste transfer lines from the Batch Transfer Tank in the building to Tank 8D-3 in WMA 3 would be removed and disposed of as LLW if they have been exposed to radioactivity; otherwise, they would remain in place.

7.6.3 Removing Remaining Floor Slabs and Foundations and Gravel Pads

All remaining concrete floor slabs and foundations would be removed, including those associated with the Lag Storage Building, Lag Storage Addition 1, and Lag Storage Addition 3. The Lag Storage Addition 2 hardstand would also be removed, along with the gravel pads associated with the Chemical Process Cell Waste Storage Area, the hazardous waste storage lockers, the cold hardstand area, the vitrification vault and empty container hardstand, the old/new hardstand storage area, the lag hardstand, and the Product Purification Cell box storage area.

The remaining floor slabs, foundations, and gravel pads would be removed along with the underlying soil to approximately two feet below grade, with the debris and removed soil disposed of at appropriate offsite waste disposal facilities. This work would be followed by Phase 1 final status surveys of the excavated areas and any independent verification surveys desired by the regulators. After the surveys have been completed, the excavations would be filled with earth.

7.7 WMA 6 Proposed Decommissioning Activities

This section addresses proposed decommissioning activities in WMA 6, the Central Project Premises, which is shown in Figure 3-38. These activities involve removal of the Sewage Treatment Plant, the south Waste Tank Farm Test Tower, the two demineralizer sludge ponds, the equalization basin, and the equalization tank. The demolition debris and the removed soil would be disposed of at appropriate offsite disposal facilities.

7.7.1 Removing the Sewage Treatment Plant

This wood frame structure with metal siding and roofing was used to treat sanitary waste and contains six in-ground concrete tanks, one above-ground polyethylene tank, and one above-ground stainless steel tank. This facility would be completely removed, including the underground concrete tanks, with the concrete foundation and underlying soil removed approximately two feet below grade.

After completion of this work, a Phase 1 final status survey would be performed in the excavated area and arrangements made for any independent verification surveys requested by the regulators. Experience with buildup of natural and manmade radioactivity in sewage sludge (ISCORS 2005) would be taken into account in these surveys. After completion of the surveys, the excavated area would be filled with earth.

7.7.2 Removing the Equalization Basin

The equalization basin is an earthen basin lined with Hypalon® approximately 50 feet by 125 feet by seven feet deep that has served as a replacement for the demineralizer sludge ponds.

The liner and approximately two feet of underlying soil would be removed and disposed of offsite. After completion of this work, a Phase 1 final status survey would be performed in the area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the area would be filled with earth.

7.7.3 Removing the Equalization Tank

The Equalization Tank is a 20,000-gallon underground concrete tank immediately north of the Equalization Basin that serves as a replacement for the Equalization Basin.

The tank would be demolished and approximately two feet of underlying soil removed, with this material disposed of offsite. After completion of this work, a Phase 1 final status survey would be performed in the area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the area would be filled with earth.

7.7.4 Removing the Demineralizer Sludge Ponds

The north and south demineralizer sludge ponds are separate, unlined basins excavated in the sand and gravel layer that are known to contain low-level radioactive contamination.

The area of the ponds would be excavated to a total depth of approximately five feet, with the material removed being disposed of offsite at an appropriate waste disposal facility. After completion of this work, a Phase 1 final status survey would be performed in the area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the area would be filled with earth.

7.7.5 Removing the South Waste Tank Farm Test Tower

This test tower would be removed, including its concrete foundation and underlying soil to approximately two feet below grade, with the debris and soil disposed of offsite. After completion of this work, a Phase 1 final status survey would be performed in the area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the area would be filled with earth.

7.7.6 Removing the Remaining Floor Slabs and Foundations

The remaining floor slabs and foundations in the area – including the underground structure of the Cooling Tower– would be removed, with underlying soil removed to a maximum depth of two feet below grade. After completion of this work, a Phase 1 final status survey would be performed in the area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the area would be filled with earth.

7.8 WMA 7 Proposed Decommissioning Activities

WMA 7, the NDA area, is shown in Figure 3-41. The NDA would continue to be monitored and maintained during Phase 1 and no decommissioning actions related to the NDA itself would take place in this phase of the proposed decommissioning. The only Phase 1 proposed decommissioning actions would involve removal of the remaining concrete slabs and gravel pads associated with the NDA hardstand.

These concrete slabs and gravel pads would be removed and the footprints of these areas would be excavated to a maximum of depth two feet below grade, with the debris and excavated materials disposed of offsite. Phase 1 final status surveys would be performed in the excavated areas and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, these areas would be filled with earth.

7.9 WMA 9 Proposed Decommissioning Activities

This section describes proposed decommissioning activities in the Integrated Radwaste Treatment System Drum Cell area, which is shown in Figure 3-42. Phase 1 proposed decommissioning activities in this area would involve removal of the Drum Cell, the trench soil container area, and the subcontractor maintenance area.

The Drum Cell is a pre-engineered metal building 375 feet long, 60 feet wide, and 26 feet high, with concrete shield walls, remote waste handling equipment, container storage areas, and a control room. It would be demolished by conventional means and the floor slab, gravel pad, and foundation removed, along with underlying soil to at least two feet below grade. After completion of

this work, a Phase 1 final status survey would be performed in the excavated area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the excavated area would be filled with clean earth.

The trench soil container area is located northwest of the Drum Cell. The material in this area would be removed and its footprint excavated to a maximum depth of approximately two feet below grade, with the excavated materials disposed of offsite. Phase 1 final status surveys would be performed in the excavated area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the area would be filled with clean earth.

The subcontractor maintenance area, a gravel pad near the rail spur, would be removed using the process used for the trench soil container area.

7.10 WMA 10 Proposed Decommissioning Activities

The Phase 1 proposed decommissioning activities in this WMA, the support and services area, would consist of removing the New Warehouse and the remaining concrete floor slabs and foundations, along with the former Waste Management Storage Area. WMA 10 is shown in Figure 3-43.

The New Warehouse would be removed. This structure is 80 feet wide, 250 feet long, and 21.5 feet high and rests on concrete piers and a poured concrete foundation wall. It would be demolished by conventional means and its foundation and the underlying soil removed to a maximum depth of approximately two feet below grade. After completion of this work, a Phase 1 final status survey would be performed in the excavated area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the excavated area would be filled with clean earth.

The remaining floor slabs and foundations in the area – including those for the Administration Building, the Expanded Environmental Laboratory, and the Fabrication Shop – would also be removed, with underlying soil removed to a maximum depth of approximately two feet below grade. The former Waste Management Storage Area would also be removed in the same manner. After completion of this work, a Phase 1 final status survey would be performed in each excavated area and arrangements made for any independent verification surveys requested by the regulators. After completion of the surveys, the excavated areas would be filled with earth.

The Meteorological Tower and the Security Gatehouse and fences would remain in place.

7.11 Remedial Technologies

A combination of conventional technologies and proven innovative technologies would be used to accomplish the proposed decommissioning activities specified in the preceding sections. This section summarizes these technologies in the following categories:

- Pipe cutting and other metal cutting,
- Tool positioning,
- Concrete cutting and demolition,
- Concrete decontamination,

- Demolition of structures, and
- Excavation and grading

It is not the intention of this summary of remediation technologies to preclude the use of other, better technologies that may be developed, so long as they are comparable with and equivalent to those discussed here, nor is it DOE's intention to endorse the products of particular manufacturers beyond observations about cases where those products have been successfully used. More specific information on the technologies to be used would be provided in the Decommissioning Work Plan(s).

7.11.1 Pipe Cutting and Other Metal Cutting

The following methods would be used as applicable for cutting radioactively contaminated piping and metal liners, equipment, and structural components. Methods would be selected based on efficiency and suitability for the particular applications, with consideration of factors such as personnel safety, metal thickness, and radioactive contamination control. These technologies are listed in alphabetical order.

Diamond Wire Cutting Systems

This technology is suitable for cutting thick steel plate such as that which may be used in the shielded transfer cell in the Load-In/Load Out Building. It is described below under Concrete Cutting and Demolition.

Duriron Pipe Cutting

Because Duriron is hard and brittle, Duriron wastewater piping is typically cut into sections using either a chain-type tool or a special tool provided by the piping manufacturer to score the pipe, and tapping it with a mallet to fracture it at the score mark.

Hand-Held Shear

This technology, manufactured by Res-Q-Tek, Inc., cuts stainless-steel pipes up to 1.5 inches in diameter, and has been used at DOE's Fernald site. This shear can also crimp pipes to minimize potential spillage of pipe contents.

High-Speed Clamshell Pipe Cutter

This technology can cut through large pipes up to 24 inches in diameter with minimal clearance requirements. This equipment is manufactured by Tri-Tool, Inc., and has been used at DOE's Hanford site.

Mega-Tech Hydraulic Shears

This equipment, manufactured by Mega-Tech, Inc., can be used to cut stainless steel pipes up to 1.5 inches in diameter and has been used at Argonne National Laboratory.

Nd:YAG Laser

A Lumonics two kilowatt neodymium-doped yttrium aluminum garnet (Nd:YAG) laser has been used to remotely size reduce about 300 fuel storage tubes and radioactively-contaminated converter shells from the former K-25 Gaseous Diffusion Plant site at Oak Ridge, Tennessee.

Nibblers

Electric nibblers have been found effective in cutting sheet metal in many applications. They are readily available commercially.

Pipe Cutting and Crimping System

The Burndy Lightweight Portable Crimper is an electrically powered hydraulic crimping tool that cuts smaller-diameter piping by crimping and minimizes the potential spillage of piping contents. This equipment is manufactured by Burndy, Inc, and has been used at DOE's Mound facility.

Pipe Cutting and Isolation System

This robotic technology developed by TPG Applied Technology consists of three tools: a pipe-cutting tool, a pipe-cleaning tool, and a pipe-plugging tool. This system has been used to cut pipes within storage tanks at the K-25 Plant at DOE's Oak Ridge site.

Powered Pipe Cutting Machines

Air-powered pipe cutoff machines have been found effective by the U.S. Navy in cutting stainless steel piping of varying diameters.

Reciprocating Saws and Portable Band Saws

Variable-speed electric reciprocating saws and portable band saws were found effective in cutting stainless steel piping and other metal shapes up to one-half inch thick during the decommissioning of the Barnwell Nuclear Fuel Plant. Effectiveness depends on blade type, cutting speed, and blade lubricant.

Roller Cutters

Manually operated roller cutters have been found effective by the U.S. Navy on highly-contaminated, smaller diameter piping where radiological containment is required.

Size Reduction Machine

The Mega-Tech Services size reduction machine has been used at DOE's Savannah River Site and is capable of hydraulically shearing piping from six feet below floor level to 15 feet above floor level. It can shear pipes up to four inches in diameter

Thermal Cutting Technologies

Oxy-acetylene and oxy-gasoline cutting torches have been used to cut steel pipe and plate at DOE sites. The oxy-gasoline cutting torch is specially suited for cutting carbon-steel pipes and plates, and can cut steel up to 4.5-inch in thickness at a rate three times faster than oxy-acetylene cutting. This equipment is manufactured by Petrogen International, and has been used at DOE's Oak Ridge, Fernald, and Mound sites.

7.11.2 Tool Positioning Technologies

The following three systems have been found to be useful at DOE sites:

Dual Arm Work Platform

The dual arm work platform is a remotely operated deployment platform that uses a variety of equipment to dismantle metal assemblies. Two Schilling Titan III manipulator arms provide six degrees of freedom, and are powered by a 3000 psi hydraulic system.

Each arm is capable of lifting 240 pounds, while the grippers on the end of the arms can exert 1,000 lbs of crushing force. The platform is designed to be free standing or suspended from an overhead crane. This system was used at the DOE CP-5 Research Reactor Large-Scale Demonstration Project at Argonne National Laboratory – East.

Mobile Work Platform

The Mobile Work Platform is a remote-controlled machine designed to remove pipe/conduit. A rotating turret is equipped with a folding main boom and a telescoping job boom capable of reaching 27 feet. The boom system can lift over four tons with the outriggers in place. With the dual crimper/shear attached to the jib boom, the reach extends out to 32 feet above the ground.

Rosie - Mobile Work Platform

Rosie evolved from the Remote Work Vehicle that supported cleanup work at the Three Mile Island nuclear power plant. The Rosie is a remotely operated, mobile work platform built by RedZone Robotics. It is a four-wheel drive, four-wheel steer locomotor that is capable of deploying tools weighing up to 2,000 lbs to a height of 27 ft with a telescoping boom with various end effectors.

A control console allows a single operator to remotely manipulate Rosie using video and data displays. Video displays are provided by up to ten cameras mounted on Rosie, in addition to cameras mounted in the facility. During the demonstration at the CP-5 Research Reactor, Rosie was fitted with a jackhammer and used to remove high-density concrete from the reactor's upper shield plug.

7.11.3 Concrete Cutting and Demolition

Concrete Saws

Concrete saws such as those used during highway pavement maintenance have been used effectively in cutting out sections of concrete floors during nuclear facility decommissioning. They are available from various manufacturers with carbide and diamond-impregnated saw blades ranging up to 30 or more inches in diameter.

Remote Controlled Demolition Machines

Demolition machines have been used to remotely remove and size-reduce concrete, piping, and structural steel. The Brokk remote controlled demolition machines, such as the model shown in Figure 7-12, are manufactured by Holmhed Systems AB. They can be operated remotely with a hydraulic hammer, excavating bucket, concrete crusher, and a shear. The arm has a reach of 15 feet, and can be operated remotely at distances up to 400 feet.

One was used effectively in dismantling equipment in the Vitrification Cell during cell deactivation. These machines could be used in various places in the Process Building and Vitrification Facility.

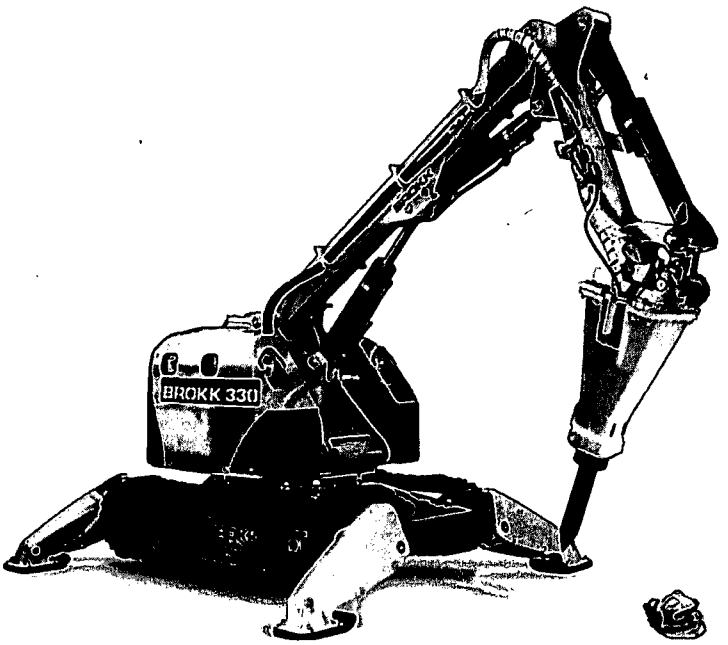

Figure 7-12 Typical Demolition Machine

Diamond Wire Cutting Systems

Diamond wire cutting utilizes diamond-impregnated wire to cut metal and concrete. The system uses a series of guide pulleys to draw the continuous wire strand through the cut. This technology has been used at numerous decommissioning projects, such as Fort St. Vrain, DOE's C Reactor Interim Safe Storage Project at the Hanford site (Trentec, Inc., Cincinnati, Ohio), and the Tokamak Fusion Test Vessel (Bluegrass Bit Co., Greenville, Alabama).

Diamond wire cuts through reinforced concrete, rebar, structural steel, and steel plate without generating large amounts of dust. The wire is cooled with either water collected in a sump, which controls any loose contamination generated during cutting, or with liquid nitrogen in situations where waste generation is a prime concern.

Jackhammers and Chipping Hammers

Pneumatic jackhammers and chipping hammers have been used on many projects to break up contaminated concrete by creating localized fractures with repeated blows. They are available from numerous manufacturers.

7.11.4 Concrete Decontamination

Contaminated concrete surfaces would be decontaminated using conventional means such as vacuuming and wiping with cloths dampened with water or non-hazardous decontamination agents. The following technologies would also be considered and used as appropriate:

Concrete Shaver

Marcrist Industries and Demolition Technologies manufacture manned and remote concrete shavers that remove surface concrete from flat and curved surfaces. The diamond-impregnated

shaving blades are ten to 12 inches wide, and each pass of the shaver can remove up to one-quarter inch of concrete at a rate of 128 square feet per hour. The Marcrist DTF-25 can shave floors to depths of 0.5 inches. Dust is contained within a HEPA-filtered vacuum system. Manned equipment has been used at the Hanford C Reactor and the remote-controlled equipment has been used at the Rancho Seco Nuclear Plant.

Concrete Spaller

This hand-held tool is used to decontaminate flat concrete walls and floors by removing concrete pieces ranging from seven to 16 inches in diameter by hydraulically expanding within pre-drilled holes. A shroud collects the pieces of concrete, while a HEPA filter controls the potential release of airborne materials. The spaller removes concrete faster, to a greater depth and at a lower cost per square foot than traditional baseline scabblers and scalers when removing to a depth of one-eighth inch or greater. Pacific Northwest National Laboratory is a manufacturer of spallers.

Centrifugal Shot Blast System

Concrete Cleaning, Inc. and Pentek manufacture manned and remotely operated centrifugal shot blast scabbling systems that use hardened steel shot at high velocities to remove the outer surface area of concrete. The concrete fragments are captured by an integrated vacuum system. This technology is used in confined space situations and for shallow depths of contamination (less than one inch).

The MOOSE®, a remotely operated floor scabbling centrifugal shot blasting system from Pentek, is capable of effectively removing concrete to a depth of 3/16 of an inch and has removed concrete to a depth of one inch with some difficulty (Figure 7-13). The technology was successfully demonstrated at DOE's Fernald facility.

Remote Dry-Ice Blasting System

The ROVCO 2 system integrates two demonstrated technologies: a remotely operated vehicle and a dry-ice (CO_2) blasting system. The vehicle transports and powers the vehicle-mounted subsystems, including the CO_2 XY orthogonal end effector (COYOTEE), cryogenesis dry-ice blasting system, and the vacuum/filtration/containment subsystems. The COYOTEE manipulates a specially designed vacuum containment workhead with the cryogenesis blasting nozzle to cover every point within a rectangular workspace. Since ROVCO 2 utilizes CO_2 gas, it has the potential to eliminate process waste resulting from the blasting material.

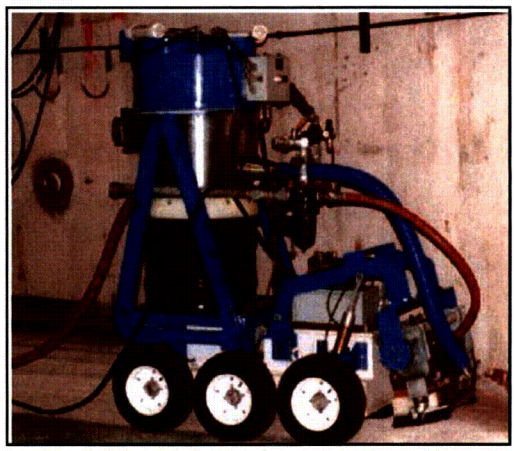

Figure 7-13. MOOSE®

Rotary Drum Planer

The rotary drum planer is widely used to remove concrete in highways and parking lots. This technology consists of a drum with replaceable tungsten-carbide teeth. The planer is attached to a

Bobcat loader and cuts a 16-inch swath up to six inches deep, providing that there is no wire or rebar present within the concrete because this metal would break the cutting teeth.

The system can be customized to be dust free by simultaneously drumming the waste with a vacuum shroud. This baseline technology has been used at numerous DOE facilities, including Fernald.

Scabblers

This manual or remote technology utilizes a series of tungsten carbide-tipped bits mounted on a hammer head that pulverize the concrete surface via mechanical impacts. The dust and debris removed from contaminated concrete surfaces are then captured by a HEPA-filtered vacuum system. This technology is suitable for removing contaminated concrete from large areas, but is less successful in corners and in concrete seams and cracks. Scabblers have been used on many decommissioning projects, including those at the Argonne National Laboratory and the Idaho National Engineering and Environmental Laboratory.

Soft Media Blast Cleaning

Soft Media Blast Cleaning uses a pneumatically propelled soft media to remove surface contaminants. The soft blast media impacts the surface with high energy, absorbing the contaminants and carrying them away from the substrate for easy disposal. This system is used for low levels of surface contamination.

Steam Vacuum Cleaning

The Kelly Decon System uses a pressurized (250 psi) superheated (up to 300°F) water stream to remove contamination from surfaces. Several of the cleaning heads integrate a vacuum hood and return line which captures and controls the steam, water droplets, and dislodged contaminants generated when the water spray impacts on the surface being cleaned. The primary application for the Kelly System has been the surface decontamination of rooms, pools, walls, large components, or similar applications related to large and/or smooth surfaces.

Robotic Hammer

This robotic jackhammering system, manufactured by Bluegrass Bit Co. of Greenville, Alabama, has been used where jack hammering is preferred, but where radiation levels preclude manned operation.

Remote-Controlled Brokk Concrete Demolition Systems

As indicated above, Brokk demolition machines such as the BM 330 model pictured in Figure 7-12, can be used effectively in concrete demolition where radiological conditions make use of remote-controlled equipment preferable.

7.11.5 Demolition of Structures

Structures would be demolished using conventional methods and proven, advanced technologies such as the following:

Backhoe Pulverizer

This machine uses air-powered or hydraulic jaws mounted on a backhoe to crush concrete and separate rebar.

Backhoe Ram

A track-mounted backhoe ram is typically used for demolition of thick concrete or cinder block. It uses a pneumatic or hydraulic moil or chisel point to deliver blows to the area of interest.

Bulldozer

Bulldozers would typically be used to push structure sections down with the blade and pull sections down using wire rope attached to the structure section.

Portable Concrete/Asphalt Crusher

The Eagle Crusher Company, Inc. manufactures a portable concrete/asphalt crusher for size-reducing concrete debris. This equipment is bulky and is setup outside and adjacent to structures. It is best suited for concrete with little or no radioactive contamination.

Track-Mounted Shear/Crusher

This hydraulic equipment (manufactured by Tiger Machine Company) is one of the baseline tools for breaking up concrete surfaces into pieces for disposal. It is effective in razing structures quickly. Criteria for using this equipment are generally the amount of surface area to be broken up and accessibility for large equipment, because the track mounted configuration limits maneuverability.

Universal Demolition Processor

This technology, made by several manufacturers (e.g., Tramac), is essentially three different technologies in one. By exchanging jaw sets, it can be a concrete pulverizer, concrete cracker (including rebar), or a shear capable of cutting thick steel plates. The universal demolition processor is attached to a standard track-mounted carrier. One benefit is that it reduces the amount of equipment on site, due to its multiple capabilities. This equipment has been used at DOE's Fernald facility and at other demolition sites (Figure 7-14)

7.11.6 Excavation and Grading

DOE would use conventional equipment to remove soil, equipment, and portions of concrete structures, such as tracked excavators. Backhoes and bulldozers would be used as needed. Similar equipment would be used for grading the site.

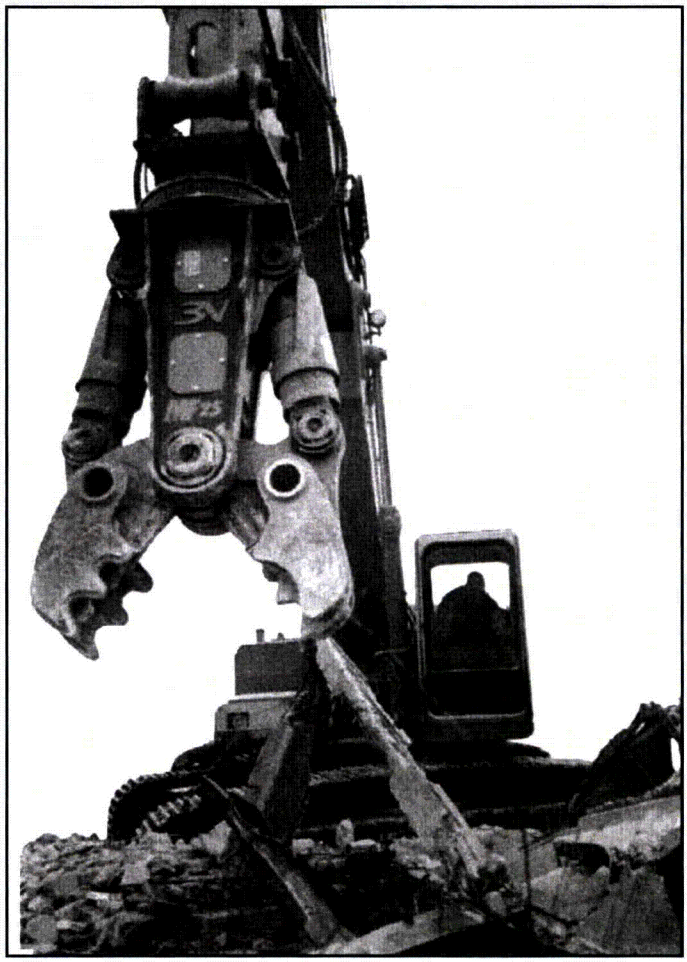

Figure 7-14. Universal Demolition Processor

7.12 Schedule

Due to the circumstances of the proposed decommissioning – such as the annual federal government funding process and the prerequisite of issuing the Record of Decision for the Decommissioning EIS – it is not practicable for DOE to provide a detailed schedule for the project at this time. Figure 7-15 provides a conceptual schedule for the project, with the basic sequence and order-of-magnitude time frames for major actions.

The dates on the schedule are contingent upon NRC approval of this plan. Before the proposed decommissioning begins, DOE would provide a more detailed schedule to NRC for information. DOE also recognizes that circumstances can change during the proposed decommissioning so that the proposed decommissioning could not be completed as outlined on the schedule. In such a case DOE would revise the schedule and provide the revised schedule to NRC.

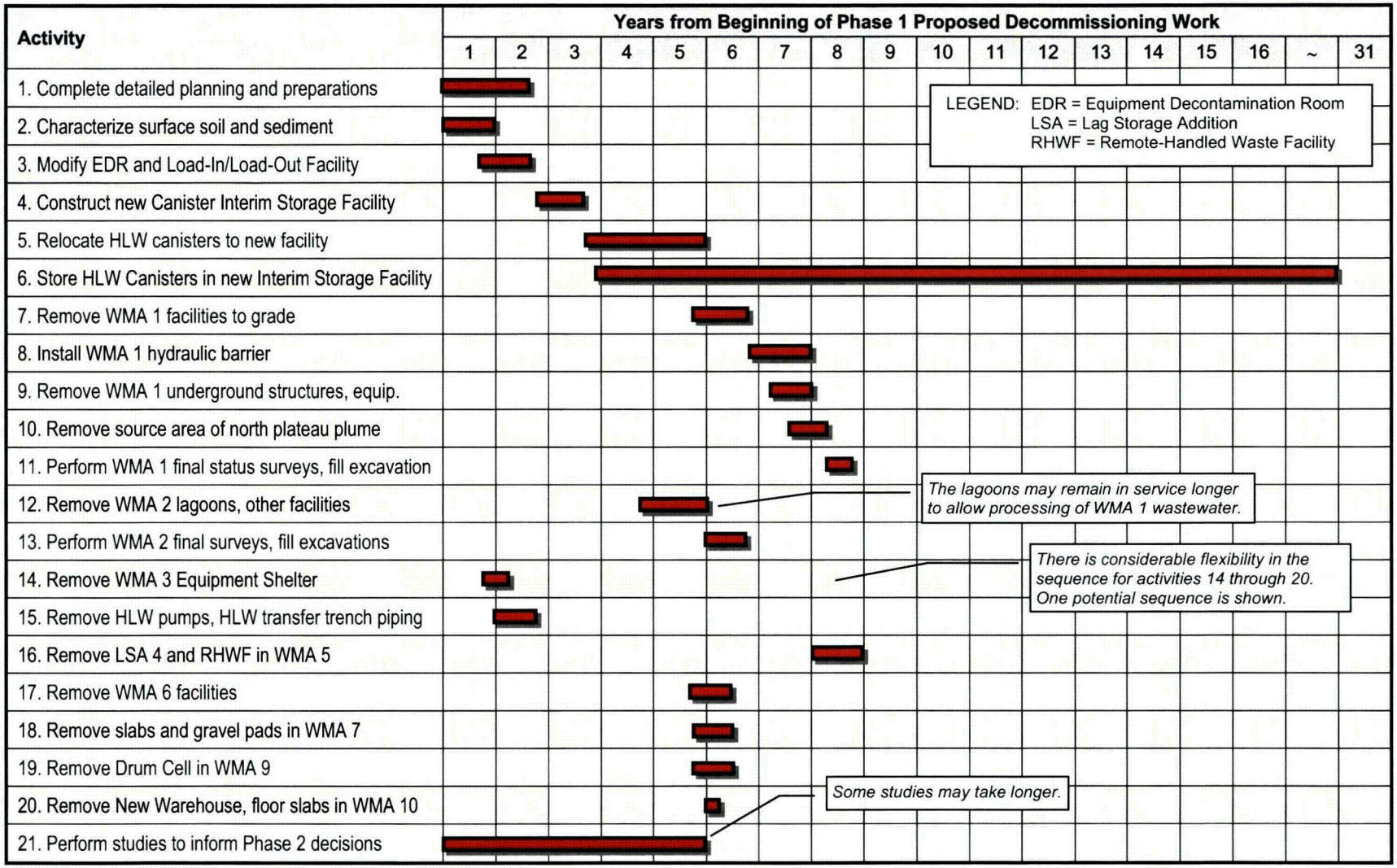

Figure 7-15. Conceptual Schedule of Phase 1 Proposed Decommissioning Activities

7.13 References

Code of Federal Regulations

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

DOE Manuals

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

Other References

EPA, 2007, *Clean Air Act Assessment Package, CAP88-PC, Version 3.0.* U.S. Environmental Protection Agency, Office of Radiation and Indoor Air, Washington, D.C., December 9, 2007.

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NRC 1997, *Standard Review Plan for Dry Cask Storage Systems* NUREG-1536, Final Report. U.S. Nuclear Regulatory Commission, Washington, D.C., January 1997.

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WVNSCO 2006, *Safety Analysis Report for Waste Processing and Support Activities*, WVNS-SAR-001, Revision 11. West Valley Nuclear Services Company, West Valley, New York, November 11, 2006.

WVNSCO and Scientech, 2000, *High-Level Waste Canister Shipout from the West Valley Demonstration Project*, Revision 2. West Valley Nuclear Services Company, West Valley, New York and Scientech, Dunedin, Florida, June 30, 2000.

8.0 QUALITY ASSURANCE PROGRAM

PURPOSE OF THIS SECTION

The purpose of this section is to describe the Quality Assurance Program for Phase 1 of the WVDP proposed decommissioning, focusing on characterization, engineering data, calculations, dose modeling, and the final status surveys. The information in this section shows how the Quality Assurance Program would be managed and implemented. It is also intended to show NRC staff how accurate, high-quality information would be provided to support Phase 1 of the proposed decommissioning.

INFORMATION IN THIS SECTION

The focus of this section is appropriate because the proposed decommissioning is being conducted under the WVDP Act as explained in Section 1. The information provided is necessarily generic in nature because contractual arrangements for the proposed decommissioning have not yet been made.

This section begins with a description of the quality assurance organization and the duties and responsibilities of the quality assurance and proposed decommissioning organizations that are associated with the Quality Assurance Program. It continues with a description of the Quality Assurance Program, control of documents, measuring and test equipment, purchased materials, and subcontractor services. The section concludes with descriptions of corrective action, audits and surveillances, and management of quality assurance records.

Because some preliminary engineering work such as dose modeling and the engineered barrier design would be completed before proposed decommissioning activities commence under this plan, this section refers to existing quality control assurance programs for those activities and briefly describes these programs.

RELATIONSHIP TO OTHER PLAN SECTIONS

To understand the scope of the Quality Assurance Program, one must consider the information in Section 1. Section 1 discusses the project background, the proposed decommissioning activities, and project management and organization.

This section provides the quality assurance requirements for the programs and activities identified in Sections 5, which addresses dose modeling, and Section 9, which deals with radiation surveys. It also applies to engineering data and calculations related to designs described in Section 7 for the Interim Storage Facility for the vitrified HLW canisters and the hydraulic barrier walls that would remain in place after Phase 1 is completed.

8.1 Quality Assurance Organization

The Quality Assurance (QA) Organization is shown in Figure 8-1. The QA Manager, who reports directly to the Decommissioning Contractor Senior Executive, manages the organization. The QA Manager provides central leadership, direction, and management to the proposed decommissioning project.

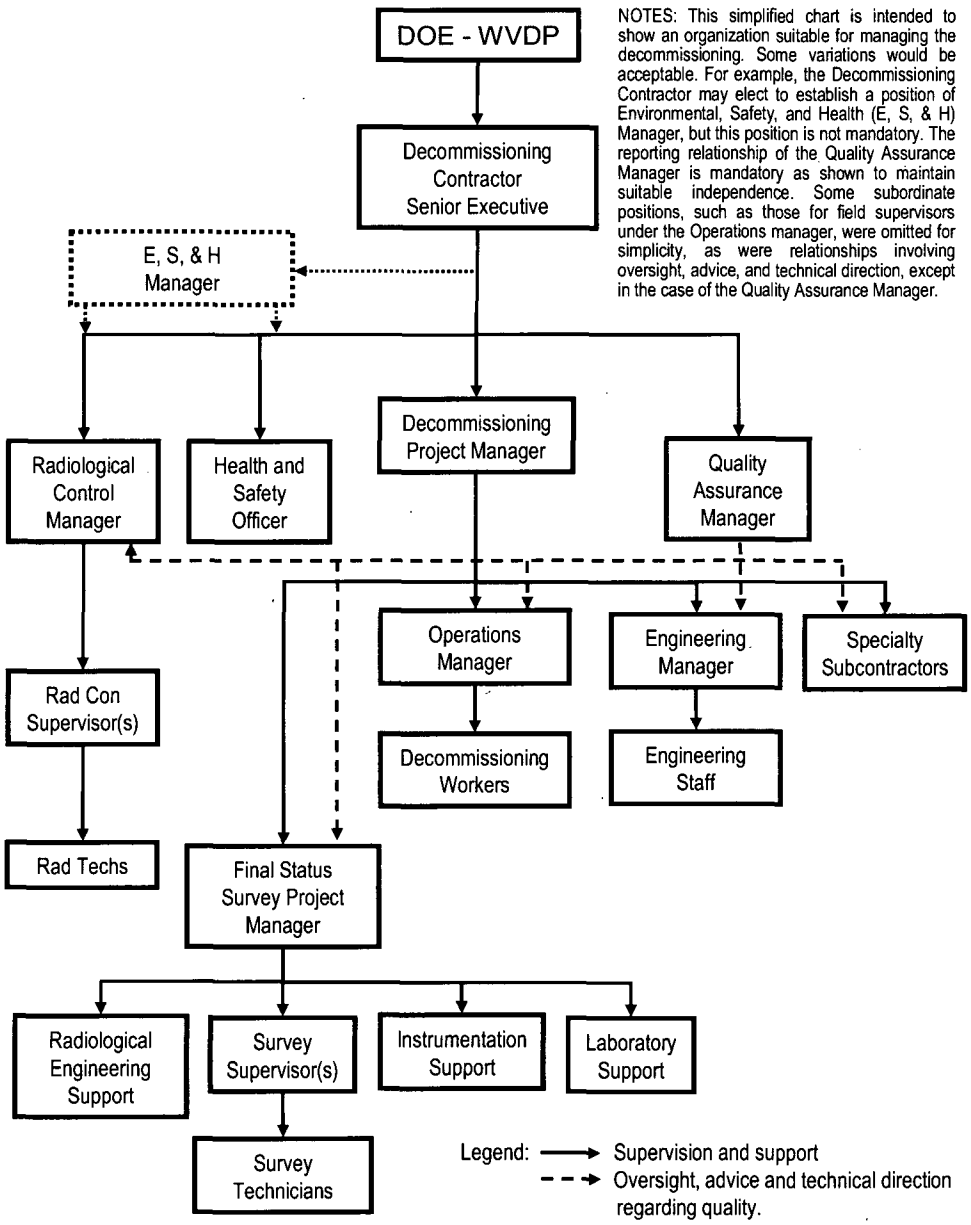

Figure 8-1. Decommissioning Organization Quality Assurance Relationships

Quality must be built into the proposed decommissioning project by project personnel. Each person in the decommissioning organization is responsible for QA related to the tasks he or she performs. To help ensure that quality is built in, QA procedures implementing the QA Program would be developed by the decommissioning organization. QA would be provided through implementation of the QA Program and project implementing procedures as it relates to QA/quality control (QC) issues.

The QA duties and responsibilities of the QA organization and the decommissioning organization are listed below.

8.1.1 Quality Assurance Organization Duties and Responsibilities

The QA Manager is responsible to:

- Develop the project QA Program manual or plan as a formal document implementing the requirements of this section and maintain this document current;
- Provide central leadership, direction, and management of the decommissioning QA Program;
- Ensure that preparation and maintenance of the QA Program are responsive to DOE and NRC QA requirements and act as the primary QA interface with DOE and NRC;
- Implement DOE and WVDP quality policies and define the direction of the QA Program with respect to these policies;
- Perform as the certifying agency for the QA Program;
- Make final interpretations of established QA requirements;
- Determine when conditions during proposed decommissioning are not in compliance with the QA Program;
- Provide input and direction for QA training;
- Provide oversight of subcontractor and vendor activities;
- Provide receipt inspection services for purchased materials;
- Evaluate the adequacy and effectiveness of the QA Program;
- Review and approve procedures implementing the requirements of the WVDP QA Program;
- Review and approve procurement documents as required;
- Perform and document independent audits, surveillances, inspections and tests as required;
- Stop unsatisfactory work and control processing and delivery of unsatisfactory materials; and
- Maintain required QA records.

8.1.2 Decommissioning Project Quality Assurance Duties and Responsibilities

Project personnel are responsible to:

- Provide the requisite level of quality in work performed;
- Develop organizational procedures implementing the requirements of the WVDP QA Program;
- Implement the policies and procedures established to support the QA Program;
- Ensure that activities affecting quality are prescribed by documented instructions, procedures, and drawings and that such activities are accomplished through implementation of these documents;
- Prepare QA Project Plans in support of characterization and the final status survey;
- Perform work safely and correctly the first time, and assure that reliability, performance, and customer satisfaction are maximized;
- Meet established requirements and recommend improvements in material and work process quality;
- Inform management of suspected unsafe or unacceptable quality conditions; and
- Stop work when it is known or suspected that work being performed could potentially result in an unsafe or unacceptable quality condition.

8.2 Assuring Quality in Preliminary Engineering Work

Some engineering work in support of the proposed decommissioning has already been performed by DOE contractors and more would be performed before this plan is approved and placed into effect. Two especially important examples of this work are dose modeling and preliminary conceptual design of engineered barriers to be installed during Phase 1 of the WVDP proposed decommissioning.

DOE ensures that QA programs used for such work meet applicable requirements, such as DOE Order 414.1C and the quality assurance requirements of Code of Federal Regulations 10 CFR 830.120. How this was accomplished for the two examples cited is as follows.

8.2.1 Dose Modeling

The dose modeling was performed by Science Applications International Corporation (SAIC) under contract to DOE.

SAIC Quality Assurance Plan and Supporting Procedures

SAIC prepared and followed a QA Project Plan that applied to the modeling work (SAIC 2008a), along with four supporting QA procedures (SAIC 2008b, 2008c, 2008d, and 2008e) that relate to the dose modeling. This plan was based on the SAIC Business Unit QA Program that was developed to meet customer requirements including those in DOE Order

414.1C, 10 CFR 830.120, and ASME NQA-1 (ASME 2000). Elements of the QA Project Plan and the supporting procedures included:

- Project organization and responsibilities,
- Personnel qualification and certification,
- Document preparation,
- Preparation of code development and verification packages,
- Performing calculations and analyses,
- Independent technical reviews by a qualified person(s),
- Documented comment resolution with formal revisions for significant changes,
- Management and independent assessment, and
- Project records.

Oversight and Review

In addition to the oversight and review provided by SAIC, DOE provided QA oversight and review of this effort, including peer review of the modeling process.

8.2.2 Engineered Barrier Design

Conceptual engineering work related to engineered barriers was performed by Washington Safety Management Solutions (WSMS) and its subcontractor URS Corporation under the requirements of the WSMS QA Plan (WSMS 2006a)[1].

WSMS Quality Assurance Program

The WSMS QA program embodies the QA criteria of 10 CFR 830.122 and DOE Order 414.1A (the earlier version of DOE Order 414.1C) and applicable DOE technical standards. The programs also use ASME NQA-1 (ASME 2000) as a basis with program enhancements from other consensus standards to ensure that the requisite level of quality for all key activities is maintained. Elements of the programs include:

- Line management responsibility for quality;
- Individual responsibility for quality at all levels;
- QA management providing planning, direction, control, and support to achieve quality objectives;
- Formal personnel training and qualification;
- A formal quality improvement process;
- Design controls, with formal design and verification processes;

[1] WSMS is now part of the Washington Division of URS Corporation.

- Work process controls;
- Procurement controls;
- Inspection and acceptance testing;
- Management assessment; and
- Independent assessment.

Contractual arrangements between WSMS and URS required URS to comply with applicable requirements of:

- The SAIC QA Project Plan that applies to proposed decommissioning preparations (SAIC 2008a), and
- The WSMS procedure for preparing technical documents and performing engineering calculations for the EIS and this plan (WSMS 2006b).

Oversight and Review

WSMS review of subcontracted work related to this plan is carried out in accordance with the WSMS QA Plan (WSMS 2006a) and the related procedure (WSMS 2006b). In addition, DOE provides independent oversight of the work performed by site contractors.

8.2.3 Other Engineering Work

DOE would ensure that other engineering data and engineering work, calculations, and modeling provided by DOE contractors in support of Phase 1 of the proposed decommissioning conforms to applicable QA requirements. For example, if another contractor(s) were to complete engineered barrier designs begun by URS and WSMS, then DOE would ensure that the QA program of the new contractor(s) is equivalent to applicable requirements in the WSMS QA Plan and the WVDP supporting procedure (WSMS 2006b).

8.3 Decommissioning Quality Assurance Program

The Decommissioning QA Program identifies and describes the integral elements of the QA activities that apply to a broad spectrum of proposed decommissioning work performed at the WVDP. The QA Program provides the framework and criteria for implementing a QA program to control activities that affect the quality of the WVDP Phase 1 proposed decommissioning.

Specifically, the QA Program would be used to plan, perform, and assess the effectiveness of project activities such as data acquisition and evaluation. It also provides the framework for the development of new or revised engineering data, calculations, and modeling associated with engineered barrier design and any revisions to the dose modeling. Activities affecting quality of the WVDP proposed decommissioning would be subject to the applicable controls of the QA Program and activities covered by the QA Program would be identified in program-defining documents.

The Decommissioning QA Program would meet the intent of 10 CFR 830.120, Subpart A, QA Requirements and the requirements of DOE Order 414.1C.

8.3.1 General Description of the Program

The WVDP Phase 1 Decommissioning QA Program would include the following elements:

- It would be established by the WVDP to govern those activities that may affect quality of the project, including the health and safety of the general public as well as project personnel.
- It would be described in a formal document that incorporates the requirements of this section.
- It shall be implemented by written procedures and carried out throughout Phase 1 of the WVDP proposed decommissioning in accordance with those procedures. The QA procedures would be consistent with regulatory and QA Program requirements.
- Activities affecting quality shall be accomplished under suitable controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied.
- The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of satisfactory implementation.
- Management of organizations participating in the program shall regularly review and assess the status, adequacy, and compliance of the parts of the program that they would be implementing.
- It shall utilize this plan and appropriate implementing QA procedures to meet its objectives.
- It would require training and qualification of workers and quality verification personnel in accordance with DOE Order 414.1C, with instruction on implementing quality assurance in proposed decommissioning activities and documentation of the objectives and content of the training or qualification, attendees, and dates of attendance.
- NRC would be notified when there are changes to the QA Program or organizational elements as approved in this plan before the revised QA Program is implemented.

8.3.2 Characterization and Final Status Survey Data

The portion of the QA Program that sets the requirements for characterization and survey data would ensure that the data sets are of the type and quality needed to demonstrate with sufficient confidence that proposed decommissioning activities can be

carried out in accordance with applicable requirements. The objective would be met through the use of the data quality control processes for data collection design, analysis, and evaluation.

The data quality control processes would ensure that: (1) the elements of the facility characterization and final status survey plans would be implemented in accordance with the approved procedures; (2) surveys would be conducted by trained personnel using calibrated instrumentation; (3) the quality of the data collected would be adequate; (4) all phases of facility characterization and final survey data acquisition and evaluation would be properly reviewed, and oversight provided; and (5) corrective actions, when identified, would be implemented in a timely manner and determined to be effective. This portion of the QA Program would be applied to all aspects of final facility characterization and status survey activities. Basic elements of the QA Program as they would be applied to characterization and survey data are discussed below.

As explained in Section 4, the underground waste tanks have previously been characterized for residual radioactivity and bounding source term estimates have been developed for other areas considered in dose modeling evaluations. Reports identified in Section 4 describe QA associated with obtaining characterization data for making source term estimates in these areas; the QA processes used were similar to those summarized below.

Training and Qualification

Personnel performing facility characterization and final status survey measurements would be trained and qualified in accordance with DOE Order 414.1C. Training would include procedures governing the performance of measurements, operation of field and laboratory instrumentation, and control of measurements and samples.

The extent of training and qualification would be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity. Records of training would be maintained in accordance with the approved course description for initial and continuing training for decommissioning.

Measurement Documentation Control

Date, instrument, location, type of measurement, and mode of operation would identify each measurement. Generation, handling, and storage of the original final status survey and facility characterization plans and data packages would be controlled. Records would be designated as quality documents and they would be maintained as such in accordance with WVDP procedures.

Survey and Sampling Methods

Areas or facilities to be characterized or surveyed would be designated as separate characterization or survey areas. Depending on its size, each area may be divided into smaller areas. The methods for determining the type and number of measurements required for each area are discussed in Section 9.

Written Procedures

Sampling and measurement tasks must be performed properly and consistently in order to assure the quality of the final results. The measurements would be performed in accordance with approved, written procedures that describe the methods and techniques used for the final facility characterization or status survey measurements and acceptance criteria to ensure that sampling and measurements are performed satisfactorily.

Control of Samples

Responsibility for the control of samples from the point of collection through the determination of the final results would be established by procedure. When control is to be transferred, chain of custody forms would accompany the sample for tracking purposes. Secure storage would be provided for archived samples.

Quality Assurance Project Plans

Quality assurance for each major task associated with characterization and the final status survey would be described in a QA Project Plan that provides a blueprint for how the quality system of this section would be applied to the particular task. Such plans would be consistent with guidance contained in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000). The applicable guidance in the *Uniform Federal Policy for Implementing Environmental Quality Systems: Evaluating, Assessing, and Documenting Environmental Data Collection/Use and Technology Programs* (DOE 2005) would also be considered.

Quality Control

Procedures would establish built-in QC verification in the processes for both field and laboratory measurements. The QC verifications would duplicate the original measurements where possible. Acceptance criteria would be established to ensure repeatability of the data. Laboratory analysis verification testing would make use of blank, spiked duplicate and replicate samples and measurements in addition to duplicates. If the acceptance criteria are not met, an investigation would be conducted to determine the cause and corrective action.

Selection, Calibration and Operation of Instrumentation

Proper selection and use of instrumentation would ensure that sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as well as assure the validity of the data. Instrument calibration would be performed with traceable sources using approved procedures. Issuance, control and operation of instruments would be conducted in accordance with WVDP procedures. Instrument operability would be verified using background and check sources as specified in Section 9.

Control of Tools and Sample Containers

New sample containers would be used for each individual sample taken. This practice would ensure the data obtained from each sample would meet QA requirements. Tools would be decontaminated after each sample and surveyed for contamination prior to taking new or additional samples.

Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, would be procured from appropriate vendors in accordance with approved quality and procurement procedures.

8.3.3 Engineering Design and Data, Calculations, and Modeling

Engineering designs and data, calculations, and modeling of engineered barrier modifications would be developed within the framework of applicable engineering requirements. The adequacy of these engineering products would be verified or validated by individuals or groups other than those who performed the work. Verification and validation work would be completed before approval and implementation.

A control process that meets the intent of the appropriate requirements of ASME NQA-1 (ASME 2000) would be implemented. Controls would be determined through a controlled process that considers environmental and quality impact.

Basic elements of the QA Program as they would be applied to engineering design modifications, engineering data, calculations, and system, structure, and component modeling are discussed below.

Design Control

The formal design process defines the control of design inputs, processes, outputs, changes, lines of communication, interfaces, and records. This process provides for timely and correct translation of design inputs into design outputs, effective coordination and interfacing of organizations participating in the design process, and acceptable and verified design outputs. Design and design modifications shall provide for the intended end use, including (but not limited to) inspection, acceptance criteria, and hazard mitigation.

Design inputs (such as engineering data) would be correctly translated into design outputs (such as specifications, drawings, procedures, and instructions). Calculations and associated design decisions would be checked for correctness during the design process. Design outputs would be verified to confirm that they would be suitable for their intended use.

Changes to final designs (including field changes and modifications and nonconforming items that would be dispositioned "use as is" or "repair") would be subjected to design control measures commensurate with those applied to the original design. These design control measures may include review of the relevant design analyses to verify their continued validity.

The acceptability of design activities and documents – including design inputs, processes, outputs, and changes – would be verified as appropriate. Computer programs would be proven through previous use, or verified through testing or simulation prior to use.

Control of Models and Calculations

Revisions to analytical and computer models that support proposed decommissioning activities would be verified to ensure they satisfy design requirements and solve the right problem (e.g., correctly model physical laws and implements system, structure, or component design rules).

Calculations that support proposed decommissioning activities would be completed, checked, reviewed, and approved prior to using their results. The process for developing calculations that support proposed decommissioning activities would require that calculations define the input data, assumptions, analytical methods, results, and conclusions. An independent reviewer would perform the verification of the correctness of the calculations including the validity of the input data and assumptions. The reviewer also would verify that any modeling of engineering barriers correctly models the design as described in the design documents. As stated above, computer programs would be proven through previous use, or verified through testing or simulation prior to use.

Written Procedures

The collection of engineering data and design, calculations, and modeling tasks must be performed properly and consistently in order to assure the quality of the final results. These tasks shall be performed in accordance with approved, written procedures. Such procedures would describe acceptable methods used for engineering tasks associated with proposed decommissioning and contain acceptance criteria to ensure that these tasks would be performed satisfactorily.

8.4 Document Control

Documents that come under the oversight of the QA Program include, but are not limited to, the QA Manual or Plan, technical and QA procedures, engineering data documents, engineering drawings, calculations, instrument calibration records, survey and characterization documents, contractor and subcontractor quality control records, and personnel training and qualification records.

Measures shall be established to control the issuance of documents that prescribe activities affecting quality, such as procedures and drawings and changes thereto. These measures shall address development of the documents by the responsible party. This would assure that documents, including changes, would be reviewed for adequacy and approved for release by authorized personnel, and would be distributed to and used at the location where the prescribed activity is to be performed. Changes to documents shall be reviewed and approved by the same organization that performed the original review and approval or by another designated responsible organization.

All QA documents would be developed, issued, revised, and retired according to the QA procedures developed for handling these documents. These QA procedures shall be controlled to assure that current copies would be made available to personnel performing the prescribed activities. Required procedures shall be reviewed by a technically competent

person other than the author, and shall be approved by a management member of the organization responsible for the prescribed activity. Significant changes to required procedures shall be reviewed and approved in the same manner as the original.

Documents affecting quality would be formally retired after their use has ended or after they are superceded by another project document. The QA Program would specify details of how this would be done.

8.5 Control of Measuring and Test Equipment

Measures shall be established to assure that tools, gauges, instruments, and other measuring and testing devices used in proposed decommissioning activities important to health and safety would be properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits. See Section 9 for a description of survey test and measuring equipment, maintenance and calibration requirements, calibration documentation, and daily check source measurements. Only properly calibrated and maintained equipment would be used for proposed decommissioning surveys and measurements. Documentation would be maintained to demonstrate that only properly calibrated and maintained equipment would be used; details of how this would be accomplished would appear in the QA Program.

8.6 Control of Purchased Material and Subcontractor Services

Measures shall be established to assure that purchased material, equipment, and services conform to the procurement documents. These measures shall include provisions, as appropriate, for vendor evaluation and selection, objective evidence of quality furnished by the vendor, inspection at the vendor source and inspection of products upon delivery.

The effectiveness of the control of contractor services shall be assessed at intervals consistent with the importance of the service. The adequacy of a vendor's QA program specified in procurement documentation shall be verified prior to use when appropriate. Vendors' adherence to their QA program shall also be verified as appropriate.

Commensurate with potential adverse impacts on quality or health and safety, material and equipment shall be inspected upon receipt at the WVDP site prior to use or storage to determine that the procurement requirements would be satisfied.

Materials, parts, or components that would be utilized for shipment of radioactive material shall be inspected upon receipt to assure that associated procurement document provisions have been satisfied. Measures shall be established for identifying nonconforming material, parts and components.

8.7 Corrective Action

Measures shall be established to assure that conditions adverse to quality such as failures, malfunctions, discrepancies, deviations, defective material and equipment, and non-conformances would be promptly identified and corrected. The identification of the condition adverse to quality, the cause of the condition and the corrective action taken shall be documented and reported to appropriate levels of management. All corrective actions shall

be reviewed and approved by the decommissioning organization line management and concurred with by the QA Manager.

8.8 Audits and Surveillances

The WVDP would perform assessments of proposed decommissioning work processes and operations through the WVDP decommissioning project organization self-assessments, audits, and surveillances. These may include, but would not be limited to, inspections/surveillances, tests, and QA audits.

The assessments would be provided by designated decommissioning project or qualified QA personnel who have sufficient authority and organizational independence to perform these assessments. These personnel would not have direct responsibilities in the areas they would be assessing. The assessments would provide (but not be limited to) the following:

- Methods to identify quality issues and problems;
- Recommendations for resolving quality issues and problems;
- Independent confirmation of resolutions and implementation of audit and surveillance findings by designated project or QA personnel;
- Tracking information on audit and surveillance findings and resolutions to trend quality issues and problems;
- Identification of improvements to proposed decommissioning project work processes, operations, procedures, and the QA Program from trending information;
- Audit and surveillance reports which document the items identified above, that would be managed and controlled by proposed decommissioning project procedures and designated project personnel;
- Information to line management and the QA Manager to ensure that further collection, analysis, or use of data would be controlled until the issue or problem is suitably resolved; and
- Information to line management and the QA Manager to ensure that further design, fabrication, construction, or operation of engineered features would be controlled until nonconforming, deficient, or unsatisfactory conditions have been suitably resolved.

8.9 Quality Assurance Records

Quality assurance records shall conform to the following requirements:

- Sufficient records shall be maintained to furnish evidence of activities affecting quality.
- Records shall be identifiable and retrievable.

- Measures shall be established which assure that qualification records of personnel performing special process activities, such as welding, nondestructive evaluation, inspection, etc., would be retained.
- Measures shall be established which assure that quality-related procurement documents would be retained.
- Measures shall be established which assure that appropriate records pertaining to audits would be retained.
- Measures shall be established which assure that records associated with radioactive material and personnel exposure controls would be retained.
- Requirements shall be established concerning record retention, such as duration, location, and assigned responsibility. Such requirements shall be consistent with the potential impact on quality, health and safety of the public, safety of project personnel, and applicable regulations.
- The QA Program would specify in particular where QA records would be stored during the proposed decommissioning and after the proposed decommissioning for the required retention period.
- QA records shall be periodically audited by the Decommissioning QA organization and stored in a designated QA records facility to be identified prior to implementation of this plan.

8.10 References

Code of Federal Regulations and Federal Register Notices

10 CFR 830.120, *Quality Assurance Requirements.*

DOE Orders, Policies, Manuals, and Standards

DOE Order 414.1C *Quality Assurance*. DOE, Washington, D. C., June 17, 2005.

Other References

ASME 2000, *Quality Assurance Program Requirements for Nuclear Facility Applications*, ASME NQA-1-2000. American Society of Mechanical Engineers, New York, 2000.

DOE 2005, *Uniform Federal Policy for Implementing Environmental Quality Systems: Evaluating, Assessing, and Documenting Environmental Data Collection/Use and Technology Programs*, DOE/EH-0667, Final Version 2. Intergovernmental Data Quality Task Force, Washington, D.C. March 2005.

NRC 2000, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, NUREG-1575, Rev 1. NRC, Washington D.C., August 2000.

SAIC 2008a, *Quality Assurance Project Plan for Preparation of (1) Decommissioning and/or Long-term Stewardship EIS for the WVDP and the Western New York Nuclear*

Services Center, (2) WVDP Decommissioning Plan Project, WV EIS/DP QAPP, Revision 1. SAIC, Germantown, Maryland, January 30, 2008.

SAIC 2008b, *Technical Procedure No. 1, Internal Code Development, Verification and Maintenance for (1) Decommissioning and/or Long-term Stewardship EIS for the WVDP and the Western New York Nuclear Services Center, (2) WVDP Decommissioning Plan Project*, WV EIS/DP TP1, Revision 1. SAIC, Germantown, Maryland, January 30, 2008.

SAIC 2008c, *Technical Procedure No. 2, Calculations and Analyses for (1) Decommissioning and/or Long-term Stewardship EIS for the WVDP and the Western New York Nuclear Services Center, (2) WVDP Decommissioning Plan Project*, WV EIS/DP TP2, Revision 1. SAIC, Germantown, Maryland, January 30, 2008.

SAIC 2008d, *Technical Procedure No. 3, Code Modification, Verification, and Maintenance for Externally Acquired Software for (1) Decommissioning and/or Long-term Stewardship EIS for the WVDP and the Western New York Nuclear Services Center, (2) WVDP Decommissioning Plan Project*, WV EIS/DP TP3, Revision 1. SAIC, Germantown, Maryland, January 30, 2008.

SAIC 2008e, *Technical Procedure No. 4, Software Configuration Management for (1) Decommissioning and/or Long-term Stewardship EIS for the WVDP and the Western New York Nuclear Services Center, (2) WVDP Decommissioning Plan Project, WV EIS/DP TP4*, Revision 1. SAIC, Germantown, Maryland, January 30, 2008.

WSMS 2006a, *Washington Safety Management Solutions LLC Quality Assurance Plan*, WSMS QA 100, Revision 0. Washington Safety Management Solutions, Aiken, South Carolina, November 13, 2006.

WSMS 2006b, *Preparation of WSMS Technical Documents for the West Valley Integrated Decommissioning and/or Long-Term Stewardship EIS and WVDP Decommissioning Plan Project*, WSMS-OPS-05-0004, Revisions 1. Washington Safety Management Solutions, West Valley, New York, August 21, 2006.

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9.0 FACILITY RADIATION SURVEYS

PURPOSE OF THIS SECTION

The purpose of this section is to describe radiation surveys to be performed in connection with Phase 1 of the WVDP proposed decommissioning.

INFORMATION IN THIS SECTION

This section first refers to the cleanup criteria for surface soil, subsurface soil, and streambed sediment that would be used to ensure that the level of remediation achieved during Phase 1 would not limit options for Phase 2 of the decommissioning. It then identifies the types of radiological surveys to be performed and the purpose of each survey. Requirements for background surveys, characterization surveys, in-process surveys, and the Phase 1 final status surveys are described.

This section outlines the survey process for each waste management area and then for environmental media. It concludes with a summary of requirements for the Phase 1 Status Survey Report.

While this section addresses all applicable requirements for facility radiation surveys, it does so in general terms because two supplemental documents would later be developed to provide additional details: a Characterization Sample and Analysis Plan and a Phase 1 Final Status Survey Plan (or multiple Phase 1 Final Status Survey Plans).

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;
- The information in Section 2 on facilities to be removed before the Phase 1 proposed decommissioning activities begin;
- The facility descriptions in Section 3;
- The information on the results of scoping and characterization surveys contained in Section 4 and Appendix B;
- The information in Section 5 on dose modeling and cleanup criteria; and
- The proposed decommissioning activities and related characterization activities described in Section 7.

The proposed characterization survey process described in this section applies to characterization surveys performed in connection with proposed decommissioning activities described in Section 7.

The survey methodology specified in this section is consistent with the provisions of NUREG-1757, Volume 2 (NRC 2006) and with the guidance found in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000). It is also consistent with DOE requirements of DOE Order 5400.5, *Radiation Protection of the Public and the Environment.*

As used in this section, the term *surveys* includes both systematic scanning and static measurements performed with an appropriately-sensitive instrument calibrated to the radiation of interest, as well as the laboratory analysis of physical samples of potentially contaminated media.

9.1 Release Criteria

Release criteria are based on the dose modeling described in Section 5 and the planned end-states for facilities and areas within the scope of the plan as discussed in Sections 1 and 7. The appearance of the Phase 1 end-state for the project premises would be similar to that shown in Figure 1-5. As explained in Section 5, derived concentration guideline levels (DCGLs) were developed for surface soil, subsurface soil and streambed sediment.

Note that DCGLs for the WVDP Phase 1 proposed decommissioning end state are expressed on the basis of 25 mrem total effective dose equivalent annually to the average member of the critical group. This annual dose is used as the basis for the cleanup criteria because the resulting DCGLs provide a conservative end state that ensures that all decommissioning options for the remainder of the project premises and the Center remain available in Phase 2.

DCGLs and Cleanup Goals

Because of the complexity of the site and the necessity to ensure that the Phase 1 proposed cleanup activities would support a range of approaches that might be used for Phase 2 of the decommissioning, cleanup goals lower than the DCGLs would be used as indicated in Section 7. These goals are identified in Table 5-14 of Section 5. The cleanup goals are referred to in this section simply as the DCGLs for consistency in terminology.

The $DCGL_W$ is the release criterion based on average concentration of radioactivity distributed over a large area. Area factors are used to adjust the $DCGL_W$ values to estimate the $DCGL_{EMC}$, the criterion for small areas of contamination elevated above the release criterion and to estimate the minimum detectable concentration for scanning surveys.

The $DCGL_W$ and $DCGL_{EMC}$ values (i.e., the cleanup goals) for 18 radionuclides of interest are expressed in Table 5-14 in Section 5. Tables 9-1, 9-2, and 9-2 provide ranges of area factors.

Table 9-1 Surface Soil Cleanup Goal Area Factors(1)

Nuclide	$DCGL_w$ 10,000 m² (pCi/g)	Area Factors ($DCGL_{EMC}/DCGL_W$)							
		5,000 m²	1,000 m²	500 m²	100 m²	50 m²	10 m²	5 m²	1 m²
Am-241	4.9E+01	1.0E+00	1.0E+00	1.9E+00	7.2E+00	1.1E+01	2.6E+01	3.8E+01	8.07E+01
C-14	3.1E+01	1.7E+00	4.3E+00	1.2E+01	1.2E+02	3.3E+02	2.9E+03	6.1E+03	3.06E+04
Cm-243	4.2E+01	1.0E+00	1.0E+00	1.4E+00	2.3E+00	2.6E+00	4.2E+00	6.3E+00	1.80E+01
Cm-244	9.4E+01	1.0E+00	1.0E+00	2.0E+00	9.2E+00	1.7E+01	5.4E+01	7.7E+01	1.31E+02
Cs-137	2.7E+01	1.1E+00	1.1E+00	1.2E+00	1.5E+00	1.6E+00	2.5E+00	3.8E+00	1.14E+01
I-129	5.8E-01	1.8E+00	1.0E+01	2.3E+01	1.3E+02	2.5E+02	1.3E+03	2.5E+03	1.27E+04
Np-237	9.6E-02	1.6E+00	8.1E+00	1.8E+01	9.8E+01	2.0E+02	9.7E+02	1.9E+03	9.31E+03
Pu-238	5.8E+01	1.0E+00	1.0E+00	2.0E+00	9.3E+00	1.7E+01	5.5E+01	7.9E+01	1.32E+02
Pu-239	5.2E+01	1.0E+00	1.0E+00	2.0E+00	9.3E+00	1.7E+01	5.5E+01	7.9E+01	1.34E+02
Pu-240	5.2E+01	1.0E+00	1.0E+00	2.0E+00	9.3E+00	1.7E+01	5.5E+01	7.9E+01	1.34E+02
Pu-241	1.6E+03	1.0E+00	1.0E+00	1.9E+00	7.4E+00	1.2E+01	2.7E+01	3.9E+01	8.25E+01
Sr-90	8.7E+00	1.7E+00	3.2E+00	6.5E+00	3.2E+01	6.3E+01	2.9E+02	5.7E+02	2.64E+03
Tc-99	2.9E+01	1.0E+00	1.0E+00	2.1E+00	1.0E+01	2.1E+01	1.0E+02	2.0E+02	1.02E+03
U-232	5.6E+00	1.6E+00	7.7E+00	1.6E+01	3.3E+01	3.6E+01	5.8E+01	8.7E+01	2.68E+02
U-233	2.0E+01	1.6E+00	8.3E+00	1.8E+01	9.9E+01	2.0E+02	9.8E+02	1.9E+03	8.87E+03
U-234	2.1E+01	1.6E+00	8.3E+00	1.8E+01	1.0E+02	2.0E+02	1.0E+03	2.0E+03	9.41E+03
U-235	1.4E+01	1.6E+00	8.1E+00	1.7E+01	7.8E+01	9.7E+01	1.5E+02	2.2E+02	6.53E+02
U-238	2.2E+01	1.6E+00	8.2E+00	1.8E+01	9.5E+01	1.8E+02	7.5E+02	1.1E+03	3.18E+03

NOTE: (1) From Table C-16 of Appendix C. The values in the second column are the cleanup goals (CG_W) from Table 5-14.

Table 9.2. Subsurface Soil Cleanup Goal Area Factors(1)

Nuclide	$DCGL_W$ 100 m² (pCi/g)	Area Factors ($DCGL_{EMC}/DCGL_W$)			
		50 m²	10 m²	5 m²	1 m²
Am-241	2.9E+03	1.4E+00	2.6E+00	3.6E+00	7.1E+00
C-14	1.9E+05	2.0E+00	9.8E+00	1.8E+01	9.1E+01
Cm-243	5.1E+02	1.1E+00	1.8E+00	2.7E+00	7.9E+00
Cm-244	8.8E+03	1.7E+00	4.1E+00	5.3E+00	7.5E+00
Cs-137	2.0E+02	1.1E+00	1.8E+00	2.7E+00	8.5E+00
I-129	1.9E+02	2.0E+00	9.5E+00	1.9E+01	9.3E+01
Np-237	1.7E+01	1.9E+00	9.3E+00	1.9E+01	9.1E+01
Pu-238	5.5E+03	1.7E+00	4.1E+00	5.3E+00	7.5E+00
Pu-239	5.0E+03	1.7E+00	4.2E+00	5.3E+00	7.6E+00
Pu-240	5.0E+03	1.7E+00	4.2E+00	5.3E+00	7.6E+00

Table 9.2. Subsurface Soil Cleanup Goal Area Factors(1)

Nuclide	$DCGL_W$ 100 m^2 (pCi/g)	Area Factors ($DCGL_{EMC}/DCGL_W$)			
		50 m^2	10 m^2	5 m^2	1 m^2
Pu-241	9.8E+04	1.4E+00	2.6E+00	3.6E+00	7.2E+00
Sr-90	1.4E+03	1.9E+00	8.1E+00	1.5E+01	6.5E+01
Tc-99	5.0E+03	2.0E+00	9.9E+00	2.0E+01	9.8E+01
U-232	5.3E+01	1.1E+00	1.9E+00	2.8E+00	8.8E+00
U-233	7.5E+02	1.9E+00	9.0E+00	1.8E+01	8.6E+01
U-234	7.7E+02	1.9E+00	9.1E+00	1.8E+01	8.8E+01
U-235	4.3E+02	1.1E+00	1.7E+00	2.6E+00	7.8E+00
U-238	8.2E+02	1.9E+00	6.8E+00	1.0E+01	2.9E+01

NOTE: (1) From Table C-47 of Appendix C. The values in the second column are the cleanup goals (CG_W) from Table 5-14.

Table 9-3. Streambed Sediment Cleanup Goal Area Factors(1)

Nuclide	$DCGL_W$ 1,000 m^2 (pCi/g)	Area Factors ($DCGL_{EMC}/DCGL_W$)					
		500 m^2	100 m^2	50 m^2	10 m^2	5 m^2	1 m^2
Am-241	1.6E+03	1.6E+00	3.0E+00	3.6E+00	5.8E+00	8.7E+00	2.5E+01
C-14	3.4E+02	2.2E+00	1.3E+01	2.8E+01	1.5E+02	3.0E+02	1.5E+03
Cm-243	3.6E+02	1.1E+00	1.2E+00	1.3E+00	2.0E+00	3.1E+00	9.1E+00
Cm-244	4.7E+03	2.0E+00	9.8E+00	1.9E+01	8.5E+01	1.6E+02	6.8E+02
Cs-137	1.3E+02	1.1E+00	1.2E+00	1.3E+00	2.1E+00	3.1E+00	9.4E+00
I-129	3.7E+02	2.0E+00	8.6E+00	1.5E+01	4.6E+01	7.7E+01	2.5E+02
Np-237	5.4E+01	1.7E+00	3.7E+00	4.6E+00	8.1E+00	1.2E+01	3.8E+01
Pu-238	2.0E+03	2.0E+00	9.9E+00	2.0E+01	9.2E+01	1.8E+02	8.1E+02
Pu-239	1.8E+03	2.0E+00	9.8E+00	1.9E+01	8.9E+01	1.7E+02	7.7E+02
Pu-240	1.8E+03	2.0E+00	9.9E+00	2.0E+01	9.3E+01	1.8E+02	8.4E+02
Pu-241	5.2E+04	1.6E+00	3.0E+00	3.7E+00	6.0E+00	9.0E+00	2.5E+01
Sr-90	9.5E+02	1.9E+00	7.1E+00	1.1E+01	2.7E+01	4.4E+01	1.4E+02
Tc-99	2.2E+05	1.8E+00	5.1E+00	7.0E+00	1.4E+01	2.1E+01	6.4E+01
U-232	2.7E+01	1.0E+00	1.2E+00	1.3E+00	2.0E+00	3.0E+00	9.5E+00
U-233	5.8E+03	1.9E+00	7.7E+00	8.7E+00	1.3E+01	2.0E+01	6.0E+01
U-234	6.1E+03	2.0E+00	9.2E+00	1.7E+01	6.2E+01	1.1E+02	4.0E+02
U-235	2.9E+02	1.0E+00	1.2E+00	1.3E+00	1.9E+00	2.9E+00	8.6E+00
U-238	1.3E+03	1.1E+00	1.4E+00	1.5E+00	2.3E+00	3.5E+00	1.1E+01

NOTE: (1) From Table C-75 of Appendix C. The values in the second column are the cleanup goals (CG_W) from Table 5-14.

A *surrogate radionuclide* is a radionuclide in a mixture of radionuclides whose concentration is more easily measured and can be used to infer the concentrations of the other radionuclides in the mixture. If actual radioactive contamination levels are below the specified concentrations of the surrogate radionuclide, then the sum of doses from all radionuclides in the mixture would fall below the dose limit of 25 mrem/y. Tables in Section 5 do not presently show $DCGL_W$ values for a surrogate radionuclide because available data on radionuclide distributions in soil and sediment are not sufficient to support this, but Section 5 may be revised after additional characterization data become available to provide such information.

As characterization and in-process surveys are performed, additional data would become available that could necessitate re-evaluation of the DCGLs, if, for example, assumptions used in development of the DCGLs were found to be incorrect based on the additional data. If such a situation develops, revised DCGLs would be calculated and this plan changed to incorporate the revised DCGLs and any related changes.

9.2 Types of Surveys and Their Purposes

Seven types of radiological surveys are associated with the WVDP Phase 1 proposed decommissioning project: (1) background surveys, (2) scoping surveys, (3) end-of-task surveys taken at the conclusion of deactivation activities, (4) characterization surveys, (5) in-process or remedial action support surveys, (6) Phase 1 final status surveys, and (7) confirmatory surveys. The nature of these surveys and, in some cases, the basic requirements are summarized here; more detail is provided further below on background surveys (9.3), characterization surveys (9.4), in-process surveys (9.5), and Phase 1 final status surveys (9.6).

9.2.1 Background Surveys

Background surveys are performed in non-impacted areas around the facility and in non-impacted buildings of construction similar to those impacted buildings of interest. Background surveys establish the baseline levels of radiation and radioactivity from radionuclides occurring in the environment or incorporated into the structural materials. Requirements for background surveys are summarized in Section 9.3 below.

9.2.2 Scoping Surveys

Scoping surveys are conducted (1) to provide preliminary data to supplement historical site assessment information needed to guide planning of characterization surveys, (2) to identify radionuclide contaminants, (3) to identify relative radionuclide ratios, and (4) to identify the general levels and extent of contaminants. As noted in Section 4, much of the existing radiological data associated with the WVDP proposed decommissioning project falls into the category of scoping survey data, although these data were generally not acquired as scoping survey data but were acquired for other operational needs. Additional scoping surveys are not planned for Phase 1 of the WVDP proposed decommissioning.

9.2.3 End-of-Task Surveys

As explained in Section 1, additional deactivation work will be completed in certain areas of the Process Building during deactivation work to be accomplished before the Phase 1 proposed decommissioning activities begin, and numerous ancillary project facilities will be removed during this period. After each area is deactivated and after each facility is removed, end-of-task or "final radiological characterization" surveys will be performed to define the resulting radiological conditions.

Such surveys are not within the scope of this plan since they will be completed before proposed decommissioning activities begin. However, their results will be considered in connection with defining characterization surveys and Phase 1 final status surveys to be performed during the proposed decommissioning.

9.2.4 Characterization Surveys

Characterization surveys include facility and site sampling, monitoring, and analysis activities to determine the extent and nature of residual contamination. They provide the basis for planning decommissioning actions, and providing technical information to develop, evaluate, and select appropriate remediation techniques. They also provide information for radiation protection purposes and for characterizing waste.

Four WVDP characterization survey programs have been completed: (1) the characterization program for the underground waste tanks, (2) the Facility Characterization Project, (3) a series of Resource Conservation and Recovery Act (RCRA) facility investigations performed in the 1990s, and (4) investigations of the north plateau groundwater plume using a Geoprobe®.[1] Additionally, routine groundwater and other environmental media sampling and analysis are performed as required by DOE Orders for annual monitoring programs. The results of these programs are summarized in Section 4. The approaches used are outlined in Section 9.7 below.

As indicated in Section 4 and Section 7, additional characterization would be performed in connection with proposed decommissioning fieldwork. The requirements for this characterization are addressed in Section 9.4.

9.2.5 In-Process Surveys

In-process surveys, also referred to as remedial action support surveys, include facility and site sampling, monitoring, and analysis activities performed in support of decontamination work. They provide information necessary for radiation protection, for guiding cleanup work, for determining when field decontamination goals have been attained, and to indicate when areas are ready for Phase 1 final status surveys. Requirements for in-process surveys are discussed in Section 9.5 below.

[1] As indicated in Section 4, additional characterization of subsurface soil in the area of the north plateau groundwater plume is being undertaken in 2008. The results of this program will become available in 2009.

9.2.6 Final Status Surveys

A final status survey using MARSSIM guidance is performed to demonstrate completion of any necessary decontamination in preparation for release of the site or facility. To reflect the phased nature of the proposed decommissioning, this plan uses the terminology "Phase 1 final status" rather than "final status". Because the decision to release or a final decision on status of the Phase 1 decommissioned areas would not be made until during Phase 2 decision making, using the terminology "final status" alone could be misinterpreted. The Phase 1 final status surveys consist of measurements and sampling to describe the radiological conditions at the close of Phase I proposed decommissioning activities. The intent is that Phase 1 final status surveys would be designed with quality, quantity and statistical objectives such that the data could be used in a MARSSIM-based "final status" evaluation in the future without a need to re-survey the areas, unless subsequent site activities influence the status. Requirements for the Phase 1 final status surveys are addressed in Section 9.6 below.

9.2.7 Confirmatory Surveys

Confirmatory surveys include limited, independent third-party measurements, sampling, and analysis to verify the results of the licensee's final status survey. Typically, confirmatory surveys conducted by NRC or its contractor consist of two components: (1) a review of the licensee's final status survey plan and report to identify any deficiencies in the planning, execution, or documentation, and (2) measurements taken at a small percentage of locations, previously surveyed by the licensee, to determine whether the licensee's results are valid and reproducible. (Note that while DOE is performing the Phase 1 final status surveys as part of its responsibilities under the WVDP Act, DOE is not the licensee for any part of the Center.).

DOE anticipates that NRC will arrange for independent in-process surveys to be performed after Phase 1 proposed decommissioning work in an area is completed. DOE also anticipates that confirmatory surveys will be performed on an area basis after the Phase 1 final status survey has been completed for that area, a strategy that experience shows to be more efficient that a single confirmatory survey at the conclusion of the project. An *area* in this context may be a group of related survey units or an entire waste management area (WMA).

To facilitate NRC in-process and confirmatory surveys, DOE would:

- Keep NRC informed of the schedule and status of decommissioning activities and the Phase 1 final status survey,
- Notify NRC when particular areas are to be ready for confirmatory surveys, and
- Prepare the portion of the Phase 1 Final Status Survey Report that addresses survey results section-by-section and provide to NRC in draft form sections that describe

DOE survey results for those areas in which NRC is to perform confirmatory surveys. Experience has shown that this practice promotes efficiency.[2]

9.3 Background Surveys

Some information on background radiation and radioactivity in non-impacted areas is available, such as that contained in annual site environmental reports (WVES and URS 2008) and that described in Section 4. Additional background measurements would be taken in connection with characterization surveys outlined in Section 9.4. These would include exposure rates and samples from non-impacted soil and building materials in appropriate background reference areas.

Applicable guidance in the MARSSIM (NRC 2000) and in NUREG-1505 (Gogolak, et al. 1997) would be considered. The background surveys would be described in detail in the Characterization Sample and Analysis Plan.

9.4 Characterization Surveys

As noted above, four formal characterization survey programs have been completed for portions of the project premises, routine sampling and analysis are performed annually, and additional characterization surveys would be performed in connection with Phase 1 proposed decommissioning activities. Characterization surveys performed in connection with Phase 1 proposed decommissioning activities would be described in more detail in a Characterization Sample and Analysis Plan that DOE or its contractor would issue prior to the decommissioning.

Characterization to be accomplished in connection with proposed decommissioning activities would be planned with the following objectives and guidance.

9.4.1 Characterization Sample and Analysis Plan Content

This plan would provide details of characterization surveys to be performed to more precisely determine the extent and the amount of residual radioactivity as proposed decommissioning activities begin.

Requirements and Guidance to be Followed

This plan would follow provisions in NUREG-1757 Volume 2 (NRC 2006) and applicable guidance of the MARSSIM (NRC 2000).

Radionuclides of Interest and Radionuclide Ratios

This plan would identify the radionuclides of interest. It would also address the variability of radionuclide ratios across the site and identify areas where the ratios need to be confirmed for use in the Phase 1 final status survey analysis.

[2] As explained in Section 9.8, DOE and the decommissioning contractor may choose to prepare multiple Phase 1 final status survey reports because of the site complexity. In this case, complete draft reports could be provided to NRC in support of the confirmatory surveys.

Waste Acceptance Criteria

This plan would identify waste acceptance criteria for those waste disposal sites proposed to be used to establish the context for the characterization measurements.

Hazardous and Toxic Materials of Interest

This plan would identify hazardous and toxic materials to be considered during the characterization, along with the applicable concentration limits, unless characterization for hazardous and toxic materials is addressed by a separate program.

Data Quality Objectives

This plan would identify data quality objectives (DQOs) for the characterization surveys, as discussed in Section 9.4.2.

Use of Characterization Data for Final Status Survey Purposes

A key objective of this plan would be to produce data for the Phase 1 final status survey of sufficient quality and quantity to serve final status survey purposes when practicable, and this matter would be addressed in the Characterization Sample and Analysis Plan.

Background Radiation and Radioactivity

The Characterization Sample and Analysis Plan would specify appropriate measurements in reference areas for materials and structures to establish background levels, taking into account available data on background radioactivity provided in Section 4, in Appendix B, and that compiled in connection with the WVDP environmental monitoring program.

Characterization Methods for Radioactivity

This plan would specify the methods to be used to collect the necessary characterization data. Among the methods considered would be:

- Exposure rate measurements
- Surface contamination scans
- Surface contamination direct measurements
- Smear surveys for removable contamination
- Debris samples (and/or smears or metal coupons) analyzed for radionuclides of interest
- Concrete surface samples including paint
- Concrete core samples
- Surface and core samples of other materials
- Soil samples
- Water samples
- Sediment samples

Other, more technically sophisticated characterization methods may be used as well, such as *in-situ* gamma spectroscopy and advanced characterization technologies that DOE has helped develop. Any new technology or instrumentation to be used would be shown to perform with sensitivities that allow detection of residual radioactivity at an appropriate fraction of the DCGLs and corresponding investigative levels.

Radiological Instrumentation

The Characterization Sample and Analysis Plan would specify the field and laboratory instruments to be used and the sensitivity of these instruments and methods. Table 9-4 shows typical field instruments to be addressed in the plan.

Table 9-4. Radiological Field Instruments

Survey Type	Instrument (or equivalent)	Characteristics	Approximate Sensitivity(2)	Remarks
Exposure rate	Eberline RO-7(1)	Ion chamber	> 1 R/h	For high-range readings.
Exposure rate	Eberline RO-2(1)	Ion Chamber	0.1 mrem/h	For low-range readings
Exposure rate	Bicron Micro Rem	Organic scintillator	Several μrem/h	For scanning soil, low potential areas.
Exposure rate	Ludlum 44-10(1)	2-inch NaI scintillator	900 cpm/μR/h	For scanning soil, low potential areas.
Exposure rate	FIDLER	5-inch diameter NaI scintillator	500 cpm per uCi/m^2	For scanning soil for low energy gamma
Alpha	Ludlum 43-89(1)	ZnS (Ag) scintillator, 100 cm^2 probe	100 dpm/100 cm^2 85 dpm/100 cm^2	Scans 100 dpm, direct measurements 85 dpm.
Beta	Ludlum 43-89(1)	ZnS (Ag) scintillator, 100 cm^2 probe	2,500 dpm/100 cm^2 800 dpm/100 cm^2	Scans 2,500 dpm, direct measurements 800 dpm.
Beta-gamma	Ludlum 44-40(1)	Geiger-Mueller (G-M) shielded pancake probe	3,300 cpm/mrem/h	For scanning in tight areas
Beta-gamma	Ludlum 44-9(1)	G-M unshielded pancake probe	3,300 cpm/mrem/h	For scanning in tight areas
Beta-gamma	Ludlum 44-6(1)	G-M sidewall detector	1,200 cpm/mrem/h	For use as a pipe probe

NOTES: (1) To be used with an appropriate scaler-ratemeter.
(2) These are approximate values based primarily on manufacturer's ratings. The sensitivities depend on background, count time, and other factors. Calculated, more precise information would be specified in the Characterization Sample and Analysis Plan.

Samples may be analyzed onsite or shipped to an offsite contract laboratory for analysis. Laboratory methods, instruments and sensitivities would be in accordance with New York State protocols for environmental analysis. Any laboratory used for environmental sample analysis would have appropriate New York State Department of Health Environmental Laboratory Approval Program certification, or equivalent.

Characterization Methods for Hazardous and Toxic Materials

This plan would specify methods used to determine the presence of hazardous and toxic materials, such as analysis for lead or polychlorinated biphenyls in paint through direct measurement by x-ray fluorescence or sampling for analysis in a laboratory, unless such surveys are covered in a separate characterization program.

Survey Locations

This plan would specify how to locate and identify sampling and measurement locations, such as how to lay out and mark appropriate size survey grids. Grid control points and positions of samples and survey readings within the grid would be located using global position system devices or conventional surveying. Class 1, Class 2, and Class 3 survey units are discussed in Section 9.6.1.

Surveys and Sampling of Individual Facilities and Areas

This plan would specify the type and extent of characterization measurements in different facilities and areas.

Surveys of Inaccessible Areas

The plan would address how areas that are inaccessible or difficult to access would be evaluated.

Characterization of Removed Materials

Characterization measurements would include those necessary for waste management purposes and the Characterization Sample and Analysis would specify the applicable requirements and guidance for characterization of materials. The decommissioning contractor would also provide a procedure for characterizing materials for waste management purposes and obtain DOE approval of this procedure. This procedure would be consistent with applicable DOE requirements and guidance, as well as any applicable State-specified waste acceptance criteria for radioactivity in the offsite landfill(s) where uncontaminated material may be disposed of. It would apply to, among other materials, surface and subsurface soil not known to have been impacted by radioactivity.

Handling Waste Generated During Characterization

The Characterization Sample and Analysis Plan would specify how to minimize and manage investigative derived waste.

Health and Safety

This plan would identify health and safety requirements associated with characterization activities; it may reference the project Health and Safety Plan for this purpose.

Quality Assurance

The Characterization Sample and Analysis Plan would address quality control and quality assurance requirements for characterization, addressing matters identified in Section 9.4.3 and referring to the Quality Assurance Project Plan as appropriate.

Supporting Procedures

This plan would specify necessary supporting procedures, such as those for obtaining, handling, preserving, and packaging samples, as well as chain of custody procedures.

Documentation

This plan would detail the requirements for formally documenting characterization data in a written report.

9.4.2 Characterization Data Quality Objectives

The Data Quality Objectives for the characterization would be detailed in the Characterization Sample and Analysis Plan; they may be briefly stated as follows:

The Problem

Available characterization data in many areas are insufficient to support proposed decommissioning activities and waste characterization and, in some cases, planning for radiation protection.

The Decision

The principal study question is what additional radiological data are needed for proposed decommissioning activities, waste management, and radiation protection. The decision statement may be expressed as follows: if collection of additional data is warranted, collect data of sufficient quality and quantity to support proposed decommissioning activities, waste characterization and/or planning for radiation protection.

Inputs to the Decision

Inputs to the decision include: (1) available data on radiological conditions; (2) professional judgment concerning data necessary to support the proposed decommissioning activities, waste management, and radiation protection; and (3) available characterization measurement methods to collect necessary additional data, such as using field instruments to determine exposure rates and contamination levels and obtaining samples of materials and having them analyzed in a laboratory.

Study Boundaries

The study boundaries include:

- The characteristics of the contaminants of interest: Various radionuclides known to be present at the site from reprocessing of spent nuclear fuel and the hazardous and toxic materials that may be present based on facility history and process history, along with the physical parameters of the facilities and areas involved, such as size, geometry, and material composition.
- The spatial boundary of the decision statement: The facilities and areas within the scope of the DP, including soil from the surface to a depth of six inches (15 cm) from the surface and, when contamination is present, down to a depth indicating the bound of sub-surface impacts.

- The temporal boundary of the problem: The data can be acquired any time before the beginning of proposed decommissioning activities in the facility or areas involved, so long as sufficient time is allowed to make preparations based on the data. Data inside facilities can be acquired without regard to conditions such as weather, temperature, and wind. Measurements and sampling in outside areas are dependent on the weather.
- Scale of decision-making: Areas of interest would generally conform to particular areas to undergo decommissioning, i.e., decisions would be made on specific areas or survey units, rather than the project premises as a whole.
- Practical constraints on data collection: These include limited access to certain areas, radiation exposure to those collecting data, availability of personnel and equipment, laboratory capabilities and capacity, and costs. Another constraint is the risk of releasing contamination to the environment and introducing new environmental contamination transport mechanisms.

Decision Rule

The decision rule on whether or not to collect data in particular areas and how much data to collect would be addressed in the Characterization Sample and Analysis Plan. It would involve the use of project experience and professional judgment to determine the adequacy of available data and the type and extent of any additional data needed.

Limits on Decision Errors

The conclusion that a facility or area has been adequately characterized is subject to the possibility of a decision error. Decisions are based on data subject to different variabilities due to choices on sample number, location, collection, and analysis. The acceptable probability of making a decision error on the adequacy of the characterization (false positive and false negative) would be addressed.

Optimizing the Design

The content of the Characterization Sample and Analysis Plan would reflect an optimum design based on the various factors considered in its preparation, including the matters discussed above.

9.4.3 Characterization Quality Requirements

The quality requirements of Section 8 would apply to characterization. The following matters would also be addressed in the Characterization Sample and Analysis Plan.

Quality Objectives for Measurements

Objectives for precision, bias, completeness, representativeness, reproducibility, comparability and statistical confidence (control charts) would be addressed.

Field Instruments

Field instruments would be calibrated in accordance with written procedures using standards traceable to the National Institute of Standards and Technology. They would be

calibrated every six months and following any substantial repair. Battery status, check source response, and background measurements would be performed prior to use each day, at the completion of use each day, and any time that instrument operation is in question. Control charts with specified limits of acceptability would be used to document and trend source response and background measurements.

Laboratory Instruments

Laboratory instruments such as alpha spectrometers, gamma spectrometers, low-background alpha-beta counters, and liquid scintillation counters would also be calibrated in accordance with written procedures using standards traceable to the National Institute of Standards and Technology. Appropriate operational checks such as background counts and reproducibility checks would be performed before use. Control charts with specified limits of acceptability would be used to document and trend source and background checks.

Offsite analytical laboratories would be required to meet all applicable quality requirements; the laboratory Quality Assurance Plan would be reviewed to ensure that applicable requirements are included. Offsite laboratories would be audited to assure quality performance.

Sample Chain of Custody

Sample chain of custody procedures would be established and followed to ensure that sample accountability and integrity are maintained. This process would include appropriate documentation utilized from the point of collection to the point where the sample is consumed in analysis, transferred to another organization, or properly disposed of.

Analytical Quality Control

Quality controls utilized for analytical chemical processes would include:

- Maintaining the quality of standards,
- Maintaining controls over sample flow,
- Controlling batch quality using method blanks,
- Using laboratory control standards traceable to the National Institute of Standards and Technology or using other industry-accepted standards or reference materials,
- Formally evaluating unacceptable results, and
- Utilizing process control charts as appropriate.

Data Quality Control

Data would be recorded in a legible manner and reviewed for matters such as accuracy of recording and transcription, procedure compliance, completeness, and consistency. Calculations would be checked and conclusions would be peer reviewed. Problems identified would be resolved before the data are utilized. Data reports and documents would be archived and maintained to comply with the Project Quality Assurance Program described in Section 8.

9.5 In-Process Surveys

In-process or remedial action support surveys would be performed while remediation is in progress to guide decontamination and determine when remediation to field goals (the cleanup goals specified in Section 5) has been attained. In-process surveys also support radiation protection.

Measurement methods and instruments used would be similar to those typically utilized during characterization and final status surveys. Where practicable, correlations between gamma exposure rates and soil radioactivity concentrations would be used to help determine when removal of target soil has been completed and to demonstrate that the instrument scan and direct measurement sensitivities are sufficient for the purpose of the in-process survey. Data reports and documents would be archived and maintained to comply with the Project Quality Assurance Program described in Section 8.

9.6 The Phase 1 Final Status Survey

As indicated previously, the Phase 1 final status survey would be accomplished in accordance with a Phase 1 Final Status Survey Plan(s). Because the proposed decommissioning work spans a significant period of time and area of the site, the Phase 1 final status survey efforts may be more readily described and controlled in several area-specific or survey unit-specific plans rather than a single, more complex plan. The use of the DQO process in the project planning cycle would ensure consistency in the design, execution, and evaluation of Phase 1 Final Status Survey Plans if multiple plans are developed.

This Phase 1 Final Status Survey Plan(s) would have an integrated design incorporating:

- Analysis of media samples from systematic positions to determine the average concentration of activity distributions in relatively large areas, and
- Surface scanning meter surveys to identify localized areas of elevated activity.

9.6.1 Phase 1 Final Status Survey Plan Content

The Phase 1 Final Status Survey Plan(s) would provide details of the Phase 1 final status surveys to demonstrate that residual radiological conditions satisfy the cleanup criteria described in Section 9.1 or to document final radiological conditions as indicated below. (The plan elements described below would apply to all Phase 1 Final Status Survey Plans if multiple plans are prepared.)

Requirements and Guidance to be Followed

The Phase 1 Final Status Survey Plan would follow provisions in NUREG-1757 Volume 2 (NRC 2006) and guidance of the MARSSIM (NRC 2000).

Overview of Survey Design

This plan would provide a brief overview of the survey design. This design would closely follow NUREG-1757 Volume 2 (NRC 2006) and the MARSSIM (NRC 2000), utilizing statistical tests to determine adequate sample density.

Radionuclides of Interest

This plan would specify the radionuclides of interest identified in Section 9.1, considering that all radionuclides may not be of interest in certain areas.

Designating Residual Radioactivity Limits and Investigative Levels

This plan would identify the cleanup criteria specified in Section 5. It would also identify investigative levels and how they were established.

Use of Characterization Data for Phase 1 Status Survey Purposes

As indicated previously, DOE plans to produce characterization data of sufficient quality to serve Phase 1 final status survey purposes when practicable for areas that appear to meet the cleanup criteria without the need for remediation, and this matter and the data of interest would be addressed in the Phase 1 Final Status Survey Plan.

Consideration of In-Process Survey Data

Any useful available data compiled during in-process surveys would be summarized in the Phase 1 Final Status Survey Plan and its use to estimate survey unit variance and confirm survey unit classification would be addressed.

Additional Radioactivity Not Accounted For During Characterization

If any radioactivity from licensed or WVDP operations is not accounted for by characterization performed previously or in connection with proposed decommissioning activities, this would be identified in the Phase 1 Final Status Survey Plan.

Classification of Areas

Different areas of the project premises facilities and areas of interest would be classified based on potential for radioactive contamination. Four classifications would be used:

Class 1: impacted areas that, prior to remediation, are expected to have concentrations of residual radioactivity that exceed the $DCGL_W$;

Class 2: impacted areas that, prior to remediation, are not likely to have concentrations of residual radioactivity that exceed the $DCGL_W$;

Class 3: any impacted areas that have a low probability of containing residual radioactivity; and

Non-impacted: areas without reasonable potential for radioactive contamination from licensed or WVDP activities.

Impacted areas are identified in Section 4 based on information available in 2008. Preliminary classification would be confirmed or adjusted based on subsequent characterization and in-process survey data.

Survey Units

Survey units are geographical areas of specified size and shape for which a separate decision would be made as to whether or not that area exceeds the regulatory limit. Areas within a survey unit would have a similar usage history and contamination potential and be contiguous areas of the same area classification.

Survey units would be specified in the Phase 1 Final Status Survey Plan. They would be identified in tables or drawings or a combination of the two. Among areas considered in designating survey units would be:

- Exposed surface areas of the WMA 1 and WMA 2 excavations before they are back-filled;
- Exposed surface areas of the excavations following removal of foundations and floor slabs; and
- Surface soil and stream sediment throughout the project premises.

In some survey units, data from characterization would be sufficient for Phase 1 final status survey purposes; this matter would be addressed in the Phase 1 Final Status Survey Plan.

Background Radiation and Radioactivity

Appropriate measurements would be taken in non-impacted background reference areas to establish background levels, taking into account available data on background summarized in Section 4, in Appendix B, that compiled in connection with the WVDP environmental monitoring program, and that collected during characterization. Media background would be subtracted from Phase 1 final status survey results.

Data Quality Objectives

Data Quality Objectives for the Phase 1 final status survey would be established as indicated in Section 9.6.2.

Survey Methods

The methods to be used to collect the necessary data in Phase 1 final status surveys would be similar to methods used in characterization surveys discussed previously. Among these are:

- Surface contamination scans,
- Direct measurements for contamination,
- Exposure rate measurements, and
- Soil and/or other media samples.

The Phase 1 Final Status Survey Plan would incorporate performance-based measurement systems, specifying the analytical sensitivity goal of each survey method. Individual methods (i.e., static surface counts) would then be translated to field procedures (instrument, detector, geometry, and count time) to assure attainment of the sensitivity required. Information necessary to perform the surveys and sampling, such as procedures for collecting and preparing samples, would be specified. Other survey methods may be used in support of the methods specified above, such as gamma scans to help identify locations of soil samples.

Radiological Instrumentation

This plan would specify the field and laboratory instruments to be used and the sensitivity of these instruments and methods. Table 9-5 shows typical field instruments to be addressed in the plan.

Table 9-5. Radiological Field Instruments for Phase 1 Final Status Survey

Survey Type	Instrument (or equivalent)	Characteristics	Approximate Sensitivity(1)	Remarks
Exposure rate	Bicron Micro Rem	Organic scintillator	Several µrem/h	For scanning soil.
Exposure rate	Ludlum 44-10	2-inch NaI scintillator	900 cpm/µR/h	For scanning soil.
Exposure Rate	FIDLER	5-inch diameter NaI scintillator	500 cpm per µCi/m^2	For scanning soil for low energy gamma

NOTE: (1) These are approximate values based primarily on manufacturer's ratings. The sensitivities depend on background, count time, and other factors. Calculated, more precise information would be specified in the Phase 1 Final Status Survey Plan.

The Phase 1 Final Status Survey Plan would specify how the minimum detectable concentration (MDC) for media samples and the MDC for scanning surveys (MDC_{scan}) would be determined for each instrument and technique using methods specified in NUREG-1757, Volume 2 (NRC 2006). It would also demonstrate that the instrument scan and direct measurement sensitivities are consistent with MARSSIM (NRC 2000) guidance and sufficient for the goals of the Phase 1 final status survey.

The laboratory instruments and methods to be utilized would also be addressed in the Phase 1 Final Status Survey Plan, along with the minimum detectable concentrations of the methods used. Instruments and methods are expected to be similar to those shown in Table 9-7.

Scan Surveys

Scan surveys of survey units of the different classifications would be performed as indicated in Table 9-6 below. The purpose of such scan surveys is to identify small areas of elevated activity.

Table 9-6. Scan Surveys for Different Survey Area Classifications

Classification	Scanning Required	Scanning Investigative Levels
Class 1	100% coverage(1)	$>DCGL_{EMC}$
Class 2	10-100% coverage(2)	$>DCGL_W$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_W$.
Class 3	Judgmental	$>DCGL_W$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_W$.
Non-impacted	None	Not applicable.

NOTES: (1) Entire surface of soil areas (and exposed building floor slabs and foundations, if any).
(2) Surveys would be both systematic and judgmental.

The derivation of scan and fixed MDCs would take into account instrument efficiencies (surface and detector), scan rates and distances over surfaces, surveyor efficiency, and minimum detectable count rate, using guidance in the MARSSIM (NRC 2000) and NUREG-1507 (Abelquist, et al. 1998).

Sample Collection and Handling

A brief description of how samples are to be collected, controlled, and handled would be provided, with reference to the detailed procedure(s) to be used for this purpose.

Survey Grids

Survey grids of appropriate size would be laid out and marked on excavations and land areas. Where practicable, grids established for characterization surveys would be re-established for use in the Phase 1 final status survey. Grid control points and positions of samples and survey readings within the grid would be located using global position system devices or conventional surveying.

Surrogate Radionuclides

Surrogate measurements focusing on Cs-137 may be used in areas where the radionuclide mix in a survey unit is consistent and Cs-137 is one of the dominant radionuclides. The Phase 1 Final Status Survey Plan would specify how this would be done in particular areas.

Surveys and Sampling of Individual Facilities and Areas

This plan would specify the process to determine the number of samples required in different areas following MARSSIM protocols. This process would include the following elements:

- Developing DQOs consistent with the requirements in Section 9.6.2,
- Utilizing as the null hypothesis (H_o) to be tested the assumption that the residual contamination exceeds the release criteria with the alternative hypothesis (H_A) being that the residual contamination meets the release criteria,
- Determining the relative shift – a ratio involving the difference between the $DCGL_W$ and the field remediation concentration goal divided by the variability in the concentration across the survey unit following remediation,
- Determining acceptable decision errors,
- Determining the number of samples needed for the Wilcoxon rank sum test (for radionuclides present in background),
- Determining the number of samples needed for the Sign test (for radionuclides not present in background), and
- Determining the number of additional samples needed if the MDC_{scan} is greater than the $DCGL_W$.

Evaluation of Results and Determination of Compliance

The measurement data would be first reviewed to confirm that the survey units were properly classified. In any cases where the results show that an area was misclassified with a less restrictive classification, the areas would be reclassified correctly, and a survey appropriate to the new classification would be performed.

Whether the measurement results demonstrate that the survey unit meets the release criteria would then be determined. The process for this and the statistical tests to be used would be specified in the Phase 1 Final Status Survey Plan, taking into account the multiple radionuclides present at the site and the different radionuclide distributions present in some areas.

If compliance is not demonstrated, then additional remediation followed by additional Phase 1 final status surveys would be performed until the release criteria are achieved.

Two radionuclides (I-129 and Np-237) in surface soil would be treated as special cases because their cleanup goals are lower than the minimum detectable concentrations in typical laboratory sample analyses. Section 7 of the MARSSIM indicates that the analytical detection limits should be 10-50 percent of the DCGL, but that higher detection sensitivities may be acceptable when lower limits are impracticable (NRC 2000). Because these two radionuclides should not appear in background soil samples, analysis at a detection limit near the DCGL would be sufficient to flag results should a sample indicate the presence of either radionuclide above its detection limit.

The Phase 1 Final Status Survey Plan would provide an alternate method for evaluating analytical results for these radionuclides that do not exceed the minimum detectable concentrations. This alternate method may involve use of an easy to detect surrogate radionuclide prevalent in surface soil, such as Cs-137 or Am-241, to infer the concentration of I-129 and Np-237. Scaling factors for spent fuel reprocessed specified in Table 4-1 would be suitable for this purpose. Another suitable alternate evaluation method could involve larger soil volumes and longer counting times for representative samples to reduce the minimum detectable concentration to a value below the cleanup goal.

The amounts of I-129 and Np-237 that might be found in surface soil contamination, if any, would be small. This conclusion is based on comparisons between the estimated amounts of these radionuclides at the site at the conclusion of spent fuel reprocessing compared to the estimated amounts of predominant radionuclides such as Sr-90 and Cs-137. Table 2-5 in Section 2 shows estimates for the radionuclide content of the underground waste tanks at the completion of reprocessing. This table shows the estimated amount of I-129 to be more than seven orders of magnitude less than the estimated Cs-137 present, with the estimated amount of Np-237 more than six orders of magnitude less that the estimated Cs-137 amount.[3]

[3] Although Np-237 is produced during radioactivity decay of Am-241, this factor is accounted for in the RESRAD model, which accounts for the progeny of the radionuclides of interest.

Health and Safety

This plan would identify health and safety requirements associated with survey activities; it may reference the project Health and Safety Plan for this purpose.

Quality Assurance

The Phase 1 Final Status Survey Plan would address quality control and quality assurance requirements for characterization, addressing matters identified in Section 9.6.3 and in Section 8, referring to the project Quality Assurance Plan as appropriate.

Supporting Procedures

This plan would specify necessary supporting procedures, such as those for obtaining and managing samples.

Documentation

This plan would detail the requirements for formally documenting and archiving Phase 1 final status survey data, in accordance with the requirements of Section 9.8.

9.6.2 Data Quality Objectives for the Phase 1 Final Status Survey

The DQOs would be detailed in the Phase 1 Final Status Survey Plan; they would involve considerations such as:

- Stating the problem: Provide adequate data of sufficient quality to determine the extent and magnitude of residual radioactive contamination.
- Identifying the decision: Will the data generated be adequate to support all survey objectives?
- Identifying inputs to the decision: Available data, including final characterization data obtained in connection with deactivation, information needed, measurement methods that would produce necessary data.
- Defining the study boundaries. Radionuclides of interest, areas of interest, necessity to obtain data to support the proposed decommissioning schedule, appropriate-sized units, limited access to certain areas, availability of personnel and equipment, laboratory analysis throughput.
- Developing a decision rule. How to make the judgment as to whether or not additional data would need to be collected.
- Specifying limits on decision error. Consider the consequences of inadequate survey data and express what is acceptable in this regard.
- Optimizing the design. Data quality assessment would be used to determine the validity and performance of the data collection design and determine the adequacy of the data set to support the decision.

9.6.3 Phase 1 Final Status Survey Quality Requirements

The quality requirements of Section 8 would apply, along with the quality requirements for the characterization survey as identified in Section 9.4.3. These matters would be addressed in the Phase 1 Final Status Survey Plan.

9.7 The Survey Process By Waste Management Area

This section outlines surveys completed and surveys to be accomplished in each WMA (9.7.1 through 9.7.11) and, separately, surveys completed and planned for environmental media across the project premises (9.7.12). Note that other considerations such as proposed decommissioning activities in adjacent areas and the impact of routes for transportation of radioactive materials on survey units and area classification would be addressed as appropriate in the Phase 1 Final Status Survey Plan(s).

9.7.1 WMA 1 Process Building and Vitrification Facility Area

Characterization surveys of the Process Building and Vitrification Facility have been performed in connection with the Facility Characterization Project. However, because radiological conditions in most building areas would change during deactivation work performed before the start of the proposed decommissioning, additional surveys would be performed as proposed decommissioning activities begin. Characterization of the contaminated soil in WMA 1 that is the source for the north plateau groundwater plume is addressed in Section 4.2; surveys related to its remediation are addressed in Section 9.7.12 below.

The Facility Characterization Project

As noted previously, the Facility Characterization Project focused on development of conservative source term estimates for various areas of the Process Building and Vitrification Facility. It followed the MARSSIM (NRC 2000) process and was carried out in accordance with the WVNSCO Characterization Management Plan (Michalczak 2004).

Description of Previous Survey Measurements. The primary process for determining the source term in a particular area involved using exposure rate measurements to quantify the amount of a surrogate gamma-emitting radionuclide such as Cs-137, and using scaling ratios to estimate the amounts of other radionuclides present. Scaling ratios were based on sample analysis, process knowledge, or other bounding assumptions. In some cases, samples were collected and the analytical results were used in calculating a source term based on surface area or volumetric computations.

The process entailed four basic steps: (1) collection and evaluation of existing data and preparation of a draft technical approach, (2) review of these data and the proposed approach by a Technical Review and Approval Panel, (3) collection of any needed data and modeling to estimate the source term, and (4) review and concurrence on the estimated source term by the Panel. Where additional data were needed, a biased sampling approach was used that typically involved field measurements such as radiation and contamination levels, along with samples of materials analyzed in a laboratory. Radiation level measurements were typically taken with a Geiger-Mueller

detector (Ludlum Model 133-6) or ion chamber (Eberline RO-20) attached to a scaler/rate meter. Smears were counted with a Tennelec gas-flow proportional counter. Detection sensitivities for the exposure rate instruments were approximately 0.1 mrem/h for the RO-20 and higher for the Model 133-6, whose scales range from 1 mR/h to 1000 R/h.

Due to the high activity associated with most of the samples, samples taken in connection with the project were analyzed in the former onsite Analytical and Process Chemistry Laboratory. Table 9-7 shows laboratory instruments and methods, along with their sensitivities.

Table 9-7. Laboratory Methods

Nuclide	Instrument/Method	WVDP Procedure	Approximate Sensitivity(1)
Am-241	Alpha and/or gamma spectrometry	ACM-2707/3104	1.0 E-05 µCi/g
C-14	Sample oxidizer and liquid scintillation	ACM-4904	1.0 E-02 µCi/g
Cm-234/244	Alpha and/or gamma spectrometry	ACM-2707/3104	1.0 E-03 µCi/g
Cs-137	Gamma spectrometry	ACM-3103/3104	1.0 E-03 µCi/g
I-129	Gamma spectrometry	ACM-3104	1.0 E-03 µCi/g
Np-237	Alpha and/or gamma spec	ACM-2707/3104	1.0 E-03 µCi/g
Sr-90	Liquid scintillation	ACM-2707/3002	1.5 E-05 µCi/g (1g sample)
Tc-99	Gas flow proportional counting	ACM-4001	1.0 E-06 µCi/g (1g sample)
Pu-238	Alpha spectrometry	ACM-2704	1.0 E-05 µCi/g
Pu-239/240	Alpha spectrometry	ACM-2704	1.0 E-05 µCi/g
Pu-241	Liquid scintillation	ACM-2707/2708	1.0 E-05 µCi/g
U-232	Alpha spectrometry	ACM-2707	1.0 E-05 µCi/g
U-233/234	Alpha spectrometry	ACM-2707	1.0 E-05 µCi/g
U-235 (-236)	Alpha spectrometry	ACM-2707	1.0 E-05 µCi/g
U-238	Alpha spectrometry	ACM-2707	1.0 E-05 µCi/g

NOTES: (1) Dependent on sample size, counting time, etc.

Formal quality assurance requirements were implemented. Data quality objectives following the MARSSIM (NRC 2000) process were used. Data collected were compiled into individual reports for the area or facility. Each report included a discussion of available historical data, the approach used to gather additional data, and the conservatively bounding source term estimate, along with all the supporting information.

Justification for Previous Survey Measurements. The focus on conservative source terms supported one of the decommissioning alternatives envisioned by DOE when the Facility Characterization Project began. This alternative would have entailed leaving most of the Process Building and Vitrification Facility in place beneath a multi-layer cap.

The focus on source term estimates rather than general radiological conditions produced information important to the performance assessment under this alternative.

The process for collection and evaluation of historical data was similar to that used for historical site assessments. Data acquired during the effort were obtained following MARSSIM quality protocols. However, these data are being treated as scoping survey data in some cases because of their limited extent.

Process Building and Vitrification Facility Characterization Surveys

In connection with proposed decommissioning activities in each area, characterization measurements would be taken as specified in the Characterization Sample and Analysis Plan. The measurements would take into account data from deactivation end-of-task surveys and fill in data gaps for areas where these surveys were not performed. Characterization measurements would be performed on the WMA 1 facilities commensurate with plans for their disposition, which is removal in each case. As indicated in Section 7, there are no plans to release these facilities from radiological controls before dismantlement or demolition, which limits characterization data needs.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination. Samples would be analyzed for specific radionuclides to confirm radionuclide distributions where such information is not already available and to provide information for radiation protection and waste characterization. Areas inaccessible to surveys would be exposed so surveys can be made where indicated in the Characterization Sample and Analysis Plan.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support planning decommissioning activities and waste management.

Process Building and Vitrification Facility In-Process Surveys

In-process surveys would be performed during remediation as specified in Section 9.5.

Process Building and Vitrification Facility Area Phase 1 Final Status Surveys

As explained previously, the final end-state of the Process Building and Vitrification Facility would involve total removal including excavation of the subsurface portions, back-filling with soil, and installing a vertical hydraulic barrier wall on the down-gradient side of the excavation footprint. Phase 1 final status surveys would be performed for exposed subsurface areas before they are backfilled; this matter would be addressed in the Phase 1 Final Status Survey Plan, which would provide details of the surveys required.

Process Building and Vitrification Facility Area Confirmatory Surveys

After Phase 1 final status surveys are completed, arrangements would be made to have any desired confirmatory surveys performed.

Characterization of Other WMA 1 Facilities

The other facilities to remain within WMA 1 after 2008 that may have been impacted by radioactivity are: (1) the 01-14 Building, (2) the Plant Office Building, (3) the Utility Room, and (4) the Utility Room Expansion. Because these facilities would be entirely or partially within the bounds of the planned excavation, characterization measurements would be performed

on these WMA 1 facilities commensurate with plans for their disposition, which is removal in each case. As indicated in Section 7, there are no plans to release these facilities from radiological controls before dismantlement or demolition, which limits characterization data needs.

Routine WVDP surveys taken through mid-2008 in these areas have typically not shown removable contamination above detection limits. However, contamination from the major acid spill during NFS operations that produced the north plateau groundwater plume is known to be present beneath the floor in the men's shower room of the Plant Office Building. And some areas in the 01-14 Building, such as areas on the third and fourth floor that contain ventilation system equipment, are not routinely surveyed.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination. Representative embedded piping in the 01-14 Building floor slab, except for sealed floor drains, would be characterized, with measurements such as (1) total beta using a suitable pipe probe (such as a Ludlum 44-6 sidewall detector) in the exposed ends of the pipe, (2) removable alpha and beta contamination in the ends of the pipe by smears, and (3) exposure rates on the accessible piping. (Note that some equipment would be removed from the 01-14 Building during deactivation.)

Characterization is not planned for the non-impacted facilities in WMA 1 – the Fire Pump House and water tank and the electrical substation.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support planning decommissioning activities and waste management.

In-Process Surveys of Other WMA 1 Facilities

In-process surveys would be performed during remediation as described in Section 9.5. However, the scope of such surveys would be minimal because of the relative low potential for contamination, except in some areas of the 01-14 Building which may contain significant contamination.

Phase 1 Final Status Surveys in Other WMA 1 Facilities

As all facilities within the Process Building excavation would be removed, the Phase 1 final status surveys would be surveys of the excavation surface in accordance with the Phase 1 Final Status Survey Plan.

Confirmatory Surveys in Other WMA 1 Areas

After Phase 1 final status surveys are completed, arrangements would be made to have any desired confirmatory surveys of these areas performed.

Characterization of Subsurface Piping in WMA 1

DOE has evaluated contaminated underground piping (Luckett, et al. 2004). This evaluation produced conservative source term estimates based on existing data, but it did not include characterization measurements. Subsurface piping within the bounds of the WMA

1 excavation would be removed, packaged and disposed of at offsite disposal facilities. There is no intent in Phase 1 of the proposed decommissioning to trace or excavate underground piping outside the bounds of the excavation.

When these lines become exposed during the course of proposed decommissioning work, measurements would be taken as necessary, for instance for waste characterization purposes for lines removed or to provide data to support Phase 2 decision-making for portions of lines remaining in place.

Description of Survey Measurements. The measurements would be taken after the interior surfaces of the lines are exposed during the course of proposed decommissioning work. Three types of measurements would be taken as appropriate: (1) total beta using a suitable pipe probe (such as a Ludlum 44-6 sidewall detector) in the exposed ends of the pipe, (2) removable alpha and beta contamination in the ends of the pipe by smears, and (3) exposure rates on the accessible piping. Where sufficient data on radionuclide distributions are not available, smears or metal coupons would be obtained and analyzed to determine the radionuclide distributions.

Justification for Survey Measurements. These measurements would provide information on interior contamination levels that would support radiation protection, waste management, and subsequent disposition determinations. The lines have a constant downward slope and ones that carried higher concentrations of radioactive liquid are made of stainless steel. This design makes contamination traps unlikely and contamination levels in areas where piping would be cut are expected to be representative of the entire length. Line 7P120 that carried THOREX waste from the Chemical Process Cell to Tank 8D-4 is expected to contain the most residual radioactivity.

In-Process Surveys Related to Subsurface Piping in WMA 1

In-process surveys would be performed during removal of piping as described in Section 9.5. Some characterization surveys would effectively be in-process surveys since they would be performed in conjunction with piping removal activities.

Phase 1 Final Status Surveys of Subsurface Piping in WMA 1

Separate Phase 1 final status surveys of the piping not encountered during excavation and subsequently abandoned in place are not planned; characterization survey data are intended to serve Phase 1 final status survey purposes.

Confirmatory Surveys of Subsurface Piping in WMA 1

Arrangements would be made for any confirmatory surveys NRC desires to be accomplished at the time when the piping ends are accessible prior to excavation backfilling.

9.7.2 WMA 2 Low-Level Waste Treatment Facility Area

Of the facilities to remain within WMA 2 after 2008 that have been impacted by radioactivity, significant characterization data are available for only one: the Old Interceptor.

Only limited data on radiological conditions are available for the others within the scope of the plan: (1) the LLW2 Building, (2) the Neutralization Pit, (3) the Solvent Dike, (4) the twin New Interceptors, and (5) the North Plateau Groundwater Pump and Treat Facility.

Note that the five lagoons in WMA 2 are addressed as environmental media in Section 9.7.12 below.

Existing Characterization Data for Old Interceptor

Description of Previous Survey Measurements on Old interceptor. Two radiation surveys taken in 2003 show levels up to 408 mrem/h (WVNSCO 2003a and WVNSCO 2003b)[4].

Justification for Previous Survey Measurements. While these surveys provided useful information, they did not completely characterize the facility, which is expected to contain contamination in depth and contamination covered by a layer of concrete added to the floor.

Characterization of WMA 2 Facilities

Characterization measurements would be performed on the WMA 2 facilities commensurate with plans for their disposition, which is removal in each case. As indicated in Section 7, there are no plans to release these facilities from radiological controls before dismantlement or demolition, which limits characterization data needs.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, total contamination, and core samples of facility surfaces in cases where they would produce information of value. Smears or samples of building materials would be obtained and analyzed to provide information on radionuclide distributions.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

In-Process Surveys of WMA 2 Area

In-process surveys would be performed during remediation as described in Section 9.5. These surveys would include the surface of the soil in excavations made during removal of the interceptors, the Neutralization Pit, and the associated valve pits.

Phase 1 Final Status Surveys in WMA 2 Areas

After proposed decommissioning activities are completed in these areas, Phase 1 final status surveys would be performed in each survey unit in accordance with the Phase 1 Final Status Survey Plan. These surveys would include the exposed soil in the large excavation made to remove Lagoons 1-3, the interceptors, the Neutralization Pit, and Solvent Dike. Also

[4] Although no radioisotope inventory report was issued for the Old Interceptor; these radiation surveys were taken for characterization purposes for the Facility Characterization Project.

considered in the Phase 1 final status surveys would be the exposed soil surfaces from removal of remaining floor slabs and foundations of facilities removed prior to the start of decommissioning: the 02 Building, the Test and Storage Building, the Vitrification Test Facility, the Maintenance Shop, the Maintenance Storage Area, the Vehicle Maintenance Shop, and the Industrial Waste Storage Area. Phase 1 final status surveys would also be performed in the excavation to remove the Maintenance Shop leach field equipment.

Confirmatory Surveys in WMA 2 Areas

After the Phase 1 final status surveys are completed, arrangements would be made to have confirmatory surveys performed. NRC or its contractor would be afforded an opportunity to perform confirmatory surveys in excavations before they are filled in.

Characterization of Subsurface Piping in WMA 2

Underground piping within WMA 2 is comprised primarily of Duriron wastewater drain lines leading to the Interceptors and interconnecting with equipment in the treatment buildings, the interceptors, and the lagoons. Also within WMA 2 is a portion of the Leachate Transfer Line from the NRC-Licensed Disposal Area (NDA).

Subsurface piping within the bounds of the WMA 2 excavations would be removed, packaged and disposed of at offsite disposal facilities. There is no intent in Phase 1 of the proposed decommissioning to trace or excavate underground piping outside the bounds of the excavations.

When these lines become exposed during excavation of the WMA 2 Facilities, during removal of the LLW2 Building floor slab and foundations, and during removal of Lagoons 4 and 5, measurements would be taken as necessary, for instance for waste characterization purposes for lines removed or to provide data to support Phase 2 decision-making for portions of lines remaining in place.

Description of Survey Measurements. Measurements would be taken after the interior surfaces of the lines are exposed when the lines are cut. Two types of measurements would be taken: (1) removable alpha and beta contamination in the end of the pipe measured by smears, and (2) exposure rates of the accessible piping.

Justification *for Survey Measurements*. These measurements would provide information to support for waste characterization purposes and to support decision-making for Phase 2 of the proposed decommissioning. .

In-Process Surveys Related to Subsurface Piping in WMA 2

In-process surveys during excavation as subsurface piping is encountered during remediation would be performed as specified in Section 9.5.

Phase 1 Final Status Surveys of Subsurface Piping in WMA 2

Separate Phase 1 final status surveys of the piping not encountered during excavation and subsequently abandoned in place are not planned; characterization survey data are intended to serve Phase 1 final status survey purposes.

Confirmatory Surveys of Subsurface Piping in WMA 2

Arrangements would be made for any confirmatory surveys NRC desires to be accomplished at the time when the piping ends are accessible, prior to the excavation being filled in.

9.7.3 WMA 3, Waste Tank Farm Area

Four facilities or groups of equipment within WMA 3 have been impacted by radioactivity and are within the scope of the plan: (1) the mobilization and transfer pumps in Tanks 8D-1, 8D-2, 8D-3, and 8D-4, (2) the piping and equipment in the HLW transfer trench, (3) the Equipment Shelter and Condensers, and (4) the Con-Ed Building. Limited data on radiological conditions are available for these facilities and this equipment as indicated in Section 4.

WMA 3 Facility Characterization Surveys

Characterization measurements would be performed in connection with proposed decommissioning activities.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination in areas of interest.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

WMA 3 Facility In-Process Surveys

In-process surveys would be performed during remediation as specified in Section 9.5.

WMA 3 Facility Phase 1 Final Status Surveys

After proposed decommissioning activities are completed in this area, Phase 1 final status surveys would be performed in accordance with the Phase 1 Final Status Survey Plan. Procedures and detection levels for scan surveys may be modified due to the higher ambient radiation levels in the area from radioactivity in the HLW tanks.

WMA 3 Confirmatory Surveys

Arrangements would be made for any confirmatory surveys desired by NRC or its contractor.

WMA 4, Construction and Demolition Debris Landfill

This landfill, which was closed in 1986, is not within the scope of the Phase 1 decommissioning work.

9.7.4 WMA 5 Waste Storage Area

Facilities within WMA 5 impacted by radioactivity and within the scope of the plan are the Remote Handled Waste Facility and Lag Storage Addition 4 and its associated Shipping

Depot. Other facilities in WMA 5 within the scope of the plan are concrete pads and foundations remaining from facilities removed prior to the start of decommissioning.

Characterization of the Remote Handled Waste Facility

Characterization measurements would be performed in this building commensurate with plans for its disposition, which is removal.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination. Representative smears would be analyzed for radionuclides of interest.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

In-Process Surveys Related to the Remote Handled Waste Facility

In-process surveys would be performed during remediation as specified in Section 9.5.

Phase 1 Final Status Surveys of the Remote Handled Waste Facility Excavation

As explained previously, this facility would be completely removed. After proposed decommissioning activities are completed, including demolition and removal of the floor slab and foundation and removal of the empty underground tank vault, Phase 1 final status surveys on the exposed excavation surface would be performed in accordance with the Phase 1 Final Status Survey Plan.

Confirmatory Surveys of the Remote Handled Waste Facility Excavation

After the Phase 1 final status surveys are completed, arrangements would be made to have any desired confirmatory surveys accomplished by the NRC or its contractor.

Characterization of Subsurface Piping in WMA 5

Within WMA 5 is underground piping running from the Remote-Handled Waste Facility to Tank 8D-3. Portions of this piping within the bounds of the building excavation would be removed, packaged and disposed of at offsite disposal facilities. As indicated in Section 7, the portion of the piping outside of the building excavation would remain in place unless it has been impacted by radioactivity.

When these lines become exposed during excavation to remove the Remote-Handled Waste Facility, measurements would be taken to confirm the radiological status for waste characterization purposes for lines removed and to provide data to support Phase 2 decision-making for the portions of the piping to remain in place.

Description of Survey Measurements. Measurements would be taken after the interior surfaces of the lines are exposed when the lines are cut. Two types of measurements would be taken: (1) removable alpha and beta contamination in the end of the pipe measured by smears, and (2) exposure rates of the accessible piping.

Justification for Survey Measurements. These measurements would provide information to support for waste characterization purposes and to support decision-making for Phase 2 of the proposed decommissioning.

In-Process Surveys Related to Subsurface Piping in WMA 5

In-process surveys during excavation as subsurface piping is encountered during remediation would be performed as specified in Section 9.5.

Phase 1 Final Status Surveys of Subsurface Piping in WMA 5

Separate Phase 1 final status surveys of the piping not encountered during excavation and subsequently abandoned in place are not planned; characterization survey data are intended to serve end Phase 1 final status survey purposes.

Confirmatory Surveys of Subsurface Piping in WMA 5

Arrangements would be made for any confirmatory surveys NRC desires to be accomplished at the time when the piping ends are accessible, prior to the excavation being filled in.

Characterization of Lag Storage Addition 4/Shipping Depot

Characterization measurements would be performed in this building commensurate with plans for its disposition, which is removal.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

In-Process Surveys Related to Lag Storage Addition 4/Shipping Depot

In-process surveys would be performed during remediation as specified in Section 9.5.

Phase 1 Final Status Surveys of the Lag Storage Addition 4/Shipping Depot Excavation

As explained previously, these facilities would be completely removed. After proposed decommissioning activities are completed in this area, including demolition and removal of the floor slab and foundation, Phase 1 final status surveys on the exposed excavation surface would be performed in accordance with the Phase 1 Final Status Survey Plan.

Confirmatory Surveys of the Lag Storage Addition 4/Shipping Depot Excavation

After Phase 1 final status surveys are completed, arrangements would be made to have any desired confirmatory surveys accomplished by the NRC or its contractor.

Phase 1 Final Status and Confirmatory Surveys of Other Floor Slabs and Foundations

Also considered in the Phase 1 final status surveys and confirmatory surveys would be the soil surfaces exposed following excavations of remaining floor slabs and foundations of

impacted facilities removed prior to the start of decommissioning. The facilities of interest are the Lag Storage Building and its additions, the Chemical Process Cell Waste Storage Area, and several hardstands and gravel pads.

After surveys specified in the Phase 1 Final Status Survey Plan are completed, the areas of interest would be made available to NRC or its contractor for any desired confirmatory surveys.

9.7.5 WMA 6 Central Project Premises

In WMA 6, the facilities to be removed during Phase 1 include the Sewage Treatment Plant, the Equalization Tank, the Equalization Basin, the two demineralizer sludge ponds, and the south Waste Tank Farm Test Tower, along with remaining floor slabs and foundations, including the underground structure of the Cooling Tower. The Equalization Basin and the two demineralizer sludge ponds are addressed along with other environmental media in Section 9.7.12.

Characterization of the Remaining Part of the Cooling Tower

The only WMA 6 structure known to have been impacted by radioactivity as of 2008 is the remaining part of the Cooling Tower. Characterization measurements would be performed in this structure commensurate with plans for its disposition, which is removal.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination. Representative smears would be analyzed for radionuclides of interest.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

Phase 1 Final Status and Confirmatory Surveys Following Removal of Floor Slabs and Foundations

After the structures and their floor slabs and foundations have been removed, the exposed soil surface of the resulting excavations would be considered in the Phase 1 final status surveys. After surveys specified in the Phase 1 Final Status Survey Plan are completed, the areas of interest would be made available to NRC or its contractor for any desired confirmatory surveys.

Phase 1 Final Status Surveys of Equalization Tank Excavation

Even though the equalization tank was not known to be impacted by radioactivity in mid-2008, as indicated in Section 7, Phase 1 final status surveys would be performed in the excavation made to remove the tank as a good practice. These surveys would be performed as specified in Phase 1 Final Status Survey Plan and would typically include measurements with a sensitive gamma detector.

After surveys specified in the Phase 1 Final Status Survey Plan are completed, the area would be made available to NRC or its contractor for any desired confirmatory surveys.

9.7.6 WMA 7 NDA and Associated Facilities

No additional characterization would be performed in the NDA itself. Table 4-10 summarizes the estimated NDA radionuclide inventory. In WMA 7, only removal of concrete and gravel pads associated with the NDA Hardstand are within the scope of this plan.

WMA 7 Facility Characterization Surveys

Characterization measurements of the hardstand would be performed in connection with proposed decommissioning activities.

Description of Planned Survey Measurements. Measurements would typically include exposure rates and material samples analyzed for radionuclides of interest.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

WMA 7 In-Process Surveys

In-process surveys would be performed during remediation as specified in Section 9.5.

WMA 7 Phase 1 Final Status Surveys

Surveys of the resulting exposed excavation surfaces would be performed in accordance with the Phase 1 Final Status Survey Plan.

WMA 7 Confirmatory Surveys

Arrangements would be made for any confirmatory surveys desired by NRC or its contractor before the excavation is filled in.

9.7.7 WMA 8, State Licensed Disposal Area

There are no facilities within WMA 8 that are within plan scope.

9.7.8 WMA 9, Radwaste Treatment System Drum Cell Area

Phase 1 proposed decommissioning activities in WMA 9 include total removal of the building, floor slabs and foundations of the Radwaste Treatment System Drum Cell, the NDA trench soil container area, and the subcontractor maintenance area.

Characterization of the Radwaste Treatment System Drum Cell Area

Characterization measurements would be performed in this building commensurate with plans for its disposition, which is removal. Characterization measurements would also be taken in the trench soil container area and the subcontractor maintenance area.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

In-Process Surveys Related to the Radwaste Treatment System Drum Cell

In-process surveys would be performed during removal activities as specified in Section 9.5.

Phase 1 Final Status Surveys of the Radwaste Treatment System Drum Cell

Following building demolition and removal of the floor slab and foundation, Phase 1 final status surveys on the exposed excavation surface would be performed in accordance with the Phase 1 Final Status Survey Plan.

Confirmatory Surveys of the Radwaste Treatment System Drum Cell Excavation

After Phase 1 final status surveys are completed, arrangements would be made to have any desired confirmatory surveys accomplished.

The NDA Trench Soil Container Area and the Subcontractor Maintenance Area

Characterization measurements would be performed in these areas commensurate with plans for their disposition, which is removal.

Description of Planned Survey Measurements. Measurements would typically include exposure rates and soil samples analyzed for radionuclides of interest.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support proposed decommissioning activities and waste management.

Other surveys of this area would include in-process surveys in accordance with Section 9.5, Phase 1 final status survey of the excavations in accordance with the Phase 1 Final Status Survey Plan, and any confirmatory surveys desired by the regulators.

9.7.9 WMA 10, Support and Services Area

Neither of the facilities within WMA 10 within plan scope, the New Warehouse and the former Waste Management Storage Area, nor the remaining concrete floor slabs and foundations to be removed, had been impacted by radioactivity as of mid-2008.

WMA 10 Facility Characterization Surveys

Characterization measurements would be performed in these facilities, floor slabs, and foundations in connection with proposed decommissioning activities.

Description of Planned Survey Measurements. Measurements would typically include exposure rates, removable contamination, and total contamination.

Justification for Planned Survey Measurements. These are the appropriate measurements necessary to facilitate radiation protection and support decommissioning activities and waste management.

WMA 10 Facility In-Process Surveys

In-process surveys would be performed during remediation as specified in Section 9.5.

WMA 10 Facility Phase 1 Final Status Surveys

Phase 1 final status surveys on the exposed excavation surfaces would be performed in accordance with the Phase 1 Final Status Survey Plan.

Limited Phase 1 final status surveys would be performed in the Security Gatehouse as a good practice because of the proximity of this facility to the Process Building. These surveys would be judgmental in scope and include scan surveys with a sensitive gamma detector such as a Bicron Micro Rem instrument.

Confirmatory Surveys of WMA 10 Facilities

Arrangements would be made for any confirmatory surveys desired by NRC or its contractor.

9.7.10 WMA 11, Bulk Storage Warehouse and Hydrofracture Test Well Area

No facilities in WMA 11 are within plan scope. Neither characterization nor Phase 1 final status surveys are planned in this area.

9.7.11 WMA 12, Balance of the Site

No facilities in WMA 12 are within plan scope. Neither characterization nor Phase 1 final status surveys are planned in this area.

9.7.12 Environmental Media

Environmental media to be considered includes soil, sediment, groundwater, and surface water on the project premises.

Existing Characterization Data

Description of Previous Survey Measurements. As explained in Section 4.2, existing data on radioactivity in environmental media comes from three principal sources: (1) the site environmental monitoring program, (2) a series of RCRA facility investigations completed in the mid-1990s, and (3) Geoprobe® investigations of the north plateau groundwater plume. Data are also available on surface radiation levels that are indicative of soil contamination in some areas from 1984 and earlier aerial surveys and a 1990 overland survey that measured gamma radiation levels.

As explained in Section 4.2, data on radioactivity in environmental media were obtained using methods such as laboratory analysis of soil and groundwater samples and measuring exposure rates using sensitive gamma detectors.

Justification for Previous Survey Measurements. The measurements were made for several purposes, including regular monitoring of the environment and specific investigations related to hazardous materials and the north plateau groundwater plume.

Soil and Sediment Characterization Surveys

Surface soil, subsurface soil, and sediments in the Phase 1 areas would be surveyed and sampled for laboratory analysis. Subsurface soil in the non-source area of the plume, in the plume impacted areas, and Phase 2 areas would not be addressed at this time.

Description of Survey Measurements. The process to be utilized would include:

- Consideration of available characterization data;
- The use of marked grids, such as 100 feet by 100 feet, in areas where systematic measurements are made;
- Surface scans for gamma activity in areas likely to contain residual contamination;
- Surface soil samples;
- Subsurface soil samples where indicated by contamination potential, including locations of subsurface features such as tanks and process lines;
- Additional subsurface samples in the top portion of the Lavery till in the WMA 1 and WMA 2 excavation footprints as specified in Section 7.2.2; and
- Sediment samples where indicated by contamination potential, including sediment in Erdman Brook and the portion of Franks Creek within the project premises security fence.

Special attention would be paid to the lagoons, basins, and discharge ponds, including the area of Lagoon 1 where previously buried radioactive debris would be removed. The experience of other DOE sites such as Mound, Fernald, and Ashtabula that have extensive experience with contaminated soil characterization would be considered. Details would appear in the Characterization Sample and Analysis Plan.

Justification for Survey Measurements. These measurements would provide information on soil and sediment contamination to support decontamination activities, facilitate radiation protection, and waste disposal plans.

Phase 1 Final Status Surveys of Soil Areas and Areas Containing Sediment

Description of Survey Measurements. Phase 1 final status surveys would be performed as specified in the Phase 1 Final Status Survey Plan in the excavation made to remove the Equalization Basin and the two demineralizer sludge ponds. Remediation of surface soil and streambed sediment may also be accomplished in Phase 1, as explained in Section 7. If this is done, Phase 1 final status surveys of the remediated areas would be performed. The process to be utilized would be similar to that for characterization surveys, with details included in the Phase 1 Final Status Survey Plan. The same grids would be reestablished and used where practicable. Characterization data would be considered in the survey design and used for Phase 1 final status survey purposes where practicable.

Justification for Survey Measurements. These measurements would provide information on soil and sediment contamination to demonstrate that release criteria are achieved as applicable.

Confirmatory Surveys of Soil Areas and Areas Containing Sediment

Arrangements would be made for confirmatory surveys by NRC or its contractor after the Phase 1 final status surveys are completed.

Groundwater

Radioactivity in groundwater would continue to be monitored during Phase 1 of the proposed decommissioning by laboratory analysis of samples drawn from the network of monitoring wells. Appendix D addresses monitoring of groundwater following the completion of Phase 1 proposed decommissioning activities. No separate characterization or Phase 1 final status surveys would be performed for groundwater.

Surface Water/Stream Sediment

Radioactivity in surface water and associated stream sediment would continue to be monitored during the decommissioning in connection with the environmental monitoring and control program outlined in Section 1.8 and Appendix D. No separate characterization or Phase 1 final status surveys would be performed.

9.8 Phase 1 Final Status Survey Report Requirements

The requirements for Phase 1 Final Status Survey Report would be identified in the Phase 1 Final Status Survey Plan. As indicated previously, because of the site complexity there may be multiple Phase 1 Final Status Survey Plans. Consequently there may be multiple Phase 1 Final Status Survey Reports. The content and coverage of the plans and reports would be determined using the DQO Process in the project planning cycle. These report requirements would include the following.

9.8.1 Overview of Results

The report would summarize the results of the surveys.

9.8.2 Discussion of Changes

The report would include a discussion of any changes that were made in the Phase 1 final status survey from what was proposed in this plan or other prior submittals.

9.8.3 Description of How Numbers of Samples Were Determined

The report would include a description of the method by which the number of samples was determined for each survey unit.

9.8.4 Sample Number Determination Values

The report would include a summary of the values of site parameters and data statistics used to determine the number of samples and a justification for these values.

9.8.5 Results for each Survey Unit

The report would include the survey results for each survey unit, including:

- The number of samples taken for the survey unit;
- A map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units and random locations shown for Class 3 survey units and reference areas;
- The measured sample concentrations;

- The statistical evaluation of the measured concentrations;
- Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation;
- A discussion of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of $DCGL_W$ and any actions taken to reduce them, if any, upon detection[5]; and
- A statement that a given survey unit satisfied the $DCGL_W$ and the elevated measurement comparison if any sample points exceeded the $DCGL_W$.

9.8.6 Survey Unit Changes

The report would include a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity.

9.8.7 Actions Taken for Failed Survey Units

If a survey unit fails, a description of the investigation conducted to ascertain the reason for the failure and a discussion of the impact that the failure has on the conclusion that the facility is ready for Phase 1 final radiological surveys would be included in the report.

9.8.8 Impact of Survey Unit Failures

For any survey units that fail, the report would include a discussion of the impact that the reason for the failure has on other survey unit information.

9.9 References

DOE Orders, Policies, Manuals, and Standards

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

Other References

Abelquist, et al. 1998, *Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, NUREG–1507. Abelquist, E., W. Brown, and G. Powers, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1998.

Gogolak, et al. 1997, *A Nonparametrical Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*, NUREG-1505, Revision 1. Gogolak, C.V, G. Powers, and A. Huffert, U.S. Nuclear Regulatory Commission, Washington, DC, 1997.

Luckett, et al. 2004, *Radioisotope Inventory Report for Underground Lines and Low Level Waste Tanks at the West Valley Demonstration Project*, WSMS-WVNS-04-0001,

[5] This would include application of the as low as reasonably achievable (ALARA) principal as discussed in Section 6.

Revision 0. Luckett, L.W., J. Fazio, and S. Marschke, Washington Safety Management Solutions, West Valley, New York, July 6, 2004.

Michalczak 2004, *Characterization Management Plan for the Facility Characterization Project*, WVDP-403, Revision 3. West Valley Nuclear Services Company, West Valley, New York, January 16, 2004.

NRC 2000, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, NUREG-1575, Revision 1. NRC, Washington, DC, August, 2000. (Also EPA 4-2-R-97-016, Revision 1, U.S. Environmental Protection Agency and DOE-EH-0624, Revision 1, DOE)

NRC 2006, *Consolidated NMSS Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria, Final Report,* NUREG 1757 Volume 2, Revision 1. U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, DC, September, 2006.

WVNSCO 2003a, *Radiological Survey Report 120396.* West Valley Nuclear Services Company, West Valley, New York, June 11, 2003.

WVNSCO 2003b, *Radiological Survey Report 1121097.* West Valley Nuclear Services Company, West Valley, New York, August 4, 2003.

WVES and URS 2008, *West Valley Demonstration Project Annual Site Environmental Report, Calendar Year 2007*, WVES and URS Group, Inc., West Valley, New York, December 2008.

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APPENDIX A
DECOMMISSIONING PLAN ANNOTATED CHECKLIST

PURPOSE OF THIS APPENDIX

The purpose of this appendix is to assist NRC staff in review of the plan by providing the checklist used in its preparation, annotated to show where each applicable topic is addressed.

INFORMATION IN THIS APPENDIX

This appendix provides in Table A-1 a comparison between the major topics of the decommissioning plan evaluation checklist found in Appendix D to Volume 1 of NUREG-1757, *Consolidated Decommissioning Guidance, Decommissioning Process for Materials Licensees* (NRC 2006), and the major sections of this plan.

It then replicates the NUREG-1757 Appendix D checklist and identifies:

- The topics that do not apply to this plan based on discussions between NRC and DOE that took place in a decommissioning plan scoping meeting held on May 19, 2008 (NRC 2008), which are marked NA for not applicable;
- The section and page number in this plan where each applicable topic is addressed; and
- The cases where NRC has agreed that DOE procedures (i.e., DOE regulations, orders, and technical standards) can be cited in the plan instead of providing details called for by the NRC checklist (NRC 2008).

RELATIONSHIP TO OTHER PARTS OF THE PLAN

This appendix shows how the other parts of this plan address the applicable topics of the NRC decommissioning plan evaluation checklist.

Table A-1. NUREG-1757 Checklist – Phase 1 Decommissioning Plan Comparison

NUREG-1757 Checklist		WVDP Phase 1 Decommissioning Plan	
Sec	**Subject**	**Sec**	**Subject**
I	Executive Summary		Executive Summary
		1	Introduction
II	Facility Operating History	2	Facility Operating History
III	Facility Description	3	Facility Description
IV	Radiological Status of Facility	4	Radiological Status of Facility
V	Dose Modeling	5	Dose Modeling
VI	Environmental Information	3	Facility Description
VII	ALARA Analysis	6	ALARA Analysis
VIII	Planned Decommissioning Activities	7	Planned Decommissioning Activities
IX	Project Management and Organization	1.6	Project Management and Organization
X	Health and Safety	1.7	Health and Safety
XI	Environmental Monitoring and Control	1.8	Environmental Monitoring and Control
XII	Radioactive Waste Management Program	1.9	Radioactive Waste Management Program
XIII	Quality Assurance Program	8	Quality Assurance Program
XIV	Facility Radiation Surveys	9	Facility Radiation Surveys
XV	Financial Assurance		Not applicable.
XVI	Restricted Release/Alternate Criteria		Not applicable.
		App A	Decommissioning Plan Annotated Checklist
		App B	Environmental Radioactivity Data
		App C	Details of DCGL Development and Integrated Dose Analysis
		App D	Engineered Barriers and Post Remediation Activities

The annotated NUREG-1757 decommissioning plan evaluation checklist begins on the next page. Acronyms and abbreviations used in the checklist are as follows:

ES = Executive Summary NA = not applicable

CONTENT	SECTION	PAGE
I. EXECUTIVE SUMMARY		
☐ The name and address of the licensee or owner of the site	ES	ES-3
☐ The location and address of the site	ES	ES-3
☐ A brief description of the site and immediate environs	ES	ES-4
☐ A summary of the licensed activities that occurred at the site	ES	ES-10
☐ The nature and extent of contamination at the site	ES	ES-12
☐ The decommissioning objective proposed by the licensee (i.e., restricted or unrestricted use)	ES	ES-16
☐ The DCGLs for the site, the corresponding doses from these DCGLs, and the method that was use to determine the DCGLs *[Note that cleanup goals below the DCGLs are the criteria to be used for remediation activities in Phase 1. These are specified in Table ES-2.]*	Table ES-1 Table ES-2	ES-17 ES-18
☐ A summary of the ALARA evaluations performed to support the decommissioning	ES	ES-19
~~☐ If the licensee requests license termination under restricted conditions, the restrictions the licensee intends to use to limit doses as required in 10 CFR Part 20.1403 or 20.1404, and a summary of institutional controls and financial assurance~~	NA	NA
~~☐ If the licensee requests license termination under restricted conditions or using alternate criteria, a summary of the public participation activities undertaken by the licensee to comply with 10 CFR Part 20.1403(d) or 20.1404(a)(4)~~	NA	NA
☐ The proposed initiation and completion dates of decommissioning	ES	ES-19
☐ Any post-remediation activities (such as ground water monitoring) that the licensee proposes to undertake prior to requesting license termination	ES	ES-19
~~☐ A statement that the licensee is requesting that its license be amended to incorporate the DP~~	NA	NA

CONTENT	SECTION	PAGE
1. Introduction		
Because of the complexities of the project, DOE has included an Introduction section. It addresses matters such as the purpose of the plan and the scope of the Phase 1 proposed decommissioning activities. It explains the background of the project, including the relationship between the plan and the Decommissioning EIS and the general responsibilities of the organizations involved. It describes the site conditions that would be in effect at the time the proposed decommissioning activities begin, i.e., the interim end state. It explains the relationship between Phase 1 and Phase 2.		
The Introduction also briefly addresses the following matters covered by DOE procedures:		
• *Project management,*		
• *Health and safety,*		
• *Environmental monitoring and control, and*		
• *The radioactive waste management program.*		
II. FACILITY OPERATING HISTORY		
II.a. LICENSE NUMBER/STATUS/AUTHORIZED ACTIVITIES		
☐ ~~The radionuclides and maximum activities of radionuclides authorized and used under the current license~~	NA	NA
☐ ~~The chemical forms of the radionuclides authorized and used under the current license~~	NA	NA
☐ ~~A detailed description of how the radionuclides are currently being used at the site~~	NA	NA
☐ ~~The location(s) of use and storage of the various radionuclides authorized under current licenses~~	NA	NA
☐ ~~A scale drawing or map of the building or site and environs showing the current locations of radionuclide use at the site~~	NA	NA
☐ ~~A list of amendments to the license since the last license renewal~~	NA	NA
II.b. LICENSE HISTORY		
☐ The radionuclides and maximum activities of radionuclides authorized and used under all previous licenses	2.1 Table 2-1 Table 2-2 Table 2-3	2-2 2-2 2-3 2-3

CONTENT	SECTION	PAGE
☐ The chemical forms of the radionuclides authorized and used under all previous licenses	Table 2-1 Table 2-2 Table 2-6 Table 2-7 Table 2-8 Table 2-9	2-2 2-3 2-11 2-13 2-14 2-18
☐ A detailed description of how the radionuclides were used at the site	2.2.1 2.1.2	2-5 2-15
☐ The location(s) of use and storage of the various radionuclides authorized under all previous licenses	2.2.1 2.1.2	2-5 2-15
☐ A scale drawing or map of the site, facilities, and environs showing previous locations of radionuclide use at the site	Figure 2-3 Figure 2-4	2-22 2-23
II.c. PREVIOUS DECOMMISSIONING ACTIVITIES		
☐ A list or summary of areas at the site that were remediated in the past *Also addresses additional remediation planned to achieve the interim end state.*	2.2 Table 2-11 Table 2-13 Figure 2-5	2-19 2-20 2-26 2-24
☐ A summary of the types, forms, activities, and concentrations of radionuclides that were present in previously remediated areas	Table 2-11 Table 2-13	2-20 2-26
☐ The activities that caused the areas to become contaminated	2.1.1 2.1.2	2-5 2-15
☐ The procedures used to remediate the areas, and the disposition of radioactive material generated during the remediation	2.2.1 2.2.2	2-20 2-20
☐ A summary of the results of the final radiological evaluation of the previously remediated area	Table 2-13 2.2.2 Table 4-5 Table 4-6 Table 4-8	2-26 2-30 4-16 4-16 4-18
☐ A scale drawing or map of the site, facilities, and environs showing the locations of previous remedial activity	Figure 2-5	2-24
II.d. SPILLS *Does not include spills inside facilities that did not impact the environment.*		
☐ A summary of areas at the site where spills (or uncontrolled releases) of radioactive material occurred in the past	2.3	2-33

CONTENT	SECTION	PAGE
☐ The types, forms, activities, and concentrations of radionuclides involved in the spill or uncontrolled release	Table 2-16 Table 2-17 Table 2-18	2-35 2-39 2-41
☐ A scale drawing or map of the site, facilities, and environs showing the locations of spills *The locations of major spills are shown in the Figureures listed. The locations of minor spills are identified in Table 2-17 (page 2-39) and Table 2-18 (page 2-41).*	Figure 2-3 Figure 2-4 Figure 2-6 Figure 2-7	2-22 2-23 2-34 2-38
II.e. PRIOR ONSITE BURIALS		
☐ A summary of areas at the site where radioactive material has been buried in the past	2.4	2-42
☐ The types, forms, activities and concentrations of waste and radionuclides in the former burial	Table 2-19 Table 2-20 Table 2-21	2-43 2-44 2-45
☐ A scale drawing or map of the site, facilities, and environs showing the locations of former burials	Figure 2-3 Figure 2-4	2-22 2-23
III. FACILITY DESCRIPTION *This section incorporates information from the DEIS. The SDA is not addressed.*		
III.a. SITE LOCATION AND DESCRIPTION		
☐ The size of the site in acres or square meters	3.1.2	3-2
☐ The State and county in which the site is located	3.1.1	3-2
☐ The names and distances to nearby communities, towns, and cities	3.1.1 3.2.2	3-2 3-29
☐ A description of the contours and features of the site	3.1.2 Figure 3-3 Figure 3-4	3-2 3-91 3-92
☐ The elevation of the site	3.1.2	3-2
☐ A description of property surrounding the site, including the location of all off-site wells used by nearby communities or individuals	3.1.4 3.2.1	3-24 3-28
☐ The location of the site relative to prominent features such as rivers and lakes	Figure 3-1 Figure 3-2	3-89 3-90

CONTENT	SECTION	PAGE
☐ A map that shows the detailed topography of the site using a contour interval	Figure 3-3 Figure 3-4	3-91 3-92
☐ The location of the nearest residences and all significant facilities or activities near the site	3.1.4	3-24
☐ A description of the facilities (e.g., buildings, parking lots, and fixed equipment) at the site	3.1.3	3-3
III.b. POPULATION DISTRIBUTION		
☐ A summary of the current population in and around the site, by compass vectors	3.2 Figure 3-44	3-28 3-126
☐ A summary of the projected population in and around the site by compass vectors *[Projections not available by compass vector.]*	3.2.2	3-30
III.c. CURRENT/FUTURE LAND USE		
☐ A description of the current land uses in and around the site	3.3.1 Figure 3-45	3-33 3-127
☐ A summary of anticipated land uses	3.3.2	3-36
III.d. METEOROLOGY AND CLIMATOLOGY		
☐ A description of the general climate of the region	3.4.1	3-38
☐ Seasonal and annual frequencies of severe weather phenomena	3.4.2	3-39
☐ Weather-related radionuclide transmission parameters	3.4.3	3-40
☐ Routine weather-related site deterioration parameters	3.4.4	3-40
☐ Extreme weather-related site deterioration parameters	3.4.4	3-41
☐ A description of the local (site) meteorology	3.4.5	3-41
☐ The National Ambient Air Quality Standards Category of the area in which the facility is located and, if the facility is not in a Category 1 zone, the closest and first downwind Category 1 Zone	3.4.5	3-45
III.e. GEOLOGY AND SEISMOLOGY		
☐ A detailed description of the geologic characteristics of the site and the region around the site	3.5	3-45

CONTENT	SECTION	PAGE
☐ A discussion of the tectonic history of the region, regional geomorphology, physiography, stratigraphy, and geochronology	3.5	3-45
☐ A regional tectonic map showing the site location and its proximity to tectonic structures	Figure 3-55	3-137
☐ A description of the structural geology of the region and its relationship to the site geologic structure	3.5	3-45
☐ A description of any crustal tilting, subsidence, karst terrain, landsliding, and erosion	3.5.3	3-49
☐ A description of the surface and subsurface geologic characteristics of the site and its vicinity	3.5	3-45
☐ A description of the geomorphology of the site	3.5.3	3-49
☐ A description of the location, attitude, and geometry of all known or inferred faults in the site and vicinity	3.5.4	3-52
☐ A discussion of the nature and rates of deformation	3.5.3	3-49
☐ A description of any man-made geologic features such as mines or quarries	3.1.1	3-2
☐ A description of the seismicity of the site and region	3.5.5	3-58
☐ A complete list of all historical earthquakes that have a magnitude of 3 or more, or a modified Mercalli intensity of IV or more within 200 miles of the site	3.5.5 Table 3-15	3-58 3-58
III.f. SURFACE WATER HYDROLOGY		
☐ A description of site drainage and surrounding watershed fluvial features	3.6.1	3-62
☐ Water resource data including maps, hydrographs, and stream records from other agencies (e.g., U.S. Geological Survey and U.S. Army Corps of Engineers)	3.6.1 Figure 3-3	3-63 3-91
☐ Topographic maps of the site that show natural drainages and man-made features	Figure 3-3 Figure 3-4	3-91 3-92
☐ A description of the surface water bodies at the site and surrounding areas	3.6.1	3-62
☐ A description of existing and proposed water control structures and diversions (both upstream and downstream) that may influence the site	none	-

CONTENT	SECTION	PAGE
☐ Flow-duration data that indicate minimum, maximum, and average historical observations for surface water bodies in the site areas	3.6.1	3-64
☐ Aerial photography and maps of the site and adjacent drainage areas identifying features such as drainage areas, surface gradients, and areas of flooding	Figure 3-3 Figure 3-4	3-91 3-92
☐ An inventory of all existing and planned surface water users, whose intakes could be adversely affected by migration of radionuclides from the site	3.6.4	3-65
☐ Topographic and/or aerial photographs that delineate the 100-year floodplain at the site	Figure 3-4	3-92
☐ A description of any man-made changes to the surface water hydrologic system that may influence the potential for flooding at the site	*No such changes*	-
III.g. GROUND WATER HYDROLOGY		
☐ A description of the saturated zone	3.7.1	3-67
☐ Descriptions of monitoring wells	3.7.2 4.2.8 Figure 4-12 Table B-15	3-679 4-56 4-61 B-41
☐ Physical parameters	3.7.3	3-70
☐ A description of ground water flow directions and velocities	3.7.1 3.7.1 Figure 3-62 Figure 3-63 Figure 3-64 Figure 3-65	3-68 3-69 3-144 3-145 3-146 3-147
☐ A description of the unsaturated zone	3.7.4	3-70
☐ Information on all monitor stations including location and depth	Table B-15	B-41
☐ A description of physical parameters	3.7.3	3-70
☐ A description of the numerical analyses techniques used to characterize the unsaturated and saturated zones	3.7.7	3-72
☐ The distribution coefficients of the radionuclides of interest at the site	3.7.8 Table 3-20	3-73 3-76

CONTENT	SECTION	PAGE
III.h. NATURAL RESOURCES		
☐ A description of the natural resources occurring at or near the site	3.8	3-79
☐ A description of potable, agricultural, or industrial ground or surface waters	3.8.3	3-80
☐ A description of economic, marginally economic, or subeconomic known or identified natural resources as defined in U.S. Geological Survey Circular 831	3.8	3-79
☐ Mineral, fuel, and hydrocarbon resources near and surrounding the site which, if exploited, would effect the licensee's dose estimates	*none*	-
IV. RADIOLOGICAL STATUS OF FACILITY		
Information on residual radioactivity and radiation levels in facilities is provided at a summary level consistent with DOE having primary responsibility for the health and safety aspects of the facility removal activities. Additional characterization would be performed in connection with the proposed decommissioning activities as specified in Section 9.		
IV.a CONTAMINATED STRUCTURES		
☐ A list or description of all structures at the facility where licensed activities occurred that contain residual radioactive material in excess of site background levels	4.1.2 Figure 4-1 Figure 4-2 Figure 4-3 Figure 4-4 Figure 4-5	4-5 4-7 4-8 4-9 4-10 4-11
☐ A summary of the structures and locations at the facility that the licensee has concluded have not been impacted by licensed operations and the rationale for the conclusion	4.1.3	4-12
~~☐ A list or description of each room or work area within each of these structures~~	NA	NA
~~☐ A summary of the background levels used during scoping or characterization surveys~~	NA	NA
~~☐ A summary of the locations of contamination in each room or work area~~	NA	NA
~~☐ A summary of the radionuclides present at each location, the maximum and average radionuclide activities in dpm/100 cm^2, and, if multiple radionuclides are present, the radionuclide ratios~~	NA	NA
~~☐ The mode of contamination for each surface (i.e., whether the radioactive material is present only on the surface of the material or if it has penetrated the material)~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~The maximum and average radiation levels in mrem/hr in each room or work area~~	NA	NA
☐ ~~A scale drawing or map of the rooms or work areas showing the locations of radionuclide material contamination~~	NA	NA
IV.b. CONTAMINATED SYSTEMS AND EQUIPMENT		
☐ ~~A list or description and the location of all systems or equipment at the facility that contain residual radioactive material in excess of site background levels~~	NA	NA
☐ ~~A summary of the radionuclides present in each system or on the equipment at each location, the maximum and average radionuclide activities in dpm/100cm^2, and, if multiple radionuclides are present, the radionuclide ratios~~	NA	NA
☐ ~~The maximum and average radiation levels in mrem/hr at the surface of each piece of equipment~~	NA	NA
☐ ~~A summary of the background levels used during scoping or characterization surveys~~	NA	NA
☐ ~~A scale drawing or map of the rooms or work areas showing the locations of the contaminated systems or equipment~~	NA	NA
IV.c. SURFACE SOIL CONTAMINATION		
Information provided focuses on the project premises using existing data, which are not available for all locations on the project premises. Contamination in stream sediment is also addressed.		
☐ A list or description of all locations at the facility where surface soil contains residual radioactive material in excess of site background levels	4.2.3 Figure 4-6	4-29 4-31
☐ A summary of the background levels used during scoping or characterization surveys	4.2.2 Table 4-11 Figure B-1 Table B-1	4-25 4-26 B-3 B-4
☐ A summary of the radionuclides present at each location, the maximum and average radionuclide activities in pCi/gm, and, if multiple radionuclides are present, the radionuclide ratios	4.2.3 4.2.5	4-29 4-35
☐ The maximum and average radiation levels in mrem/hr at each location *[Data are not available at sample locations.]*	4.2.6	4-48

CONTENT	SECTION	PAGE
☐ A scale drawing or map of the site showing the locations of radionuclide material contamination in surface soil	Figure 4-6	4-31
IV.d. SUBSURFACE SOIL CONTAMINATION		
Information provided focuses on the project premises using existing data, which are not available for all locations on the project premises.		
☐ A list or description of all locations at the facility where subsurface soil contains residual radioactive material in excess of site background levels	4.2.4 Figure 4-7 Figure 4-8	4-30 4-32 4-34
☐ A summary of the background levels used during scoping or characterization surveys	4.2.2	4-25
☐ A summary of the radionuclides present at each location, the maximum and average radionuclide activities in pCi/gm, and, if multiple radionuclides are present, the radionuclide ratios	4.2.4 4.2.5	4-30 4-35
☐ The depth of the subsurface soil contamination at each location	Figure 4-8 4.2.5	4-34 4-35
☐ A scale drawing or map of the site showing the locations of subsurface soil contamination	Figure 4-7 Figure 4-8	4-32 4-34
IV.e. SURFACE WATER		
[Information provided focuses on the project premises using existing data, which are not available for all locations on the project premises.]		
☐ A list or description of all surface water bodies at the facility that contain residual radioactive material in excess of site background levels	4.2.7 Figure 4-11	4-53 4-54
☐ A summary of the background levels used during scoping or characterization surveys	Table 4-11	4-26
☐ A summary of the radionuclides present in each surface water body and the maximum and average radionuclide activities in becquerel per liter (Bq/L) (picocuries per liter (pCi/L)	Table 4-24	4-55
IV.f. GROUND WATER		
Information provided focuses on the project premises.		
☐ A summary of the aquifer(s) at the facility that contain residual radioactive material in excess of site background levels	4.2.8	4-56
☐ A summary of the background levels used during scoping or characterization surveys	Table 4-11	4-26

CONTENT	SECTION	PAGE
☐ A summary of the radionuclides present in each aquifer and the maximum and average radionuclide activities in Becquerel per liter (Bq/L) (picocuries per liter (pCi/L))	Table 4-25	4-57
V. DOSE MODELING		
V.a. UNRESTRICTED RELEASE USING SCREENING CRITERIA		
Screening criteria are not used.		
V.a.1. Unrestricted Release Using Screening Criteria for Building Surface Residual Radioactivity		
~~☐ The general conceptual model (for both the source term and the building environment) of the site~~	NA	NA
~~☐ A summary of the screening method (i.e., running DandD or using the look-up Tables) used in the DP~~	NA	NA
V.a.2. Unrestricted Release Using Screening Criteria for Surface Soil Residual Radioactivity		
~~☐ Justification on the appropriateness of using the screening approach (for both the source term and the environment) at the site~~	NA	NA
~~☐ A summary of the screening method (i.e., running DandD or using the look-up Tables) used in the DP~~	NA	NA
V.b. UNRESTRICTED RELEASE USING SITE-SPECIFIC INFORMATION		
Although no remediated areas would be released for unrestricted use during Phase 1, information specified in this subsection is provided for development of DCGLs and cleanup goals for surface soil, subsurface soil, and streambed sediment. The level of detail provided is similar to that in the Decommissioning EIS.		
☐ Source term information including nuclides of interest, configuration of the source, and areal variability of the source	5.1.2	5-2
☐ Description of the exposure scenario including a description of the critical group	5.2.1 Figure 5-7 Figure 5-8 Figure 5-9	5-20 5-20 5-24 5-28
☐ Description of the conceptual model of the site including the source term, physical features important to modeling the transport pathways, and the critical group	5.2.1 Figure 5-7 Figure 5-8 Figure 5-9	5-20 5-20 5-24 5-28
☐ Identification/description of the mathematical model used (e.g., hand calculations, DandD Screen v1.0, and RESRAD v5.81)	5.2.2	5-31

CONTENT	SECTION	PAGE
☐ Description of the parameters used in the analysis	Table C-1	C-3
☐ Discussion about the effect of uncertainty on the results	5.2.4	5-35
☐ Input and output files or printouts, if a computer program was used	App C Related CD	C-1
V.c. RESTRICTED RELEASE USING SITE-SPECIFIC INFORMATION *Although Phase 1 proposed decommissioning activities would not result in a restricted release, this plan provides a limited site-wide integrated dose assessment to help place the Phase 1 proposed decommissioning activities involving remediation of soil in the WMA 1 and WMA 2 excavations into context with regard to supporting potential Phase 2 decommissioning alternatives. Information provided on the topics in this subsection is limited to that necessary to support this assessment. The level of detail is similar to that in the Decommissioning EIS.*		
☐ Source term information including nuclides of interest, configuration of the source, areal variability of the source, and chemical forms	5.1.2	5-2
☐ A description of the exposure scenarios, including a description of the critical group for each scenario	5.2.1 Figure 5-7 Figure 5-8 Figure 5-9	5-20 5-20 5-24 5-28
☐ A description of the conceptual model(s) of the site that includes the source term, physical features important to modeling the transport pathways, and the critical group for each scenario	5.2.1 Figure 5-7 Figure 5-8 Figure 5-9	5-120 5-20 5-24 5-28
☐ Identification/description of the mathematical model(s) used (e.g., hand calculations and RESRAD v5.81)	5.2.2	5-31
☐ A summary of parameters used in the analysis	Table C-1	C-3
☐ A discussion about the effect of uncertainty on the results	5.2.4	5-35
☐ Input and output files or printouts, if a computer program was used	App C Related CD	C-1
V.d. RELEASE INVOLVING ALTERNATE CRITERIA *DOE would not use alternative criteria.*		
~~☐ Source term information including nuclides of interest, configuration of the source, areal variability of the source, and chemical forms~~	NA	NA
~~☐ A description of the exposure scenarios, including a description of the critical group for each scenario~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~A description of the conceptual model(s) of the site that includes the source term, physical features important to modeling the transport pathways, and the critical group for each scenario~~	NA	NA
☐ ~~Identification/description of the mathematical model(s) used (e.g., hand calculations and RESRAD v5.81)~~	NA	NA
☐ ~~A summary of parameters used in the analysis~~	NA	NA
☐ ~~A discussion about the effect of uncertainty on the results~~	NA	NA
☐ ~~Input and output files or printouts, if a computer program was used~~	NA	NA
VI. ENVIRONMENTAL INFORMATION		
☐ Environmental information described in NUREG-1748	3	3-1[1]
☐ For an EIS, the environmental information is reviewed by the EPAD EIS project manager	Noted	-
VII. ALARA ANALYSIS *The ALARA analysis focuses on the DCGLs for surface and subsurface soil and streambed sediment.*		
☐ A description of how the licensee will achieve a decommissioning goal below the dose limit	6.2	6-4
☐ A quantitative cost benefit analysis	6.3 6.4	6-6 6-9
☐ A description of how costs were estimated	6.3.2	6-6
☐ A demonstration that the doses to the average member of the critical group are ALARA	6.3 6.4	6-6 6-9
VIII. PLANNED DECOMMISSIONING ACTIVITIES *The remediation tasks are described in general terms. Every room and area is not addressed since decontamination would be limited and the facilities would be demolished. Typical remediation techniques to be used are described in Section 7.11, starting on page 7-40. More detail would be provided later in the Decommissioning Work Plan(s). Measures for preventing contamination or recontamination of the site due to proposed decommissioning activities are addressed in Section 7.2.2 on page 7-6.*		

[1] Section 3 provides a detailed description of the affected environment. All of the information specified in NUREG-1748 is contained in the Decommissioning EIS.

CONTENT	SECTION	PAGE
VIII.a. CONTAMINATED STRUCTURES		
☐ A summary of the remediation tasks planned for each room or area in the contaminated structure, in the order in which they will occur	7.3.3 to 7.3.9	7-14 to 7-27
☐ A description of the remediation techniques that will be employed in each room or area of the contaminated structure	7.11	7-40
~~☐ A summary of the radiation protection methods and control procedures that will be employed in each room or area~~	NA	NA
~~☐ A summary of the procedures already authorized under the existing license and those for which approval is being requested in the DP~~	NA	NA
☐ A commitment to conduct decommissioning activities in accordance with written, approved procedures	7.2.2	7-5
☐ A summary of any unique safety or remediation issues associated with remediating the room or area	7.2.2	7-6
~~☐ For Part 70 licensees, a summary of how the licensee will ensure that the risks addressed in the facility's Integrated Safety Analysis will be addressed during decommissioning~~	NA	NA
VIII.b. CONTAMINATED SYSTEMS AND EQUIPMENT		
☐ A summary of the remediation tasks planned for each system in the order in which they will occur, including which activities will be conducted by licensee staff and which will be performed by a contractor	7.3.3 to 7.3.9	7-14 to 7-27
☐ A description of the techniques that will be employed to remediate each system in the facility or site	7-11	7-40
~~☐ A description of the radiation protection methods and control procedures that will be employed while remediating each system~~	NA	NA
☐ A summary of the equipment that will be removed or decontaminated and how the decontamination will be accomplished	7.3 7.4.2 7.5	7-9 7-28 7-34
~~☐ A summary of the procedures already authorized under the existing license and those for which approval is being requested in the DP~~	NA	NA
☐ A commitment to conduct decommissioning activities in accordance with written, approved procedures	7.2.2	7-5
☐ A summary of any unique safety or remediation issues associated with remediating any system or piece of equipment	7.2.2	7-6

CONTENT	SECTION	PAGE
☐ ~~For Part 70 licensees, a summary of how the licensee will ensure that the risks addressed in the facility's Integrated Safety Analysis will be addressed during decommissioning~~	NA	NA
VIII.c. SOIL		
☐ A summary of the removal/remediation tasks planned for surface and subsurface soil at the site in the order in which they will occur, including which activities will be conducted by licensee staff and which will be performed by a contractor	7.3.8 7.4.3 7.7.4	7-19 7-29 7-39
☐ A description the techniques that will be employed to remove or remediate surface and subsurface soil at the site	7.3.8 7.4.3 7.7.4 7.11	7-19 7-29 7-39 7-47
☐ ~~A description of the radiation protection methods and control procedures that will be employed during soil removal/ remediation~~	NA	NA
☐ ~~A summary of the procedures already authorized under the existing license and those for which approval is being requested in the DP~~	NA	NA
☐ A commitment to conduct decommissioning activities in accordance with written, approved procedures	7.2.2	7-5
☐ A summary of any unique safety or removal/remediation issues associated with remediating the soil	7.2.2	7-6
☐ ~~For Part 70 licensees, a summary of how the licensee will ensure that the risks addressed in the facility's Integrated Safety Analysis will be addressed during decommissioning~~	NA	NA
VIII.d. SURFACE AND GROUND WATER		
Surface water removed from the lagoons would be remediated in Phase 1 of the proposed decommissioning, and groundwater removed from the WMA 1 and WMA 2 excavations would be treated also.		
☐ A summary of the remediation tasks planned for ground and surface water in the order in which they will occur, including which activities will be conducted by licensee staff and which will be performed by a contractor	7.3.8 7.4.3	7-25 7-33
☐ A description of the remediation techniques that will be employed to remediate the ground or surface water	7.3.8 7.4.3	7-25 7-33
☐ ~~A description of the radiation protection methods and control procedures that will be employed during ground or surface water remediation~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~A summary of the procedures already authorized under the existing license and those for which approval is being requested in the DP~~	NA	NA
☐ A commitment to conduct decommissioning activities in accordance with written, approved procedures	7.2.2	7-5
☐ A summary of any unique safety or remediation issues associated with remediating the ground or surface water	7.2.2	7-6
VIII.e. SCHEDULES		
☐ A Gantt or PERT chart detailing the proposed remediation tasks in the order in which they will occur	Figure 7-15	7-49
☐ A statement acknowledging that the dates in the schedule are contingent upon NRC approval of the DP	7.12	7-48
☐ A statement acknowledging that circumstances can change during decommissioning, and, if the licensee determines that the decommissioning cannot be completed as outlined in the schedule, the licensee will provide an updated schedule to NRC	7.12	7-48
☐ ~~If the decommissioning is not expected to be completed within the timeframes outlined in NRC regulations, a request for alternative schedule for completing the decommissioning~~	NA	NA
IX. PROJECT MANAGEMENT AND ORGANIZATION		
~~*This section focuses on project management and organization related to the final status surveys.*~~ *Matters in this section are addressed by the DOE procedures identified in Section 1.6.*		
IX.a. DECOMMISSIONING MANAGEMENT ORGANIZATION		
☐ ~~A description of the decommissioning organization~~	NA	NA
☐ ~~A description of the responsibilities of each of these decommissioning project units~~	NA	NA
☐ ~~A description of the reporting hierarchy within the decommissioning project management organization~~	NA	NA
☐ ~~A description of the responsibility and authority of each unit to ensure that decommissioning activities are conducted in a safe manner and in accordance with approved written procedures~~	NA	NA

CONTENT	SECTION	PAGE
IX.b. DECOMMISSIONING TASK MANAGEMENT		
~~A description of the manner in which the decommissioning tasks are managed~~	NA	NA
~~A description of how individual decommissioning tasks are evaluated and how the Radiation Work Permits (RWPs) are developed for each task~~	NA	NA
~~A description of how the RWPs are reviewed and approved by the decommissioning project management organization~~	NA	NA
~~A description of how RWPs are managed throughout the decommissioning project~~	NA	NA
~~A description of how individuals performing the decommissioning tasks are informed of the procedures in the RWP~~	NA	NA
IX.c. DECOMMISSIONING MANAGEMENT POSITIONS AND QUALIFICATIONS		
~~A description of the duties and responsibilities of each management position in the decommissioning organization and the reporting responsibility of the position~~	NA	NA
~~A description of the duties and responsibilities of each chemical, radiological, physical, and occupational safety-related position in the decommissioning organization and the reporting responsibility of each position~~	NA	NA
~~A description of the duties and responsibilities of each engineering, quality assurance, and waste management position in the decommissioning organization and the reporting responsibility of each position~~	NA	NA
~~The minimum qualifications for each of the positions describe above, and the qualifications of the individuals currently occupying the positions~~	NA	NA
~~A description of all decommissioning and safety committees~~	NA	NA
IX.d. RADIATION SAFETY OFFICER		
~~A description of the health physics and radiation safety education and experience required for individuals acting as the licensee's RSO~~	NA	NA
~~A description of the responsibilities and duties of the RSO~~	NA	NA
~~A description of the specific authority of the RSO to implement and manage the licensee's radiation protection program~~	NA	NA

CONTENT	SECTION	PAGE
IX.e. TRAINING		
~~A description of the radiation safety training that the licensee will provide to each employee~~	NA	NA
~~A description of any daily worker "jobside" or "tailgate" training that will be provided at the beginning of each workday or job task to familiarize workers with job-specific procedures or safety requirements~~	NA	NA
~~A description of the documentation that will be maintained to demonstrate that training commitments are being met~~	NA	NA
IX.f. CONTRACTOR SUPPORT		
~~A summary of decommissioning tasks that will be performed by contractors~~	NA	NA
~~A description of the management interfaces that will be in place between the site's management and onsite supervisors, and contractor management and onsite supervisors~~	NA	NA
~~A description of the oversight responsibilities and authority that the licensee will exercise over contractor personnel~~	NA	NA
~~A description of the training that will be provided to contractor personnel by the licensee and the training that will be provided by the contractor~~	NA	NA
~~A commitment that the contractor will comply with all radiation safety and license requirements at the facility~~	NA	NA
X. HEALTH AND SAFETY PROGRAM DURING DECOMMISSIONING: RADIATION SAFETY CONTROLS AND MONITORING FOR WORKERS *Matters in this section are addressed by the DOE procedures identified in Section 1.7.*		
X.a. AIR SAMPLING PROGRAM		
~~A description which demonstrates that the air sampling program is representative of the workers breathing zones~~	NA	NA
~~A description of the criteria which demonstrates that air samplers with appropriate sensitivities will be used, and that samples will be collected at appropriate frequencies~~	NA	NA
~~A description of the conditions under which air monitors will be used~~	NA	NA
~~A description of the criteria used to determine the frequency of calibration of the flow meters on the air samplers~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~A description of the action levels for air sampling results~~	NA	NA
☐ ~~A description of how minimum detecTable activities (MDA) for each specific radionuclide that may be collected in air samples are determined~~	NA	NA
X.b. RESPIRATORY PROTECTION PROGRAM		
☐ ~~A description of the process controls, engineering controls, or procedures to control concentrations of radioactive materials in air~~	NA	NA
☐ ~~A description of the evaluation which will be performed when it is not practical to apply engineering controls or procedures~~	NA	NA
☐ ~~A description of the considerations used which demonstrates respiratory protection equipment is appropriate for a specific task based on the guidance on assigned protection factors~~	NA	NA
☐ ~~A description of the medical screening and fit testing required before workers will use any respirator that is assigned a protection factor~~	NA	NA
☐ ~~A description of the written procedures maintained to address all the elements of the respiratory protection program~~	NA	NA
☐ ~~A description of the use, maintenance, and storage of respiratory protection devices~~	NA	NA
☐ ~~A description of the respiratory equipment users training program~~	NA	NA
☐ ~~A description of the considerations made when selecting respiratory protection equipment~~	NA	NA
X.c. INTERNAL EXPOSURE DETERMINATION		
☐ ~~A description of the monitoring to be performed to determine worker exposure~~	NA	NA
☐ ~~A description of how worker intakes are determined using measurements of quantities of radionuclides excreted from, or retained in the human body~~	NA	NA
☐ ~~A description of how worker intakes are determined by measurements of the concentrations of airborne radioactive materials in the workplace~~	NA	NA
☐ ~~A description of how worker intakes for an adult, a minor, and a declared pregnant woman (DPW) are determined using any combination of the measurements above, as may be necessary~~	NA	NA

CONTENT	SECTION	PAGE
~~A description of how worker intakes are converted into committed effective dose equivalent~~	NA	NA
X.d. EXTERNAL EXPOSURE DETERMINATION		
~~A description of the individual-monitoring devices which will be provided to workers~~	NA	NA
~~A description of the type, range, sensitivity, and accuracy of each individual-monitoring device~~	NA	NA
~~A description of the use of extremity and whole-body monitors when the external radiation field is non-uniform~~	NA	NA
~~A description of when audible-alarm dosimeters and pocket dosimeters will be provided~~	NA	NA
~~A description of how external dose from airborne radioactive material is determined~~	NA	NA
~~A description of the procedure to insure that surveys necessary to supplement personnel monitoring are performed~~	NA	NA
~~A description of the action levels for worker's external exposure, and the technical bases and actions to be taken when they are exceeded~~	NA	NA
X.e. SUMMATION OF INTERNAL AND EXTERNAL EXPOSURES		
~~A description of how the internal and external monitoring results are used to calculate TODE and TEDE doses to occupational workers~~	NA	NA
~~A description of how internal doses to the embryo/fetus, which is based on the intake of an occupationally exposed DPW will be determined~~	NA	NA
~~A description of the monitoring of the intake of a DPW, if determined to be necessary~~	NA	NA
~~A description of the program for the preparation, retention, and reporting of records for occupational radiation exposures~~	NA	NA
X.f. CONTAMINATION CONTROL PROGRAM		
~~A description of the written procedures to control access to, and stay time in, contaminated areas by workers, if they are needed~~	NA	NA
~~A description of surveys to supplement personnel monitoring for workers during routine operations, maintenance, clean-up activities, and special operations~~	NA	NA

CONTENT	SECTION	PAGE
~~A description of the surveys which will be performed to determine the baseline of background radiation levels and radioactivity from natural sources for areas where decommissioning activities will take place~~	NA	NA
~~A description in matrix or Tableular form which describes contamination action limits (that is, actions taken to either decontaminate a person, place, or area, restrict access, or modify the type or frequency of radiological monitoring)~~	NA	NA
~~A description (included in the matrix or Table mentioned above) of proposed radiological contamination guidelines for specifying and modifying the frequency for each type of survey used to assess the reduction of total contamination~~	NA	NA
~~A description of the procedures used to test sealed sources, and to insure that sealed sources are leaked tested at appropriate intervals~~	NA	NA
X.g. INSTRUMENTATION PROGRAM		
~~A description of the instruments to be used to support the health and safety program~~	NA	NA
~~A description of instrumentation storage, calibration, and maintenance facilities for instruments used in field surveys~~	NA	NA
~~A description of the method used to estimate the MDC or MDA (at the 95 percent confidence level) for each type of radiation to be detected~~	NA	NA
~~A description of the instrument calibration and quality assurance procedures~~	NA	NA
~~A description of the methods used to estimate uncertainty bounds for each type of instrumental measurement~~	NA	NA
~~A description of air sampling calibration procedures or a statement that the instruments will be calibrated by an accredited laboratory~~	NA	NA
X.h. NUCLEAR CRITICALITY SAFETY		
~~A description of how the NCS functions, including management responsibilities and technical qualifications of safety personnel, will be maintained when needed throughout the decommissioning process~~	NA	NA
~~A description of how an awareness of procedures and other items relied on for safety will be maintained throughout decommissioning among all personnel, with access to systems that may contain fissionable material in sufficient amounts for criticality~~	NA	NA

CONTENT	SECTION	PAGE
~~A summary of the review of NCSA's or the ISA indicating either that the process needs no new safety procedures or requirements, or that new requirements or analysis have been performed~~	NA	NA
~~A summary of any generic NCS requirements to be applied to general decommissioning, decontamination, or dismantlement operations, including those dealing with systems that may unexpectedly contain fissionable material~~	NA	NA
X.i. HEALTH PHYSICS AUDITS, INSPECTIONS, AND RECORDKEEPING PROGRAM		
~~A general description of the annual program review conducted by executive management~~	NA	NA
~~A description of the records to be maintained of the annual program review and executive audits~~	NA	NA
~~A description of the types and frequencies of surveys and audits to be performed by the RSO and RSO staff~~	NA	NA
~~A description of the process used in evaluating and dealing with violations of NRC requirements or license commitments identified during audits~~	NA	NA
~~A description of the records maintained of RSO audits~~	NA	NA
XI. ENVIRONMENTAL MONITORING AND CONTROL PROGRAM		
Matters in this section are to be addressed by the DOE procedures identified in Section 1.8.		
XI.a. ENVIRONMENTAL ALARA EVALUATION PROGRAM		
~~A description of ALARA goals for effluent control~~	NA	NA
~~A description of the procedures, engineering controls, and process controls to maintain doses ALARA~~	NA	NA
~~A description of the ALARA reviews and reports to management~~	NA	NA
XI.b. EFFLUENT MONITORING PROGRAM		
~~A demonstration that background and baseline concentrations of radionuclides in environmental media have been esTablelished through appropriate sampling and analysis~~	NA	NA
~~A description of the known or expected concentrations of radionuclides in effluents~~	NA	NA
~~A description of the physical and chemical characteristics of radionuclides in effluents~~	NA	NA

CONTENT	SECTION	PAGE
~~☐ A summary or diagram of all effluent discharge locations~~	NA	NA
~~☐ A demonstration that samples will be representative of actual releases~~	NA	NA
~~☐ A summary of the sample collection and analysis procedures~~	NA	NA
~~☐ A summary of the sample collection frequencies~~	NA	NA
~~☐ A description of the environmental monitoring recording and reporting procedures~~	NA	NA
~~☐ A description of the quality assurance program to be established and implemented for the effluent monitoring program~~	NA	NA
XI.c. EFFLUENT CONTROL PROGRAM		
~~☐ A description of the controls that will be used to minimize releases of radioactive material to the environment~~	NA	NA
~~☐ A summary of the action levels and a description of the actions to be taken should a limit be exceeded~~	NA	NA
~~☐ A description of the leak detection systems for ponds, lagoons, and tanks~~	NA	NA
~~☐ A description of the procedures to ensure that releases to sewer systems are controlled and maintained to meet the requirements of 10 CFR 20.2003~~	NA	NA
~~☐ A summary of the estimates of doses to the public from effluents and a description of the method used to estimate public dose~~	NA	NA
XII. RADIOACTIVE WASTE MANAGEMENT PROGRAM		
Matters in this section are to be addressed by the DOE procedures identified in Section 1.9.		
XII.a. SOLID RADWASTE		
~~☐ A summary of the types of solid radwaste that are expected to be generated during decommissioning operations~~	NA	NA
~~☐ A summary of the estimated volume, in cubic feet, of each solid radwaste type summarized in Line 1 above~~	NA	NA
~~☐ A summary of the radionuclides (including the estimated activity of each radionuclide) in each estimated solid radwaste type summarized in Line 1 above~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~A summary of the volumes of Class A, B, C, and Greater-than-Class C solid radwaste that will be generated by decommissioning operations~~	NA	NA
☐ ~~A description of how and where each of the solid radwaste summarized in Line 1 above will be stored onsite prior to shipment for disposal~~	NA	NA
☐ ~~A description of how the each of the solid radwastes summarized in Line 1 above will be treated and packaged to meet disposal site acceptance criteria prior to shipment for disposal~~	NA	NA
☐ ~~If appropriate, how the licensee intends to manage volumetrically contaminated material~~	NA	NA
☐ A description of how the licensee will prevent contaminated soil, or other loose solid radwaste, from being re-disbursed after exhumation and collection	7.2.2	7-6
☐ ~~The name and location of the disposal facility that the licensee intends to use for each solid radwaste type summarized in Line 1 above~~	NA	NA
XII.b. LIQUID RADWASTE		
☐ ~~A summary of the types of liquid radwaste that are expected to be generated during decommissioning operations~~	NA	NA
☐ ~~A summary of the estimated volume, in liters, of each liquid radwaste type summarized in Line 1 above~~	NA	NA
☐ ~~A summary of the radionuclides (including the estimated activity of each radionuclide) in each liquid radwaste type summarized in Line 1 above~~	NA	NA
☐ ~~A summary of the estimated volumes of Class A, B, C, and Greater-than-Class C liquid radwaste that will be generated by decommissioning operations~~	NA	NA
☐ ~~A description of how and where each of the liquid radwastes summarized in Line 1 above will be stored onsite prior to shipment for disposal~~	NA	NA
☐ ~~A description of how the each of the liquid radwastes summarized in Line 1 above will be treated and packaged to meet disposal site acceptance criteria prior to shipment for disposal~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~The name and location of the disposal facility that the licensee intends to use for each liquid radwaste type summarized in Line 1 above~~	NA	NA
XII.c. MIXED WASTE		
☐ ~~A summary of the types of solid and liquid mixed waste that are expected to be generated during decommissioning operations~~	NA	NA
☐ ~~A summary of the estimated volumes in cubic feet of each solid mixed waste type summarized in Line 1 above, and in liters for each liquid mixed waste~~	NA	NA
☐ ~~A summary of the radionuclides (including the estimated activity of each radionuclide) in each type of mixed waste type summarized in Line 1 above~~	NA	NA
☐ ~~A summary of the estimated volumes of Class A, B, C, and Greater than Class C mixed waste that will be generated by decommissioning operations~~	NA	NA
☐ ~~A description of how and where each of the mixed wastes summarized in Line 1 above will be stored onsite prior to shipment for disposal~~	NA	NA
☐ ~~A description of how the each of the mixed wastes summarized in Line 1 above will be treated and packaged to meet disposal site acceptance criteria prior to shipment for disposal~~	NA	NA
☐ ~~The name and location of the disposal facility that the licensee intends to use for each mixed waste type summarized in Line 1 above~~	NA	NA
☐ ~~A discussion of the requirements of all other regulatory agencies having jurisdiction over the mixed waste~~	NA	NA
☐ ~~A demonstration the that the licensee possesses the appropriate EPA or State permits to generate, store, and/or treat the mixed wastes~~	NA	NA
XIII. QUALITY ASSURANCE PROGRAM		
This section focuses on characterization surveys, the final status survey, engineering data, calculations, and dose modeling.		
XIII.a. ORGANIZATION		
☐ A description of the QA program management organization	8.1 Figure 8-1	8-2 8-2

CONTENT	SECTION	PAGE
☐ A description of the duties and responsibilities of each unit within the organization and how delegation of responsibilities is managed within the decommissioning program	8.1.1 8.1.2	8-3 8-4
☐ A description of how work performance is evaluated	8.2	8-4
☐ A description of the authority of each unit within the QA program	8.1.1 8.1.2	8-3 8-4
☐ An organization chart of the QA program organization	Figure 8-1	8-2
XIII.b. QUALITY ASSURANCE PROGRAM		
☐ A commitment that activities affecting the quality of site decommissioning will be subject to the applicable controls of the QA program and activities covered by the QA program are identified on *program defining documents*	8.3.1	8-7
☐ A brief summary of the company's [DOE's] corporate QA policies	8.3.1	8-7
☐ A description of provisions to ensure that technical and quality assurance procedures required to implement the QA program are consistent with regulatory, licensing, and QA program requirements and are properly documented and controlled	8.3	8-6
☐ A description of the management reviews, including the documentation of concurrence in these quality-affecting procedures	8.1.1 8.2.1 8.2.2	8-3 8-5 8-6
☐ A description of the quality-affecting procedural controls of the principal contractors	8.2.1 8.2.2 8.2.3 8.3.2	8-4 8-5 8-6 8-7
☐ A description of how NRC will be notified of changes (a) for review and acceptance in the accepted description of the QA program as presented or referenced in the DP before implementation and (b) in organizational elements within 30 days after the announcement of the changes	8.3.1	8-7
☐ A description is provided of how management regularly assesses the scope, status, adequacy, and compliance of the QA program	8.8	8-12
☐ A description of the instruction provided to personnel responsible for performing activities affecting quality	8.2.1 8.2.2 8.2.3 8.3.2	8-4 8-5 8-6 8-8

CONTENT	SECTION	PAGE
☐ A description of the training and qualifications of personnel verifying activities	8.3.1	8-7
☐ For formal training and qualification programs, documentation includes the objectives and content of the program, attendees, and date of attendance	8.9	8-13
☐ A description of the self-assessment program to confirm that activities affecting quality comply with the QA program	8.8	8-13
☐ A commitment that persons performing self-assessment activities are not to have direct responsibilities in the area they are assessing	8.8	8-13
☐ A description of the organizational responsibilities for ensuring that activities affecting quality are (a) prescribed by documented instructions, procedures, and drawings and (b) accomplished through implementation of these documents	8.1.1 8.1.2	8-3 8-4
☐ A description of the procedures to ensure that instructions, procedures, and drawings include quantitative acceptance criteria and qualitative acceptance criteria for determining that important activities have been satisfactorily performed	8.3.1	8-7
XIII.c. DOCUMENT CONTROL		
☐ A summary of the types of QA documents that are included in the program	8.4	8-11
☐ A description of how the licensee develops, issues, revises, and retires QA documents	8.4	8-11
XIII.d. CONTROL OF MEASURING AND TEST EQUIPMENT		
☐ A summary of the test and measurement equipment used in the program	8.5	8-12
☐ A description of how and at what frequency the equipment will be calibrated	8.5 9.4.3	8-12 9-11
☐ A description of the daily calibration checks that will be performed on each piece of test or measurement equipment	8.5	8-12
☐ A description of the documentation that will be maintained to demonstrate that only properly calibrated and maintained equipment was used during the decommissioning	8.5	8-12

CONTENT	SECTION	PAGE
XIII.e. CORRECTIVE ACTION		
☐ A description of the corrective action procedures for the facility, including a description of how the corrective action is determined to be adequate	8.7	8-12
☐ A description of the documentation maintained for each corrective action and any follow-up activities by the QA organization after the corrective action is implemented	8.7	8-12
XIII.f. QUALITY ASSURANCE RECORDS		
☐ A description of the manner in which the QA records will be managed	8.9	8-13
☐ A description of the responsibilities of the QA organization	8.1.1	8-3
☐ A description of the QA records storage facility	8.9	8-14
XIII.g. AUDITS AND SURVEILLANCES		
☐ A description of the audit program	8.8	8-14
☐ A description of the records and documentation generated during the audits and the manner in which the documents are managed	8.8	8-14
☐ A description of all follow-up activities associated with audits or surveillances	8.8	8-14
☐ A description of the trending/tracking that will be performed on the results of audits and surveillances	8.8	8-14
XIV. FACILITY RADIATION SURVEYS		
XIV.a. RELEASE CRITERIA		
The Phase 1 DP focuses on DCGLs for surface soil. subsurface soil, and streambed sediment. DCGLs are provided in Section 5 only to avoid duplication. Note that cleanup goals below the DCGLs are specified in Section 5 in Table 5-14 on page 5-48 – these are the criteria to be used for remediation activities in Phase 1.		
☐ A summary Table or list of the $DCGL_W$ for each radionuclide and impacted media of concern *[Table 5-14 provides the cleanup goals.]*	Table 5-14	5-48
If Class 1 survey units are present, a summary Table or list of area factors that will be used for determining a $DCGL_{EMC}$ for each radionuclide and media of concern	Table 9-1 Table 9-2 Table 9-3	9-3 9-3 9-4

CONTENT	SECTION	PAGE
☐ If Class 1 survey units are present, the $DCGL_{EMC}$ values for each radionuclide and medium of concern	Table 5-14	5-48
☐ If multiple radionuclides are present, the appropriate $DCGL_W$ for the survey method to be used *[A $DCGL_W$ for a surrogate radionuclide would be developed if practicable after additional characterization data are obtain during Phase 1 proposed decommissioning activities.]*	NA	NA
XIV.b. CHARACTERIZATION SURVEYS		
☐ A description and justification of the survey measurements for impacted media	9.2.4 9.4 9.7	9-6 9-8 9-22
☐ A description of the field instruments and methods that were used for measuring concentrations and the sensitivities of those instruments and methods	9.4 Table 9-4	9-10 9-10
☐ A description of the laboratory instruments and methods that were used for measuring concentrations and the sensitivities of those instruments and methods	9.4.1 9.4.3	9-10 9-14
☐ The survey results, including tables or charts of the concentrations of residual radioactivity measured *[The report of additional characterization to be performed early in Phase 1 of the proposed decommissioning would present data in tables and figures similar to those in Section 2 and Section 4.]*	Table 2-10 Table 2-19 Table 4-3 Table 4-4 Table 4-5 Table 4-6 Table 4-8 Table 4-9	2-19 2-43 4-15 4-15 4-16 4-16 4-18 4-20
☐ Maps or drawings of the site, area, or building, showing areas classified as non-impacted or impacted *[The drawings provided in Section 4 would be confirmed or revised when additional characterization data become available early in Phase 1 of the proposed decommissioning.]*	Figure 4-1 Figure 4-2 Figure 4-3 Figure 4-4 Figure 4-5	4-7 4-8 4-9 4-10 4-11
☐ Justification for considering areas to be non-impacted *[The justification provided in Section 4 would be confirmed or revised when additional characterization data become available early in Phase 1 of the proposed decommissioning.]*	4.1.3	4-12
☐ A discussion of why the licensee considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected *[The subsections of Section 9.7 provide justification for both previous and planned characterization measurements by WMA.]*	9.7	9-22

CONTENT	SECTION	PAGE
☐ For areas and surfaces that are inaccessible or not readily accessible, a discussion of how they were surveyed or why they did not need to be surveyed	9.4.1	9-11
☐ For sites, areas, or buildings with multiple radionuclides, a discussion justifying the ratios of radionuclides that will be assumed in the final status survey or an indication that no fixed ratio exists and each radionuclide will be measured separately	9.4.1	9-8
XIV.c. IN-PROCESS SURVEYS		
☐ A description of field screening methods and instrumentation	9.5	9-15
☐ A demonstration that field screening should be capable of detecting residual radioactivity at the DCGL *[As indicated in Section 9.5, methods and instruments for in-process surveys would be similar to those used during characterization and final status surveys. The field instruments suitable for scanning soil would not be able to detect non-gamma emitting radionuclides.]*	9.5 Table 9-5	9-15 9-18
XIV.d. FINAL STATUS SURVEY DESIGN *Phase 1 final status surveys would be performed in cases where the proposed decommissioning activities would make an area inaccessible for later final status surveys and confirmatory surveys. These surveys would be managed as final status surveys although a potential for recontamination may exist in certain areas. Details would be provided in the Final Status Survey Plan.*		
☐ A brief overview describing the final status survey design	9.6.1	9-15
☐ A description and map or drawing of impacted areas of the site, area, or building classified by residual radioactivity levels (Class 1, 2, or 3) and divided into survey units with an explanation of the basis for division into survey units *[Survey units would be specified in the Final Status Survey Plan as indicated in Section 9.6.1 on page 9-17.]*	9.6.1	9-17
☐ A description of the background reference areas and materials, if they will be used, and a justification for their selection *[Details would appear in the Final Status Survey Plan.]*	9.6.1	9-17
☐ A summary of the statistical tests that will be used to evaluate the survey results *[Details would appear in the Final Status Survey Plan.]*	9.6.1	9-20
☐ A description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide	Table 9-5 9.6.1	9-18 9-18

CONTENT	SECTION	PAGE
☐ For in-situ sample measurements made by field instruments, a description of the instruments, calibration, operational checks, sensitivity, and sampling methods, with a demonstration that the instruments and methods have adequate sensitivity *[The only field instruments planned for use are the instruments in Table 9-5 on page 9-18.]*	Table 9-5 9.6.1	9-18 9-18
☐ A description of the analytical instruments for measuring samples in the laboratory, as well as calibration, sensitivity, and methods with a demonstration that the instruments and methods have adequate sensitivity	9.6.1	9-18
☐ A description of how the samples to be analyzed in the laboratory will be collected, controlled, and handled	9.6.1	9-19
☐ A description of the final status survey investigation levels and how they were determined	9.6.1	9-16
☐ A summary of any significant additional residual radioactivity that was not accounted for during site characterization	9.6.1	9-16
☐ A summary of direct measurement results and/or soil concentration levels in units that are comparable to the DCGL, and if data is used to estimate or update the survey unit	9.6.1	9-16
☐ A summary of the direct measurements or sample data used to both evaluate the success of remediation and to estimate the survey unit variance	9.6.1	9-17
XIV.e. FINAL STATUS SURVEY REPORT *DOE is addressing each checklist topic as a requirement for the report.*		
☐ An overview of the results of the final status survey	9.8.1	9-37
☐ A discussion of any changes that were made in the final status survey from what was proposed in the DP or other prior submittals	9.8.2	9-37
☐ A description of the method by which the number of samples was determined for each survey unit	9.8.3	9-37
☐ A summary of the values used to determine the number of samples and a justification for these values	9.8.4	9-37
☐ The survey results for each survey unit include:	9.8.5	9-37
— The number of samples taken for the survey unit;	9.8.5	9-37

CONTENT	SECTION	PAGE
— A description of the survey unit, including (a) a map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units and random locations shown for Class 3 survey units and reference areas, and (b) a discussion of remedial actions and unique features;	9.8.5	9-37
— The measured sample concentrations in units that are comparable to the DCGL;	9.8.5	9-37
— The statistical evaluation of the measured concentrations;	9.8.5	9-38
— Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation;	9.8.5	9-38
— A discussion of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of $DCGL_W$; and	9.8.5	9-38
— A statement that a given survey unit satisfied the $DCGL_W$ and the elevated measurement comparison if any sample points exceeded the $DCGL_W$.	9.8.5	9-38
☐ A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity (e.g., material not accounted for during site characterization)	9.8.6	9-38
☐ A description of how ALARA practices were employed to achieve final activity levels	9.8.5	9-38
☐ If a survey unit fails, a description of the investigation conducted to ascertain the reason for the failure and a discussion of the impact that the failure has on the conclusion that the facility is ready for final radiological surveys and that it satisfies the release criteria	9.8.7	9-38
☐ If a survey unit fails, a discussion of the impact that the reason for the failure has on other survey unit information	9.8.8	9-38
XV. FINANCIAL ASSURANCE		
This matter is not applicable to the Phase 1 DP consistent with 10 CFR 30.35(f)(4).		
XV.a. COST ESTIMATE		
~~☐ A cost estimate that appears to be based on documented and reasonable assumptions~~	NA	NA

CONTENT	SECTION	PAGE
XV.b. CERTIFICATION STATEMENT		
~~☐ The certification statement is based on the licensed possession limits and the applicable quantities specified in 10 CFR 30.35, 40.36, or 70.25~~	NA	NA
~~☐ The licensee is eligible to use a certification of financial assurance and, if eligible, that the certification amount is appropriate~~	NA	NA
~~☐ The financial assurance mechanism supplied by the licensee consists of one or more of the following instruments:~~ ~~Trust fund;~~ ~~Escrow account;~~ ~~Government fund;~~ ~~Certificate of deposit;~~ ~~Deposit of government securities;~~ ~~Surety bond;~~ ~~Letter of credit;~~ ~~Line of credit;~~ ~~Insurance policy;~~ ~~Parent company guarantee;~~ ~~Self guarantee;~~ ~~External sinking fund;~~ ~~Statement of intent; or~~ ~~By special arrangements with a government entity assuming custody or ownership of the site.~~	NA	NA
XV.c. FINANCIAL MECHANISM		
~~☐ The financial assurance mechanism is an originally signed duplicate~~	NA	NA
~~☐ The wording of the financial assurance mechanism is identical to the recommended wording provided in Appendix F of this document~~	NA	NA
~~☐ For a licensee regulated under 10 CFR Part 72, a means is identified in the DP for adjusting the financial assurance funding level over any storage and surveillance period~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~The amount of financial assurance coverage provided by the licensee for site control and maintenance is at least as great as that calculated using the formula provided in this NUREG~~	NA	NA
XVI. RESTRICTED USE/ALTERNATE CRITERIA		
Because there would be no facility or property release associated with the Phase 1 of the proposed decommissioning, this section does not apply.		
XVI.a. RESTRICTED USE		
XVI.a.1. Eligibility Demonstration		
☐ ~~A demonstration that the benefits of dose reduction are less than the cost of doses, injuries, and fatalities~~	NA	NA
☐ ~~A demonstration that the proposed residual radioactivity levels at the site are ALARA~~	NA	NA
XVI.a.2. Institutional Controls		
DOE would continue to manage the project premises after completion of the Phase 1 proposed decommissioning work until the actions required by the WVDP Act have been completed. DOE's site management plan for the post-Phase 1 period would provide de facto institutional control of the site during this period. Accordingly, DOE would briefly describe this plan, addressing the topics identified as applicable below as they apply to the post-Phase 1 period under DOE control.		
☐ ~~A description of the legally enforceable institutional control(s) and an explanation of how the institutional control is a legally enforceable mechanism~~	NA	NA
☐ ~~A description of any detriments associated with the maintenance of the institutional control(s)~~	NA	NA
☐ ~~A description of the restrictions on present and future landowners~~	NA	NA
☐ A description of the entities enforcing, and their authority to enforce, the institutional control(s)	App D	D-23
☐ A description of the design features of the site that support institutional controls	App D	D-23
☐ A discussion of the durability of the institutional control(s), including the performance of any engineered barriers used	App D	D-5
☐ ~~A description of the activities that the entity with the authority to enforce the institutional controls may undertake to enforce the institutional control(s)~~	NA	NA

CONTENT	SECTION	PAGE
☐ ~~A description of the manner in which the entity with the authority to enforce the institutional control(s) will be replaced if that entity is no longer willing or able to enforce the institutional control(s) (this may not be needed for Federal or State entities)~~	NA	NA
☐ ~~A description of the duration of the institutional control(s), the basis for the duration, the conditions that will end the institutional control(s), and the activities that will be undertaken to end the institutional control(s)~~	NA	NA
☐ ~~A description of the plans for corrective actions that may be undertaken in the event the institutional control(s) fail~~	NA	NA
☐ ~~A description of the records pertaining to the institutional controls, how and where will they will be maintained, and how the public will have access to the records~~	NA	NA
XVI.a.3. Site Maintenance and Financial Assurance		
☐ ~~A demonstration that an appropriately qualified entity has been provided to control and maintain the site~~	NA	NA
☐ A description of the site maintenance and control program and the basis for concluding that the program is adequate to control and maintain the site	App D	D-10
☐ ~~A description of the arrangement or contract with the entity charged with carrying out the actions necessary to maintain control at the site~~	NA	NA
☐ ~~A demonstration that the contract or arrangement will remain in effect for as long as feasible, and include provisions for renewing or replacing the contract~~	NA	NA
☐ ~~A description of the manner in which independent oversight of the entity charged with maintaining the site will be conducted and what entity will conduct the oversight~~	NA	NA
☐ ~~A demonstration that the entity providing the oversight has the authority to replace the entity charged with maintaining the site~~	NA	NA
☐ ~~A description of the authority granted to the third party to perform, or have performed, any necessary maintenance activities~~	NA	NA
☐ ~~Unless the entity is a government entity, a demonstration that the third party is not the entity holding the financial assurance mechanism~~	NA	NA
☐ ~~A demonstration that sufficient records evidencing to official actions and financial payments made by the third party are open to public inspection~~	NA	NA

CONTENT	SECTION	PAGE
~~A description of the periodic site inspections that will be performed by the third party, including the frequency of the inspections~~	NA	NA
~~A copy of the financial assurance mechanism provided by the licensee~~	NA	NA
~~A demonstration that the amount of financial assurance provided is sufficient to allow an independent third party to carry out any necessary control and maintenance activities~~	NA	NA
XVI.a.4. Obtaining Public Advice		
This section does not apply because public advice is not being sought under the provisions of 10 CFR 20.1403(d) to support license termination under restricted conditions.		
~~A description of how individuals and institutions that may be affected by the decommissioning were identified and informed of the opportunity to provide advice to the licensee~~	NA	NA
~~A description of the manner in which the licensee obtained advice from these individuals or institutions~~	NA	NA
~~A description of how the licensee provided for participation by a broad cross-section of community interests in obtaining the advice~~	NA	NA
~~A description of how the licensee provided for a comprehensive, collective discussion on the issues by the participants represented~~	NA	NA
~~A copy of the publicly available summary of the results of discussions, including individual viewpoints of the participants on the issues, and the extent of agreement and disagreement among the participants~~	NA	NA
~~A description of how this summary has been made available to the public~~	NA	NA
~~A description of how the licensee evaluated the advice, and the rationale for incorporating or not incorporating the advice from affected members of the community into the DP~~	NA	NA
XVI.a.5. Dose Modeling and ALARA Demonstration		
~~A summary of the dose to the average member of the critical group when radionuclide levels are at the DCGL with institutional controls in place, as well as the estimated doses if they are no longer in place~~	NA	NA
~~A summary of the evaluation performed pursuant to Chapter 6 of Volume 2 of this NUREG series, demonstrating that these doses are ALARA~~	NA	NA

CONTENT	SECTION	PAGE
~~If the estimated dose to the average member of the critical group could exceed 100 mrem/y (but would be less than 500 mrem/y) when the radionuclide levels are at the DCGL, a demonstration that the criteria in 10 CFR 20.1403(e) have been met~~	NA	NA
XVI.b. ALTERNATE CRITERIA		
~~A summary of the dose in TEDE(s) to the average member of the critical group when the radionuclide levels are at the DCGL (considering all man-made sources other than medical)~~	NA	NA
~~A summary of the evaluation performed pursuant to Chapter 6 of Volume 2 of this NUREG series demonstrating that these doses are ALARA~~	NA	NA
~~An analysis of all possible sources of exposure to radiation at the site and a discussion of why it is unlikely that the doses from all man-made sources, other than medical, will be more than 1 mSv/y (100 mrem/y)~~	NA	NA
~~A description of the legally enforceable institutional control(s) and an explanation of how the institutional control is a legally enforceable mechanism~~	NA	NA
~~A description of any detriments associated with the maintenance of the institutional control(s)~~	NA	NA
~~A description of the restrictions on present and future landowners~~	NA	NA
~~A description of the entities enforcing and their authority to enforce the institutional control(s)~~	NA	NA
~~A discussion of the durability of the institutional control(s)~~	NA	NA
~~A description of the activities that the party with the authority to enforce the institutional controls will undertake to enforce the institutional control(s)~~	NA	NA
~~A description of the manner in which the entity with the authority to enforce the institutional control(s) will be replaced if that entity is no longer willing or able to enforce the institutional control(s)~~	NA	NA
~~A description of the duration of the institutional control(s), the basis for the duration, the conditions that will end the institutional control(s), and the activities that will be undertaken to end the institutional control(s)~~	NA	NA
~~A description of the corrective actions that will be undertaken in the event the institutional control(s) fail~~	NA	NA
~~A description of the records pertaining to the institutional controls, how and where they will be maintained, and how the public will have access to the records~~	NA	NA

CONTENT	SECTION	PAGE
~~☐ A description of how individuals and institutions that may be affected by the decommissioning were identified and informed of the opportunity to provide advice to the licensee~~	NA	NA
~~☐ A description of the manner in which the licensee obtained advice from affected individuals or institutions~~	NA	NA
~~☐ A description of how the licensee provided for participation by a broad cross-section of community interests in obtaining the advice~~	NA	NA
~~☐ A description of how the licensee provided for a comprehensive, collective discussion on the issues by the participants represented~~	NA	NA
~~☐ A copy of the publicly available summary of the results of discussions, including individual viewpoints of the participants on the issues and the extent of agreement and disagreement among the participants~~	NA	NA
~~☐ A description of how this summary has been made available to the public~~	NA	NA
~~☐ A description of how the licensee evaluated advice from individuals and institutions that could be affected by the decommissioning and the manner in which the advice was addressed~~	NA	NA

References

NRC 2006, NUREG-1757, *Consolidated Decommissioning Guidance*, Volume 1, Revision 2. U.S. Nuclear Regulatory Commission, Washington, D.C., September 2006.

NRC 2008, *Summary of a Meeting Between NRC and DOE on the WVDP Phase 1 Decommissioning Plan*, May 19, 2008.

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APPENDIX B
ENVIRONMENTAL RADIOACTIVITY DATA

PURPOSE OF THIS APPENDIX

The purpose of this appendix is to provide information on radioactivity in environmental media to supplement information in Section 4.2. This appendix discusses how radionuclide-specific and media-specific background values were developed and describes the methods used to determine whether specific areas of the site have been impacted (i.e., contain media with radioactivity concentrations in excess of background).

INFORMATION IN THIS APPENDIX

This appendix identifies locations used in establishing background radioactivity concentrations and methods used for calculating these concentrations. It also provides tables of background summary data for each environmental medium, explains methods used to evaluate concentrations exceeding background in onsite environmental media, provides tables of radionuclide ratios, and provides summary data of radioactivity concentrations and status with respect to background at onsite routine monitoring locations. Supplementary data for groundwater sampling points (e.g., location coordinates, sample depth, geologic unit) are also provided.

RELATIONSHIP TO OTHER PARTS OF THE PLAN

The information in this appendix supplements that provided in Section 4.2 and supports planning for additional characterization of soil and sediment to be performed early in Phase 1 of the proposed decommissioning in accordance with the Characterization Sample and Analysis Plan described in Section 9.

1.0 Locations Used for Background Calculations

Samples of surface soil, sediment, surface water, and groundwater are routinely collected from background locations (i.e., "control" or "rference" locations) as part of the WVDP *Environmental Monitoring Program Plan* (WVES 2008a) and the WVDP *Groundwater Monitoring Plan* (WVES 2008b). Environmental radiation measurements are also taken with thermoluminescent dosimeters (TLDs) at background locations as described in the *Environmental Monitoring Program Plan*. Location designators beginning with a "W" indicate a water sample. Those beginning with an "S" indicate soil or sediment samples. A designator beginning with a "D" indicates direct measurement of environmental exposure.

1.1 Surface Soil

Surface soil samples were collected annually until 2004, when the collection period was reduced to once every three years. Data from only two background locations were available. One (SFGRVAL, located at the air sampling station in Great Valley) is the primary (and current) background location. The other (SFNASHV, located at the former air sampling station at Nashville) was discontinued in 2003. (See Figure B-1.) Therefore, few data points were available to calculate surface soil backgrounds.

To increase the number of data points for estimating background radionuclide concentrations, data from soil collected at other offsite sampling locations (i.e., at perimeter locations and in the nearby communities of West Valley and Springville) were evaluated for the possibility of using data from each in soil background calculations. Data sets for each radionuclide from each soil sampling location (1995-2007) were statistically compared with the comparable data set from the primary background location, SFGRVAL, using the nonparametric Mann-Whitney U-test (Sheskin 1997). The null hypothesis being tested was that the median of the test data set was higher than the median at the reference data set (SFGRVAL) (one-tailed test, $P<0.05$). The results are summarized in Table B-1 below, with the sample locations shown in Figure B-1 or B-2. (Note that, at the 0.05 level, the possibility of making an incorrect decision regarding the status of the location with respect to background could have occurred by chance alone five percent of the time.)

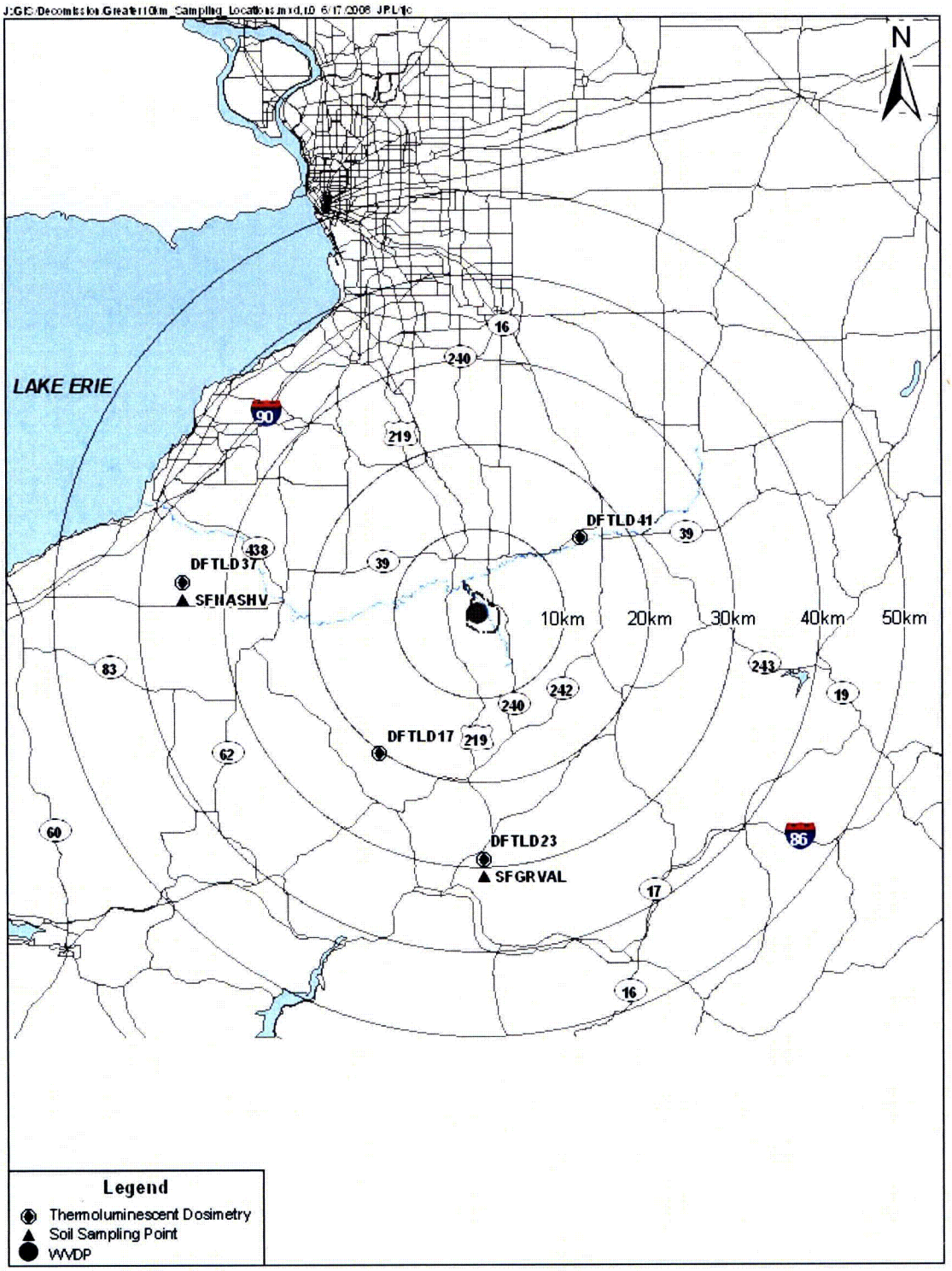

Figure B-1. Background Sampling Locations More Than 10 Kilometers From the WVDP

Table B-1. Summary of Comparisons of Radionuclide Data from Test Surface Soil Locations vs. SFGRVAL Background

Location	Radionuclide Measurement										
	Gross alpha	Gross Beta	Sr-90	Cs-137	U-232	U-233/ 234	U-235/ 236	U-238	Pu-238	Pu-239/ 240	Am-241
SFGRVAL vs.											
SFNASHV	NS	NS	NS	NS	---	---	---	---	NS	NS	NS
SFFXVRD	NS	NS	NS	NS	---	---	---	---	NS	NS	NS
SFTCORD	NS	**Higher**	NS	NS	---	---	---	---	NS	NS	NS
SFRT240	NS	NS	NS	NS	---	---	---	---	NS	NS	NS
SFSPRVL	NS	NS	NS	NS	---	---	---	---	NS	NS	NS
SFWEVAL	NS	NS	NS	NS	---	---	---	---	NS	NS	NS
SFBOEHN	NS	NS	NS	NS	NS	**Higher**	NS	NS	NS	NS	NS
SFRSPRD	NS	NS	NS	**Higher**	NS	NS	NS	NS	NS	NS	NS
SFBLKST	NS	**Higher**	NS	NS	---	---	---	---	NS	NS	NS

KEY: **Higher** = Null hypothesis was not rejected; results higher than background (P<0.05).
NS = Null hypothesis was rejected; results were not significantly higher than background.
--- = Constituent was not measured at this location.

LOCATION CODES: SFGRVAL = Background at Great Valley;
SFNASHV = Background at Nashville in the town of Hanover;
SFTCORD = Perimeter at Thomas Corners Road;
SFRT240 = Perimeter at Route 240;
SFSPRVL = Community at Springville;
SFWEVAL = Community at West Valley;
SFBOEHN = Perimeter at Boehn Road;
SFRSPRD = Perimeter at Rock Springs Road;
SFBLKST = Perimeter at Bulk Storage Warehouse.
(Location SFNASHV was discontinued in 2003; locations SFTCORD, SFBOEHN, and SFBLKST were discontinued 2005.)
See Figures B-1 and B-2 for sample locations.

If data were determined not to be statistically higher than background (i.e., unlikely to have been impacted by the WVDP, indicated by "NS" results in the above table), the data were pooled with data from Great Valley and included in background calculations.

As discussed in Section 4.2.1 of this plan, data were extracted from the WVDP Laboratory Information Management System. Samples from which the data were taken had been collected and analyzed in accordance with controlled sampling plans and defined quality assurance protocols. All data used for background calculations were independently validated and approved.

Although not all analyses were performed by the same laboratories over the years, before a laboratory was awarded a contract, analytical procedures were reviewed, laboratories were audited by WVDP personnel familiar with radioanalytical methods, and

performance on proficiency samples for the radionuclides of interest were examined for acceptability. Analysis of gamma-emitting radionuclides – i.e., U-232, U-233/234, U-235/236, U-238, Pu-238, Pu-239/240, and Am-241 – was done by alpha spectrometry to meet contractual detection limits. After contracts were awarded, laboratories were contractually required to participate in formal crosscheck programs and perform acceptably. During the term of the contracts, laboratories were routinely audited by WVDP personnel to ensure that contractually required standards were maintained.

1.2 Subsurface soil

Data from only two boreholes (BH-38 on the north plateau and BH-39 on the south plateau) were available for this calculation. The boreholes were driven into areas of the WVDP classified as non-impacted as part of a Resource Conservation and Recovery Act (RCRA) Facility Investigation (RFI) soil characterization study in 1993. (See Figure B-3.) Although samples were taken from three depths at each borehole, the surficial samples (0-2 feet depth) were classified as surface soil for the purposes of this plan. Therefore, only two samples from each borehole, a total of four samples, were classified as subsurface soil. Although subsurface soil background values were calculated from these four data points, they were not used as reference values because there were too few points and because the onsite locations were potentially affected by historical activities at the site. Surface soil background results were used to evaluate the presence of radionuclide concentrations in excess of background in subsurface soil samples.

1.3 Surface Water and Sediment

The routine Environmental Monitoring Program background locations were used as the source of background data. Both surface water and sediment background data were taken from samples collected at Buttermilk Creek upstream of the WVDP (surface water monitoring point WFBCBKG and sediment monitoring point SFBCSED) and at Bigelow Bridge on Cattaraugus Creek upstream of the point where Buttermilk Creek, containing effluent from the WVDP, flows into Cattaraugus Creek (surface water point WFBIGBR and sediment point SFBISED). (See Figure B-2.)

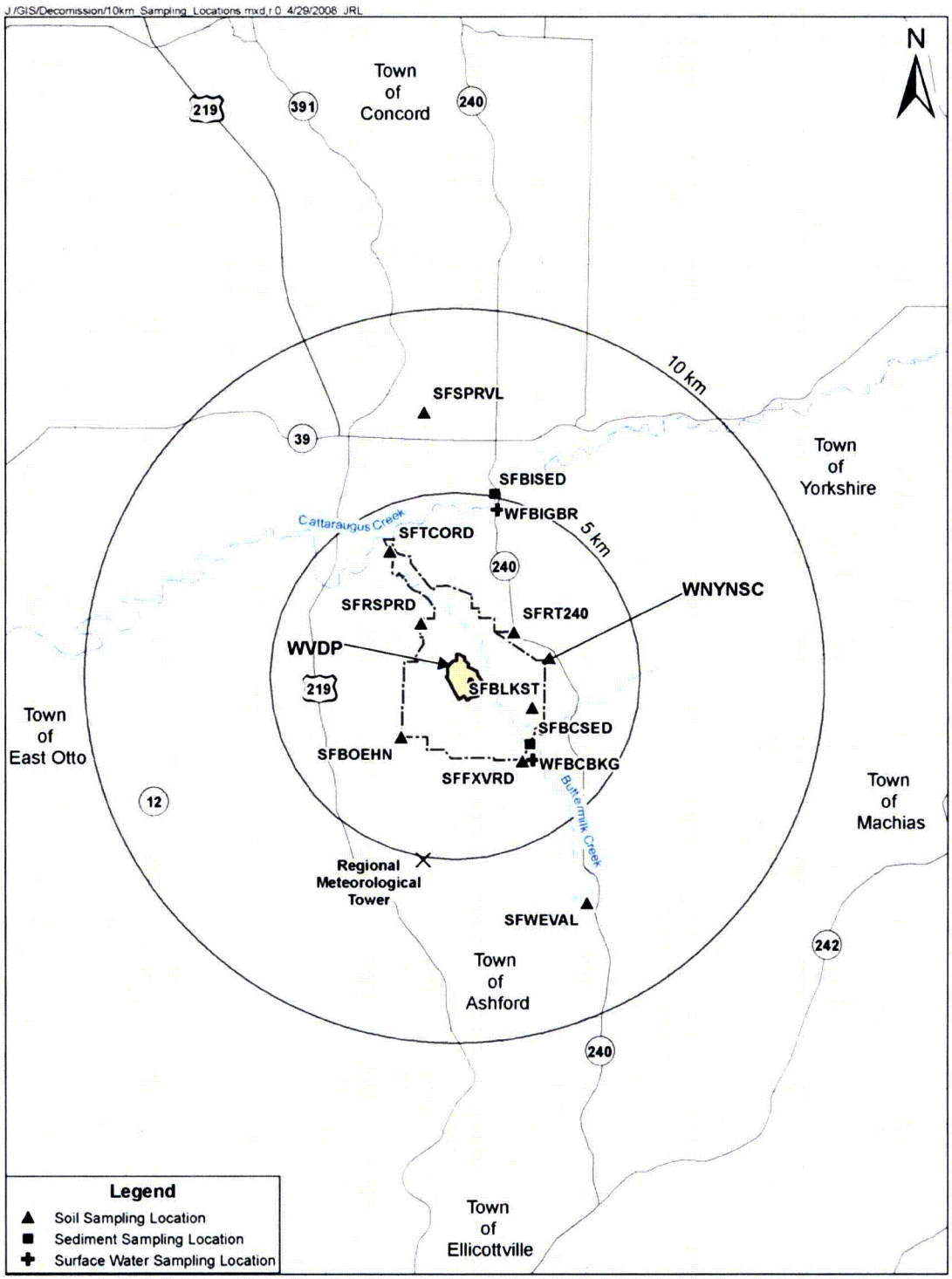

Figure B-2. Sampling Locations Within 10 Kilometers of the WVDP Used for Background Calculations

1.4 Groundwater

The routine background locations from the Groundwater Monitoring Program were used as the source of background data. (See Figure B-3.) Radionuclide concentrations were taken from monitoring wells WNWNB1S, WNW0204, WNW0301, WNW0401, WNW0405, WNW0706, WNW0901, and WNW0908, which serve(d) as upgradient reference locations for the following geologic units: the sand and gravel (S&G) unit (WNWNB1S, WNW0301, WNW0401, and WNW0706); the Lavery till sand (LTS) unit (WNW0204); the unweathered Lavery till (ULT) unit (WNW0405); the Kent recessional sequence (KRS) unit (WNW0901); and the weathered Lavery till (WLT) unit (WNW0908).

Because few background data points were available for most radionuclides in groundwater and no background isotopic data (or very limited data) were available for groundwater from some of the geological units (e.g., the Lavery till sand and the Kent recessional sequence), data sets for the various units were combined to calculate one overall site groundwater background value for each radionuclide. Potential implications of pooling the data were considered to be minimal because most of the data sets were comprised largely of nondetect values as shown in Table B-7, and because, when positive detects were noted (with the exception of naturally occurring radionuclides), they were usually below (or slightly higher than) the contractual detection limits.

1.5 Gamma Radiation Measurements From TLDs

TLD data were taken from four background locations (three no longer active) over the 1986-2007 time period. (See Figure B-1.) Measurements were taken at:

(1) The current background location (DFTLD23), located 18 miles (29 km) south of the WVDP at the Great Valley air sampler;

(2) The five-points landfill (DFTLD17), located 12 miles (19 km) southwest of the Site;

(3) The former air sampling location at Nashville in the town of Hanover (DFTLD37), located 23 miles (37 km) northwest of the Site; and

(4) Sardinia-Savage Road (DFTLD41), 15 miles (24 km) northeast of the Site.

Quarterly exposure rates (in mR/qtr) and hourly exposure rates (in mR/h) were calculated.

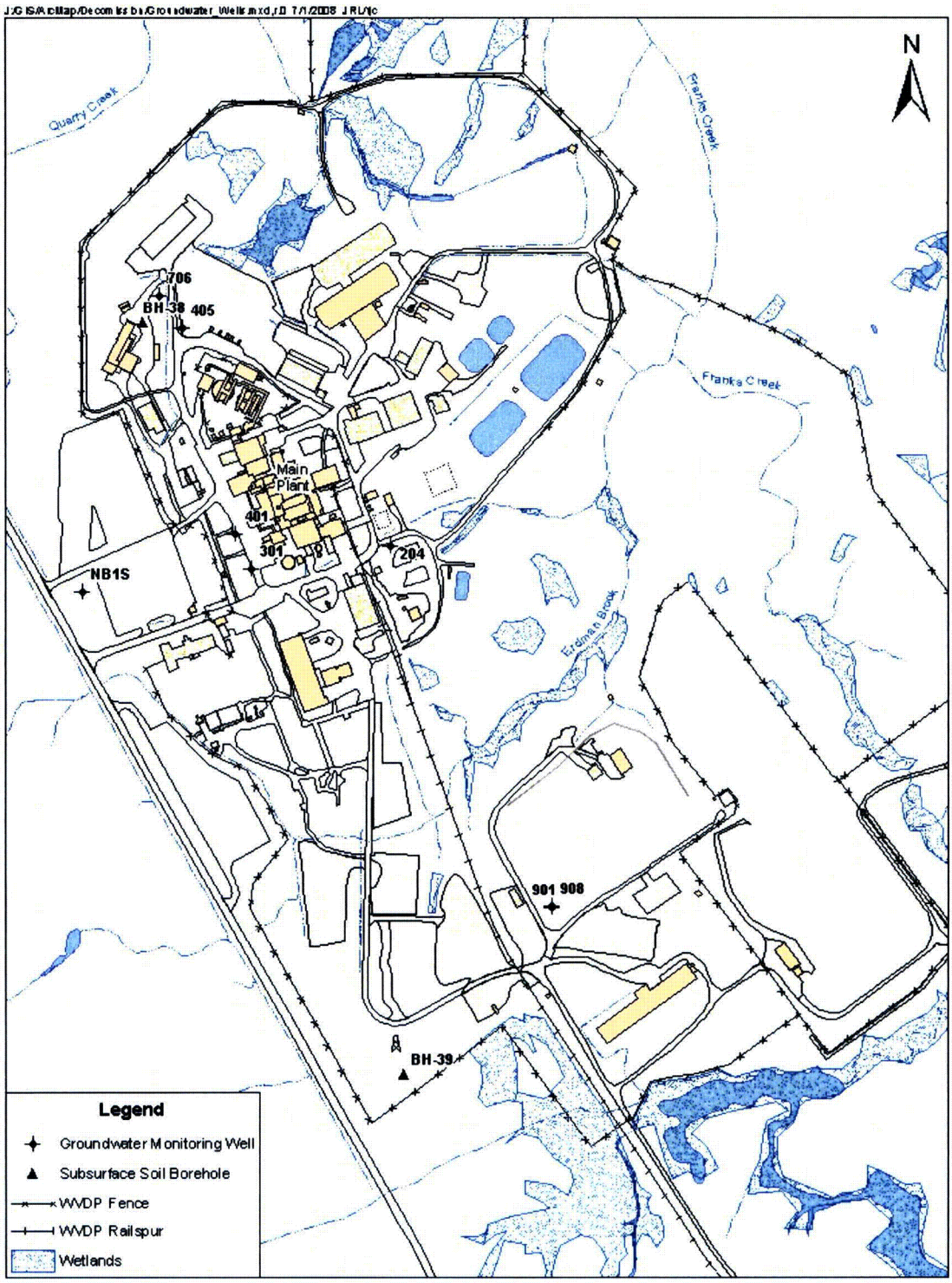

Figure B-3. Onsite Groundwater Wells and Subsurface Soil Boreholes Used as Background

2.0 Methods Used for Background Calculations

Radionuclides for which backgrounds were estimated were selected with consideration of (1) radionuclides of interest from the Facility Characterization Project, as listed in Decommissioning Plan section 4.1.1, and (2) radionuclides that are routinely monitored in environmental media at the WVDP, for which sufficient data were available to develop a reliable estimate of background. (See Section 4.2.2 of this plan for a more detailed discussion of how background constituents were selected.)

Once radionuclides and locations applicable to each environmental medium had been defined, sample results were extracted from the Laboratory Information Management System database using the Environmental Affairs Trend Tool. As part of the extraction process, data from duplicate samples (i.e., separate samples of one medium collected at the same place and time; co-located samples) were combined into a single result for use in calculations, as were data from replicate samples (i.e., recounts or splits of the same sample). Calculations to combine results from duplicates and replicates, using protocols defined in controlled WVDP Procedure EM-11 (WVNSCO 2004b), were automatically done by the Environmental Affairs Trend Tool during data extraction.

Extracted data files were block copied into Microsoft Excel® spreadsheets and the information identified in Table B-2 was summarized for each environmental medium.

Table B-2. Summary Information for Environmental Medium Background Calculations

Item	Explanatory Notes
Constituent	Gross measurement, radionuclide measurement, or direct radiation measurement
Average result	In the LIMS database, individual radionuclide concentration measurements are represented by a result term plus or minus an associated uncertainty term. The average result is the direct average of result terms from all samples in the data set, including negative numbers and zeros.
Uncertainty associated with the average result	The uncertainty term associated with the average result is calculated from the sample uncertainty terms in accordance with Procedure EM-11 per the following formula: $\text{uncertainty} = \text{SQRT}((\text{uncertainty}_1^2 + \ldots + \text{uncertainty}_N^2) / N)$ where uncertainty_1 = the uncertainty term from sample 1 uncertainty_N = the uncertainty term from sample N N = the total number of samples SQRT = square root
Median	To estimate the median of each data set, each sample result±uncertainty was assigned a single result equal to the larger of the result or the uncertainty term. Using the Excel® median function, the median was selected from the set of single values. If more than half the sample results were nondetects, the median was assigned a "<" sign, indicating that the median represented a nondetect value.

Table B-2. Summary Information for Environmental Medium Background Calculations

Item	Explanatory Notes
	Note that if a data set is symmetric, the average and median will be the same. However, if the distribution is skewed to the right (that is, it contains a large number of low values and a few high values), the average will usually be higher than the median. For this reason, with asymmetrically distributed data sets (as is often the case with environmental data) the median may be the more reliable estimator of central tendency.
Maximum	The maximum was selected from only the results indicating that activity had been detected. If no activity had been detected in any of the samples from that data set, the maximum was set equal to the highest uncertainty term and assigned a "<" sign, indicating that it was a nondetect.
N	Total number of samples. (Duplicate samples were counted as one, as were replicate samples.)
% NDs	If the uncertainty term for a sample was larger than the result (i.e., the range around the result term included zero), the radionuclide was considered not detected (ND) in that sample. Total number of ND samples divided by the total number of samples was expressed as a percentage.
Years	The period of years from which the data set was taken.
Data source locations	A listing of the sampling locations from which background data were taken.

Soil and sediment data, as extracted from the Laboratory Information Management System, were in units of µCi/g (dry weight). Surface water and groundwater data were in units of µCi/mL. All calculations were performed in units as extracted from the Laboratory Information Management System. Environmental dosimetry readings were in mR/qtr. For comparisons with onsite sample results, background data were then converted to the units specified in the Decommissioning Plan using the following conversion factors:

Soil and sediment: 1 µCi/g = 1E+06 pCi/g

Water: 1 µCi/mL = 1E+09 pCi/L

3.0 Background Summary Data for Each Environmental Medium

Summary tables of background values (in units of pCi/g per unit dry weight [soil or sediment], pCi/L [surface water and groundwater], or mR/quarter [environmental exposure]) used to evaluate data from onsite sampling locations are presented in the following tables.

Table B-3. Surface Soil Background Radionuclide Concentrations for the WVDP[(1),(2)]

Constituent	Avg. Concentration (pCi/g) Result	±	Uncertainty	Median (pCi/g)	Maximum (pCi/g)	N	% NDs	Years	Data Source Locations
Gross alpha	1.34E+01	±	3.58E+00	1.29E+01	2.73E+01	104	0%	1995-2007	SFGRVAL, SFNASHV, SFFXVRD, SFTCORD, SFRT240, SFSPRVL, SFWEVAL, SFBOEHN, SFRSPRD, SFBLKST
Gross beta	2.03E+01	±	3.11E+00	2.00E+01	4.00E+01	84	0%	1995-2007	SFGRVAL, SFNASHV, SFFXVRD, SFRT240, SFSPRVL, SFWEVAL, SFBOEHN, SFRSPRD
Sr-90	1.51E-01	±	1.46E-01	9.48E-02	3.10E+00	104	25%	1995-2007	SFGRVAL, SFNASHV, SFFXVRD, SFTCORD, SFRT240, SFSPRVL, SFWEVAL, SFBOEHN, SFRSPRD, SFBLKST
Cs-137	4.50E-01	±	6.68E-02	4.17E-01	1.21E+00	93	0%	1995-2007	SFGRVAL, SFNASHV, SFFXVRD, SFTCORD, SFRT240, SFSPRVL, SFWEVAL, SFBOEHN, SFBLKST
U-232	5.52E-03	±	2.80E-02	< 2.35E-02	1.89E-02	32	97%	1995-2007	SFGRVAL, SFBOEHN, SFRSPRD
U-233/234	7.79E-01	±	1.15E-01	7.88E-01	9.39E-01	22	0%	1995-2007	SFGRVAL, SFRSPRD
U-235/236	5.98E-02	±	3.36E-02	5.24E-02	2.18E-01	32	9%	1995-2007	SFGRVAL, SFBOEHN, SFRSPRD
U-238	7.79E-01	±	1.13E-01	7.87E-01	9.31E-01	32	0%	1995-2007	SFGRVAL, SFBOEHN, SFRSPRD
Pu-238	5.39E-03	±	1.38E-02	< 1.21E-02	4.02E-02	92	86%	1996-2007	SFGRVAL, SFNASHV, SFFXVRD, SFTCORD, SFRT240, SFSPRVL, SVWEVAL, SFBOEHN, SFRSPRD, SFBLKST
Pu-239/240	2.01E-02	±	1.79E-02	1.55E-02	2.34E-01	104	44%	1995-2007	SFGRVAL, SFNASHV, SFFXVRD, SFTCORD, SFRT240, SFSPRVL, SFWEVAL, SFBOEHN, SFRSPRD, SFBLKST
Am-241	1.45E-02	±	1.92E-02	< 1.62E-02	1.93E-01	104	64%	1995-2007	SFGRVAL, SFNASHV, SFFXVRD, SFTCORD, SFRT240, SFSPRVL, SFWEVAL, SFBOEHN, SFRSPRD, SFBLKST

LEGEND: N = Number of samples
ND = Nondetect

NOTES: (1) Soil samples collected at air samplers at background locations (SFGRVAL = Great Valley; SFNASHV = Nashville), perimeter locations (SFFXVRD = Fox Valley Road; SFTCORD = Thomas Corners Road; SFRT240 = Route 240; SFBOEHN = Boehn Road; SFRSPRD = Rock Springs Road; SFBLKST = Bulk Storage Warehouse), and community locations (SFSPRVL = Springville; SFWEVAL = West Valley).

(2) Data from perimeter and community samplers were pooled with data from background locations if they were not statistically higher than background.

Table B-4. Sediment Background Radionuclide Concentrations for the WVDP[1]

Constituent	Average concentration (pCi/g)			Median (pCi/g)	Maximum (pCi/g)	N	% NDs	Years	Data Source Locations
	Result	±	Uncertainty						
Gross alpha	1.02E+01	±	3.28E+00	9.21E+00	2.18E+01	22	0%	1995-2006	SFBCSED, SFBISED
Gross beta	1.74E+01	±	3.01E+00	1.64E+01	2.71E+01	23	0%	1995-2007	SFBCSED, SFBISED
Sr-90	1.49E-02	±	4.91E-02	< 3.35E-02	1.57E-01	23	65%	1995-2007	SFBCSED, SFBISED
Cs-137	3.50E-02	±	2.50E-02	3.75E-02	7.84E-02	23	30%	1995-2007	SFBCSED, SFBISED
U-232	1.15E-02	±	5.50E-02	< 3.10E-02	3.92E-02	23	87%	1995-2007	SFBCSED, SFBISED
U-233/234	5.99E-01	±	1.19E-01	6.59E-01	8.58E-01	23	4%	1995-2007	SFBCSED, SFBISED
U-235/236	5.31E-02	±	3.67E-02	4.57E-02	2.78E-01	23	22%	1995-2007	SFBCSED, SFBISED
U-238	6.11E-01	±	1.19E-01	6.52E-01	9.01E-01	23	4%	1995-2007	SFBCSED, SFBISED
Pu-238	1.67E-02	±	1.79E-02	< 1.41E-02	1.29E-01	23	74%	1995-2007	SFBCSED, SFBISED
Pu-239/240	1.08E-02	±	1.37E-02	< 1.22E-02	6.07E-02	23	83%	1995-2007	SFBCSED, SFBISED
Am-241	1.07E-02	±	1.83E-02	< 1.41E-02	8.60E-02	23	74%	1995-2007	SFBCSED, SFBISED

LEGEND: N = Number of samples
ND = Nondetect

NOTE: (1) Sediment samples were collected at upstream sampling locations on Buttermilk Creek (SFBCSED) and Cattaraugus Creek (SFBISED).

Table B-5. Subsurface Soil Background Radionuclide Concentrations for the WVDP(1)

Constituent	Average concentration (pCi/g) Result	±	Uncertainty	Median (pCi/g)	Maximum (pCi/g)	N(2)	% NDs	Years	Data Source Locations
Gross alpha	1.40E+01	±	5.52E+00	1.40E+01	1.50E+01	4	0%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
Gross beta	5.28E+01	±	3.77E+00	5.15E+01	6.10E+01	4	0%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
Sr-90	3.20E-02	±	4.00E-02	< 3.20E-02	< 6.00E-02	4	100%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
Cs-137	1.02E-02	±	2.35E-02	< 2.30E-02	< 2.70E-02	4	100%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
U-232	4.21E-03	±	1.42E-02	< 1.02E-02	< 2.40E-02	4	100%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
U-233/234	1.53E-01	±	3.61E-02	1.55E-01	1.70E-01	4	0%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
U-235/236	6.05E-03	±	9.45E-03	< 1.02E-02	1.14E-02	4	75%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
U-238	1.12E-01	±	3.13E-02	1.15E-01	1.40E-01	4	0%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
Pu-238	2.53E-03	±	1.08E-02	< 7.14E-03	< 1.83E-02	4	100%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
Pu-239/240	1.26E-03	±	1.04E-02	< 6.19E-03	< 1.83E-02	4	100%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)
Am-241	2.96E-03	±	8.41E-03	< 7.96E-03	< 1.07E-02	4	100%	1993	BH-38 (north plateau), BH-39 (south plateau) (except for 0-2' sample)

LEGEND: N = Number of samples
ND = Nondetect

NOTE: (1) Subsurface soil background samples were collected in 1993 at borehole 38 on the north plateau (BH-38), and at borehole 39 on the south plateau (BH-39). Two samples were collected from each. (The 0-2' depth sample from each was not included in the subsurface background calculation. It was classified as a surface soil sample.)

(2) Surface soil background concentrations in Table B-3 were used to evaluate subsurface soil samples because too few subsurface soil background data were available.

Table B-6. Surface Water Background Radionuclide Concentrations for the WVDP

Constituent	Average concentration (pCi/L)			Median (pCi/L)	Maximum (pCi/L)	N	% NDs	Years	Data Source Locations
	Result	±	Uncertainty						
Gross alpha	4.74E-01	±	1.28E+00	< 9.55E-01	5.43E+00	387	74%	1991-2007	WFBCBKG, WFBIGBR
Gross beta	2.64E+00	±	1.43E+00	2.34E+00	2.03E+01	388	12%	1991-2007	WFBCBKG, WFBIGBR
H-3	1.35E+01	±	8.43E+01	< 8.21E+01	6.33E+02	388	85%	1991-2007	WFBCBKG, WFBIGBR
C-14	1.19E+01	±	4.44E+01	< 1.33E+01	4.05E+02	68	81%	1991-2007	WFBCBKG
Sr-90	2.00E+00	±	1.61E+00	9.04E-01	1.23E+01	251	47%	1991-2007	WFBCBKG, WFBIGBR
Tc-99	-4.40E-01	±	1.80E+00	< 1.80E+00	7.25E+00	52	85%	1995-2007	WFBCBKG
I-129	1.39E-01	±	8.71E-01	< 7.86E-01	2.02E+00	68	90%	1991-2007	WFBCBKG
Cs-137	6.31E-01	±	5.98E+00	< 4.15E+00	1.01E+01	250	95%	1991-2007	WFBCBKG, WFBIGBR
U-232	1.81E-02	±	8.91E-02	< 4.28E-02	2.60E-01	68	87%	1991-2007	WFBCBKG
U-233/234	1.10E-01	±	7.02E-02	9.94E-02	2.98E-01	61	16%	1992-2007	WFBCBKG
U-235/236	1.71E-02	±	4.07E-02	< 3.28E-02	1.00E-01	67	82%	1991-2007	WFBCBKG
U-238	7.44E-02	±	6.35E-02	5.72E-02	4.00E-01	68	35%	1991-2007	WFBCBKG
Pu-238	1.45E-02	±	6.24E-02	< 3.10E-02	1.02E-01	68	93%	1991-2007	WFBCBKG
Pu-239/240	9.17E-03	±	3.50E-02	< 2.71E-02	1.98E-01	68	91%	1991-2007	WFBCBKG
Am-241	5.42E-02	±	7.15E-02	< 3.27E-02	2.20E+00	68	81%	1991-2007	WFBCBKG

LEGEND: N = Number of samples
ND = Nondetect
WFBCBKG = Buttermilk Creek background; WFBIGBR = Cattaraugus Creek background at Bigelow Bridge.

Table B-7. Groundwater Background Radionuclide Concentrations for the WVDP

Constituent	Average concentration (pCi/L)			Median (pCi/L)	Maximum (pCi/L)	N	% NDs	Years	Data Source Locations
	Result	±	Uncertainty						
Gross alpha	1.06E+00	±	5.69E+00	< 2.59E+00	2.19E+01	566	87%	1991-2007	WNW-NB1S, -0204, -0301, -0401, -0405, -0706, -0901, -0908
Gross beta	6.19E+00	±	5.11E+00	4.56E+00	2.82E+01	566	28%	1991-2007	WNW-NB1S, -0204, -0301, -0401, -0405, -0706, -0901, -0908
H-3	2.11E+01	±	8.55E+01	< 8.58E+01	9.41E+02	566	81%	1991-2007	WNW-NB1S, -0204, -0301, -0401, -0405, -0706, -0901, -0908
C-14	4.95E+00	±	2.63E+01	< 2.66E+01	7.43E+00	56	98%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
Sr-90	2.69E+00	±	1.35E+00	2.44E+00	7.38E+00	56	16%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
Tc-99	-3.71E-01	±	1.91E+00	< 1.85E+00	3.98E+00	56	96%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
I-129	2.39E-01	±	7.38E-01	< 6.01E-01	1.58E+00	56	86%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
Cs-137	1.75E+00	±	2.39E+01	< 2.22E+01	1.90E+01	258	98%	1991-2007	WNW-NB1S, -0204, -0301, -0401, -0405, -0706, -0901, -0908
U-232	2.28E-02	±	1.00E-01	< 4.92E-02	3.78E-01	56	88%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
U-233/234	4.88E-01	±	1.94E-01	1.60E-01	8.20E+00	56	13%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
U-235/236	4.52E-02	±	6.03E-02	< 5.00E-02	1.93E-01	56	71%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
U-238	3.18E-01	±	1.48E-01	1.21E-01	5.30E+00	56	21%	1993-2007	WNW-NB1S, -0401, -0405, -0706, -0908
Pu-238	5.94E-02	±	9.59E-02	< 4.65E-02	2.20E-01	6	83%	1993-1994	WNW-NB1S, -0405, -0908
Pu-239/240	4.95E-02	±	8.35E-02	< 5.28E-02	2.70E-01	6	83%	1993-1994	WNW-NB1S, -0405, -0908
Am-241	4.32E-02	±	4.76E-02	< 3.81E-02	1.80E-01	6	83%	1993-1994	WNW-NB1S, -0405, -0908

Legend: N = Number of samples

ND = Nondetect

"WNW" locations refer to individual wells that serve as groundwater backgrounds for solid waste management units in the groundwater monitoring program.

Table B-8. Background Environmental Radiation Levels at the WVDP

Constituent	Average (mR/quarter) Result ± Uncertainty	Median	Maximum	N	Years	Data Source Locations(1)
Environmental radiation	19.3 ± 7.1	19.2	35.0	264	1986-2007	DFTLD23, DFTLD17, DFTLD37, DFTLD41

NOTE: (1) Background locations: DFTLD17 (Five Point Landfill); DFTLD23 (Great Valley); DFTLD37 (Dunkirk); DFTLD41 (Sardinia-Savage Road).

4.0 Methods for Evaluating Concentrations Above Background in Onsite Environmental Media

Data from onsite sampling were available in three forms:

(1) Single observations or measurements with no associated uncertainty term (for example, a sediment concentration from 1988 presented in a historical report);

(2) A radionuclide concentration result, plus or minus an associated uncertainty term, from a sample collected as part of a one-time sampling project (i.e., the RFI soil, sediment, and subsurface soil survey done in 1993); and

(3) Multi-year data sets from samples collected at specified locations as part of the routine Environmental Monitoring or Groundwater Monitoring programs.

4.1 Single-Value Observations

Single-value observations were directly compared with the maximum result from the applicable background radionuclide-medium combination. For example, a Cs-137 concentration from lagoon sediment, as reported in WVNSCO 1994, was compared directly with the maximum Cs-137 concentration observed in background sediment. A value higher than the background result was classified as exceeding background.

4.2 Single Samples With Specified Uncertainty

A single-sample result reported with an associated uncertainty term, such as the result from a sample collected as part of the 1993 RFI investigation, was compared with background using the relative errors ratio test. This test (as described in WVDP procedure EM-74, WVNSCO 2004a) is primarily used as a data validation tool to test the acceptability of results from duplicate samples (i.e., to determine the likelihood that the samples could have come from the same population).

In the relative errors ratio test, one sample result (plus or minus its associated uncertainty term) is compared another sample result (plus or minus its associated uncertainty term). To perform the relative errors ratio calculation, the absolute value of the difference between the two sample results is divided by the sum of the squares of the estimated standard deviations (as based on the error terms) from each. If the result is not greater than 1.96 (approximating a 95 percent confidence interval), the two samples would be considered acceptable as duplicates. In other words, the samples could have been drawn from the same population (the test sample could have been drawn from the background population) if the confidence intervals bracketing the result terms from the two samples overlap.

For purposes of the current evaluation, each onsite sample result was tested against the mean (plus or minus the associated uncertainty term) of the applicable radionuclide/ medium background value. As noted earlier, because little information was available for subsurface soil, the surface soil background values were used to evaluate the status of subsurface soil. If the test sample result met the three following conditions, the result was classified as exceeding background:

- The radionuclide was detected,

- The relative errors ratio value was greater than 1.96, and
- The result term for the sample was higher than the average result term for the background.

Areas with radiological concentrations exceeding background, as determined by the RER calculation, are summarized in Decommissioning Plan Figures 4-6 (surface soil and sediment), 4-7 (subsurface soil), and 4-13 (Geoprobe® groundwater). Maximum above-background concentrations for specific radionuclides at locations in each WMA are summarized in Decommissioning Plan Section 4.2.5, Tables 4-12 through 4-22 (surface soil, sediment, and subsurface soil), and Decommissioning Plan Section 4.2.8, Table 4-26 (Geoprobe® groundwater).

4.3 Data From Routine Monitoring Locations

Radionuclide concentration data sets from routine monitoring locations were compared with applicable background data sets using the nonparametric Mann-Whitney "U" test. As recommended in MARSSIM, a nonparametric test was used because environmental data are usually not normally distributed and because there are often a significant number of results lower than detectable concentrations. Both conditions were true of the WVDP data sets examined in this evaluation.

Because of the larger number of observations available for these comparisons, the "U" test was more sensitive at detecting concentrations exceeding background at a specific location than was the RER test that considered only one measurement. Note that trends (i.e., increasing or decreasing radionuclide concentrations) were not evaluated as part of this exercise, which focused only on comparisons with background. (Data trends at the WVDP are routinely evaluated and conclusions summarized in formal reports associated with the Environmental Monitoring and Groundwater Monitoring Programs.)

The Mann-Whitney U test, similar to the Wilcoxon Rank Sum test used in MARSSIM, is a rank-based test. The null hypothesis being tested was that the median of the tested data set was higher than the median at the background location (one-tailed test, $P<0.05$). To perform the test, data sets were assembled for radionuclide concentrations at each of the onsite routine monitoring points (soil/sediment sampling locations, surface water sampling locations, and routine groundwater sampling locations). So that the data could be ranked, each radionuclide measurement was assigned a single value. All "detect" values (i.e., the result term was larger than the uncertainty term) were set equal to the result term of the measurement; all "nondetect" values (i.e., the uncertainty term was larger than the result term) were set equal to zero. In this way, all nondetect values received the same rank. (Note that summary statistics, such as averages, had already been calculated for each data set. The arbitrarily assigned zero values were used only for ranking purposes.)

The two data sets (test location and background reference location) were then combined into one data set and the results ranked in numerical order from the smallest to the largest. From the assigned ranks, the test statistic (i.e., "U") was calculated for each (Sheskin 1997). The normal approximation for larger sample sizes ("z") was also calculated. Critical values of "U" and "z" were taken from statistical tables in Sheskin 1997.

If the "U" value was lower than the critical value of "U" (or, for larger numbers of samples, if the "z" value exceeded the critical level of "z"), and the mean rank from the test data set was greater than that from the background data set, then the null hypothesis (i.e., that the median of the test data set exceeded that of the background data set) was not rejected. In other words, at a 95% confidence level, it was likely that the median of the test data set exceeded that of the background data set.

Locations where results from routine monitoring locations exceeded background are summarized by waste management area and radionuclide in section 4.2, Table 4-17 (sediment from sampling location SNSWAMP), Table 4-18 (sediment from sampling location SNSW74A), Table 4-22 (sediment from sampling location SNSP006), Table 4-24 (routine onsite surface water monitoring locations), and Table 4-25 (routine groundwater monitoring locations).

Direct onsite measurements of environmental radiation (TLD results), for which the data sets approximate a normal distribution, were compared with background measurements using the one-way analysis of variance (ANOVA) Excel® function ($p<0.05$). If the "F" statistic exceeded the critical value of "F," and the average from the test data set exceeded the background average, measurements from the test location were determined to exceed background. Results are summarized in section 4.2, Table 4-23.

5.0 Radionuclide Ratios to Cs-137

The concentrations of hard-to-measure radionuclides in a medium are often estimated on the basis of their relationship to a more easily measured nuclide, such as Cs-137, as defined in a well-characterized distribution. As discussed in Section 4.1.4 of this plan, two primary distributions have been identified at the WVDP: (1) the Spent Nuclear Fuel distribution — applicable to nuclear fuel prior to reprocessing, and (2) the Batch 10 distribution — applicable to the high-level waste after the uranium and plutonium had been extracted. Comparable ratios from the two distributions are presented in Table 4-3. As shown in Table 4.3 of this plan, Sr-90 may comprise a larger relative fraction of the total radioactivity in the "feed and waste" category (i.e., before waste reprocessing), while a larger relative fraction of Am-241 may be more characteristic of the "product" category (i.e., after waste reprocessing).

If surface soil, sediment or subsurface soil samples contained both Cs-137 and other radionuclides at above-background concentrations, the ratio of each above-background radionuclide to Cs-137 was calculated. Only data from the same discrete samples were used to calculate ratios. Ratios in surface soil, sediment, and subsurface soil are summarized by WMA in Tables B-9, B-10, and B-11, respectively. For each medium, the following information is listed:

- Number of samples for which each nuclide exceeded background,
- Minimum ratio,
- Median ratio,
- Maximum ratio,

- Concentration of Cs-137 (in pCi/g dry) in the sample with the maximum ratio, and
- Location at which the maximum ratio was observed.

With respect to environmental concentrations exceeding background, the ratio of a radionuclide to Cs-137 may help to better trace the source of the activity. For instance, the area of elevated Sr-90 concentrations on the north plateau downgradient of the Process Building has been traced to a leak of radioactively contaminated acid in the late 1960s. This plume is characterized by high Sr-90-to-Cs-137 ratios.

6.0 Supplementary Data for Onsite Monitoring Locations

Summary statistics were calculated for radiological constituents measured at all routine monitoring locations on the WVDP site, sediment for the years 1995 through 2007, and surface water and groundwater for the last ten years (1998 through 2007). Constituents exceeding background levels at each location are presented in section 4.2. Complete results, including those from locations determined to be non-impacted, are presented in the following tables for onsite sediment (Table B-12), surface water (B-13), and groundwater (B-14).

Supplementary information about routine groundwater monitoring locations (i.e., location coordinates, surface elevation, construction material of the well or trench, diameter of the well [if applicable], screened interval, and geologic unit monitored) are summarized in Table B-15. Similar information for special Geoprobe® groundwater sampling points is provided in Table B-16.

Note that only routine monitoring locations included in the current Groundwater Monitoring Program were included in the evaluation presented in Section 4.2.8 of this plan. A large number of points at which groundwater had been sampled in the past were not included in this evaluation. For completeness, information on excluded points is summarized in Table B-17. Reasons for exclusion included:

- The well was dry;
- No radiological data were available;
- Data were not validated (e.g., piezometers, surface elevation points, wells for the north plateau groundwater recovery system, wells used to evaluate the pilot permeable treatment wall);
- Wells had been dropped from the groundwater program because existing coverage was considered sufficient (e.g., more than twenty wells discontinued in 1995); or
- Sampling points were located in areas outside the scope of the Phase 1 Decommissioning Plan (e.g., groundwater seeps outside the process premises, wells from WMA 8 [New York State-Licensed Disposal Area]).

7.0 References

Sheskin 1997, *Handbook of Parametric and Nonparametric Statistical Procedures*. Sheskin, D.J., CRC Press LLC, 1997.

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WVES 2008b, *Groundwater Monitoring Plan*, WVDP-239, Revision 12. West Valley Environmental Services LLC, West Valley, New York, February 12, 2008.

WVNSCO 1994, *Environmental Information Document, Volume IV: Soils Characterization*, WVDP-EIS-008, Revision 0. West Valley Nuclear Services Company, West Valley, New York, September 15, 1994.

WVNSCO 2004a, *Radioanalytical Data Validation*, EM-74, Revision 8. West Valley Nuclear Services Company, West Valley, New York, November 22, 2004.

WVNSCO 2004b, *Documentation and Reporting of Environmental Data*, EM-11, Revision 8. West Valley Nuclear Services Company, West Valley, New York, December 27, 2004.

Table B-9. Radionuclides in Surface Soil: Ratios to Cs-137[1]

Area[2]	Radionuclide	N	Minimum	Median	Maximum	Cs-137 (pCi/g)[3]	Location of Maximum Ratio
WMA 2	Sr-90	5	0.015	0.28	1.4	8.5E-01	Surface soil near Lagoons 4 and 5 (BH-04)
WMA 3	U-238	1	0.047	0.047	0.047	2.2E+01	Surface soil near Waste Tank Farm
	Am-241	1	0.011	0.011	0.011	2.2E+01	Surface soil near Waste Tank Farm
WMA 4	Sr-90	3	0.29	0.96	9.5	1.2E+00	CDDL soil (6-12" depth, 1990)
WMA 5	Sr-90	2	0.019	0.047	0.075	1.1E+01	Surface soil near RHWF (BH-38)
	Pu-238	1	0.0033	0.0033	0.0033	1.1E+01	Surface soil near RHWF (BH-38)
	Pu-239/240	1	0.015	0.015	0.015	1.1E+01	Surface soil near RHWF (BH-38)
	Am-241	4	0.026	0.033	0.073	1.2E+01	LSA 3 & 4 footers (1990)
WMA 6	Sr-90	12	0.036	0.094	1.7	2.9E+00	Rail spur by FRS (1994)
WMA 7	Sr-90	8	0.11	1.9	8.3	1.1E+00	NDA Surface Soil (1994)
	Pu-238	1	0.021	0.021	0.021	4.1E+00	Surface soil by the NDA Interceptor Trench (BH-42)
	Pu-239/240	1	0.022	0.022	0.022	4.1E+00	Surface soil by the NDA Interceptor Trench (BH-42)
	Am-241	1	0.037	0.037	0.037	4.1E+00	Surface soil by the NDA Interceptor Trench (BH-42)
WMA 12	Sr-90	4	0.14	0.25	0.29	4.5E+00	Surface soil near WMA 2 and WMA 6 (BH-16)

NOTES: (1) Ratios were calculated from samples for which both Cs-137 and the nuclide of interest exceeded background, with ratios rounded to two significant digits or nearest integer.

(2) No surface soil data were available for WMA 1. No radionuclides exceeded background in WMA 9. Only Cs-137 exceeded background in WMA 10.

(3) Cs-137 concentration at the location with the maximum ratio.

LEGEND: BH = bore hole CDDL = Construction and Demolition Debris Landfill FRS = Fuel Receiving and Storage LSA = Lag Storage Addition N = number of samples RHWF = Remote-Handled Waste Facility.

Table B-10. Radionuclides in Sediment: Ratios to Cs-137[1]

Area[2]	Radionuclide	N	Minimum	Median	Maximum	Cs-137 (pCi/g)[3]	Location of Maximum Ratio
WMA 2	Sr-90	41	0.0063	0.065	144	1.0E+01	Sediment from the Solvent Dike (1986)
	U-232	1	0.0054	0.0054	0.0054	1.4E+03	Lagoon 3 sediment (1994)
	U-233/234	2	0.0032	0.030	0.056	1.7E+01	Sediment from drainage downgradient of Solvent Dike (ST-28)
	U-235/236	7	0.000010	0.000076	0.011	1.7E+01	Sediment from drainage downgradient of Solvent Dike (ST-28)
	U-238	28	0.000052	0.0014	0.057	2.1E+01	Lagoon 3 sediment (1990)
	Pu-238	10	0.00028	0.0015	0.018	4.4E+04	Lagoon 2 shoreline sediment (1990)
	Pu-239/240	9	0.00051	0.0011	0.019	1.7E+01	Sediment from drainage downgradient of Solvent Dike (ST-28)
	Am-241	29	0.00058	0.0019	4.2	1.0E+01	Sediment from the Solvent Dike (1986)
WMA 4	Sr-90	18	0.041	0.80	16	3.1E+00	Sediment from drainage through CDDL (ST-30)
	U-233/234	9	0.036	0.11	1.4	6.6E-01	Sediment at Northeast Swamp (SNSWAMP)
	U-235/236	2	0.023	0.14	0.27	6.6E-01	Sediment at Northeast Swamp (SNSWAMP)
	U-238	9	0.036	0.12	1.3	6.6E-01	Sediment at Northeast Swamp (SNSWAMP)
	Pu-238	10	0.0057	0.022	0.057	5.2E+00	Sediment at Northeast Swamp (SNSWAMP)
	Pu-239/240	13	0.0089	0.033	0.21	1.1E+01	Sediment at Northeast Swamp (SNSWAMP)
	Am-241	14	0.010	0.056	0.22	2.1E+00	Sediment at Northeast Swamp (SNSWAMP)
WMA 5	Sr-90	15	0.026	0.13	3.3	6.4E-01	Sediment at North Swamp (SNSW74A)
	U-233/234	4	0.12	0.37	0.75	1.1E+00	Sediment at North Swamp (SNSW74A)
	U-235/236	1	0.047	0.047	0.047	2.7E+00	Sediment at North Swamp (SNSW74A)
	U-238	4	0.15	0.34	2.0	4.7E-01	Sediment at North Swamp (SNSW74A)
	Pu-238	1	0.015	0.015	0.015	3.8E+00	Sediment at North Swamp (SNSW74A)
	Pu-239/240	9	0.019	0.035	0.096	4.7E-01	Sediment at North Swamp (SNSW74A)
	Am-241	11	0.0011	0.057	0.087	6.4E-01	Sediment at North Swamp (SNSW74A)

Table B-10. Radionuclides in Sediment: Ratios to Cs-137[1]

Area[2]	Radionuclide	N	Minimum	Median	Maximum	Cs-137 (pCi/g)[3]	Location of Maximum Ratio
WMA 6	Sr-90	3	0.062	0.27	0.59	5.9E-01	Sediment from south Demineralizer Sludge Pond (ST-36)
WMA 7	Sr-90	1	3.7	3.7	3.7	9.0E-01	Sediment from drainage near Interceptor Trench (ST-23)
	Pu-238	1	0.096	0.096	0.096	9.0E-01	Sediment from drainage near Interceptor Trench (ST-23)
	Am-241	1	0.046	0.046	0.046	9.0E-01	Sediment from drainage near Interceptor Trench (ST-23)
WMA 12	Sr-90	33	0.022	0.058	0.59	2.7E-01	Sediment from Franks Creek (ST-13) near burial areas
	U-232	2	0.0010	0.0021	0.0031	3.5E+01	Sediment from Erdman Brook (ST-19) after Lagoon 3 discharge
	U-233/234	3	0.034	0.038	0.075	1.1E+01	Sediment from Franks Creek at fence line (SNSP006)
	U-238	4	0.0094	0.035	0.058	1.4E+01	Sediment from Franks Creek at fence line (SNSP006)
	Pu-238	10	0.00070	0.0034	0.042	5.9E+01	Sediment from Erdman Brook (ST-20) after drainage from WMA 2
	Pu-239/240	7	0.00068	0.0029	0.012	5.9E+01	Sediment from Erdman Brook (ST-20) after drainage from WMA 2
	Am-241	18	0.0012	0.0047	0.033	4.3E+01	Sediment from Erdman Brook (ST-22) downgradient of NDA

NOTES: (1) Ratios were calculated from samples for which both Cs-137 and the nuclide of interest exceeded background, with the ratios rounded to two significant digits or the nearest integer.

(2) No sediment data were available for WMAs 1, 3, or 9. Only Cs-137 exceeded background in WMA 10.

(3) Cs-137 concentration at the location with the maximum ratio.

LEGEND: CDDL = Construction and Demolition Debris Landfill N = number of samples

Table B-11. Radionuclides in Subsurface Soil: Ratios to Cs-137[1]

Area[2]	Radionuclide	N	Minimum	Median	Maximum	Cs-137 (pCi/g)[3]	Location of Maximum Ratio
WMA 1	Sr-90	6	33	449	1594	1.75E+00	Geoprobe® location GP-72 at 17-19' depth
	Am-241	1	0.026	0.026	0.026	3.3E+03	Laundry line breach (2004)
WMA 2	Sr-90	16	0.037	1.3	78	7.2E-01	Near Solvent Dike (BH-11 at 8-10' depth)
	U-232	9	0.0050	0.016	0.081	1.6E+01	Maintenance shop leach field (BH-35 at 6-8' depth)
	U-233/234	5	0.0046	0.019	5.0	7.2E-01	Near Solvent Dike (BH-11 at 8-10' depth)
	U-235/236	4	0.000038	0.0011	0.74	7.2E-01	Near Solvent Dike (BH-11 at 8-10' depth)
	U-238	5	0.00052	0.013	3.1	7.2E-01	Near Solvent Dike (BH-11 at 8-10' depth)
	Pu-238	13	0.0049	0.019	0.089	1.9E+00	Between Interceptors and inactive Lagoon 1 (BH-14 at 14-16' depth)
	Pu-239/240	13	0.0046	0.031	0.10	7.2E-01	Near Solvent Dike (BH-11 at 8-10' depth)
	Am-241	13	0.010	0.047	0.15	7.2E-01	Near Solvent Dike (BH-11 at 8-10' depth)
WMA 4	Sr-90	1	0.73	0.73	0.73	7.3E-01	CDDL (BH-27, 2-4' depth)
WMA 6	Sr-90	2	1.1	67	133	4.3E+00	Near the FRS (BH-19A at 12-14' depth)
	Pu-238	2	0.025	0.030	0.035	4.3E+00	Near the FRS (BH-19A at 12-14' depth)
	Pu-239/240	2	0.040	0.043	0.047	4.3E+00	Near the FRS (BH-19A at 12-14' depth)
	Am-241	2	0.19	0.20	0.20	2.4E+00	Near the Utility Room (BH-17 at 14-16' depth)
WMA 7	Am-241	39	0.024	0.035	0.077	4.4E+00	NDA rolloff (excavated subsurface soil, analyzed in 1997)

NOTES: (1) Ratios were calculated from samples for which both Cs-137 and the nuclide of interest exceeded background, with radios rounded to two significant digits or nearest integer.

(2) No subsurface soil data were available for WMAs 3 and 9. No Cs-137 results exceeding background were found in WMAs 5,10, 12.

(3) Cs-137 concentration at the location with the maximum ratio.

LEGEND: CDDL = Construction and Demolition Debris Landfill FRS = Fuel Receiving and Storage N = Number of samples

Table B-12. Summary of Radionuclide Results from Routine Onsite Sediment Monitoring Locations

WMA	Monitoring Point	Constituent	N	Median (pCi/g)	Average (pCi/g) Result	±	Uncertainty	Maximum (pCi/g)	Exceeded Background?(1)
WMA 4	SNSWAMP Sediment at northeast swamp drainage	Gross alpha	13	1.73E+01	1.68E+01	±	3.95E+00	2.26E+01	Yes
		Gross beta	13	5.43E+01	5.51E+01	±	4.66E+00	8.98E+01	Yes
		Sr-90	17	2.35E+00	5.20E+00	±	4.97E-01	2.98E+01	Yes
		Cs-137	17	7.40E+00	9.99E+00	±	1.39E+00	3.14E+01	Yes
		U-232	17	<2.19E-02	9.20E-03	±	3.41E-02	4.79E-02	No
		U-233/234	16	8.21E-01	7.24E-01	±	1.79E-01	1.13E+00	Yes
		U-235/236	16	5.82E-02	5.94E-02	±	5.38E-02	1.76E-01	No
		U-238	16	7.93E-01	7.06E-01	±	1.65E-01	1.14E+00	Yes
		Pu-238	10	2.79E-01	2.62E-01	±	6.87E-02	4.32E-01	Yes
		Pu-239/240	17	2.26E-01	2.58E-01	±	7.10E-02	6.42E-01	Yes
		Am-241	17	4.59E-01	5.13E-01	±	1.22E-01	1.29E+00	Yes
WMA 5	SNSW74A Sediment at north swamp drainage	Gross alpha	13	1.19E+01	1.29E+01	±	3.06E+00	2.20E+01	Yes
		Gross beta	13	2.33E+01	2.35E+01	±	2.97E+00	3.47E+01	Yes
		Sr-90	17	3.28E-01	4.67E-01	±	8.73E-02	2.10E+00	Yes
		Cs-137	17	2.55E+00	2.83E+00	±	2.54E-01	8.82E+00	Yes
		U-232	17	<2.16E-02	8.57E-03	±	2.53E-02	4.23E-02	No
		U-233/234	16	7.18E-01	6.24E-01	±	1.74E-01	1.06E+00	No
		U-235/236	16	5.49E-02	5.59E-02	±	4.05E-02	1.26E-01	No
		U-238	17	6.82E-01	6.36E-01	±	1.80E-01	1.35E+00	No
		Pu-238	10	2.37E-02	2.30E-02	±	1.88E-02	5.59E-02	No
		Pu-239/240	17	6.17E-02	6.52E-02	±	4.13E-02	1.92E-01	Yes
		Am-241	17	6.10E-02	9.01E-02	±	5.09E-02	2.58E-01	Yes

Table B-12. Summary of Radionuclide Results from Routine Onsite Sediment Monitoring Locations

WMA	Monitoring Point	Constituent	N	Median (pCi/g)	Average (pCi/g)			Maximum (pCi/g)	Exceeded Background?(1)
					Result	±	Uncertainty		
WMA 12	SNSP006 Sediment from Franks Creek at security fence	Gross alpha	13	1.10E+01	1.01E+01	±	2.84E+00	1.32E+01	No
		Gross beta	13	4.27E+01	5.01E+01	±	4.09E+00	1.60E+02	Yes
		Sr-90	17	8.38E-01	1.49E+00	±	2.29E-01	9.98E+00	Yes
		Cs-137	17	1.30E+01	2.10E+01	±	2.75E+00	9.76E+01	Yes
		U-232	17	4.07E-02	4.01E-02	±	6.81E-02	1.43E-01	Yes
		U-233/234	16	6.40E-01	6.05E-01	±	1.78E-01	1.02E+00	No
		U-235/236	16	4.56E-02	3.87E-02	±	5.46E-02	1.04E-01	No
		U-238	17	6.07E-01	5.53E-01	±	1.68E-01	9.15E-01	No
		Pu-238	10	3.17E-02	4.29E-02	±	2.58E-02	1.40E-01	Yes
		Pu-239/240	17	2.60E-02	2.97E-02	±	2.54E-02	1.08E-01	Yes
		Am-241	17	4.34E-02	6.51E-02	±	4.78E-02	2.40E-01	Yes

NOTE: (1) Using the nonparametric Mann-Whitney "U" Test, the data set of sediment background results (summarized in Table B-4) was compared with the data set from each of the sampling locations. See Appendix B, Section 4.3.

Table B-13. Summary of Radionuclide Results from Routine Onsite Surface Water Monitoring Locations

WMA	Monitoring Point	Constituent	N	Median (pCi/L)(2)	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?(1)
					Result	±	Uncertainty		
WMA 2	WNSP001 Lagoon 3 Discharge Weir	Gross alpha	232	1.75E+01	1.92E+01	±	1.32E+01	1.01E+02	Yes
		Gross beta	433	2.56E+02	3.01E+02	±	2.25E+01	8.18E+02	Yes
		H-3	231	2.47E+03	2.75E+03	±	1.42E+02	7.17E+03	Yes
		C-14	62	<2.82E+01	1.35E+01	±	2.24E+01	4.75E+01	Yes
		Sr-90	231	9.88E+01	1.21E+02	±	7.42E+00	3.19E+02	Yes
		Tc-99	197	6.53E+01	7.90E+01	±	4.79E+01	3.36E+02	Yes
		I-129	62	2.13E+00	2.44E+00	±	1.48E+00	1.04E+01	Yes
		Cs-137	231	6.10E+01	7.57E+01	±	1.88E+01	3.29E+02	Yes
		U-232	62	8.02E+00	8.98E+00	±	9.91E-01	2.14E+01	Yes
		U-233/234	62	5.04E+00	5.49E+00	±	6.20E-01	1.36E+01	Yes
		U-235/236	62	2.62E-01	2.75E-01	±	1.21E-01	5.84E-01	Yes
		U-238	62	3.76E+00	3.82E+00	±	4.87E-01	7.57E+00	Yes
		Pu-238	62	6.53E-02	1.53E-01	±	6.78E-02	1.62E+00	Yes
		Pu-239/240	62	5.17E-02	1.34E-01	±	6.19E-02	1.39E+00	Yes
		Am-241	62	6.79E-02	1.18E-01	±	6.01E-02	9.74E-01	Yes
WMA 4	WNSWAMP Northeast Swamp Drainage	Gross alpha	450	<1.87E+00	2.86E-01	±	2.28E+00	7.25E+00	No
		Gross beta	451	3.01E+03	3.24E+03	±	5.33E+01	9.98E+03	Yes
		H-3	451	1.13E+02	1.13E+02	±	8.21E+01	5.20E+02	Yes
		C-14	34	<1.58E+01	2.13E+00	±	2.09E+01	3.72E+01	No
		Sr-90	121	1.52E+03	1.70E+03	±	3.14E+01	5.16E+03	Yes
		I-129	34	<9.05E-01	5.39E-01	±	9.28E-01	1.29E+00	No
		Cs-137	120	<2.43E+00	6.76E-01	±	3.33E+00	5.74E+00	No
		U-232	34	<6.42E-02	7.47E-03	±	1.59E-01	9.76E-02	No
		U-233/234	34	1.73E-01	1.97E-01	±	1.36E-01	9.27E-01	Yes
		U-235/236	34	<4.20E-02	2.54E-02	±	5.77E-02	8.82E-02	No
		U-238	34	1.01E-01	1.21E-01	±	1.07E-01	7.21E-01	Yes
		Pu-238	34	<3.11E-02	1.20E-02	±	9.54E-02	1.50E-01	No
		Pu-239/240	34	<2.90E-02	1.48E-02	±	6.65E-02	1.44E-01	No
		Am-241	34	<3.42E-02	2.86E-02	±	9.57E-02	1.79E-01	No

Table B-13. Summary of Radionuclide Results from Routine Onsite Surface Water Monitoring Locations

WMA	Monitoring Point	Constituent	N	Median (pCi/L)(2)	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?(1)
					Result	±	Uncertainty		
WMA 5	WNSW74A North Swamp Drainage	Gross alpha	450	<2.17E+00	3.88E-02	±	3.09E+00	7.89E+00	No
		Gross beta	450	1.17E+01	1.21E+01	±	4.34E+00	4.24E+01	Yes
		H-3	450	<8.18E+01	-2.14E+00	±	8.07E+01	2.80E+02	No
		C-14	34	<1.40E+01	-7.72E-01	±	1.94E+01	1.50E+01	No
		Sr-90	120	5.52E+00	5.46E+00	±	1.89E+00	1.25E+01	Yes
		I-129	34	<7.10E-01	2.09E-01	±	7.37E-01	1.31E+00	No
		Cs-137	120	<7.08E+00	1.20E+00	±	8.85E+00	1.18E+01	No
		U-232	34	<4.83E-02	8.38E-03	±	6.79E-02	6.22E-02	No
		U-233/234	34	1.54E-01	1.64E-01	±	8.44E-02	3.54E-01	Yes
		U-235/236	34	<3.70E-02	1.89E-02	±	3.99E-02	1.38E-01	No
		U-238	34	1.01E-01	1.04E-01	±	6.65E-02	2.00E-01	Yes
		Pu-238	34	<2.10E-02	1.43E-02	±	3.36E-02	1.16E-01	No
		Pu-239/240	34	<2.39E-02	4.73E-03	±	2.73E-02	<6.94E-02	No
		Am-241	34	<2.81E-02	1.68E-02	±	3.17E-01	8.63E-02	No
WMA 6	WNSP007 Sanitary Waste Discharge	Gross alpha	324	<2.62E+00	1.37E-01	±	3.32E+00	4.80E+00	No
		Gross beta	324	1.45E+01	1.53E+01	±	5.02E+00	4.05E+01	Yes
		H-3	324	<8.25E+01	2.26E+01	±	8.18E+01	1.53E+03	No
		Sr-90	14	3.11E+00	3.38E+00	±	1.75E+00	1.17E+01	Yes
		Cs-137	35	<2.92E+00	8.12E-01	±	3.94E+00	4.44E+00	No
	WNCOOLW Cooling Tower Water	Gross alpha	73	<1.91E+00	5.65E-01	±	2.03E+00	5.81E+00	No
		Gross beta	73	6.83E+00	9.05E+00	±	3.64E+00	3.43E+01	Yes
		H-3	73	<8.17E+01	2.86E+00	±	7.94E+01	4.27E+02	No
		Sr-90	10	1.60E+00	1.50E+00	±	1.40E+00	4.68E+00	No
		Cs-137	31	<7.20E+00	8.61E-01	±	8.32E+00	9.15E+00	No

Table B-13. Summary of Radionuclide Results from Routine Onsite Surface Water Monitoring Locations

WMA	Monitoring Point	Constituent	N	Median (pCi/L)(2)	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?(1)
					Result	±	Uncertainty		
WMA 12	WNSP006 Franks Creek at security fence	Gross alpha	471	<1.50E+00	9.49E-01	±	1.61E+00	1.07E+01	No
		Gross beta	471	3.53E+01	4.44E+01	±	3.99E+00	1.94E+02	Yes
		H-3	471	<8.54E+01	1.36E+02	±	8.33E+01	2.25E+03	Yes
		C-14	40	<1.85E+01	-1.31E+00	±	2.09E+01	2.06E+01	No
		Sr-90	120	1.87E+01	1.98E+01	±	2.99E+00	4.96E+01	Yes
		Tc-99	40	<2.09E+00	3.28E+00	±	2.15E+00	5.24E+01	Yes
		I-129	40	<7.04E-01	3.26E-01	±	7.25E-01	1.65E+00	No
		Cs-137	120	<8.02E+00	6.32E+00	±	9.50E+00	7.33E+01	Yes
		U-232	40	3.17E-01	3.16E-01	±	1.34E-01	7.51E-01	Yes
		U-233/234	40	3.66E-01	3.73E-01	±	1.31E-01	6.87E-01	Yes
		U-235/236	40	<4.41E-02	3.26E-02	±	4.61E-02	9.57E-02	No
		U-238	40	2.54E-01	2.77E-01	±	1.12E-01	7.43E-01	Yes
		Pu-238	40	<3.36E-02	2.14E-02	±	3.39E-02	1.36E-01	Yes
		Pu-239/240	40	<2.79E-02	1.13E-02	±	3.02E-02	6.62E-02	No
		Am-241	40	<3.30E-02	3.23E-02	±	3.69E-02	1.60E-01	No
	WNSP005 Facility yard drainage	Gross alpha	140	<2.71E+00	1.22E+00	±	3.24E+00	1.85E+01	No
		Gross beta	140	1.50E+02	1.63E+02	±	9.11E+00	4.53E+02	Yes
		H-3	140	<8.28E+01	3.78E+01	±	8.23E+01	1.25E+03	Yes
		Sr-90	35	9.61E+01	1.02E+02	±	6.52E+00	1.98E+02	Yes
		Cs-137	14	<1.91E+00	9.28E-01	±	2.19E+00	<3.69E+00	No
	WNNDADR Drainage between NDA and SDA	Gross alpha	130	<1.34E+00	8.22E-01	±	1.40E+00	5.84E+00	No
		Gross beta	136	1.74E+02	1.83E+02	±	6.45E+00	4.06E+02	Yes
		H-3	546	1.00E+03	1.16E+03	±	1.02E+02	4.02E+03	Yes
		Sr-90	41	8.48E+01	8.40E+01	±	5.45E+00	1.22E+02	Yes
		I-129	34	<8.12E-01	2.62E-01	±	8.53E-01	1.15E+00	No
		Cs-137	120	<6.67E+00	5.99E-01	±	8.48E+00	1.86E+01	No

Table B-13. Summary of Radionuclide Results from Routine Onsite Surface Water Monitoring Locations

WMA	Monitoring Point	Constituent	N	Median (pCi/L)(2)	Average (pCi/L) Result	±	Uncertainty	Maximum (pCi/L)	Exceeded Background?(1)
WMA 12	WNERB53 Erdman Brook north of burial areas	Gross alpha	401	<1.45E+00	1.56E-01	±	1.65E+00	2.51E+00	No
		Gross beta	401	1.73E+01	1.81E+01	±	2.92E+00	4.37E+01	Yes
		H-3	403	<8.31E+01	3.08E+01	±	8.11E+01	3.46E+02	Yes
		Sr-90	14	8.23E+00	8.04E+00	±	1.98E+00	9.91E+00	Yes
		Cs-137	14	<2.07E+00	7.52E-01	±	3.96E+00	2.41E+00	No
	WNFRC67 Franks Creek east of burial areas	Gross alpha	99	<7.00E-01	9.41E-02	±	7.56E-01	3.89E+00	No
		Gross beta	99	2.63E+00	2.56E+00	±	1.50E+00	9.00E+00	No
		H-3	99	<8.31E+01	3.08E+01	±	8.11E+01	3.46E+02	Yes
		Sr-90	19	<1.17E+00	5.00E-01	±	1.09E+00	3.42E+00	No
		Cs-137	19	<2.13E+00	5.50E-01	±	2.58E+00	2.26E+00	No

NOTES: (1) Using the nonparametric Mann-Whitney "U" Test, the data set of surface water background results (summarized in Table B-6) was compared with the data set from each of the above sampling locations. See Appendix B, Section 4.3.

(2) 1 pCi/L = 3.7E-02 Bq/L

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?[4]
					Result	±	Uncertainty		
WMA 1	WP-A S&G	Gross alpha	12	<3.56E-01	1.71E-01	±	2.12E+00	1.82E+00	No
		Gross beta	12	2.41E+01	3.09E+01	±	4.55E+00	5.44E+01	Yes
		H-3	12	1.18E+04	1.12E+04	±	6.24E+02	1.26E+04	Yes
WMA 2	WNW0103 S&G	Gross alpha	40	<7.32E+00	1.06E+00	±	1.01E+01	1.25E+01	No
		Gross beta	40	1.45E+02	1.85E+02	±	1.93E+01	5.53E+02	Yes
		H-3	40	<8.42E+01	5.19E+01	±	8.12E+01	2.02E+02	No
	WNW0104 S&G	Gross alpha	40	<3.86E+00	2.23E-01	±	5.95E+00	5.04E+00	No
		Gross beta	40	5.88E+04	5.63E+04	±	1.64E+03	1.01E+05	Yes
		H-3	40	3.73E+02	3.91E+02	±	8.65E+01	7.53E+02	Yes
	WNW0105 S&G	Gross alpha	41	<4.21E+00	1.04E+00	±	7.17E+00	4.60E+00	No
		Gross beta	41	3.88E+04	3.30E+04	±	1.54E+03	1.02E+05	Yes
		H-3	40	3.57E+02	3.72E+02	±	9.12E+01	7.09E+02	Yes
	WNW0106 S&G	Gross alpha	40	<2.50E+00	1.94E+00	±	3.44E+00	1.31E+01	No
		Gross beta	40	1.64E+01	8.22E+01	±	7.99E+00	5.76E+02	Yes
		H-3	40	9.56E+02	1.04E+03	±	1.00E+02	1.82E+03	Yes
	WNW0107 ULT	Gross alpha	40	<1.85E+00	8.97E-01	±	1.88E+00	5.71E+00	No
		Gross beta	40	7.00E+00	8.23E+00	±	2.63E+00	2.22E+01	Yes
		H-3	40	3.74E+02	4.78E+02	±	9.04E+01	9.85E+02	Yes
	WNW0108 ULT	Gross alpha	40	1.64E+00	1.47E+00	±	1.46E+00	4.31E+00	Yes
		Gross beta	40	2.49E+00	2.42E+00	±	1.90E+00	5.36E+00	No
		H-3	40	1.17E+02	1.10E+02	±	8.38E+01	2.47E+02	Yes
	WNW0110 ULT	Gross alpha	40	<1.49E+00	1.01E+00	±	1.61E+00	4.39E+00	No
		Gross beta	40	2.32E+00	2.23E+00	±	1.95E+00	7.92E+00	No
		H-3	40	1.31E+03	1.28E+03	±	1.08E+02	1.66E+03	Yes
	WNW0111 S&G	Gross alpha	40	<4.38E+00	3.15E+00	±	5.06E+00	1.03E+01	Yes
		Gross beta	40	5.55E+03	5.87E+03	±	1.40E+02	1.18E+04	Yes
		H-3	40	1.97E+02	2.34E+02	±	8.39E+01	7.97E+02	Yes

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?[4]
					Result	±	Uncertainty		
WMA 2	WNW0116 S&G	Gross alpha	40	<3.08E+00	8.94E-01	±	4.35E+00	7.03E+00	No
		Gross beta	40	8.69E+02	1.98E+03	±	1.55E+02	9.51E+03	Yes
		H-3	40	1.67E+02	1.88E+02	±	8.24E+01	4.66E+02	Yes
	WNW0205 S&G	Gross alpha	35	<4.87E+00	4.37E-01	±	7.67E+00	<2.73E+01	No
		Gross beta	35	1.61E+01	1.66E+01	±	8.39E+00	4.08E+01	Yes
		H-3	35	<8.14E+01	9.44E+00	±	8.02E+01	2.09E+02	No
	WNW0206 LTS	Gross alpha	35	<2.47E+00	6.69E-01	±	3.33E+00	5.02E+00	No
		Gross beta	35	<3.16E+00	1.95E+00	±	3.53E+00	6.11E+00	No
		H-3	35	<8.18E+01	2.94E+01	±	7.96E+01	2.07E+02	No
	WNW0408 S&G	Gross alpha	40	<3.58E+00	-7.91E+00	±	9.05E+00	6.44E+00	No
		Gross beta	39	3.96E+05	4.01E+05	±	3.04E+03	6.28E+05	Yes
		H-3	40	1.52E+02	1.86E+02	±	1.13E+02	2.21E+03	Yes
		C-14	10	<2.16E+01	-7.20E-01	±	2.27E+01	<3.42E+01	No
		Sr-90	10	1.54E+05	1.54E+05	±	1.73E+02	2.53E+05	Yes
		Tc-99	10	1.57E+01	1.70E+01	±	3.28E+00	2.51E+01	Yes
		I-129	10	<9.94E-01	7.65E-02	±	2.53E+00	9.46E-01	No
		Cs-137	10	<4.01E+00	-3.24E-01	±	4.29E+00	<6.72E+00	No
		U-232	10	<6.32E-02	6.31E-02	±	2.04E-01	5.31E-02	No
		U-233/234	10	4.51E-01	5.34E-01	±	2.22E-01	1.27E+00	Yes
		U-235/236	10	<5.44E-02	8.34E-02	±	9.98E-02	3.11E-01	No
		U-238	10	2.87E-01	3.11E-01	±	1.57E-01	4.82E-01	Yes
		Pu-238	2	<6.83E-02	2.09E-02	±	7.45E-02	<9.80E-02	No
		Pu-239/240	2	<6.56E-02	7.70E-03	±	6.65E-02	<7.68E-02	No
		Am-241	2	4.60E-02	3.60E-02	±	4.72E-02	5.90E-02	No
	WNW0501	Gross alpha	40	<4.79E+00	4.82E-01	±	8.34E+00	6.10E+00	No

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?[4]
					Result	±	Uncertainty		
WMA 2	S&G	Gross beta	40	1.93E+05	1.91E+05	±	2.61E+03	3.24E+05	Yes
		H-3	40	1.35E+02	1.25E+02	±	8.37E+01	3.15E+02	Yes
		Sr-90	10	9.18E+04	9.33E+04	±	2.43E+02	1.48E+05	Yes
	WNW0502 S&G	Gross alpha	40	<4.40E+00	7.94E-01	±	8.04E+00	1.46E+01	No
		Gross beta	40	1.68E+05	1.64E+05	±	2.80E+03	2.33E+05	Yes
		H-3	40	1.33E+02	1.44E+02	±	8.36E+01	4.98E+02	Yes
		Sr-90	10	8.36E+04	8.27E+04	±	2.05E+02	1.16E+05	Yes
	WNW8603 S&G	Gross alpha	41	<5.02E+00	3.92E-01	±	7.89E+00	9.30E+00	No
		Gross beta	41	5.66E+04	4.81E+04	±	1.20E+03	9.01E+04	Yes
		H-3	40	3.37E+02	3.43E+02	±	8.79E+01	5.81E+02	Yes
	WNW8604 S&G	Gross alpha	35	<4.68E+00	1.07E+00	±	7.83E+00	9.00E+00	No
		Gross beta	35	4.12E+04	4.57E+04	±	1.12E+03	1.04E+05	Yes
		H-3	35	3.48E+02	3.76E+02	±	8.38E+01	6.41E+02	Yes
	WNW8605 S&G	Gross alpha	40	9.11E+00	8.46E+00	±	7.66E+00	2.08E+01	Yes
		Gross beta	40	1.09E+04	1.10E+04	±	1.73E+02	1.62E+04	Yes
		H-3	40	3.70E+02	4.19E+02	±	8.68E+01	1.27E+03	Yes
	WP-C S&G	Gross alpha	12	<3.95E-01	9.03E-01	±	2.74E+00	<6.92E+00	No
		Gross beta	12	2.44E+01	4.16E+01	±	5.48E+00	1.19E+02	Yes
		H-3	12	4.91E+04	4.75E+04	±	1.56E+03	6.61E+04	Yes
	WP-H S&G	Gross alpha	13	6.08E+00	7.90E+01	±	2.33E+01	7.42E+02	Yes
		Gross beta	13	6.97E+03	7.23E+03	±	1.87E+02	1.25E+04	Yes
		H-3	13	2.99E+03	3.42E+03	±	5.00E+02	7.38E+03	Yes
WMA 3	WNW8609 S&G	Gross alpha	40	<3.10E+00	-3.75E-01	±	5.55E+00	3.84E+00	No
		Gross beta	40	1.51E+03	1.37E+03	±	4.15E+01	2.28E+03	Yes
		H-3	40	4.51E+02	4.66E+02	±	9.10E+01	7.88E+02	Yes
		Sr-90	20	7.99E+02	7.17E+02	±	2.07E+01	1.12E+03	Yes
WMA 4	WNW0801	Gross alpha	40	<3.85E+00	6.31E-02	±	6.49E+00	5.45E+00	No

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L) Result	±	Uncertainty	Maximum (pCi/L)	Exceeded Background?[4]
WMA 4	S&G	Gross beta	40	7.95E+03	8.59E+03	±	2.72E+02	1.46E+04	Yes
		H-3	40	1.51E+02	1.64E+02	±	8.24E+01	3.82E+02	Yes
		Sr-90	40	4.13E+03	4.33E+03	±	4.73E+01	7.99E+03	Yes
	WNW0802	Gross alpha	40	<1.33E+00	1.05E+00	±	2.03E+00	1.66E+01	No
	S&G	Gross beta	40	9.94E+00	3.47E+01	±	5.14E+00	2.84E+02	Yes
		H-3	40	<1.05E+02	9.00E+01	±	8.00E+01	4.20E+02	Yes
	WNW0803	Gross alpha	40	<3.01E+00	9.79E-01	±	3.38E+00	8.96E+00	No
	S&G	Gross beta	40	1.48E+01	1.51E+01	±	4.69E+00	2.50E+01	Yes
		H-3	40	1.84E+02	1.60E+02	±	8.46E+01	3.42E+02	Yes
	WNW0804	Gross alpha	40	<2.04E+00	6.00E-01	±	2.87E+00	6.54E+00	No
	S&G	Gross beta	40	2.58E+02	2.86E+02	±	1.07E+01	6.89E+02	Yes
		H-3	40	1.19E+02	1.14E+02	±	7.98E+01	3.60E+02	Yes
	WNW8612	Gross alpha	40	<2.62E+00	3.33E-01	±	3.34E+00	4.57E+00	No
	S&G	Gross beta	41	<3.58E+00	1.57E+00	±	3.60E+00	5.91E+00	No
		H-3	40	4.21E+02	4.33E+02	±	8.88E+01	8.46E+02	Yes
WMA 5	WNW0406	Gross alpha	40	<2.22E+00	1.54E-01	±	2.58E+00	4.49E+00	No
	S&G	Gross beta	40	7.44E+00	8.08E+00	±	3.49E+00	1.67E+01	Yes
		H-3	40	1.17E+02	1.06E+02	±	8.42E+01	4.38E+02	Yes
		C-14	10	<2.65E+01	-2.04E+00	±	2.36E+01	2.72E+01	No
		Sr-90	10	1.92E+00	2.15E+00	±	1.45E+00	4.57E+00	No
		Tc-99	11	2.19E+00	2.53E+00	±	1.91E+00	8.50E+00	Yes
		I-129	10	<8.91E-01	3.48E-01	±	9.17E-01	1.72E+00	No
		Cs-137	10	<6.41E+00	-9.30E-01	±	7.35E+00	<1.48E+01	No
		U-232	10	<4.55E-02	2.47E-02	±	1.24E-01	<3.59E-01	No
		U-233/234	10	1.37E-01	1.42E-01	±	1.05E-01	2.67E-01	No
		U-235/236	10	<3.97E-02	2.32E-02	±	5.51E-02	6.92E-02	No
		U-238	10	8.08E-02	8.87E-02	±	8.17E-02	1.92E-01	No

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?[4]
					Result	±	Uncertainty		
WMA 5	WNW0409 ULT	Gross alpha	40	<1.01E+00	9.39E-01	±	9.94E-01	2.32E+00	Yes
		Gross beta	40	2.56E+00	2.36E+00	±	1.37E+00	4.38E+00	No
		H-3	40	<8.01E+01	-3.82E+00	±	7.86E+01	2.10E+02	No
	WNW0602A S&G	Gross alpha	35	<1.37E+00	4.04E-01	±	1.60E+00	2.51E+00	No
		Gross beta	35	1.21E+01	1.32E+01	±	2.87E+00	3.46E+01	Yes
		H-3	35	2.15E+02	2.18E+02	±	8.88E+01	4.88E+02	Yes
	WNW0604 S&G	Gross alpha	41	<2.04E+00	3.35E-01	±	2.45E+00	3.10E+00	No
		Gross beta	41	6.06E+00	6.29E+00	±	2.97E+00	1.29E+01	Yes
		H-3	40	<8.14E+01	1.99E+01	±	8.01E+01	2.07E+02	No
	WNW0605 S&G	Gross alpha	35	<1.54E+00	4.40E-01	±	1.59E+00	1.13E+01	No
		Gross beta	35	4.83E+01	5.07E+01	±	3.98E+00	8.82E+01	Yes
		H-3	35	<8.08E+01	1.59E+01	±	7.86E+01	1.44E+02	No
	WNW0704 ULT/S&G	Gross alpha	40	<1.93E+00	1.75E-01	±	2.25E+00	2.23E+00	No
		Gross beta	40	8.05E+00	8.20E+00	±	3.05E+00	1.34E+01	Yes
		H-3	40	<8.20E+01	-1.69E+01	±	8.24E+01	2.16E+02	No
	WNW0707 ULT/S&G	Gross alpha	40	<1.15E+00	3.09E-01	±	1.35E+00	4.40E+00	No
		Gross beta	40	4.17E+00	4.16E+00	±	1.98E+00	9.85E+00	No
		H-3	40	<8.22E+01	-1.89E+01	±	8.11E+01	1.05E+02	No
	WNW1303 ULT	Gross alpha	19	<9.42E-01	1.19E+00	±	2.06E+00	5.46E+00	No
		Gross beta	19	2.17E+00	2.24E+00	±	2.25E+00	9.38E+00	No
		H-3	19	<8.25E+01	-4.98E+01	±	2.09E+02	1.26E+02	No
	WNW1304 S&G	Gross alpha	19	<6.14E+00	-8.58E-01	±	8.32E+00	6.92E+00	No
		Gross beta	19	<8.20E+00	4.92E+00	±	8.11E+00	1.33E+01	No
		H-3	19	<9.44E+01	2.36E+01	±	2.16E+02	1.60E+02	No
		C-14	18	<3.03E+01	2.02E+00	±	2.92E+01	3.69E+01	No
		Sr-90	18	1.60E+00	1.93E+00	±	1.28E+00	6.33E+00	No
		Tc-99	18	<1.94E+00	1.25E-01	±	1.91E+00	2.62E+00	No

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L) Result	±	Uncertainty	Maximum (pCi/L)	Exceeded Background?[4]
WMA 5		I-129	18	<7.52E-01	3.39E-01	±	1.33E+00	2.83E+00	No
		Cs-137	18	<2.77E+00	7.11E-01	±	4.88E+00	2.52E+00	No
		U-232	18	<3.73E-02	-1.09E-02	±	6.74E-02	<2.17E-01	No
		U-233/234	18	2.66E-01	2.93E-01	±	1.26E-01	5.65E-01	Yes
		U-235/236	18	<4.07E-02	3.85E-02	±	5.31E-02	1.77E-01	No
		U-238	18	1.91E-01	2.15E-01	±	1.05E-01	5.77E-01	Yes
	WNW8607 S&G	Gross alpha	40	<2.36E+00	-7.83E-02	±	4.40E+00	9.45E+00	No
		Gross beta	40	2.57E+01	2.75E+01	±	5.30E+00	7.63E+01	Yes
		H-3	40	<8.47E+01	1.97E+01	±	8.30E+01	2.04E+02	No
WMA 7	WNW0902 KRS	Gross alpha	20	1.46E+00	1.34E+00	±	1.34E+00	5.44E+00	Yes
		Gross beta	20	2.70E+00	2.76E+00	±	1.64E+00	4.92E+00	No
		H-3	20	<8.08E+01	-3.35E+01	±	8.18E+01	1.18E+02	No
	WNW0909 WLT	Gross alpha	26	<3.24E+00	1.16E+00	±	3.83E+00	1.14E+01	No
		Gross beta	34	3.74E+02	3.70E+02	±	1.40E+01	6.44E+02	Yes
		H-3	30	8.23E+02	1.54E+03	±	1.20E+02	3.95E+03	Yes
		C-14	10	<2.49E+01	7.23E+00	±	2.39E+01	3.53E+01	No
		Sr-90	17	1.87E+02	1.83E+02	±	8.33E+00	2.21E+02	Yes
		Tc-99	11	<1.86E+00	1.31E+00	±	1.82E+00	5.01E+00	Yes
		I-129	11	6.21E+00	6.30E+00	±	1.88E+00	9.65E+00	Yes
		Cs-137	10	<5.51E+00	1.09E+00	±	6.42E+00	<1.28E+01	No
		U-232	12	<5.99E-02	6.37E-02	±	1.62E-01	5.26E-01	No
		U-233/234	12	5.97E-01	7.42E-01	±	2.40E-01	1.34E+00	Yes
		U-235/236	11	6.71E-02	7.66E-02	±	7.65E-02	2.48E-01	No
		U-238	12	4.72E-01	5.44E-01	±	1.97E-01	1.03E+00	Yes

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L) Result	±	Uncertainty	Maximum (pCi/L)	Exceeded Background?[4]
WMA 7	WNW0910 ULT	Gross alpha	25	<2.53E+00	1.88E+00	±	2.29E+00	3.45E+00	Yes
		Gross beta	25	3.80E+01	1.46E+02	±	8.51E+00	1.54E+03	Yes
		H-3	24	<8.06E+01	-1.24E+01	±	8.05E+01	2.39E+02	No
	WNNDATR WLT	Gross alpha	160	2.22E+00	2.08E+00	±	2.11E+00	1.06E+01	Yes
		Gross beta	166	1.45E+02	1.75E+02	±	8.36E+00	5.51E+02	Yes
		H-3	164	3.65E+03	5.00E+03	±	2.28E+02	1.99E+04	Yes
		C-14	20	<2.18E+01	3.02E-01	±	2.39E+01	1.33E+01	No
		Sr-90	28	5.84E+01	7.85E+01	±	5.55E+00	2.84E+02	Yes
		Tc-99	21	<1.94E+00	6.32E-01	±	1.89E+00	5.12E+00	No
		I-129	41	<9.14E-01	8.44E-01	±	9.35E-01	7.00E+00	Yes
		Cs-137	140	<6.80E+00	7.20E-01	±	8.88E+00	1.50E+01	No
		U-232	21	<7.12E-02	5.11E-02	±	1.18E-01	4.72E-01	No
		U-233/234	21	1.67E+00	1.51E+00	±	2.81E-01	2.11E+00	Yes
		U-235/236	21	1.06E-01	1.35E-01	±	9.47E-02	3.04E-01	Yes
		U-238	21	1.30E+00	1.22E+00	±	2.50E-01	1.73E+00	Yes
	WNW8610 KRS	Gross alpha	20	<2.21E+00	6.60E-01	±	2.88E+00	6.35E+00	No
		Gross beta	20	4.41E+00	4.79E+00	±	3.09E+00	9.91E+00	No
		H-3	20	<8.17E+01	-3.80E+01	±	7.96E+01	1.46E+02	No
	WNW8611 KRS	Gross alpha	21	<1.98E+00	1.23E+00	±	2.25E+00	4.50E+00	No
		Gross beta	21	<2.71E+00	2.83E+00	±	2.81E+00	1.67E+01	No
		H-3	20	<8.15E+01	-4.98E+01	±	8.08E+01	8.44E+01	No
WMA 9	WNW1005 WLT	Gross alpha	20	<2.49E+00	1.97E+00	±	2.92E+00	4.69E+00	No
		Gross beta	20	<3.52E+00	2.36E+00	±	2.98E+00	5.14E+00	No
		H-3	20	<8.36E+01	1.24E+01	±	8.14E+01	2.01E+02	No

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L) Result	±	Uncertainty	Maximum (pCi/L)	Exceeded Background?[4]
WMA 9	WNW1006 WLT	Gross alpha	20	<5.10E+00	4.24E+00	±	5.50E+00	1.02E+01	Yes
		Gross beta	20	<6.80E+00	4.58E+00	±	5.68E+00	1.03E+01	No
		H-3	20	<8.20E+01	-1.81E+01	±	8.24E+01	1.67E+02	No
WMA 10	WNW0302 S&G	Gross alpha	36	<5.51E+00	8.24E-01	±	9.02E+00	1.55E+00	No
		Gross beta	36	<7.22E+00	4.13E+00	±	8.13E+00	1.27E+01	No
		H-3	36	<8.23E+01	3.72E+01	±	8.11E+01	1.87E+02	No
	WNW0402 S&G	Gross alpha	35	<5.13E+00	5.02E-01	±	6.93E+00	7.45E+00	No
		Gross beta	35	<5.64E+00	2.53E+00	±	6.56E+00	8.33E+00	No
		H-3	35	<8.21E+01	2.73E+01	±	8.05E+01	1.99E+02	No
	WNW0403 S&G	Gross alpha	35	<2.11E+00	3.85E-01	±	2.45E+00	5.94E+00	No
		Gross beta	35	5.76E+00	6.17E+00	±	3.26E+00	1.06E+01	No
		H-3	35	<8.22E+01	2.20E+01	±	7.97E+01	1.92E+02	No
	WNW1008B KRS	Gross alpha	20	<1.08E+00	7.09E-01	±	1.12E+00	3.11E+00	No
		Gross beta	20	2.68E+00	3.15E+00	±	1.46E+00	9.18E+00	No
		H-3	20	<8.04E+01	-2.23E+01	±	7.96E+01	7.81E+01	No
	WNW1008C WLT	Gross alpha	20	<1.51E+00	8.13E-02	±	1.48E+00	<1.89E+00	No
		Gross beta	20	<1.86E+00	1.15E+00	±	2.00E+00	3.03E+00	No
		H-3	20	<8.15E+01	-1.06E+00	±	8.10E+01	1.33E+02	No
	WNW1301 ULT	Gross alpha	1	<1.48E+01	1.43E+01	±	1.48E+01	<1.48E+01	No
		Gross beta	1	<1.02E+01	-1.04E+01	±	1.02E+01	<1.02E+01	No
		H-3	1	<8.61E+02	-6.09E+02	±	8.61E+02	<8.61E+02	No
	WNW1302 S&G	Gross alpha	19	<3.69E+00	1.00E+00	±	5.69E+00	4.88E+00	No
		Gross beta	19	<5.62E+00	2.76E+00	±	6.44E+00	6.47E+00	No
		H-3	19	<9.37E+01	-4.07E+01	±	2.05E+02	1.15E+02	No

Table B-14. Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations[1]

WMA	Monitoring Point[2]	Constituent	N	Median (pCi/L)[3]	Average (pCi/L)			Maximum (pCi/L)	Exceeded Background?[4]
					Result	±	Uncertainty		
WMA 12	WNW0903 KRS	Gross alpha	20	<1.90E+00	3.35E-01	±	2.26E+00	4.29E+00	No
		Gross beta	20	<2.42E+00	2.30E+00	±	2.62E+00	9.21E+00	No
		H-3	20	<8.20E+01	-5.34E+01	±	8.16E+01	1.62E+02	No
	WNW0906 WLT	Gross alpha	20	<1.78E+00	1.47E+00	±	1.72E+00	4.19E+00	No
		Gross beta	20	4.50E+00	4.92E+00	±	2.22E+00	1.41E+01	No
		H-3	20	<8.43E+01	3.80E+00	±	8.23E+01	1.55E+02	No

NOTES: (1) See Figure 4-12 in Section 4 of this plan for the locations of monitoring wells where concentrations exceed background.

(2) Geologic unit is indicated below each monitoring point.

(3) 1 pCi/L = 3.7E-02 Bq/L.

(4) Data sets for radiological constituents in groundwater were compared with data sets from background wells using the nonparametric Mann-Whitney "U" test, as described in Appendix B, Section 4.3.

LEGEND: S&G = Sand and Gravel; ULT = unweathered Lavery till; KRS = Kent Recessional Sequence; WLT = weathered Lavery till; LTS = Lavery till sand.

Table B-15. Groundwater Monitoring Locations: Coordinates, Depth, Screened Interval, and Geologic Unit

Monitoring Location(1)	North Coordinate(2)	East Coordinate(2)	Surface Elevation (ft)	Well Construction Material	Well Diameter (in)	Depth to Screen Top (ft)	Depth to Screen Bottom (ft)	Geologic Unit of Screened Interval
WNW0103	893013.68	1129469.99	1399.99	ST. STL.	2	6	21	S&G-TBU
WNW0104	893295.07	1129574.51	1399.29	ST. STL.	2	8	23	S&G-TBU/SWS
WNW0105	893536.70	1129768.63	1385.59	ST. STL.	2	13	28	S&G-TBU/SWS
WNW0106	893495.37	1129926.24	1383.73	ST. STL.	2	9.5	14.5	S&G-TBU
WNW0107	893399.05	1130060.32	1376.40	ST. STL.	2	8	28	ULT
WNW0108	893110.00	1129915.26	1381.66	ST. STL.	2	13	33	ULT
WNW0110	893024.67	1129881.74	1387.74	ST. STL.	2	13	33	ULT
WNW0111	892874.91	1129694.33	1392.54	ST. STL.	2	6	11	S&G-TBU
WNW0116	893518.81	1129560.10	1387.39	ST. STL.	2	6	11	S&G-TBU
WNW0204	892670.48	1129380.67	1406.83	ST. STL.	2	38	43	LTS
WNW0205	892696.37	1129528.87	1398.32	ST. STL.	2	6	11	S&G-TBU
WNW0206	892705.65	1129535.43	1398.39	ST. STL.	2	32.8	37.8	LTS
WNW0301	892593.20	1128914.31	1418.44	ST. STL.	2	6	16	S&G-TBU
WNW0302	892599.05	1128910.79	1418.46	ST. STL.	2	23	28	S&G-SWS
WNW0401	892708.28	1128864.51	1418.57	ST. STL.	2	6	16	S&G-TBU
WNW0402	892702.84	1128867.50	1419.34	ST. STL.	2	24	29	S&G-SWS
WNW0403	892865.78	1128790.38	1419.66	ST. STL.	2	8	13	S&G-TBU
WNW0405	893405.48	1128685.08	1408.56	ST. STL.	2	7.5	12.5	ULT
WNW0406	893250.04	1128992.47	1405.85	ST. STL.	2	11.8	16.8	S&G-TBU
WNW0408	893074.34	1129214.81	1405.56	ST. STL.	2	28	38	S&G-TBU/SWS
WNW0409	893256.53	1128988.16	1404.34	ST. STL.	2	44	54	ULT
WNW0501	893186.25	1129277.65	1402.18	ST. STL.	2	23	33	S&G-SWS
WNW0502	893325.38	1129406.73	1397.45	ST. STL.	2	8	18	S&G-TBU/SWS
WNW0602A	893403.75	1129244.07	1397.27	PVC	2	5	15	S&G-TBU

Table B-15. Groundwater Monitoring Locations: Coordinates, Depth, Screened Interval, and Geologic Unit

Monitoring Location(1)	North Coordinate(2)	East Coordinate(2)	Surface Elevation (ft)	Well Construction Material	Well Diameter (in)	Depth to Screen Top (ft)	Depth to Screen Bottom (ft)	Geologic Unit of Screened Interval
WNW0604	893576.30	1128926.84	1398.95	ST. STL.	2	6	11	S&G-TBU
WNW0605	893815.08	1129254.11	1383.90	ST. STL.	2	6	11	S&G-TBU
WNW0704	893763.67	1128814.82	1395.36	ST. STL.	2	5.5	15.5	ULT
WNW0706	893512.77	1128608.18	1409.03	ST. STL.	2	6	11	S&G-TBU
WNW0707	893896.47	1128617.53	1396.26	ST. STL.	2	6	11	ULT
WNW0801	893679.20	1129555.29	1383.51	ST. STL.	2	7.5	17.5	S&G-TBU
WNW0802	893904.53	1129687.61	1377.50	ST. STL.	2	6	11	S&G-TBU
WNW0803	893914.79	1129907.88	1370.17	ST. STL.	2	8	18	S&G-SWS
WNW0804	893751.72	1129982.56	1373.04	ST. STL.	2	4	9	S&G-TBU
WNW0901	891449.83	1129923.88	1392.72	ST. STL.	2	121	136	KRS
WNW0902	891671.96	1129774.24	1390.46	ST. STL.	2	118	128	KRS
WNW0903	892064.50	1129974.91	1380.69	ST. STL.	2	118	133	KRS
WNW0906	891945.99	1129796.90	1384.55	ST. STL.	2	5	10	WLT
WNW0908	891453.85	1129920.53	1392.94	ST. STL.	2	6	21	WLT
WNW0909	892085.66	1130121.37	1372.99	ST. STL.	2	8	23	WLT
WNW0910	892088.89	1130128.11	1372.69	PVC	2	25	30	ULT
WNW1005	890964.33	1130017.26	1389.68	ST. STL.	2	9	19	WLT
WNW1006	891264.17	1130206.69	1392.32	ST. STL.	2	10	20	WLT
WNW1008B	890904.46	1129534.09	1402.35	ST. STL.	2	46	51	KRS
WNW1008C	890914.13	1129545.20	1402.43	ST. STL.	2	8	18	WLT
WNW1301	893111.93	1128386.20	1429.49	PVC	2	20	30	ULT
WNW1302	893111.83	1128386.64	1429.47	PVC	2	5	8	S&G-TBU
WNW1303	893400.10	1128599.38	1414.65	PVC	2	23	38	ULT
WNW1304	893405.10	1128595.82	1414.36	PVC	2	6	10	S&G-TBU

Table B-15. Groundwater Monitoring Locations: Coordinates, Depth, Screened Interval, and Geologic Unit

Monitoring Location(1)	North Coordinate(2)	East Coordinate(2)	Surface Elevation (ft)	Well Construction Material	Well Diameter (in)	Depth to Screen Top (ft)	Depth to Screen Bottom (ft)	Geologic Unit of Screened Interval
WNW8603	893537.65	1129716.56	1385.45	PVC	4	8.25	23.25	S&G-TBU/SWS
WNW8604	893396.47	1129624.90	1390.41	PVC	4	6	21	S&G-TBU/SWS
WNW8605	892864.58	1129650.32	1393.19	PVC	4	5.5	10.5	S&G-TBU
WNW8607	893392.16	1128904.17	1405.03	PVC	4	11	16	S&G-TBU
WNW8609	893126.56	1129091.64	1407.07	PVC	4	12.7	22.7	S&G-TBU
WNW8610	891896.52	1130392.29	1376.88	STL.	2	97.33	112.33	KRS
WNW8611	892067.89	1130297.10	1376.34	STL.	2	103.5	118.5	KRS
WNW8612	893983.30	1130028.31	1367.76	PVC	4	6.6	16.6	S&G-TBU/SWS
WNWNB1S	892513.28	1128353.79	1447.08	ST. STL.	2	8	13	S&G-TBU
WNNDATR	892068.35	1130126.06	1374.89	CONCRETE	60	0	0	WLT
WP-A	892883.92	1129232.58	1408.34	IRON	2	29	33	S&G-TBU/SWS
WP-C	892986.95	1129411.57	1400.89	IRON	2	19	23	S&G-TBU
WP-H	892925.41	1129367.85	1405.38	IRON	2	13	17	S&G-TBU

NOTES: (1) Radiological data from the current monitoring locations, as listed in the 2008 Groundwater Monitoring Program, were evaluated for the WVDP Phase 1 DP. Monitoring point WNNDATR is an interceptor trench.

(2) Western New York State Planar Coordinate System

LEGEND: STL = steel, ST.STL = stainless steel, PVC = polyvinyl chloride, S&G = sand and gravel, TBU = thick bedded unit, SWS = slack water sequence, ULT = unweathered Lavery till, LTS = Lavery till sand, KRS = Kent recessional sequence, WLT = weathered Lavery till.

Table B-16. Location, Elevation, and Depth of Geoprobe® Groundwater Sampling Points

Location Code	Year Sampled	North Coordinate(1)	East Coordinate(1)	Surface Elevation (ft)	Sample Depths (ft) and Geologic Units(2)
GP01	1994	893754.94	1129433.58	1375.00	04-06
GP0197	1997	893527.20	1129733.08	1382.35	00-04, 04-08, 08-12, 12.5-14, 12-16, 16-20, 17.5-19, 20-24, 22.5-24, 24-28 (ULT)
GP02	1994	893701.98	1129480.46	1378.95	06-08
GP0297	1997	893527.37	1129689.35	1383.08	00-04, 04-08, 08-12, 12.5-14, 12-16, 16-20, 17.5-19, 20-24, 24-28. 25.5-27
GP03	1994	893684.86	1129546.39	1380.07	08-10, 13-15
GP0397	1997	893527.23	1129662.34	1383.08	00-04, 04-08, 08-12, 10.5-12, 12-16, 15.5-17, 16-20, 20.5-22, 20-24, 24.5-26, 24-28, 28-32 (ULT)
GP04	1994	893587.10	1129609.73	1381.96	10-12
GP0497	1997	893529.48	1129630.86	1383.10	08.5-10, 13.5-15, 18.5-20, 23-24.5
GP05	1994	893556.85	1129746.34	1391.59	15-17, 20-22, 25-27
GP0597	1997	893531.83	1129600.53	1383.51	08.5-10, 13.5-15
GP06	1994	893523.31	1129743.01	1382.59	15-17, 20-22, 25-27
GP0697	1997	893635.51	1129508.65	1381.39	08.5-10, 13.5-15, 17.5-19
GP07	1994	893623.69	1129777.03	1378.60	07.5-09.5
GP0797	1997	893633.61	1129535.22	1380.88	08.5-10, 13.5-15, 18.5-20
GP08	1994	893485.68	1129640.70	1384.66	09-11, 14-16, 19-21
GP0897	1997	893629.21	1129567.72	1380.15	08.5-10, 12.5-14.5, 17.5-18.5
GP09	1994	893446.05	1129609.75	1385.81	09-11, 14-16, 19-21
GP0997	1997	893630.01	1129599.46	1379.30	08.5-10, 13.5-15
GP10	1994	893495.08	1129514.19	1386.41	09-11
GP1097	1997	893628.00	1129624.69	1379.01	08.5-10, 13.5-15, 18.5-20
GP11	1994	893514.96	1129468.64	1386.51	08-10
GP1197	1997	893625.73	1129664.22	1378.57	08.5-10, 13.5-15, 17.5-19, 23.4-25
GP12	1994	893594.08	1129526.20	1382.41	07-09
GP1297	1997	893623.09	1129706.63	1378.15	00-04, 04-08, 07.5-09, 08-12, 12.5-14, 12-16, 16-20, 17.5-19, 20-24, 22-23.5, 24-28 (ULT)
GP13	1994	893422.90	1129419.73	1390.67	10-12
GP1397	1997	893621.53	1129744.33	1377.93	09-10.5, 13.5-15, 18.5-20
GP13A	1994	893385.24	1129395.73	1392.97	11-13, 15-17, 16-18
GP14	1994	893179.41	1129370.33	1399.11	15-17, 20-22, 25-27, 30-32
GP1497	1997	893619.43	1129784.76	1378.09	00-04, 04-08, 08-09.5, 08-12, 12-16, 16-20 (ULT)

Table B-16. Location, Elevation, and Depth of Geoprobe® Groundwater Sampling Points

Location Code	Year Sampled	North Coordinate(1)	East Coordinate(1)	Surface Elevation (ft)	Sample Depths (ft) and Geologic Units(2)
GP15	1994	893222.77	1129158.76	1402.57	15-17
GP1597	1997	893662.03	1129761.57	1376.85	08-10, 13-15, 18-20
GP16	1994	893217.10	1129056.60	1402.66	15-17, 20-22
GP1697	1997	893662.85	1129707.70	1377.19	08-10, 12-15, 18-20
GP17	1994	893055.18	1129446.69	1399.01	12-14
GP1797	1997	893733.87	1130014.29	1370.09	08-10, 13-15
GP18	1994	892932.47	1129283.29	1404.16	18-20, 21.5-23.5
GP1897	1997	893666.65	1129642.75	1387.08	08-10, 13-15, 17.5-19.5
GP1898	1998	892929.53	1129281.76	1403.99	12-14, 16-19, 22-24
GP1997	1997	893528.51	1129675.56	1383.27	00-04, 04-08, 08-12, 12-16, 14-16, 16-20, 19-21, 20-22, 22-24, 24-26, 26-28, 28-30
GP20	1994	893141.44	1129083.93	1403.07	15-17
GP2097	1997	893529.48	1129645.74	1383.35	00-04, 04-08, 08-12, 12-14, 12-16, 16-20, 17-19, 20-24, 22-24, 24-28
GP2197	1997	893531.19	1129615.48	1383.43	00-04, 04-08, 08-12, 12-16, 13-15, 16-20, 20-24, 23-25, 24-28 (ULT), 28-32 (ULT), 32-36 (ULT)
GP2297	1997	893462.46	1129692.02	1384.93	12-14, 17-19, 22-24
GP23	1994	892960.50	1129165.19	1409.41	20-22, 22.5-24.5, 27-29, 32-34
GP2397	1997	893512.71	1129715.96	1383.06	12-14, 16-19, 22-24
GP2397	1998	892980.83	1129165.77	1408.96	17-19, 22-24, 25-29, 32-34
GP24	1994	893006.32	1129151.08	1408.99	17-19, 22-24, 26-28, 30-32
GP2497	1997	893506.39	1129771.02	1382.83	00-04, 04-08, 08-12, 12-16, 14-16, 16-20, 19-21, 20-24, 24-26, 24-28, 28-30, 30-32 (ULT)
GP2597	1997	893804.22	1129989.94	1368.40	08-10
GP26	1994	892992.21	1129084.84	1409.63	17-19
GP2697	1997	893671.61	1129961.64	1375.36	04.5-06.5, 09-11, 14-16
GP27	1994	892960.10	1129096.04	1408.86	16-18, 21-23, 26-28
GP2797	1997	893576.18	1129713.16	1381.18	12-14, 16-19, 22-24
GP28	1994	892855.87	1129220.94	1408.08	16-18, 21-23, 26-28, 31-33
GP2897	1997	893579.60	1129663.78	1381.44	12-14, 16-19, 22-24
GP29	1994	892783.34	1129163.61	1410.01	15-17, 21-23, 27-29, 33-35
GP2997	1997	893583.58	1129622.59	1381.56	12-14

Table B-16. Location, Elevation, and Depth of Geoprobe® Groundwater Sampling Points

Location Code	Year Sampled	North Coordinate(1)	East Coordinate(1)	Surface Elevation (ft)	Sample Depths (ft) and Geologic Units(2)
GP2998	1998	892781.53	1129163.00	1409.81	17-19, 19-21, 21-23, 22-24, 23-25, 25-27, 27-29, 29-31, 31-33, 33-35, 34-36, 35-37, 37-38 (ULT), 38-39 (ULT), 39-40 (ULT), 40-41 (ULT)
GP30	1994	892835.65	1129144.49	1409.32	18-20, 22-24, 27-29, 32-34
GP3098	1998	892829.94	1129141.96	1409.18	18-20, 20-22, 22-24, 23-27, 23-37, 24-26, 26-28, 28-30, 30-32, 32-34, 34-36, 36-36.5, 36.5-37 (ULT), 37-37.5 (ULT), 37.5-38 (ULT), 38-38.5 (ULT), 38.5-39 (ULT), 39-39.5 (ULT), 39.5-40 (ULT)
GP31	1994	893269.27	1129335.71	1396.59	12-14, 17-19
GP32	1994	893827.03	1129487.70	1372.83	05-07
GP32A	1994	893831.75	1129475.59	1372.45	05-07
GP33	1994	893813.09	1129337.41	1375.73	05-07
GP33A	1994	893819.60	1129347.72	1375.24	05-07
GP35	1994	893858.20	1129143.23	1384.48	04-06
GP36	1994	893815.85	1128971.59	1387.17	03.5-05.5
GP37	1994	893720.92	1128930.11	1389.11	05-07
GP38	1994	893594.09	1128959.27	1392.71	06.5-08.5
GP39	1994	893498.24	1128979.05	1396.44	06-08, 10-12
GP40	1994	893459.75	1129103.74	1394.08	08-10, 13-15
GP41	1994	893388.58	1129138.49	1396.59	14-16
GP42	1994	893362.12	1129180.49	1395.96	11-13
GP43	1994	893334.39	1129257.32	1396.17	12-14
GP44	1994	893003.49	1129551.08	1393.29	09-11, 14-16
GP45	1994	892995.79	1129523.66	1394.34	10-12, 15-17, 18.5-20.5
GP46	1994	892968.45	1129466.90	1397.24	12-14, 17-19
GP47	1994	892969.21	1129522.40	1394.24	11-13, 16-18
GP48	1994	892924.74	1129842.93	1386.88	07-09
GP50	1994	892833.51	1129852.05	1384.55	08-10
GP51	1994	893825.87	1129561.74	1374.48	06.5-08.5
GP52	1994	893859.57	1129634.30	1374.21	08-10
GP53	1994	893278.77	1128978.62	1401.62	14-16
GP56	1994	892704.20	1129025.11	1410.49	06-08, 15.5-17.5
GP59	1994	892859.54	1129363.33	1399.83	09-11, 17-19
GP60	1994	892870.18	1129409.83	1400.01	12-14, 17-19

Table B-16. Location, Elevation, and Depth of Geoprobe® Groundwater Sampling Points

Location Code	Year Sampled	North Coordinate(1)	East Coordinate(1)	Surface Elevation (ft)	Sample Depths (ft) and Geologic Units(2)
GP61	1994	893875.01	1129563.26	1372.91	06-08
GP62	1994	893933.30	1129567.59	1371.20	04-06
GP64	1994	893781.92	1129295.55	1379.81	09-11
GP66	1994	893125.94	1129318.33	1403.62	17-19, 22-24, 26-28, 30-32
GP67	1994	893186.02	1129410.00	1399.12	15-17, 20-22, 25-27, 30-32
GP68	1994	893199.21	1129449.59	1398.42	15-17, 20-22, 25-27, 30-32
GP69	1994	892721.81	1129189.75	1410.10	19-21, 29-31, 34-36
GP70	1994	892815.80	1129223.19	1409.19	16-18, 21-23, 26-28
GP71	1994	892845.53	1129242.84	1406.51	16-18, 21-23, 25-27
GP72	1994	892873.33	1129179.42	1409.41	16-18, 21-23, 20-32
GP7298	1998	892873.12	1129178.71	1409.17	17-19, 19-21, 21-23, 22-24, 23-25, 25-27, 27-29, 29-31, 31-33, 32-34, 33-35, 35-37, 37-39 (ULT), 39-41 (ULT)
GP73	1994	892908.21	1129176.59	1410.51	21-23, 26-28, 30-32
GP7398	1998	892899.43	1129186.81	1410.00	18-20, 20-22, 22-24, 24-26, 25-27, 26-28, 28-30, 30-32, 32-34, 34-36, 35-37, 36-38, 38-40, 40.5-41 (ULT), 40-45.5 (ULT), 41.5-42 (ULT), 41-41.5 (ULT)
GP74	1994	892906.72	1129072.17	1409.69	18-20, 23-25, 28-30
GP75	1994	892804.03	1129071.55	1410.49	19-21, 23-25, 27-29
GP76	1994	892829.00	1129049.17	1414.49	19-21, 23-25, 27-29
GP77	1994	892748.07	1129075.00	1414.49	19-21, 19-23, 27-29, 31-33
GP78	1994	892841.92	1129109.44	1414.48	19-21, 19-23, 23-25, 27-29, 31-33
GP7898	1998	892831.03	1129127.81	1409.70	19-21, 20-22, 21-23, 23-25, 24-27, 25-27, 27-29, 29-31, 30-32, 31-33, 33-35, 35-37
GP79	1994	892757.54	1129099.11	1414.49	21-23, 25-27, 29-31
GP80	1994	892809.20	1129126.66	1414.48	25-27, 30-32, 34-39, 35-35, 35-37
GP8098	1998	892792.03	1129125.21	1414.28	22-24, 24-26, 26-28, 27-29, 28-30, 30-32, 32-34, 34-36, 36-38, 38-40, 40-42 (ULT)
GP8198	1998	893048.83	1129217.96	1403.98	15-17, 20-22, 25-27, 30-32, 35-37
GP8298	1998	892996.19	1129315.09	1402.13	12-14, 17-19, 20-24
GP8398	1998	892982.69	1129187.54	1407.43	17-19, 19-21, 20-22, 21-23, 23-25, 25-27, 27-29, 29-31, 31-33, 32-34, 33-35, 35-37
GP8698	1998	892845.57	1129161.24	1409.02	18-20, 20-22, 22-24, 24-26, 24-27, 26-28, 28-30, 30-32, 32-34, 34-36, 35-37, 36-38, 38-39, 39-39.5, 39.5-40 (ULT), 40-40.5 (ULT), 40.5-41 (ULT), 41-41.5 (ULT), 41.5-42 (ULT)
GP8798	1998	892813.15	1129225.60	1408.43	15-17, 20-22, 25-27, 28-32

Table B-16. Location, Elevation, and Depth of Geoprobe® Groundwater Sampling Points

Location Code	Year Sampled	North Coordinate(1)	East Coordinate(1)	Surface Elevation (ft)	Sample Depths (ft) and Geologic Units(2)
GP8898	1998	893533.28	1129528.60	1384.14	07-09, 12-14
GP8998	1998	893722.00	1129516.58	1379.09	06-08, 11-13, 16-18
GP9098	1998	893826.72	1129596.32	1373.46	03-05, 08-10
GP9198	1998	893875.44	1129596.20	1372.82	03-05
GP9298	1998	893811.26	1129533.79	1373.71	04-06, 09-11, 14-16, 18.5-21
GP9398	1998	893821.48	1129568.33	1372.62	04-06, 09-11, 14-16
GP9498	1998	893874.66	1129532.98	1372.01	03-05, 08-10, 12-15

NOTES: (1) Western New York State Planar Coordinate System

(2) All screened intervals were within the Sand and Gravel (S&G) unit except for those from the Unweathered Lavery Till unit, designated as "ULT."

Table B-17. Groundwater Points Excluded from the Evaluation[1]

Sampling Location	North Coordinate [2]	East Coordinate [2]	Surface Elevation (ft)	Elevation at Top of Screened Interval (ft)	Elevation at Bottom of Screened Interval (ft)	Geologic Unit of Screened Interval
NDA WP-A	892047.61	1130117.37	1375.47	1355.27	1348.77	ULT
NDA WP-B	892045.71	1130112.17	1375.45	1360.25	1357.75	WLT
NDA WP-C	892006.26	1130115.39	1378.47	1367.67	1362.17	WLT
NP0101	893602.56	1129427.10	1386.10	1379.60	1374.60	S&G
NP0102	893577.38	1129428.82	1389.40	1381.90	1376.90	S&G
NP0103	893586.49	1129466.86	1385.10	1376.60	1371.60	S&G
NP0104	893621.36	1129460.64	1384.10	1379.60	1369.60	S&G
NP0105	893528.03	1129853.06	1382.50	1374.50	1359.50	S&G
NP0106	893598.16	1129779.73	1380.70	1369.70	1364.70	S&G
NP0107	893542.52	1129601.69	1384.10	1375.60	1370.60	S&G
NP0108	893518.32	1129601.99	1385.30	1376.30	1371.30	S&G
NP0109	893543.29	1129552.36	1384.30	1376.30	1369.30	S&G
NP0110	893573.10	1129628.57	1383.50	1373.50	1370.50	S&G
NP0111	893609.48	1129621.28	1381.40	1366.40	1363.40	S&G
NP0112	893605.26	1129622.72	1381.50	1373.50	1368.50	S&G
NP0113	893578.74	1129574.71	1383.00	1373.00	1368.00	S&G
NP0114	893564.04	1129564.66	1383.50	1375.50	1370.50	S&G
NP0115	893484.80	1129685.67	1385.60	1366.60	1359.60	S&G
NP0116	893490.96	1129688.62	1385.30	1373.80	1368.80	S&G
NP0117	893446.35	1129634.45	1386.40	1368.40	1363.40	S&G
NP0118	893439.47	1129630.61	1386.60	1375.60	1370.60	S&G
NP0119	893526.14	1129664.12	1385.10	1364.10	1359.10	S&G
NP0120	893526.24	1129655.74	1385.30	1371.30	1366.30	S&G
NP0121	893518.59	1129668.60	1384.60	1373.60	1358.60	S&G
NP0122	893512.26	1129663.29	1384.60	1377.60	1362.60	S&G
NP0123	893513.46	1129649.40	1384.90	1370.90	1365.90	S&G
NP0124	893512.56	1129653.52	1384.70	1365.70	1360.70	S&G
NP0125	893518.72	1129631.75	1384.60	1377.60	1362.60	S&G
NP0126	893513.83	1129634.52	1384.70	1377.70	1362.70	S&G
NP0127	893561.96	1129508.64	1386.10	1379.60	1369.60	S&G
NP0128	893611.18	1129516.76	1382.80	1375.80	1365.80	S&G
NP0129	893585.08	1129529.17	1383.40	1376.40	1366.40	S&G
NP0130	893629.71	1129576.60	1381.00	1374.00	1364.00	S&G
NP0131	893535.80	1129735.81	1383.00	1366.00	1356.00	S&G
NP0132	893556.54	1129690.68	1383.70	1364.70	1360.70	S&G
NP0133	893616.82	1129670.92	1379.90	1364.90	1354.90	S&G
PTWRP	893516.03	1129663.87	1384.88	1380.88	1360.88	S&G

Table B-17. Groundwater Points Excluded from the Evaluation[1]

Sampling Location	North Coordinate [2]	East Coordinate [2]	Surface Elevation (ft)	Elevation at Top of Screened Interval (ft)	Elevation at Bottom of Screened Interval (ft)	Geologic Unit of Screened Interval
PZ01	893501.64	1129644.29	1385.10	1378.10	1363.10	S&G
PZ02	893502.55	1129658.76	1385.10	1378.10	1363.10	S&G
PZ03	893509.15	1129639.29	1384.60	1377.60	1362.60	S&G
PZ04	893508.56	1129664.33	1384.70	1377.70	1362.70	S&G
PZ05	893519.11	1129676.77	1384.40	1377.40	1362.40	S&G
PZ06	893538.60	1129638.19	1384.30	1377.30	1362.30	S&G
PZ07	893537.58	1129663.80	1384.00	1377.00	1362.00	S&G
PZ08	893516.74	1129643.87	1385.40	1368.40	1365.40	S&G
PZ09	893516.34	1129651.79	1385.40	1367.90	1365.40	S&G
PZ10	893521.60	1129632.18	1384.60	1375.60	1372.60	S&G
RW01	893556.21	1129506.87	1384.43	1379.43	1369.43	S&G
RW02	893559.26	1129478.22	1384.38	1380.38	1370.38	S&G
RW03	893565.07	1129493.51	1385.28	1380.28	1370.28	S&G
WNGSEEP	893765.77	1130322.30	1356.89	NA	NA	S&G
WNGSP04	893866.63	1130309.52	NA	NA	NA	S&G
WNGSP06	893960.73	1130283.50	NA	NA	NA	S&G
WNGSP11	894065.05	1130090.45	NA	NA	NA	S&G
WNGSP12	894171.90	1130050.85	NA	NA	NA	S&G
WNNDATR	892068.35	1130126.06	1372.49	NA	NA	WLT
WNSE007	893850.15	1129578.86	1371.11	NA	NA	S&G
WNSE008	893791.04	1130002.44	1368.52	NA	NA	S&G
WNSE009	893683.63	1129699.74	1378.11	NA	NA	S&G
WNSE011	893838.93	1129534.25	1373.08	NA	NA	S&G
WNW0109	892972.05	1129830.09	1386.84	1373.84	1353.84	ULT
WNW0114	893452.77	1129988.66	1377.01	1368.01	1348.01	ULT
WNW0115	893525.49	1129564.84	1384.19	1366.19	1356.19	ULT
WNW0201	892419.73	1129383.16	1408.19	1398.19	1388.19	S&G
WNW0202	892407.19	1129390.47	1407.95	1374.95	1369.95	LTS
WNW0203	892670.42	1129376.09	1404.62	1396.62	1386.62	S&G
WNW0207	892503.34	1129677.53	1396.11	1390.11	1385.11	S&G
WNW0208	892488.90	1129674.25	1396.26	1378.26	1373.26	LTS
WNW0305	892630.33	1129176.24	1410.38	1394.38	1379.38	S&G
WNW0306	892633.70	1129174.87	1410.32	1344.32	1329.32	KRS
WNW0307	892634.87	1129177.55	1410.53	1404.53	1394.53	S&G
WNW0404	892871.77	1128786.30	1416.69	1390.19	1380.19	S&G
WNW0407	893250.92	1128996.78	1402.40	1336.90	1326.90	ULT
WNW0410	892868.61	1128789.26	1416.64	1348.64	1338.64	KRS

Table B-17. Groundwater Points Excluded from the Evaluation[1]

Sampling Location	North Coordinate [2]	East Coordinate [2]	Surface Elevation (ft)	Elevation at Top of Screened Interval (ft)	Elevation at Bottom of Screened Interval (ft)	Geologic Unit of Screened Interval
WNW0411	892694.15	1128869.23	1416.27	1370.27	1350.27	KRS
WNW0601	893810.70	1129256.11	1381.14	1377.14	1375.14	S&G
WNW0603	893519.08	1128736.33	1401.14	1393.14	1388.14	S&G
WNW0701	893501.78	1128611.97	1406.52	1383.52	1378.52	ULT
WNW0702	893775.67	1128516.08	1397.68	1369.68	1359.68	ULT
WNW0703	893887.50	1128622.76	1393.12	1382.12	1372.12	ULT
WNW0705	893779.24	1128509.78	1397.87	1391.87	1376.87	ULT
WNW0904	892066.15	1129984.19	1377.95	1361.95	1351.95	ULT
WNW0905	892131.67	1130069.18	1373.56	1355.56	1350.56	S&G
WNW0907	891901.62	1129774.48	1382.27	1376.27	1366.27	WLT
WNW1001	890969.42	1130010.26	1387.55	1281.55	1271.55	KRS
WNW1002	891267.67	1130208.43	1389.76	1291.76	1276.76	KRS
WNW1003	891303.20	1130437.01	1387.65	1259.65	1249.65	KRS
WNW1004	891085.15	1130459.09	1383.89	1290.89	1275.89	KRS
WNW1007	891306.41	1130433.26	1387.55	1374.55	1364.55	WLT
WNW1101A	891062.41	1130830.41	1379.37	1373.37	1363.37	WLT
WNW1101B	891060.33	1130826.90	1379.42	1359.42	1349.42	ULT
WNW1101C	891058.61	1130823.07	1379.13	1285.13	1270.13	KRS
WNW1102A	891508.74	1131146.27	1382.71	1375.71	1365.71	WLT
WNW1102B	891514.11	1131142.06	1382.59	1361.59	1351.59	ULT
WNW1103A	891925.14	1130822.28	1379.90	1373.90	1363.90	WLT
WNW1103B	891929.54	1130818.73	1379.83	1358.83	1343.83	ULT
WNW1103C	891934.64	1130815.86	1379.51	1273.51	1258.51	KRS
WNW1104A	892289.10	1130545.05	1376.12	1372.12	1357.12	WLT
WNW1104B	892285.42	1130549.21	1376.10	1355.10	1340.10	ULT
WNW1104C	892282.05	1130553.29	1375.96	1261.96	1251.96	KRS
WNW1105A	892608.51	1130294.17	1365.80	1354.80	1344.80	ULT
WNW1105B	892608.20	1130289.77	1366.01	1345.01	1330.01	ULT
WNW1106A	891960.87	1130374.92	1374.36	1368.36	1358.36	WLT
WNW1106B	891964.09	1130372.02	1374.32	1353.62	1343.62	ULT
WNW1107A	892368.58	1130256.16	1377.16	1373.16	1358.16	WLT
WNW1108A	891312.43	1130600.10	1380.93	1374.93	1364.93	WLT
WNW1109A	891929.92	1130329.31	1374.86	1368.86	1358.86	WLT
WNW1109B	891934.27	1130326.01	1374.02	1358.02	1343.02	ULT
WNW1110A	892100.29	1130691.11	1377.05	1367.05	1357.05	WLT
WNW1111A	891654.21	1131042.28	1380.22	1369.22	1359.22	ULT
WNW80-4	893687.98	1129428.98	1386.55	1373.98	1368.98	S&G

Table B-17. Groundwater Points Excluded from the Evaluation[1]

Sampling Location	North Coordinate [2]	East Coordinate [2]	Surface Elevation (ft)	Elevation at Top of Screened Interval (ft)	Elevation at Bottom of Screened Interval (ft)	Geologic Unit of Screened Interval
WNW834D	893670.95	1129435.35	1380.48	1256.18	1249.98	KRS
WNW834E	893670.95	1129435.35	1381.64	NA	NA	BR
WNW8606	892694.89	1129523.46	1396.49	1390.89	1385.89	S&G
WNW8608	893250.67	1128985.62	1401.59	1394.59	1384.59	S&G
WNW90I7	891913.54	1130323.78	NA	NA	NA	WLT
WNW96I1	891991.27	1130117.11	1379.89	1374.89	1369.89	WLT
WNW96I2	891915.18	1130305.03	1380.41	1374.91	1369.91	WLT
WNW96I3	891898.75	1129901.48	1380.32	1372.32	1367.32	WLT
WNW96I4	891872.40	1129910.29	1381.36	1374.36	1369.36	WLT
WNWEW-1	893578.98	1129453.22	1384.91	1379.91	1371.91	S&G
WNWEW-4	893546.14	1129515.19	1384.17	1380.17	1368.17	S&G
WNWWP-4	893486.96	1129473.70	1387.63	1379.63	1377.63	S&G
WP01	893485.51	1129520.87	1386.57	1378.57	1376.57	S&G
WP02	893566.19	1129521.75	1383.10	1376.10	1373.10	S&G
WP03	893513.64	1129490.62	1385.88	1377.88	1375.88	S&G
WP05	893584.51	1129490.37	1383.91	1376.91	1373.91	S&G
WP06	893548.40	1129479.09	1384.94	1377.94	1374.94	S&G
WP07	893520.93	1129467.36	1386.08	1378.08	1376.08	S&G
WP08	893500.03	1129447.32	1387.34	1379.34	1377.34	S&G
WP09	893591.43	1129438.20	1384.81	1377.81	1374.81	S&G
WP10	893533.21	1129414.87	1390.47	1383.47	1380.47	S&G
WP11	893537.89	1129741.98	1382.08	1370.08	1367.08	S&G
WP12	893552.47	1129785.92	1381.68	1369.68	1366.68	S&G
WP13	893603.74	1129840.46	1379.78	1367.78	1364.78	S&G
WP14	893561.33	1129744.79	1381.38	1369.38	1366.38	S&G
WP15	893530.52	1129536.70	1384.08	1377.08	1374.08	S&G
WP16	893591.77	1129669.06	1381.61	1365.61	1362.61	S&G
WP17	893631.05	1129660.29	1379.01	1371.01	1368.01	S&G
WP18	893627.96	1129702.66	1378.66	1370.66	1367.66	S&G
WP20D	892845.95	1129162.30	1409.60	1379.60	1376.6	S&G
WP20S	892844.41	1129162.58	1409.60	1388.60	1385.60	S&G
WP21	893534.74	1129529.93	1384.50	1377.50	1374.50	S&G
WP22	893723.11	1129517.68	1379.80	1365.80	1362.80	S&G
WP23	893809.43	1129533.65	1374.60	1366.60	1363.60	S&G
WP24	893874.64	1129534.13	1372.50	1364.50	1361.50	S&G
WP25	893522.25	1129629.76	1384.70	1377.70	1362.70	S&G
WP26	893511.05	1129650.65	1384.50	1377.50	1362.50	S&G

Table B-17. Groundwater Points Excluded from the Evaluation(1)

Sampling Location	North Coordinate (2)	East Coordinate (2)	Surface Elevation (ft)	Elevation at Top of Screened Interval (ft)	Elevation at Bottom of Screened Interval (ft)	Geologic Unit of Screened Interval
WP27	893519.23	1129672.49	1384.40	1377.40	1362.40	S&G
WP28	893513.60	1129644.17	1384.60	1377.60	1362.60	S&G
WP29	893519.34	1129643.90	1385.10	1378.10	1363.10	S&G
WP30	893526.35	1129644.34	1385.20	1378.20	1363.20	S&G
WP31	893519.50	1129651.73	1385.40	1378.40	1363.40	S&G
WP32	893520.70	1129651.71	1385.40	1378.40	1363.40	S&G
WP33	893522.25	1129651.70	1385.40	1378.40	1363.40	S&G
WP34	893526.13	1129651.67	1385.40	1378.40	1363.40	S&G
WP35	893538.42	1129651.63	1384.00	1377.00	1362.00	S&G
WP36	893513.55	1129659.28	1384.70	1377.70	1362.70	S&G
WP37	893519.29	1129659.11	1385.30	1378.30	1363.30	S&G
WP38	893520.62	1129659.08	1385.40	1378.40	1363.40	S&G
WP39	893522.08	1129659.00	1385.40	1378.40	1363.40	S&G
WP40	893526.27	1129659.35	1385.30	1378.30	1363.30	S&G

NOTES: (1) This table lists points that were not included in the evaluation for DP section 4.2 because: a) no radiological data were available; b) data from that point were not validated (e.g., piezometers, surface elevation points, wells for the north plateau groundwater recovery system, wells for evaluation of the permeable treatment wall); c) sampling was dropped from the groundwater program because coverage was considered sufficient and no additional sampling was required (e.g., several points discontinued in 1995); d) the well was dry; or e) the sampling point was from an area outside the scope of the Phase 1 DP (e.g., groundwater seeps outside the process premises, wells from WMA 8).

(2) Western New York State Planar Coordinate System

LEGEND: S&G = sand and gravel, ULT = unweathered Lavery till, WLT = weathered Lavery till, LTS = Lavery till sand, KRS = Kent recessional sequence, BR = bedrock.

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APPENDIX C

DETAILS OF DCGL DEVELOPMENT AND THE INTEGRATED DOSE ASSESSMENT

PURPOSE OF THIS APPENDIX

The purpose of this appendix is to provide supporting information related to development of derived concentration guideline levels (DCGLs) and the limited integrated dose assessment performed to ensure that cleanup criteria for surface soil, subsurface soil, and streambed sediment used in Phase 1 of the proposed decommissioning would support any decommissioning approach that may be selected for Phase 2.

INFORMATION IN THIS APPENDIX

This appendix provides the following information:

- Table C-1 in Section 1 provides a complete list of RESRAD input parameters, except for distribution coefficients, and the bases for these parameters.
- Table C-2 in Section 1 provides a list of distribution coefficients and their bases.
- Table C-3 in Section 1 provides the exposure pathways considered in the analysis.
- Table C-4 in Section 1 provides data on measured radionuclide concentrations in the Lavery till in the area of the large excavations in Waste Management Area 1 and Waste Management Area 2.
- Section 2 describes the information that comprises Attachment 1, which supports the calculation of DCGL and Cleanup Goal values presented in Section 5 of the Decommissioning Plan.
- Attachment 1 provides electronic RESRAD input and output files for the three base cases (surface soil, subsurface soil, and streambed sediment), the limited integrated dose analysis, and the input parameter sensitivity analyses performed, along with the associated Microsoft Excel spreadsheets.

RELATIONSHIP TO OTHER PLAN SECTIONS

This appendix provides supporting information for Section 5. Information provided in Section 5 and in Section 1 on the project background will help place the information in this appendix into context.

1.0 Tabulated Data

Table C-1 identifies input parameters used in the RESRAD models, except for the distribution coefficients, which are included in Table C-2. Input parameters are provided for the three source exposure scenarios: surface soil (SS), subsurface soil (SB), and stream bank sediment (SD). The RESRAD input parameters presented in Table C-1 were selected as discussed in Section 5.

Distribution coefficients (K_d) are presented in Table C-2 for chemical elements of the 18 radionuclides and their decay progeny for each of the three analyses (SS, SB and SD) for each of the modeled media (contaminated zone, unsaturated zone and saturated zone) used in RESRAD. The conceptual models assume the sand and gravel unit is representative of the three RESRAD zones, except that in the SB and SD analyses, the contaminated zone is assumed to be represented by the Lavery till. The table includes the RESRAD default value, the specific value input into the RESRAD model for $DCGL_W$ calculations, either measured site-specific or reference values (as identified in Note 1 to table C-2), and the range of values used in the sensitivity analysis. The K_d values were selected to represent the central tendency of the site-specific data or were based on specific soil strata characteristics where available. Variability/uncertainty in the K_d values was addressed through the sensitivity analysis.

The exposure pathways presented in Table C-3 were based on the critical groups identified for each of the source media. The resident farmer was the critical receptor for soil exposure and the recreationist was identified as the critical receptor for stream bank sediment exposure.

The data in Table C-4 are the basis for the maximum radionuclide concentration data in Table 5-1. These data comprise the available characterization data for radionuclides in the Lavery till within the footprints of the large excavations for the Process Building-Vitrification area and the Low-Level Waste Treatment Facility area that are described in Section 7.

Preliminary dose assessments have been performed for the remediated WMA 1 and WMA 2 excavations. These assessments made use of the maximum measured radioactivity concentration in the Lavery till for each radionuclide as summarized in Table C-4, and the results of modeling to develop DCGLs for 25 mrem per year as shown in Table 5-8. The results were as follow:

WMA 1, a maximum of 1.0 mrem a year

WMA 2, a maximum of 0.08 mrem a year

Given the limited data available, these results must be viewed as order-of-magnitude estimates. However, they do suggest that actual potential doses from the two remediated areas are likely to be substantially below 25 mrem per year.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Area of contaminated zone (m^2)	1.00E+04	1.00E+04	SS	Assumed area of 10,000 m^2 for subsistence farmer scenario; garden is 2,000 m^2.
	1.00E+04	1.00E+02	SB	Assumed area of 100 m^2 for excavated contaminated cistern cuttings scenario.
	1.00E+04	1.00E+03	SD	Assumed 1000 m^2 area along stream bank (3 m wide by ~330 m length).
Thickness of contaminated zone (m)	2.00E+00	1.00E+00	SS, SD	Assumed surface soil contaminated zone thickness.
	2.00E+00	3.00E-01	SB	Assumed thickness of contaminated cistern cuttings spread on surface.
Length parallel to aquifer flow (m)	1.00E+02	1.00E+02	SS	Assumed. Only applicable for non-dispersion model.
Time since placement of material (y)	0.00E+00	0.00E+00	All	Only non-zero if K_d values are not available. (Site-specific K_ds are available).
Cover depth (m)	0.00E+00	0.00E+00	All	No cover considered.
Density of cover material (g/cm^3)	0.00E+00	not used	All	No cover considered.
Cover depth erosion rate (m/y)	0.00E+00	not used	All	No cover considered.
Density of contaminated zone (g/cm^3)	1.50E+00	1.70E+00	All	WVNSCO 1993a and WVNSCO 1993c.
Contaminated zone erosion rate (m/y)	1.00E-03	0.00E+00	All	Assumed for no source depletion.
Contaminated zone total porosity	4.00E-01	3.60E-01	All	WVNSCO 1993c.
Contaminated zone field capacity	2.00E-01	2.00E-01	All	WVNSCO 1993c.
Contaminated zone hydraulic conductivity (m/y)	1.00E+01	1.40E+02	All	Average for Sand and Gravel Thick Bedded Unit (4.43E-03 cm/s from Table 3-19) divided by 10 to provide vertical conductivity that accounts for potential anisotropy (DEIS Appendix E, Table E-3).
Contaminated zone b parameter	5.30E+00	1.40E+00	All	Yu, et al. 2000, Att. C table 3.5-1, mean for loamy sand (ln(mean)=0.305).
Average annual wind speed (m/sec)	2.00E+00	2.60E+00	All	WVNSCO 1993d.
Humidity in air (g/m^3)	8.00E+00	not used	All	Applicable for tritium exposures only.
Evapotranspiration coefficient	5.00E-01	5.50E-01	All	Evapotranspiration and runoff coefficients selected to achieve infiltration rate of 0.42 m/y (25% of applied water) for surface soil model.
Precipitation (m/y)	1.00E+00	1.16E+00	All	WVNSCO 1993d.
Irrigation (m/y)	2.00E-01	4.70E-01	SS, SB	Beyeler, et al. 1999.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
	2.00E-01	0.00E+00	SD	Not applicable for non-farming scenario.
Irrigation mode	overhead	overhead	All	Site-specific.
Runoff coefficient	2.00E-01	6.00E-01	All	Runoff and evapotranspiration coefficients selected to achieve infiltration rate of 0.42 m/y (25% of applied water) for surface soil model.
Watershed area for nearby stream or pond (m^2)	1.00E+06	1.37E+07	All	Based on drainage area of site of 13.7 km^2 or ~5.2 mi^2 .
Accuracy for water/soil computations	1.00E-03	1.00E-03	All	Default assumed.
Saturated zone density (g/cm^3)	1.50E+00	1.70E+00	All	WVNSCO 1993a and WVNSCO 1993c.
Saturated zone total porosity	4.00E-01	3.60E-01	All	WVNSCO 1993c.
Saturated zone effective porosity	2.00E-01	2.50E-01	All	WVNSCO 1993c.
Saturated zone field capacity	2.00E-01	2.00E-01	All	WVNSCO 1993c.
Saturated zone hydraulic conductivity (m/y)	1.00E+02	1.40E+03	All	Average for Sand and Gravel Thick Bedded Unit (4.43E-03 cm/s from Table 3-19)
Saturated zone hydraulic gradient	2.00E-02	3.00E-02	All	WVNSCO 1993b.
Saturated zone b parameter	5.30E+00	1.40E+00	All	Yu, et al. 2000, Att. C table 3.5-1, mean for loamy sand (ln(mean)=0.305).
Water table drop rate (m/y)	1.00E-03	0.00E+00	All	Site Specific.
Well pump intake depth (m below water table)	1.00E+01	5.00E+00	SS	Assumption based on site hydrogeology. Only applicable to non-dispersion model.
Model: Non-dispersion (ND) or Mass-Balance (MB)	ND	ND	SS	Applicable to areas >1,000 m2 (Yu, et.al. 2001, p.E-18)
	MB	MB	SB, SD	Applicable to areas <1,000 m2 (Yu, et. al. 2001, pE-18)
Well pumping rate (m^3/y)	2.50E+02	5.72E+03	SS, SB	Based on 2.9 m^3/y drinking water (2 L/d per 4 people for 365 days), 329 m^3/y household water (225 L/d per 4 people for 365 day), 385 m^3/y livestock watering (5 beef cattle at 50 L/d, 5 milk cows 160 L/d) and 5,000 m^3/y for irrigation of 10,000 m^2 (at rate of 0.5 m/y) from Yu, et al. 2000, Attachment C, Section 3.10.
	2.50E+02	0.00E+00	SD	Not applicable for non-farming scenario.
Number of unsaturated zone strata	1.00E+00	1.00E+00	All	Assumed.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Unsaturated zone thickness (m)	4.00E+00	2.00E+00	SS, SB	Site specific.
	4.00E+00	0.00E+00	SD	Assumed saturated for stream bank.
Unsaturated zone soil density (g/cm^3)	1.50E+00	1.70E+00	All	WVNSCO 1993a and WVNSCO 1993c.
Unsaturated zone total porosity	4.00E-01	3.60E-01	All	WVNSCO 1993c.
Unsaturated zone effective porosity	2.00E-01	2.50E-01	All	WVNSCO 1993c.
Unsaturated zone field capacity	2.00E-01	2.00E-01	All	WVNSCO 1993c.
Unsaturated zone hydraulic conductivity (m/y)	1.00E+01	1.40E+02	All	Average for Sand and Gravel Thick Bedded Unit (4.43E-03 cm/s from Table 3-19) divided by 10 to provide vertical conductivity that accounts for potential anisotropy (DEIS Appendix E, Table E-3).
Unsaturated zone b parameter	5.30E+00	1.40E+00	All	Yu, et al. 2000, Att. C table 3.5-1, mean for loamy sand (ln(mean)=0.305).
Distribution coefficients – radionuclides				
Contaminated zone (mL/g)	varies	Site specific	All	See Table C-2 for distribution coefficients.
Unsaturated zone 1 (mL/g)	varies	Site specific	All	See Table C-2 for distribution coefficients.
Saturated zone (mL/g)	varies	Site specific	All	See Table C-2 for distribution coefficients.
Plant Transfer Factor	varies	Chemical-specific	All	Default values assumed.
Fish Transfer Factor	Varies	Chemical-specific	SD	Default values assumed.
Leach rate (1/y)	varies	not used	All	Using site-specific Kd values instead of assigning leach rate.
Solubility constant	varies	not used	All	Using site-specific Kd values instead of assigning solubility constant.
Inhalation rate (m^3/y)	8.40E+03	8.40E+03	SS, SB	Beyeler, et al. 1999.
Mass loading for inhalation (g/m^3)	1.00E-04	2.50E-05	SS, SB	Yu, et al. 2000, Att. C Table 4.6-1 value represents ~60th percentile of distribution.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Exposure duration (y)	3.00E+01	1.00E+00	All	Yearly dose estimates calculated.
Filtration factor, inhalation	4.00E-01	4.00E-01	SS, SB	RESRAD default assumes 40% of outdoor concentration indoors (Yu, et al. 2000).
Shielding factor, external gamma	7.00E-01	2.73E-01	SS, SB	Yu, et al. 2000, Att. C Figure 7.10-1, mean of distribution approximates a frame house with slab or basement.
Fraction of time spent indoors	5.00E-01	6.60E-01	SS, SB	Yu, et al. 2000, Att. C Figure 7.6-2, value represents ~50th percentile of distribution.
	5.00E-01	0.00E+00	SD	Assumed.
Fraction of time spent outdoors	2.50E-01	2.50E-01	SS, SB	RESRAD default value used.
	2.50E-01	1.20E-02	SD	Based on 104 hours/year (2 hours/day, 2 day/week, 26 weeks/y) spent on the stream bank over 8760 residence hours per year (24 hr/day, 365 days/y)
Shape factor flag, external gamma	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Fruits, vegetables and grain consumption (kg/y)	1.60E+02	1.78E+02	SS, SB	Yu, et al. 2000, Att C. Table 5.4-2, value is mean of fruit, nonleafy and grains.
Leafy vegetable consumption (kg/y)	1.40E+01	2.46E+01	SS, SB	Yu, et al. 2000, Att C. Table 5.4-2, value is difference between mean total veg (74.6 kg/d) and nonleafy (50kg/d).
Milk consumption (L/y)	9.20E+01	1.01E+02	SS, SB	Yu, et al. 2000, Att C. Table 5.3-2, value is mean of all years.
Meat and poultry consumption (kg/y)	6.30E+01	6.50E+01	All	Beyeler, et al. 1999.
Fish consumption (kg/y)	5.40E+00	9.00E+00	SD	Exposure Factors Handbook (EPA, 1999). The value represents the 95th percentile of fish consumption by recreational anglers
Other seafood consumption (kg/y)	9.00E-01	0.00E+00	SD	Assumes only fish consumed from the stream
Soil ingestion rate (g/y)	3.65E+01	1.83E+01	All	Yu, et al. 2000, Att C. Figure 5.6-1, value represents mean of distribution for resident farmer (50 mg/d).
Drinking water intake (L/y)	5.10E+02	7.30E+02	SS, SB	Beyeler, et al. 1999.
	5.10E+02	1.00E+00	SD	Based on 104 hour/year exposure and 10 mL/hr for wading scenario (http://www.epa.gov/Region4/waste/ots/healtbul.htm)
Contamination fraction of drinking water	1.0	1.0	All	Assumed. For streambed sediment, this is 100% of incidental ingestion.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Contamination fraction of household water	1.0	1.0	SS, SB	Assumed.
Contamination fraction of livestock water	1.0	1.0	SS, SB	Assumed.
Contamination fraction of groundwater	1.0	0	SD	All water ingested is from surface water.
Contamination fraction of irrigation water	1.0	1.0	SS, SB	Assumed.
Contamination fraction of aquatic food	1.0	1.0	SD	Assumed.
Contamination fraction of plant food	-1	-1.0	SS, SB	Value of -1.0 allows RESRAD to calculate the fraction based on the source area.
Contamination fraction of meat	-1	-1.0	All	Value of -1.0 allows RESRAD to calculate the fraction based on the source area.
Contamination fraction of milk	-1	-1.0	SS, SB	Value of -1.0 allows RESRAD to calculate the fraction based on the source area.
Livestock fodder intake for meat (kg/day)	6.80E+01	2.73E+01	SS, SB	Beyeler, et al. 1999.
	6.80E+01	2.25E+00	SD	Assumption for deer.
Livestock fodder intake for milk (kg/day)	5.50E+01	6.42E+01	SS, SB	Beyeler, et al. 1999.
Livestock water intake for meat (L/day)	5.00E+01	5.00E+01	All	Beyeler, et al. 1999, assumed for venison exposure to sediment source.
Livestock water intake for milk (L/day)	1.60E+02	1.60E+02	SS, SB	RESRAD default value used.
Livestock soil intake (kg/day)	5.00E-01	5.00E-01	All	RESRAD default, assumed for venison exposure to sediment source.
Mass loading for foliar deposition (g/m^3)	1.00E-04	4.00E-04	SS, SB	Beyeler, et al. 1999.
Depth of soil mixing layer (m)	1.50E-01	1.50E-01	SS, SB	Beyeler, et al. 1999.
Depth of roots (m)	9.00E-01	9.00E-01	SS, SB	RESRAD default, represents crops with short growing seasons.
Drinking water fraction from ground water	1.0	1.0	All	Assumed.
Household water fraction from ground water	1.0	1.0	SS, SB	Assumed.
Livestock water fraction from ground water	1.0	1.0	SS, SB	Assumed.
Irrigation fraction from ground water	1.0	1.0	SS, SB	Assumed.
Wet weight crop yield for non-leafy (kg/m^2)	7.00E-01	1.75E+00	SS, SB	Yu, et al. 2000, Att. C Figure 6.5-1 value is mean of distribution.
Wet weight crop yield for leafy (kg/m^2)	1.50E+00	1.50E+00	SS, SB	RESRAD default.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Wet weight crop yield for fodder (kg/m^2)	1.10E+00	1.10E+00	SS, SB	RESRAD default.
Growing season for non-leafy (years)	1.70E-01	1.70E-01	SS, SB	RESRAD default.
Growing season for leafy (years)	2.50E-01	2.50E-01	SS, SB	RESRAD default.
Growing season for fodder (years)	8.00E-02	8.00E-02	SS, SB	RESRAD default.
Translocation factor for non-leafy	1.00E-01	1.00E-01	SS, SB	RESRAD default.
Translocation factor for leafy	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Translocation factor for fodder	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Dry foliar interception fraction for non-leafy	2.50E-01	2.50E-01	SS, SB	RESRAD default.
Dry foliar interception fraction for leafy	2.50E-01	2.50E-01	SS, SB	RESRAD default.
Dry foliar interception fraction for fodder	2.50E-01	2.50E-01	SS, SB	RESRAD default.
Wet foliar interception fraction for non-leafy	2.50E-01	2.50E-01	SS, SB	RESRAD default.
Wet foliar interception fraction for leafy	2.50E-01	6.70E-01	SS, SB	Yu, et al. 2000, Att. C Figure 6.7-1 represent the most likely value.
Wet foliar interception fraction for fodder	2.50E-01	2.50E-01	SS, SB	RESRAD default.
Weathering removal constant (1/y)	2.00E+01	1.80E+01	SS, SB	Yu, et al. 2000, Att. C Figure 6.6-1 represent the most likely value
Carbon-14-related exposure parameters				
C-12 concentration in water (g/cc)	2.00E-05	2.00E-05	All	RESRAD default.
C-12 concentration in soil (g/g)	3.00E-02	3.00E-02	All	RESRAD default.
Fraction of vegetable carbon from soil	2.00E-02	2.00E-02	All	RESRAD default.
Fraction of vegetable carbon from air	9.80E-01	9.80E-01	All	RESRAD default.
C-14 evasion layer thickness in soil (m)	3.00E-01	3.00E-01	All	RESRAD default.
C-14 evasion flux rate from soil (1/sec)	7.00E-07	7.00E-07	All	RESRAD default.
C-12 evasion flux rate from soil (1/sec)	1.00E-10	1.00E-10	All	RESRAD default.
Fraction of grain in beef cattle feed	0.8	0.8	All	RESRAD default.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Fraction of grain in milk cow feed	0.2	0.2	All	RESRAD default.
Storage times of contaminated foodstuff (days)				
Fruits, non-leafy vegetables, and grain	1.40E+01	1.40E+01	SS, SB	RESRAD default.
Leafy vegetables	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Milk	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Meat	2.00E+01	2.00E+01	SS, SB	RESRAD default.
Fish	7.00E+00	7.00E+00	SD	RESRAD default.
Crustacea and mollusks	7.00E+00	7.00E+00	Not used	RESRAD default.
Well water	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Surface water	1.00E+00	1.00E+00	SS, SB	RESRAD default.
Livestock fodder	4.50E+01	4.50E+01	SS, SB	RESRAD default
Radon-related exposure parameters				
Thickness of building foundation (m)	1.50E-01	not used	All	Applicable for Radon exposures only
Bulk density of building foundation (g/cc)	2.40E+00	not used	All	Applicable for Radon exposures only.
Total porosity of cover material	4.00E-01	not used	All	Applicable for Radon exposures only.
Total porosity of building foundation	1.00E-01	not used	All	Applicable for Radon exposures only.
Volumetric water constant of the cover material	5.00E-02	not used	All	Applicable for Radon exposures only.
Volumetric water constant of the foundation	3.00E-02	not used	All	Applicable for Radon exposures only.
Diffusion coefficient for radon gas (m^2/sec)				
in cover material	2.00E-06	not used	All	Applicable for Radon exposures only.
in foundation material	3.00E-07	not used	All	Applicable for Radon exposures only.
in contaminated zone soil	2.00E-06	not used	All	Applicable for Radon exposures only.
Radon vertical dimension of mixing (m)	2.00E+00	not used	All	Applicable for Radon exposures only.

Table C-1. RESRAD Input Parameters

RESRAD Parameter (Units)	Default	Value	Medium	Comment/Reference
Average building air exchange rate (1/hr)	5.00E-01	not used	All	Applicable for Radon exposures only.
Height of building or room (m)	2.50E+00	not used	All	Applicable for Radon exposures only.
Building indoor area factor	0.00E+00	not used	All	Applicable for Radon exposures only.
Building depth below ground surface (m)	-1	not used	All	Applicable for Radon exposures only.
Emanating power of Rn-222 gas	2.50E-01	not used	All	Applicable for Radon exposures only.
Emanating power of Rn-220 gas	1.50E-01	not used	All	Applicable for Radon exposures only.

LEGEND: SS = surface soil, SB = subsurface soil, SD = streambed sediment.

Table C-2. Soil/Water Distribution Coefficients

Radionuclide	RESRAD Default (mL/g)	Surface Soil DCGL Contaminated Zone (mL/g)	Subsurface Soil DCGL Contaminated Zone (mL/g)	Sediment DCGL Contaminated Zone (mL/g)	Unsaturated[2] Zone (mL/g)	Saturated[3] Zone (mL/g)
Principal Elements						
Americium	20	1900[4] (420 - 111,000)	4000[5] (420 - 111,000)	4000[5] (420 - 111,000)	1900[4] (420 - 111,000)	1900[4] (420 - 111,000)
Carbon	0	5[4] (0.7 - 12)	7[5] (0.7 - 12)	7[5] (0.7 - 12)	5[4] (0.7 - 12)	5[4] (0.7 - 12)
Curium[6]	calculated	calculated	calculated	calculated	calculated	calculated
Cesium	4600	280[4] (48 - 4800)	480[5] (48 - 4800)	480[5] (48 - 4800)	280[4] (48 - 4800)	280[4] (48 - 4800)
Iodine[6]	calculated	1[4] (0.4 - 3.4)	2[7] (0.4 - 3.4)	2[7] (0.4 - 3.4)	1[4] (0.4 - 3.4)	1[4] (0.4 - 3.4)
Neptunium[6]	calculated	2.3[8] (0.5 - 5.2)	3[5] (0.5 - 5.2)	3[5] (0.5 - 5.2)	2.3[8] (0.5 - 5.2)	2.3[8] (0.5 - 5.2)
Plutonium	2000	2600[8] (5 - 27,900)	3000[5] (5 - 27,900)	3000[5] (5 - 27,900)	2600[8] (5 - 27,900)	2600[8] (5 - 27,900)

Table C-2. Soil/Water Distribution Coefficients

Radionuclide	RESRAD Default (mL/g)	Surface Soil DCGL Contaminated Zone (mL/g)	Subsurface Soil DCGL Contaminated Zone (mL/g)	Sediment DCGL Contaminated Zone (mL/g)	Unsaturated(2) Zone (mL/g)	Saturated(3) Zone (mL/g)
Strontium	30	6.16(8) (1 - 32)	15(5) (1 - 32)	15(5) (1 - 32)	6.16(8) (1 - 32)	6.16(8) (1 - 32)
Technetium	0	0.1(4) (0.01 - 4.1)	4.1(7) (1 - 10)	4.1(7) (1 - 10)	0.1(4) (0.01 - 4.1)	0.1(4) (0.01 - 4.1)
Uranium	50	35(4) (15 - 350)	10(7) (1 - 100)	10(7) (1 - 100)	35(4) (15 - 350)	35(4) (15 - 350)
Progeny Elements(9)						
Actinium	20	20	20	20	20	20
Lead	100	100	100	100	100	100
Protactinium	50	50	50	50	50	50
Radium	70	70	70	70	70	70
Thorium	60,000	60,000	60,000	60,000	60,000	60,000

NOTES: (1) Sources of K_d values considered included Table 3-20; NUREG-5512 (Beyeler, et al. 1999), Table 6.7; RESRAD User's Guide (Yu, et al. 2001), Tables E-3, E-4; and Sheppard and Thibault 1990. Values in parentheses are the bounds used in the sensitivity evaluation, selected considering site-specific and literature values to reflect a reasonable range.

(2) Sediment model assumes no unsaturated zone. Values used for surface and subsurface soil evaluation only.

(3) Values presented here are those used for surface soil DCGLs based on the non-dispersion model. Saturated zone distribution coefficients are not utilized by RESRAD for the mass-balance groundwater model.

(4) From Sheppard and Thibault 1990, for sand.

(5) Site specific value for the unweathered Lavery till (see Section 3.7.8, Table 3-20).

(6) RESRAD default for this radionuclide is to allow the code to calculate the distribution coefficient based on correlation with plant root uptake transfer factor.

(7) Site specific value for the Lavery till (see Section 3.7.8, Table 3-20).

(8) Site specific value for the sand and gravel unit (see Section 3.7.8, Table 3-20).

(9) Progeny K_ds were not included in the sensitivity analysis; RESRAD default values were used in all cases.

Table C-3 Scenario exposure pathways for WVDP DCGL development

Exposure Pathways	Resident Farmer (surface soil and Lavery Till source)	Recreationist (sediment source)
Incidental ingestion of source	●	●
External exposure to source	●	●
Inhalation of airborne source	●	○
Ingestion of groundwater impacted by source	●	x
Ingestion of milk impacted by soil and water sources	●	x
Ingestion of beef impacted by soil and water sources	●	x
Ingestion of produce impacted by soil and water sources	●	x
Incidental ingestion of surface water impacted by source	○	●
Ingestion of fish impacted by source	○	●
Ingestion of venison impacted by sediment and water sources	○	●

LEGEND:

● - Pathway is considered complete and is included in DCGL development.

○ - Pathway is considered potentially complete but unlikely, and is not included in DCGL development.

x - Pathway is considered incomplete and is not included in DCGL development.

Table C-4. Radiological Concentrations from Lavery Till Samples in the WMA 1 and WMA 2 Excavation Areas[1]

Location	Nuclide	Result (pCi/g)	Depth (ft)
BH-17 (WMA 6, 1993) (possibly some from sand & gravel at sample top)	Sr-90	1.1E-01	26-28
	Cs-137	2.6E-02	26-28
	U-232	< 3.2E-03	26-28
	U-233/234	1.6E-01	26-28
	U-235	< 5.8E-03	26-28
	U-235/236	< 6.9E-03	26-28
	U-238	1.1E-01	26-28
	Pu-238	< 4.3E-03	26-28
	Pu-239/240	< 4.3E-03	26-28
	Pu-241	1.3E+00	26-28
	Am-241	< 9.6E-03	26-28
BH-21A (WMA 1, 1993) (possibly some from sand & gravel at sample top)	Sr-90	4.5E+02	36-38
	Cs-137	< 3.0E-02	36-38
	U-232	< 7.4E-03	36-38
	U-233/234	8.6E-02	36-38

Table C-4. Radiological Concentrations from Lavery Till Samples in the WMA 1 and WMA 2 Excavation Areas[1]

Location	Nuclide	Result (pCi/g)	Depth (ft)
	U-235	< 5.1E-03	36-38
	U-235/236	< 7.2E-03	36-38
	U-238	7.1E-02	36-38
	Pu-238	< 4.8E-03	36-38
	Pu-239/240	< 4.8E-03	36-38
	Pu-241	< 1.1E+00	36-38
	Am-241	< 7.2E-03	36-38
GP30 (WMA 1, 1998)	Sr-90	6.6E+00	36.5-37
	Sr-90	4.2E+00	37-37.5
	Sr-90	6.3E+00	37.5-38
	Sr-90	5.5E+01	38-38.5
	Sr-90	5.9E+01	38.5-39
	Sr-90	3.4E+01	39-39.5
	Sr-90	2.9E+01	39.5-40
GP73 (WMA 1, 1998)	Sr-90	1.9E+00	40-40.5
	Sr-90	1.8E+00	40.5-41
	Sr-90	5.2E+00	41-41.5
	Sr-90	8.4E+00	41.5-42
GP80 (WMA 1, 1998)	C-14	< 8.6E-02	40-42
	Sr-90	1.3E+01	40-42
	Tc-99	< 2.6E-01	40-42
	I-129	< 2.3E-01	40-42
	Cs-137	< 2.2E-02	40-42
	Pu-241	< 2.1E+00	40-42
GP86 (WMA 1, 1998)	Sr-90	2.2E+00	39-39.5
	Sr-90	1.0E+00	39.5-40
	Sr-90	3.0E+00	40-40.5
	Sr-90	1.0E+01	40.5-41
	Sr-90	4.1E+01	41-41.5
	Sr-90	3.0E+01	41.5-42
BH-05 (WMA 2, 1993)	Sr-90	8.5E-01	12-14
	Cs-137	4.5E-01	12-14
	U-232	1.2E-02	12-14
	U-233/234	1.8E-01	12-14
	U-235	< 5.9E-03	12-14
	U-235/236	< 8.3E-03	12-14
	U-238	1.1E-01	12-14

Table C-4. Radiological Concentrations from Lavery Till Samples in the WMA 1 and WMA 2 Excavation Areas[1]

Location	Nuclide	Result (pCi/g)	Depth (ft)
	Pu-238	1.0E-02	12-14
	Pu-239/240	< 5.9E-03	12-14
	Pu-241	< 1.3E+00	12-14
	Am-241	3.0E-02	12-14
BH-07 (WMA 2, 1993) (possibly some from sand & gravel at sample top)	Sr-90	1.3E-01	12-14
	Cs-137	7.5E-02	12-14
	U-232	< 8.7E-03	12-14
	U-233/234	2.2E-01	12-14
	U-235	< 6.6E-03	12-14
	U-235/236	< 7.6E-03	12-14
	U-238	1.5E-01	12-14
	Pu-238	< 4.7E-03	12-14
	Pu-239/240	< 6.2E-03	12-14
	Pu-241	9.5E-01	12-14
	Am-241	< 5.1E-03	12-14
BH-08 (WMA 2, 1993) (possibly some from sand & gravel at sample top)	Sr-90	1.8E+02	10-12
	Cs-137	2.5E+02	10-12
	U-232	1.9E+01	10-12
	U-233/234	9.7E+00	10-12
	U-235	3.2E-01	10-12
	U-235/236	5.0E-01	10-12
	U-238	1.3E+01	10-12
	Pu-238	3.9E+00	10-12
	Pu-239/240	7.6E+00	10-12
	Pu-241	2.7E+01	10-12
	Am-241	1.1E+01	10-12
BH-12 (WMA 2, 1993) (possibly some from sand & gravel at sample top)	Sr-90	1.8E-01	14-16
	Cs-137	< 2.2E-02	14-16
	U-232	< 6.0E-03	14-16
	U-233/234	1.1E-01	14-16
	U-235	< 7.0E-03	14-16
	U-235/236	1.3E-02	14-16
	U-238	9.7E-02	14-16
	Pu-238	< 4.9E-03	14-16
	Pu-239/240	< 4.9E-03	14-16
	Pu-241	< 1.0E+00	14-16
	Am-241	< 4.6E-03	14-16

Table C-4. Radiological Concentrations from Lavery Till Samples in the WMA 1 and WMA 2 Excavation Areas[1]

Location	Nuclide	Result (pCi/g)	Depth (ft)
BH-13 (WMA 2, 1993) (possibly some from sand & gravel at sample top)	Sr-90	1.8E-01	18-20
	Cs-137	2.7E+00	18-20
	U-232	1.6E-02	18-20
	U-233/234	8.5E-02	18-20
	U-235	< 5.1E-03	18-20
	U-235/236	< 8.2E-03	18-20
	U-238	5.3E-02	18-20
	Pu-238	2.4E-02	18-20
	Pu-239/240	2.6E-02	18-20
	Pu-241	< 8.1E-01	18-20
	Am-241	9.5E-02	18-20
BH-14 (WMA 2, 1993) (possibly some from sand & gravel at sample top)	Sr-90	1.8E+01	14-16
	Cs-137	1.9E+00	14-16
	U-232	2.0E-02	14-16
	U-233/234	1.9E-01	14-16
	U-235	< 7.9E-03	14-16
	U-235/236	< 1.1E-02	14-16
	U-238	2.8E-01	14-16
	Pu-238	1.7E-01	14-16
	Pu-239/240	1.6E-01	14-16
	Pu-241	< 1.1E+00	14-16
	Am-241	1.1E-01	14-16

NOTE: (1) Data are from the 1993 RCRA facility investigation and the Geoprobe® studies described in Section 4.

2.0 Information Provided in Attachment 1

Other information associated with the dose modeling is provided in Attachment 1. As explained in Section 5, the dose calculations were performed using RESRAD 6.4 and the results were exported to Microsoft Excel for post-processing. Attachment 1 provides:

- RESRAD input files to verify input parameters and model setup,
- RESRAD output files to verify input parameters and results,
- Excel result files containing (1) RESRAD output results (exported from the RESRAD summary report), (2) summaries of data [maximum dose-source ratios (DSRs) and times of maxima], (3) calculation of $DCGL_W$ values from the maximum DSRs, (4) calculation of area factors and $DCGL_{EMC}$ values, and (5) summary of sensitivity results

DCGL development was based on entering unit source concentrations (1pCi/g) for 18 radionuclides into RESRAD to generate DSRs in units of mrem/y per pCi/g (RESRAD

output results based on unit concentrations can be interpreted as either the dose or DSR, and the terms are used interchangeably in this document). The individual, peak DSRs are then used to generate DCGLs for each radionuclide based on the following equation:

DCGL (pCi/g) = Dose Limit (mrem/y) / Maximum DSR (mrem/y per pCi/g) (Eq.1)

The dose limit of 25 mrem/y and maximum DSRs were used as the basis for developing the DCGLs. Further details regarding the Attachment 1 files are presented below. Because of the uncertainty in the actual distributions and mixtures of radionuclides in the environmental media, the DCGL for each radionuclide is calculated individually. Following characterization, the working cleanup levels for mixtures can be developed using the sum of fractions method discussed in Chapter 5 of the MARSSIM.

2.1 Input Parameters Tables

The parameters input to the RESRAD model include:

- Base case values for the $DCGL_W$ calculations,
- Modification of source area only for $DCGL_{EMC}$ calculations, and
- Variation of key parameters to evaluate model sensitivity

The Excel file "WV Sensitivity Parameters Table.xls" (Table C.5) provides a summary of the following parameters which were varied to evaluate model sensitivity.

- Surface Soil Sources
  - Indoor/outdoor time fraction
  - Source thickness
  - Unsaturated zone thickness
  - Irrigation/well pumping rate
  - Soil/water distribution coefficients
  - Hydraulic conductivity (Vertical/Horizontal)
  - Runoff/Evapotranspiration coefficients/ Infiltration rate
  - Depth of well intake
  - Length of contaminated area parallel to aquifer flow
  - Plant transfer factors
  - Use of mass balance instead of non-dispersion groundwater model
- Subsurface Soil Sources (subsurface soil distributed on the surface):
  - Indoor/outdoor time fraction
  - Source thickness
  - Unsaturated zone thickness
  - Irrigation/well pumping rate

- Soil/water distribution coefficients
- Hydraulic conductivity (Vertical/Horizontal)
- Runoff/Evapotranspiration coefficients/ Infiltration rate
- Plant transfer factors

- Stream Bank Sediment sources:
  - Outdoor time fraction
  - Source thickness
  - Unsaturated zone thickness
  - Soil/water distribution coefficients
  - Runoff/Evapotranspiration coefficients/ Infiltration rate
  - Plant transfer factors
  - Fish transfer factors

These sensitivity parameters were selected based on preliminary model simulations and consideration of parameter priorities presented in Table 4.2 of NUREG-6697, Attachment B (Yu, et al. 2000). The parameters selected for analysis are discussed further below.

Sensitivity parameter values were selected to represent a reasonable range in order to provide bounds on the uncertainty in the DCGL calculations. The basis for particular parameter values are discussed below.

Indoor/Outdoor fraction – varied from 0.45/0.45 to 0.8/0.1 from the base case values of 0.66/0.25. The lower indoor fraction represents equal time indoors and outdoors, while the higher fraction was selected to represent a farmer spending inordinate amounts of time indoors.

Source thickness – for surface soil and sediment, varied from 0.5 to 3m to bound the base case value of 1m with potential thicknesses resulting from remedial activities and to account for potential source erosion uncertainty. For subsurface soil, varied from 0.1 to 1 m to bound the base case value of 0.3 m. The subsurface source thickness is dependent on the amount of material excavated during well/cistern installation, and depths less than the base case would correspond with a smaller source area for a given excavated volume.

Unsaturated zone thickness – varied from 1 to 5 m to bound the 2 m base case value with the range possible for the site. The range of results also provides an assessment of potential source erosion uncertainty.

Irrigation/well pumping rate - varied from 0.2/2720 to 0.8/8720 (m/y)/(m^3/y) to bound the base case of 0.5/5720 (m/y)/(m^3/y). The irrigation rate and well pump rate are directly related and the range reflects changes in crop irrigation only. For all cases, the assumed household and livestock water ingestion rates were held constant. This parameter is applicable to soil exposure only, not to sediment exposure

Soil/Water distribution coefficients – varied for each radionuclide based on site-specific data where available. If a range of site-specific distribution coefficients was not available (as was the case for the majority of radionuclides), values were selected from the literature to provide a bound on the base case uncertainty. The conceptual models assume the sand and gravel unit is representative of the three RESRAD zones (contaminated, unsaturated and saturated), except that in the SB and SD analyses, the contaminated zone is assumed to be represented by the Lavery till.

Hydraulic conductivity – for the contaminated and unsaturated zone, varied the vertical conductivity from 1 m/y (3.2E-06 cm/s) to 350 m/y (1.1E-03 cm/s) to bound the base case value of 140 m/y (4.4E-04 cm/s) which is the average for the sand and gravel unit divided by 10 to account for anisotropy (DEIS Appendix E, Table E-3). Similarly for the saturated zone, the horizontal conductivity was varied from 10 to 3500 m/yr from the base case of 1400 m/y. The conceptual model assumes the sand and gravel unit is representative of the unsaturated and saturated zone. The upper bound value is that used in the DEIS and is included for comparison.

Runoff/evapotranspiration coefficient – varied from 0.2/0.5 to 0.8/0.8 to bound the base case of 0.6/0.55. The base case was selected to achieve infiltration rate of 0.42 m/y which corresponds to 25% of the applied water (DEIS Appendix E). The upper bounds are assumed values and the lower bounds for these parameters represent the RESRAD defaults.

Depth of well intake – applicable to non-dispersion model only (surface soil base case). Varied from 3 to 10 m to bound the base case value of 5m. The lower bound represents the minimum for a 1 m contaminated thickness and 2 m unsaturated zone. The upper bound represents the upper end of observed thickness of the saturated zone on site.

Length of contaminated area parallel to aquifer flow - applicable to non-dispersion model only (surface soil base case). Varied from 50 m to 200 m to bound the base case of 100 m.

Plant transfer factors – varied from the constituent specific base cases by increasing and decreasing each parameter an order of magnitude.

Fish transfer factors – applicable for sediment source evaluation. Values varied from the constituent specific base cases by increasing and decreasing each parameter an order of magnitude.

Groundwater model – the surface soil base case non-dispersion model is varied to provide results for the mass balance model for comparison. The RESRAD User's Manual suggests the non-dispersion model for areas >1,000 m^2 (Yu et al. 2001, p.E-18).

2.2 RESRAD Input Files

The following RESRAD input files are provided to allow verification of input parameters and reproduction of the output files and summary graphics:

- $DCGL_W$ input files:

- WV Surface – 10k Base.RAD (Surface soil source of 10,000 m^2)
- WV Subsurface – 100 Base.RAD (Subsurface material as a surface source of 100 m^2)
- WV Sediment - 1k Base.RAD (Sediment source of 1,000 m^2)

- $DCGL_{EMC}$ input files (varying only source area from $DCGL_W$ files):
  - Surface Soil Source
    - WV Surface - 5k EMC.RAD (5,000 m^2 source)
    - WV Surface - 1k EMC.RAD (1,000 m^2 source)
    - WV Surface - 500 EMC.RAD (500 m^2 source)
    - WV Surface - 100 EMC.RAD (100 m^2 source)
    - WV Surface - 50 EMC.RAD (50 m^2 source)
    - WV Surface - 10 EMC.RAD (10 m^2 source)
    - WV Surface - 5 EMC.RAD (5 m^2 source)
    - WV Surface - 1 EMC.RAD (1 m^2 source)
  - Subsurface Source
    - WV Subsurface - 50 EMC.RAD (50 m^2 source)
    - WV Subsurface - 10 EMC.RAD (10 m^2 source)
    - WV Subsurface - 5 EMC.RAD (5 m^2 source)
    - WV Subsurface - 1 EMC.RAD (1 m^2 source)
  - Stream Bank Sediment Source
    - WV Sediment - 500 EMC.RAD (500 m^2 source)
    - WV Sediment - 100 EMC.RAD (100 m^2 source)
    - WV Sediment - 50 EMC.RAD (50 m^2 source)
    - WV Sediment - 10 EMC.RAD (10 m^2 source)
    - WV Sediment - 5 EMC.RAD (5 m^2 source)
    - WV Sediment - 1 EMC.RAD (1 m^2 source)

  Note: sediment source area width was maintained at 3 m when varying areas to represent assumed stream bank configuration.

- Sensitivity analysis input files:
  - Surface soil Source
    - WV Surface - SENS1.RAD (decreased indoor fraction)
    - WV Surface - SENS2.RAD (increased indoor fraction)
    - WV Surface - SENS3.RAD (decreased source layer thickness)

- WV Surface - SENS4.RAD (increased source layer thickness)
- WV Surface - SENS5.RAD (decreased unsaturated zone thickness)
- WV Surface - SENS6.RAD (increased unsaturated zone thickness)
- WV Surface - SENS7.RAD (decreased well pumping rate)
- WV Surface - SENS8.RAD (increased well pumping rate)
- WV Surface - SENS9.RAD (decreased K_d values)
- WV Surface - SENS10.RAD (increased K_d values)
- WV Surface - SENS11.RAD (decreased K value)
- WV Surface - SENS12.RAD (increased K value)
- WV Surface - SENS13.RAD (decreased runoff/evapotranspiration)
- WV Surface - SENS14.RAD (increased runoff/evapotranspiration)
- WV Surface - SENS15.RAD (decreased well intake depth)
- WV Surface - SENS16.RAD (increased well intake depth)
- WV Surface - SENS17.RAD (decreased length parallel to flow)
- WV Surface - SENS18.RAD (increased length parallel to flow)
- WV Surface - SENS19.RAD (decreased plant transfer factors)
- WV Surface – SENS20.RAD (increased plant transfer factors)
- WV Surface - SENS21.RAD (mass balance groundwater model)

– Subsurface Soil Source

- WV Subsurface - SENS1.RAD (decreased indoor fraction)
- WV Subsurface - SENS2.RAD (increased indoor fraction)
- WV Subsurface - SENS3.RAD (decreased source layer thickness)
- WV Subsurface - SENS4.RAD (increased source layer thickness)
- WV Subsurface - SENS5.RAD (decreased unsaturated zone thickness)
- WV Subsurface - SENS6.RAD (increased unsaturated zone thickness)
- WV Subsurface - SENS7.RAD (decreased well pumping rate)
- WV Subsurface - SENS8.RAD (increased well pumping rate)
- WV Subsurface - SENS9.RAD (decreased K_d values)
- WV Subsurface - SENS10.RAD (increased K_d values)
- WV Subsurface - SENS11.RAD (decreased K value)
- WV Subsurface - SENS12.RAD (increased K value)
- WV Subsurface - SENS13.RAD (decreased runoff/evapotranspiration)

    - WV Subsurface - SENS14.RAD (increased runoff/evapotranspiration)
    - WV Subsurface - SENS15.RAD (decreased plant transfer factors)
    - WV Subsurface – SENS16.RAD (increased plant transfer factors)
  - Sediment Source
    - WV Sediment - SENS1.RAD (decreased outdoor fraction)
    - WV Sediment - SENS2.RAD (increased outdoor fraction)
    - WV Sediment - SENS3.RAD (decreased source layer thickness)
    - WV Sediment - SENS4.RAD (increased source layer thickness)
    - WV Sediment - SENS5.RAD (increased unsaturated zone thickness)
    - WV Sediment - SENS6.RAD (largest unsaturated zone thickness)
    - WV Sediment - SENS7.RAD (decreased K_d values)
    - WV Sediment - SENS8.RAD (increased K_d values)
    - WV Sediment – SENS9.RAD (decreased runoff/evapotranspiration)
    - WV Sediment – SENS10.RAD (increased runoff/evapotranspiration)
    - WV Sediment - SENS11.RAD (decreased plant transfer factors)
    - WV Sediment – SENS12.RAD (increased plant transfer factors)
    - WV Sediment - SENS13.RAD (decreased fish transfer factors)
    - WV Sediment – SENS14.RAD (increased fish transfer factors)

The dose results from the above input files were the basis for calculation of $DCGL_W$ and $DCGL_{EMC}$ values. The DCGLs were calculated in Excel spreadsheets, based on exported data from the RESRAD summary output report. The following section describes the RESRAD output files, which are provided for informational purposes.

2.3 RESRAD Output Files

The RESRAD output files are provided to allow review of results without running the simulations. For the $DCGL_W$ simulations, summary, detailed, daughter, and concentration reports are included in the QA files. The summary report is also available for the $DCGL_{EMC}$ simulations. As indicated in the previous section, DCGL calculations are based on data exported from the RESRAD summary output report. RESRAD output files generated are as follows;

- $DCGL_W$ output files:
  - Surface Soil Source
    - WV Surface – 10k Base_sum.TXT (summary report)
    - WV Surface – 10k Base_ det.TXT (detailed report)

        - WV Surface – 10k Base _dtr.TXT (daughter report)
        - WV Surface – 10k Base _conc.TXT (concentration report)
    - Subsurface Soil Source
        - WV Subsurface – 100 Base_sum.TXT (summary report)
        - WV Subsurface – 100 Base_det.TXT (detailed report)
        - WV Subsurface – 100 Base_dtr.TXT (daughter report)
        - WV Subsurface – 100 Base_conc.TXT (concentration report)
    - Sediment Source
        - WV Sediment – 1k Base_sum.TXT (summary report)
        - WV Sediment – 1k Base_det.TXT (detailed report)
        - WV Sediment – 1k Base_dtr.TXT (daughter report)
        - WV Sediment – 1k Base_conc.TXT (concentration report)
- $DCGL_{EMC}$ output files (varying only source area from $DCGL_w$ files):
    - Surface Soil Source
        - WV Surface - 5k EMC_sum.TXT (5,000 m^2 source)
        - WV Surface - 1k EMC_ sum.TXT (1,000 m^2 source)
        - WV Surface - 500 EMC_sum.TXT (500 m^2 source)
        - WV Surface - 100 EMC_sum.TXT (100 m^2 source)
        - WV Surface - 50 EMC_sum.TXT (50 m^2 source)
        - WV Surface - 10 EMC_sum.TXT (10 m^2 source)
        - WV Surface - 5 EMC_sum.TXT (5 m^2 source)
        - WV Surface - 1 EMC_sum.TXT (1 m^2 source)
    - Subsurface Soil Source
        - WV Subsurface - 50 EMC_sum.TXT (50 m^2 source)
        - WV Subsurface - 10 EMC_sum.TXT (10 m^2 source)
        - WV Subsurface - 5 EMC_sum.TXT (5 m^2 source)
        - WV Subsurface - 1 EMC_sum.TXT (1 m^2 source)
    - Sediment Source
        - WV Sediment - 500 EMC_sum.TXT (500 m^2 source)
        - WV Sediment - 100 EMC_sum.TXT (100 m^2 source)
        - WV Sediment - 50 EMC_sum.TXT (50 m^2 source)
        - WV Sediment - 10 EMC_sum.TXT (10 m^2 source)

    - WV Sediment - 5 EMC_sum.TXT (5 m^2 source)
    - WV Sediment - 1 EMC_sum.TXT (1 m^2 source)

- Sensitivity analysis output files:
  - Surface Soil Source
    - WV Surface - SENS1_sum.TXT (decreased indoor fraction)
    - WV Surface - SENS2_sum.TXT (increased indoor fraction)
    - WV Surface - SENS3_sum.TXT (decreased source layer thickness)
    - WV Surface - SENS4_sum.TXT (increased source layer thickness)
    - WV Surface - SENS5_sum.TXT (decreased unsaturated zone thickness)
    - WV Surface - SENS6_sum.TXT (increased unsaturated zone thickness)
    - WV Surface - SENS7_sum.TXT (decreased well pumping rate)
    - WV Surface - SENS8_sum.TXT (increased well pumping rate)
    - WV Surface - SENS9_sum.TXT (decreased K_d values)
    - WV Surface - SENS10_sum.TXT (increased K_d values)
    - WV Surface - SENS11_sum.TXT (decreased K value)
    - WV Surface - SENS12_sum.TXT (increased K value)
    - WV Surface - SENS13_sum.TXT (decreased runoff/evapotranspiration)
    - WV Surface - SENS14_sum.TXT (increased runoff/evapotranspiration)
    - WV Surface - SENS15_sum.TXT (decreased well intake depth)
    - WV Surface - SENS16_sum.TXT (increased well intake depth)
    - WV Surface - SENS17_sum.TXT (decreased length parallel to flow)
    - WV Surface - SENS18_sum.TXT (increased length parallel to flow)
    - WV Surface - SENS19_sum.TXT (decreased plant transfer factors)
    - WV Surface – SENS20_sum.TXT (increased plant transfer factors)
    - WV Surface - SENS21_sum.TXT (mass balance groundwater model)
  - Subsurface Soil Source
    - WV Subsurface - SENS1_sum.TXT (decreased indoor fraction)
    - WV Subsurface - SENS2_sum.TXT (increased indoor fraction)
    - WV Subsurface - SENS3_sum.TXT (decreased source layer thickness)
    - WV Subsurface - SENS4_sum.TXT (increased source layer thickness)
    - WV Subsurface - SENS5_sum.TXT (decreased unsaturated zone thickness)

- WV Subsurface - SENS6_sum.TXT (increased unsaturated zone thickness)
- WV Subsurface - SENS7_sum.TXT (decreased well pumping rate)
- WV Subsurface - SENS8_sum.TXT (increased well pumping rate)
- WV Subsurface - SENS9_sum.TXT (decreased K_d values)
- WV Subsurface - SENS10_sum.TXT (increased K_d values)
- WV Subsurface - SENS11_sum.TXT (decreased K value)
- WV Subsurface - SENS12_sum.TXT (increased K value)
- WV Subsurface - SENS13_sum.TXT (decreased runoff/evapotranspiration)
- WV Subsurface - SENS14_sum.TXT (increased runoff/evapotranspiration)
- WV Subsurface - SENS15_sum.TXT (decreased plant transfer factors)
- WV Subsurface – SENS16_sum.TXT (increased plant transfer factors)

– Stream Bank Sediment Source

- WV Sediment - SENS1_sum.TXT (decreased outdoor fraction)
- WV Sediment - SENS2_sum.TXT (increased outdoor fraction)
- WV Sediment - SENS3_sum.TXT (decreased source layer thickness)
- WV Sediment - SENS4_sum.TXT (increased source layer thickness)
- WV Sediment - SENS5_sum.TXT (increased unsaturated zone thickness)
- WV Sediment - SENS6_sum.TXT (largest unsaturated zone thickness)
- WV Sediment - SENS7_sum.TXT (decreased K_d values)
- WV Sediment - SENS8_sum.TXT (increased K_d values)
- WV Sediment – SENS9_sum.TXT (decreased runoff/evapotranspiration)
- WV Sediment – SENS10_sum.TXT (increased runoff/evapotranspiration)
- WV Sediment - SENS11_sum.TXT (decreased plant transfer factors)
- WV Sediment – SENS12_sum.TXT (increased plant transfer factors)
- WV Sediment - SENS13_sum.TXT (decreased fish transfer factors)
- WV Sediment – SENS14_sum.TXT (increased fish transfer factors)

The following section presents the methods used to generate DCGLs from the RESRAD model output previously described.

2.4 Excel Result Files

The outputs of the RESRAD simulations (the DSR for each of the radionuclides at various future times) were exported to Excel from the RESRAD summary output report (specifically, the DSR values in the table presented at the bottom of page 45 of each

RESRAD summary report). For each simulation, dose results were exported for each of the 18 radionuclides, which includes the simulation year and dose (for that year) for each radionuclide. These have been generated for $DCGL_W$, $DCGL_{EMC}$, and sensitivity simulations for each source media and isotope. The peak dose for each radionuclide is identified and used as the basis for the DCGL calculation as follows;

$DCGL_w$ = Dose Limit / Peak radionuclide DSR (Eq.2)

Specific Excel result files are described below.

2.4.1 Surface Soil DCGLs

Surface soil DCGLs were calculated to conform with the annual dose limit for large areas ($DCGL_W$), smaller areas of elevated concentrations ($DCGL_{EMC}$), and to evaluate the sensitivity of the model to variations in specific parameters. The files associated with these calculations are described below.

Surface Soil $DCGL_W$ Values

The soil $DCGL_W$ values were calculated based on resident farmer exposure for a 10,000 m^2 source area and results from the RESRAD summary output report are presented in the Excel file "WVDP Surface DCGLs.XLS" in the sheet "Base" (Table C-6). The input files for the surface soil evaluation are presented in Section 2.2. These surface soil $DCGL_W$ values are the basis for calculation of surface soil area factors and $DCGL_{EMC}$ values.

Surface Soil $DCGL_{EMC}$ Values

The $DCGL_W$ values calculated on the Excel summary sheet previously discussed serve as the base case for subsequent $DCGL_{EMC}$ development; $DCGL_{EMC}$ values are based on varying the source area from the 10,000 m^2 value used for the $DCGL_W$ as discussed in Chapter 5 of the MARSSIM. The Excel file "WVDP Surface DCGLs.XLS" has sheets for each of the source areas used to generate the $DCGL_{EMC}$ (Tables C-7 to C-14). The sheet "Summary" in the Excel file "WV Surface DCGLs.XLS" summarizes the $DCGL_{EMC}$ (Table C-15) and Soil Area Factors (TableC-16) for each of the 18 radionuclides and selected source areas (ranging from 1 to 10,000 m^2).

Surface Soil $DCGL_W$ Sensitivity Analysis

The surface soil $DCGL_W$ sensitivity to key parameters was assessed by varying the input values for specific parameters and tabulating the results. The Excel file "WV Surface DCGL Sensitivity.XLS" contains the DSRs and DCGLs for each of 18 radionuclides from the RESRAD summary report output for each of the sensitivity simulations. Results of each run are in sheets SENS1 through SENS21 (Tables C-17 to C-37). Also included in the file are a summarization of the calculated DCGLs (Table C-38) and a summary of the percent change from the base case (Table C-39) for each of the sensitivity runs (also presented in Table 5-9). Table C-40 below presents a summary of the surface soil sensitivity results.

Table C-40 Summary of Surface Soil DCGL Sensitivity Analysis

Parameter	Run	Change in Sensitivity Parameter	Minimum		Maximum	
			Change	Nuclide(s)	Chang e	Nuclide(s)
Indoor/Outdoor Fraction	1	-32%	-23%	U-232	0%	C-14 I-129 Np-237 Tc-99
	2	21%	0%	C-14 I-129 Np-237 Tc-99 U-234	30%	U-232
Source Thickness	3	-50%	9%	Cs-137	238%	C-14
	4	200%	-58%	C-14	0%	Am-241 Cm-243 Cm-244 Pu-239 Pu-240
Unsaturated Zone Thickness	5	-50%	-10%	Tc-99	6%	U-235
	6	150%	-4%	U-235	10%	Tc-99
Irrigation/Pump Rate	7	-57%	-1%	U-232	52%	I-129
	8	70%	-31%	I-129	2%	U-232
Distribution Coefficients (Kd)	9	lower	-100%	Pu-239	6%	U-232
	10	higher	-4%	U-232	1146%	U-234
Hydraulic Conductivity	11	-99%	0%	Sr-90	1873%	I-129
	12	150%	0%	Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232	122%	U-235
Runoff/Evapo-transporation Coefficient	13	-69%	-28%	U-234	3%	U-232
	14	64%	-3%	U-232	123%	Np-237
Depth of Well Intake	15	-40%	-42%	I-129	0.1%	U-232
	16	100%	0%	Am-241 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240	93%	Np-237
Length Parallel to Aquifer Flow	17	-50%	0%	Am-241 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240	78%	U-235
	18	100%	-44%	U-235	0.1%	U-232
Plant Transfer Factors	19	-90%	-4%	I-129	387%	Sr-90
	20	900%	-90%	Sr-90	-6%	I-129
Mass Balance Model	21	-69%	-81%	U-234	0%	U-232

2.4.2 Subsurface Soil (Lavery till) DCGLs

To evaluate an excavation that would expose the resident farmer to subsurface material, DCGLs were developed to address this potential future source. It is possible that a farmer may install a cistern or well to access groundwater, and in the excavation process, contaminated Lavery till material from the subsurface may be spread on the ground surface and be a source of exposure. The following subsections discuss the files associated with this calculation.

Subsurface Soil $DCGL_W$ Values

The subsurface $DCGL_w$ values are presented in the Excel file "WV Subsurface DCGLs.XLS" in the sheet "Base" (TableC-41), and are based on the RESRAD input file "WV Subsurface – 100 Base.RAD" and results from page 45 of the RESRAD summary output report "WV Subsurface – 100 Base.TXT".

For calculation of the distributed soil, $DCGL_W$ values for a 100 m^2 source area of Lavery till on the surface were increased by a factor of 10 to account for an assumed blending of residually contaminated till with clean overlying soil in the excavation process (assuming 0.5 m of till for each 5 m of total excavation). This factor is applied to the final RESRAD generated DCGLw as presented in the overall summary table (See "DCGL Summary" section).

The input files for the subsurface soil evaluation are discussed in Section 2.2. These Lavery Till $DCGL_W$ values are used as the basis for calculation of the subsurface soil $DCGL_{EMC}$ values and for sensitivity analysis as described below.

Subsurface Soil $DCGL_{EMC}$ Values

Calculation of $DCGL_{EMC}$ values for the subsurface Lavery till was based on the base case area of 100 m^2 used for development of the $DCGL_W$ values (after accounting for blending). The $DCGL_{EMC}$ values were generated by varying the source area. The RESRAD output for these simulations are presented and summarized in the Excel file "WV Subsurface DCGLs.XLS". The results for each source area are presented in individual sheets (Tables C-42 to C-45). The sheet "Summary" presents the $DCGL_{EMC}$ values (Table C-46) and subsurface soil area factors (Table C-47) for each of the 18 radionuclides and selected source areas (ranging from 1 to 100 m^2).

Subsurface Soil Sensitivity Analysis

The subsurface soil $DCGL_W$ sensitivity to key parameters was assessed by varying the input values for specific parameters and tabulating the results. The Excel file "WV Subsurface DCGL Sensitivity.XLS" contains the DSRs and DCGLs for each of 18 radionuclides from the RESRAD summary report output for each of the sensitivity simulations. Results of each run are in sheets SENS1 through SENS16 (Tables C-48 to C-63). Also included in the file is a summarization of the calculated DCGLs (Table C-64) and a summary of the percent change from the base case (Table C-65) for each of the sensitivity runs (also presented in Table 5-10). Table C-66 below presents a summary of the subsurface soil sensitivity results.

Table C-66 Summary of Subsurface Soil DCGL Sensitivity Analysis

Parameter	Run	Change in Sensitivity Parameter	Minimum		Maximum	
			Change	Nuclide(s)	Change	Nuclide(s)
Indoor/Outdoor Fraction	1	-32%	-25%	Cs-137	0.1%	U-234
	2	21%	-1%	U-238	35%	U-232
Source Thickness	3	-67%	10%	U-238	255%	Tc-99
	4	233%	-90%	C-14	-1%	Cs-137
Unsaturated Zone Thickness	5	-50%	-3%	Tc-99	0%	Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Pu-241 Sr-90 U-232 U-235
	6	150%	0%	Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Pu-241 Sr-90 Tc-99 U-232 U-235	1%	U-238
Irrigation/Pump Rate	7	-57%	-36%	I-129	0%	Am-241 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240
	8	70%	0%	Cm-243 Pu-238 Pu-239 Pu-240	159%	U-238
Distribution Coefficients (K_d)	9	lower	-99%	Pu-239	16%	Tc-99
	10	higher	-27%	U-232	3144%	U-234
Hydraulic Conductivity	11	-99%	-1%	U-238	3%	I-129
	12	150%	0%	Am-241 C-14 Cm-243 Cm-244 Cs-137 I-129 Np-237 Pu-238 Pu-239 Pu-240 Pu-241 Sr-90 Tc-99 U-232 U-233 U-234 U-235 U-238	0%	Am-241 C-14 Cm-243 Cm-244 Cs-137 I-129 Np-237 Pu-238 Pu-239 Pu-240 Pu-241 Sr-90 Tc-99 U-232 U-233 U-234 U-235 U-238
Runoff/Evapo-transporation Coefficient	13	-69%	-38%	U-234	16%	U-232
	14	64%	-19%	U-232	188%	U-234
Plant Transfer Factors	15	-90%	-0.4%	U-238	574%	Sr-90
	16	900%	-90%	Tc-99	-1%	U-234

2.4.3 Streambed Sediment DCGLs

DCGLs were also developed to account for potential exposure associated with stream bank sediment (including direct pathways, fish ingestion, and venison ingestion). The stream bank rather than the streambed was the focus of the analysis because the recreationist is assumed to be in direct contact with the stream bank, and not the stream bed.

Files associated with the calculations are discussed below and presented in the files attachment.

Streambed Sediment $DCGL_W$ Values

The sediment $DCGL_W$ values were calculated based on a recreationist exposure for a 1,000 m^2 source area and results from the RESRAD summary output report are presented in the Excel file "WVDP Surface DCGLs.XLS" in the sheet "Base" (Table C-67). The input files for the sediment evaluation are discussed in Section 2.2. These sediment $DCGL_W$ values are the basis for calculation of Sediment Area Factors and $DCGL_{EMC}$ values.

Streambed Sediment $DCGL_{EMC}$ Values

The $DCGL_W$ values calculated on the Excel summary sheet previously discussed serve as the base case for subsequent $DCGL_{EMC}$ development, which are based on varying the source area from the 1,000 m^2 value used for the $DCGL_W$ values. The RESRAD output for these simulations are presented and summarized in the Excel file "WV Sediment DCGLs.XLS". The results for each source area are presented in individual sheets (Tables C-68 to C-73). The sheet "Summary" presents the $DCGL_{EMC}$ values (Table C-74) and sediment area factors (Table C-75) the 18 radionuclides and selected source areas (ranging from 1 to 1,000 m^2).

Streambed Sediment Sensitivity Analysis

The sediment $DCGL_W$ sensitivity to key parameters was assessed by varying the input values and tabulating the results. The Excel file "WV Sediment DCGL Sensitivity.XLS" contains the RESRAD summary report output for each of the sensitivity simulations. Results of each run are in sheets SENS1 through SENS14 (Tables C-76 to C-89). Also included in the file is a summarization of the calculated DCGLs (Table C-90) and percent change from the base case (Table C-91) for each of the sensitivity runs (also presented in Table 5-11). Table C-92 below presents a summary of the sediment sensitivity analysis.

Table C-92 Summary of Sediment DCGL Sensitivity Analysis

Parameter	Run	Change in Sensitivity Parameter	Minimum		Maximum	
			Change	Nuclide(s)	Change	Nuclide(s)
Outdoor Fraction	1	-50%	0%	C-14	97%	U-232
	2	100%	-50%	Cm-243	0%	C-14
Source Thickness	3	-50%	0%	Cm-243	157%	C-14
	4	200%	-52%	C-14	0%	Am-241 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239

Table C-92 Summary of Sediment DCGL Sensitivity Analysis

Parameter	Run	Change in Sensitivity Parameter	Minimum		Maximum	
			Change	Nuclide(s)	Change	Nuclide(s)
						Pu-240
Unsaturated Zone Thickness	**5**	0 m to 1m	0.3%	Cs-137	83%	U-234
	6	0 m to 3 m	0.3%	Cs-137	83%	U-234
Soil/Water Distribution Coefficients (Kd)	**7**	lower	-90%	Pu-239	47%	Pu-241
	8	higher	-59%	U-233	127%	Np-237
Runoff/Evapotran-sporation Coefficient	**9**	-54%	0%	Am-241 Cm-243 Pu-238 Pu-239 Pu-240	8%	U-232
	10	78%	-29%	U-233	0%	Am-241 Cm-243 Cm-244 Pu-238 Pu-239 Pu-240
Plant Transfer Factors	**11**	-90%	-29%	U-233	82%	Sr-90
	12	900%	-82%	Sr-90	-1%	U-235
Fish Transfer Factors	**13**	-90%	-28%	U-233	99%	Np-237
	14	900%	-84%	Np-237	-3%	Cs-137

Consideration of Subsurface Lavery till as a Continuing Source to Groundwater

An evaluation of the potential for the Lavery till to act as a continuing source to groundwater was conducted and concluded the following (See section 3.7 and Table 3-19 of the body of the plan):

- A well screened entirely in the Lavery Till could not produce enough groundwater for the resident farmer scenario.
- A well screened in both the sand and gravel unit and Lavery till would likely pump mostly groundwater from the sand and gravel unit due to the much higher relative hydraulic conductivity and subsequent development of preferential flowpaths, and contain highly diluted contributions of contaminated groundwater from the Lavery Till.
- Advective movement from the Lavery Till to the overlying Sand and Gravel Unit is unlikely considering the vertical downward groundwater gradient.
- Diffusive movement from the Lavery Till to the Sand and Gravel Unit is unlikely considering the very low diffusion coefficients for radionuclides.
- Migration vertically upward from the till through the aquifer and into a well that is screened several meters above the till is unlikely.

DCGL Summary

The Excel File "WV DCGL Summary Tables.xls" (Table C-93) summarizes the DCGLs for the surface soil, subsurface soil and sediment, and presents $DCGL_W$ and $DCGL_{EMC}$ for a 1 m^2 area (also presented in Table 5-8).

Integrated Dose Assessment

In order to account for potential exposure to multiple sources, a combined dose assessment was conducted. The assessment considered which combination of exposures was likely, and concluded that the resident farmer may also spend time in recreation along the stream bank.

The Excel File "WV DCGL Summary Tables.xls" presents the calculated $DCGL_W$ and $DCGL_{EMC}$ values when considering the combined doses from surface soil (90% x 25 mrem/yr = 22.5 mrem/y) and sediment sources (10% x 25 mrem/y = 2.5 mrem/y), which are summarized in Tables C-94, C-95, and C-96 (also presented in Table 5-13). In the same Excel file, Table C-96 presents the cleanup goals to be used as the criteria for the proposed remediation activities. Values in Table C-97 represent the $DCGL_W$ and $DCGL_{EMC}$ values for surface soil and sediment (considering the combined dose), as well as cleanup goals for subsurface soil (which are 50% of the $DCGL_W$ and $DCGL_{EMC}$ values adjusted to provide a margin of confidence/safety factor for excavation success for each radionuclide (also presented in Table 5-12).

Evaluation of Institutional Control Period

After Phase 1 proposed remediation there is assumed to be a 30 year period of institutional controls (associated with storage of the HLW canisters until 2041), prior to site access by the critical receptors. During this period, radionuclide inventories will be subject to decay and leaching, which will result in site concentrations at the time of exposure that are reduced from the initial concentrations left at the time of proposed remediation. With the exception of Sr-90 and Cs-137, DCGLs were developed neglecting the effects of decay and leaching from the source during the 30 year institutional control period. The ratio of the initial concentrations in soil to the RESRAD generated soil concentration after a 30 year simulation was used to provide an evaluation of uncertainty associated with the assumption of neglecting decay/leaching. A RESRAD simulation was run using the surface soil base case without irrigation, well pumping, or plant/animal/human uptake from soil (see RESRAD input file "WV SURFACE – 10k – LCH_DCAY.RAD" and output file "WV SURFACE – 10k – LCH_DCAY_sum.txt". The RESRAD concentration output summary file (see page 8 of the file "WV SURFACE – 10k – LCH_DCAY_conc.txt") provides the soil concentration at year 30, which is then related to the initial soil concentration to quantify the effects of leaching/decay (see Excel file "WV Institutional Control.xls" Table C-98).

Evaluation of Potential Dose Drivers and Sensitivity Parameters

The impact of specific sensitivity parameters is dependent on the radionuclides that contribute the majority of the dose to the receptor. Due to limited site data, a full evaluation can not be performed until additional site characterization data is available. In the interim, Table C-99 presented below identifies the primary dose pathways for each radionuclide and indicates which of the sensitivity parameters have significant impact on the dose. This evaluation would be refined as additional site data are collected.

Table C-99 Summary of Primary Dose Pathways

Nuclide	Primary Pathway for Dose	Key Parameters(1)	Year of Peak Dose
Surface Soil			
Am-241	Water independent (plant uptake)	plant transfer factors, source thickness	0.00E+00
C-14	Water independent (plant uptake)	source thickness	0.00E+00
Cm-243	External Exposure, Water independent (plant uptake)	plant transfer factors, source thickness	0.00E+00
Cm-244	Water independent (plant uptake)	plant transfer factors, source thickness	0.00E+00
Cs-137	External Exposure	outdoor fraction, plant transfer factors	0.00E+00
I-129	Water dependent (water ingestion, plant and milk uptake)	K, Kd, runoff/evap coefficients, well intake depth, groundwater model	9.21E+00
Np-237	Water dependent (water ingestion, plant uptake)	hydraulic conductivity, Kd, runoff/evap coefficients, well intake depth, groundwater model	2.01E+01
Pu-238	Water independent (plant uptake)	Kd, plant transfer factors	0.00E+00
Pu-239	Water independent (plant uptake)	Kd, plant transfer factors	0.00E+00
Pu-240	Water independent (plant uptake)	Kd, plant transfer factors	0.00E+00
Pu-241	Water independent (plant uptake)	Kd, plant transfer factors	5.52E+01
Sr-90	Water independent (plant uptake)	source thickness, plant transfer factors, Kd, groundwater model	0.00E+00
Tc-99	Water dependent (water ingestion, plant uptake), independent (plant uptake)	source thickness, well intake depth, plant transfer factors, length parallel to flow, Kd, K, groundwater model	1.54E+00
U-232	External Exposure	outdoor fraction, plant transfer factors	8.17E+00
U-233	Water dependent (water ingestion, plant uptake)	irrigation/pump rate, Kd, runoff/evap coefficients, groundwater model	2.96E+02
U-234	Water dependent (water ingestion, plant uptake)	irrigation/pump rate, Kd, runoff/evap coefficients, groundwater model	2.96E+02
U-235	Water dependent (water ingestion, plant uptake)	irrigation/pump rate, Kd, runoff/evap coefficients, groundwater model	2.96E+02
U-238	Water dependent (water ingestion, plant uptake)	irrigation/pump rate, Kd, runoff/evap coefficients, groundwater model	2.96E+02
Subsurface Soil			
Am-241	External Exposure, Water independent (plant uptake)	source thickness, plant transfer factors	0.00E+00
C-14	Water independent (plant uptake)	source thickness	0.00E+00
Cm-243	External Exposure	outdoor fraction, source thickness	0.00E+00
Cm-244	Water independent (plant uptake)	source thickness, plant transfer factors	0.00E+00
Cs-137	External Exposure	outdoor fraction, source thickness	0.00E+00
I-129	Water dependent (water ingestion)	source thickness, irrigation/pump rate, Kd, runoff/evap coefficients	6.32E+00
Np-237	Water independent (soil ingestion, plant	source thickness, Kd, runoff/evap coefficients	1.37E+01

Table C-99 Summary of Primary Dose Pathways

Nuclide	Primary Pathway for Dose	Key Parameters(1)	Year of Peak Dose
	uptake)		
Pu-238	Water independent (plant uptake, soil ingestion and inhalation)	source thickness, Kd, plant transfer factors	0.00E+00
Pu-239	Water independent (plant uptake, soil ingestion and inhalation)	source thickness, Kd, plant transfer factors	0.00E+00
Pu-240	Water independent (plant uptake, soil ingestion and inhalation)	source thickness, Kd, plant transfer factors	0.00E+00
Pu-241	Water independent (plant uptake)	source thickness, Kd, plant transfer factors	6.14E+01
Sr-90	Water independent (plant uptake)	source thickness, Kd, plant transfer factors	0.00E+00
Tc-99	Water dependent (plant uptake)	source thickness, plant transfer factors	0.00E+00
U-232	External Exposure	outdoor fraction, source thickness	4.60E+00
U-233	Water dependent (water ingestion)	Kd, runoff/evap coefficients	1.97E+02
U-234	Water dependent (water ingestion)	Kd, runoff/evap coefficients	1.97E+02
U-235	External Exposure	outdoor fraction, source thickness, Kd	0.00E+00
U-238	Water dependent (water ingestion)	source thickness, irrigation/pump rate, Kd, runoff/evap coefficients, groundwater model	1.98E+02
Sediment			
Am-241	External Exposure, Soil ingestion, Water independent (meat uptake)	outdoor fraction	0.00E+00
C-14	Water independent (meat uptake), Water dependent (fish uptake)	source thickness, unsaturated thickness, Kd	0.00E+00
Cm-243	External Exposure	outdoor fraction	0.00E+00
Cm-244	Soil ingestion	outdoor fraction	0.00E+00
Cs-137	External Exposure	outdoor fraction	0.00E+00
I-129	Water independent (meat uptake), Water dependent (fish uptake)	unsaturated thickness, Kd, fish transfer factors	0.00E+00
Np-237	External Exposure, Water independent (meat uptake), Water dependent (fish uptake)	unsaturated thickness, Kd, fish transfer factors	0.00E+00
Pu-238	Water independent (meat uptake), Soil ingestion	outdoor fraction, Kd	0.00E+00
Pu-239	Water independent (meat uptake), Soil ingestion	outdoor fraction, Kd	2.82E-01
Pu-240	Water independent (meat uptake), Soil ingestion	outdoor fraction, Kd	1.18E-01
Pu-241	External Exposure, Water independent (meat uptake), Soil ingestion	outdoor fraction, Kd	5.78E+01
Sr-90	Water independent (meat uptake)	plant and fish transfer factors	0.00E+00
Tc-99	Water independent (meat uptake)	Kd, plant and fish transfer factors	0.00E+00
U-232	External Exposure	outdoor fraction, Kd	7.72E+00

Table C-99 Summary of Primary Dose Pathways

Nuclide	Primary Pathway for Dose	Key Parameters(1)	Year of Peak Dose
U-233	External Exposure, Water independent (meat uptake), Water dependent (fish uptake)	outdoor fraction, unsaturated thickness, Kd, plant and fish transfer factors	1.56E-01
U-234	Water independent (meat uptake), Water dependent (fish uptake)	outdoor fraction, unsaturated thickness, Kd, fish transfer factors	1.81E-01
U-235	External Exposure	outdoor fraction	0.00E+00
U-238	External Exposure	outdoor fraction, fish transfer factors	0.00E+00

NOTE: (1) Key parameters identified in sensitivity runs. As additional site characterization data becomes available, the radionuclides driving dose and parameters most critical to calculating dose can be used to refine the sensitivity analysis.

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Attachments

1. Electronic Files Described in Section 2 (provided separately)

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APPENDIX D
ENGINEERED BARRIERS AND POST-REMEDIATION ACTIVITIES

PURPOSE OF THIS APPENDIX

The purpose of this appendix is to provide additional detail on engineered barriers installed during Phase 1 of the proposed decommissioning and describe the post-remediation monitoring, maintenance, and institutional control program to be implemented for the WVDP premises following Phase 1 of the Proposed Decommissioning.

INFORMATION IN THIS APPENDIX

This appendix includes information on engineered barrier conceptual designs and the post-remediation monitoring, maintenance, and institutional control program, organized as follows:

- Section 1 describes the conceptual designs of the engineered barriers to be installed during Phase 1 proposed decommissioning;
- Section 2 describes the post-remediation site monitoring and maintenance program that would be implemented for the project premises at the conclusion of Phase 1 proposed decommissioning;
- Section 3 describes the post-remediation site institutional control program that would be implemented for the project premises at the conclusion of Phase 1 of the proposed decommissioning.

RELATIONSHIP TO OTHER PLAN SECTIONS

Information provided in Section 1 on the project background and Section 7 on proposed decommissioning activities, would help place the information in this appendix into context. The content of Appendix D, like that of other parts of the plan, is consistent with the annotated NRC decommissioning plan checklist in Appendix A, which expresses NRC's expectations for section content.

1.0 Description of Engineered Barriers

This section presents a detailed description of the conceptual designs for the engineered barriers to be installed during Phase 1 of the proposed decommissioning, supplementing the physical descriptions previously presented in Section 7. Engineered barriers would be installed at the WMA 1 and WMA 2 excavations to facilitate the removal of sub-grade structures, excavate contaminated soil to meet unrestricted release criteria, and to prevent the recontamination of the WMA 1 and WMA 2 excavated areas by the non-source area of the North Plateau Plume.

According to the NRC's Final Policy Statement (67 FR 22), engineered barriers are generally passive manmade structures or devices intended to improve a facility's ability to meet a site's performance objectives. While institutional controls are designed to restrict access, engineered barriers are usually designed to inhibit water from contacting waste, limit releases, or mitigate doses to intruders.

1.1 Waste Management Area 1

Phase 1 of the WVDP proposed decommissioning would include the removal of all above grade and sub-grade structures of WMA 1 and the removal of the underlying soils associated with the source area of the north plateau groundwater plume to a maximum depth of approximately 50 feet. The removal of the sub-grade structures and the soils of the source area of the plume would require the installation of temporary and permanent subsurface hydraulic barrier walls prior to excavation as described in Section 7. A French drain system would be installed in the backfilled excavation to prevent mounding of groundwater against the permanent barrier wall as described in Section 7. These barrier walls and the French drain system are described in greater detail below.

1.1.1 Need for Subsurface Engineered Barriers and French Drain

During Phase 1 proposed decommissioning sub-grade structures (building cells, underground piping and tanks) and underlying vadose and saturated soils associated with the source area of the North Plateau Plume in WMA 1 would be removed down to the underlying Lavery till to meet the unrestricted release criteria in 10 CFR 20.1402. Much of the WMA 1 excavation would be within the saturated sand and gravel unit within the north plateau groundwater plume.

Subsurface hydraulic barrier walls would be installed on each side of the WMA 1 excavation to:

- Isolate the excavation from the non-source area of the north plateau groundwater plume,
- Prevent groundwater intrusion into the excavation from the surrounding sand and gravel unit,
- Allow dewatering of saturated soils within the excavation,
- Facilitate removal of sub-grade structures,
- Allow excavation of subsurface soil down into the Lavery till and up to the hydraulic barrier walls,
- Allow final status surveys and NRC confirmatory surveys to be performed in the bottom and sides of the excavation, and
- Prevent recontamination of the remediated and backfilled WMA 1 excavation from the non-source area of the north plateau groundwater plume until a Phase 2 decommissioning decision is made.[1]

Subsurface soil characterization would be performed in WMA 1 before excavation begins to identify the lateral extent of subsurface soil contamination associated with the source area of the North Plateau Plume. This subsurface soil data would be used to locate the temporary interlocking sheet piling which would be driven through the uncontaminated sand and gravel unit into the underlying Lavery till on the upgradient and cross-gradient sides of the WMA 1 excavation to prevent groundwater intrusion into the excavation from upgradient sources. A

[1]The recontamination potential is low since groundwater flows northeast away from WMA 1.

permanent hydraulic barrier of slurry wall type construction would be installed on the downgradient side of the excavation in soil contaminated by the north plateau groundwater plume to act as an intrusion barrier to prevent the migration of Sr-90 contaminated groundwater from the non-source area of the north plateau groundwater plume into the WMA 1 excavation.

The permanent downgradient hydraulic barrier would:

- Prevent recontamination of the remediated and backfilled WMA 1 excavation from the non-source area of the plume until a Phase 2 decommissioning decision is made, and
- Minimize groundwater recharge to the non-source area of the plume, thereby minimizing hydraulic heads and groundwater velocity.

A French drain system would be installed adjacent and hydraulically upgradient of the permanent hydraulic barrier wall once the WMA 1 excavation has been backfilled to maintain groundwater elevations near there current levels. The French drain system would:

- Prevent groundwater mounding against, and potential overtopping of, the permanent downgradient hydraulic barrier wall;
- Maintain hydraulic heads on the upgradient side of the barrier wall that coincide with the elevation of the French drain system, that are higher than groundwater levels downgradient of the barrier wall. This would create a hydraulic gradient towards the non-source area of the north plateau groundwater plume, preventing seepage from the plume through the wall into the backfilled excavation; and
- In conjunction with the permanent downgradient hydraulic barrier, minimize groundwater recharge to the non-source area of the North Plateau Plume thereby minimizing hydraulic heads and groundwater velocity across the North Plateau.

1.1.2 Hydraulic Barrier Walls and French Drain System

The WMA 1 excavation would require the installation of approximately 2,250 linear feet of subsurface hydraulic barrier wall comprised of temporary interlocking steel sheet piling on the upgradient and cross-gradient sides of the excavation and a permanent hydraulic barrier wall on the downgradient side of the excavation before excavation begins as shown on Figure D-1.

Temporary Sheet Pile Barrier Walls

Approximately 1,500 feet of conventional interlocking sheet piles would be installed in uncontaminated soils along the upgradient and cross-gradient sides of the excavation boundary before excavation begins (Figure D-1). The piles would be driven a minimum of two feet into the underlying Lavery till to prevent groundwater from migrating beneath the piles into the WMA 1 excavation.

Contaminated soil exceeding the subsurface soil cleanup criteria specified in Section 5 would be excavated leaving a soil cut-back slope against the sheet pile walls containing soil with radionuclide concentrations below the subsurface soil clean-up criteria.[2] The soil cut-backs along the sheet pile walls would be surveyed during the Phase 1 final status surveys as specified in Sections 7 and 9 of this plan. The sheet pile barrier wall would be removed as

[2] Figure 7-8 in Section 7 of this plan shows typical excavation slopes.

specified in Section 7 once the final status survey, the independent verification survey, and backfilling of the WMA 1 excavation is completed to allow a return to typical groundwater flow patterns within the sand and gravel unit.

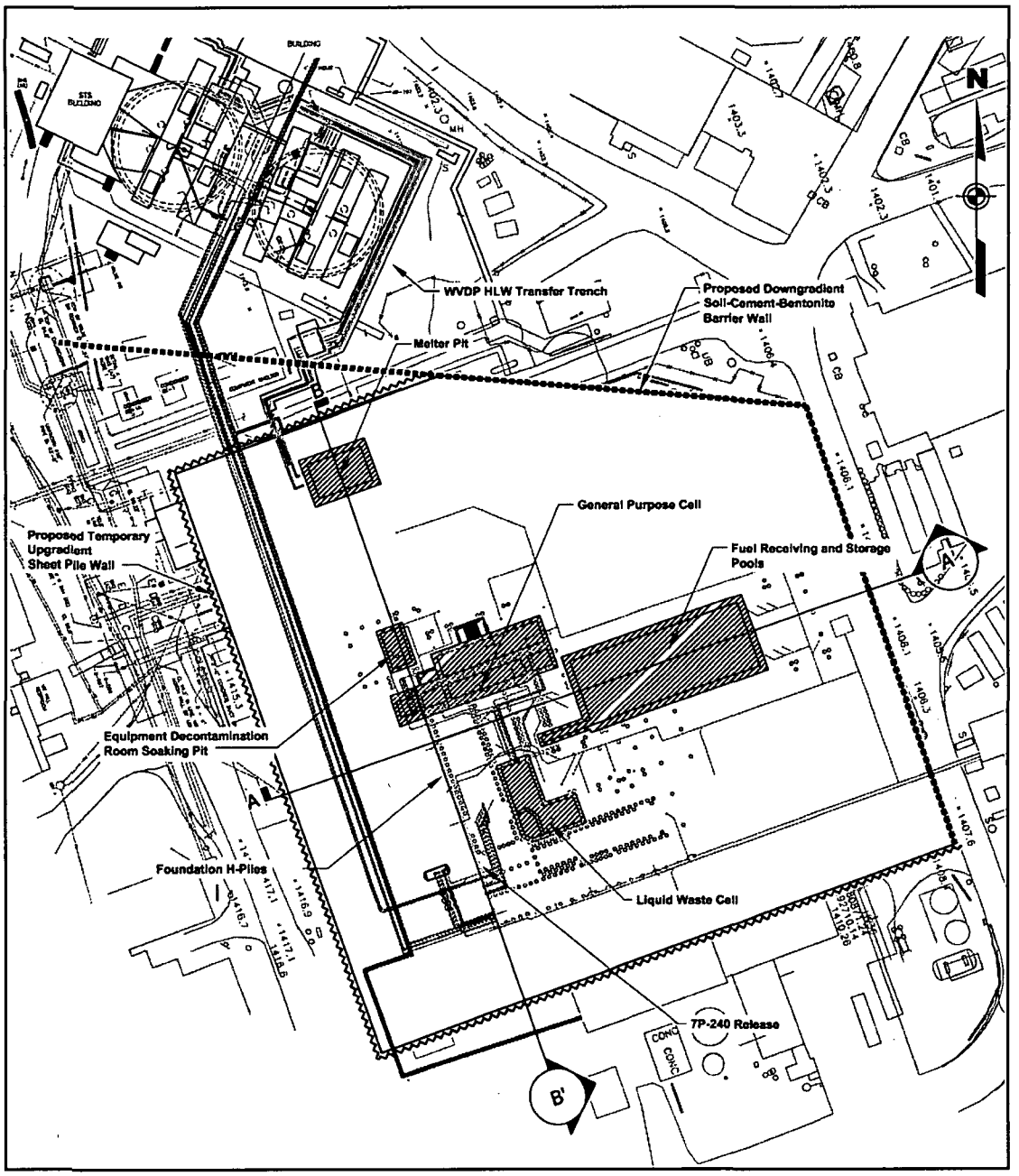

Figure D-1. Plan View of the WMA 1 Excavation

Permanent Downgradient Hydraulic Barrier Wall

The permanent hydraulic barrier wall constructed on the downgradient side of the WMA 1 excavation (Figure D-1) would be a vertical soil-cement-bentonite slurry wall installed using slurry wall trenching technology. This hydraulic barrier technology was selected because of its long history of successful usage. This wall would prevent migration of Sr-90 contaminated groundwater from the non-source area of the North Plateau Plume into the WMA 1 excavation both during excavation and after backfilling the excavation with clean fill.

The hydraulic barrier wall downgradient of the WMA 1 excavation would be installed under a carefully planned and rigorous quality control-quality assurance program as described in Section 8.

The soil-cement-bentonite barrier wall would be a mixture of 85 percent soil, five percent Portland cement, and 10 percent bentonite. The Portland cement would provide internal stability to the barrier wall and it would have an initial maximum design hydraulic conductivity of 6.0 E-06 cm/s.

The soil-cement-bentonite barrier wall would be approximately 750 feet long, two to 13 feet wide, and would be up to 50 feet deep with an average depth of 27 feet. The wall would extend through the sand and gravel unit and a minimum of two feet into the Lavery till to minimize groundwater flow beneath the bottom of the wall.

Approximately 225 feet of barrier wall outside of the excavation boundary would be two to three feet thick. The remaining 525 feet of barrier wall within the boundary of the excavation would be at least 13 feet thick to allow the excavation of subsurface soils up to and into the barrier wall. The proposed thickness would allow an excavation cut back slope of 1:2 (horizontal to vertical), which is typical of what can be achieved in most stiff clayey soils. The barrier wall material within the excavation cut-back slope would be surveyed during the Phase 1 final status survey.[3]

The upper three feet of the barrier wall would be constructed of clean backfill similar to the surrounding sand and gravel unit. This material would allow vehicular traffic over the barrier wall without damaging the underlying barrier wall.

French Drain System

A French drain system would be installed upgradient of the permanent hydraulic barrier wall during the backfilling of the WMA 1 excavation (Figure D-1). The French drain would be installed to keep groundwater levels at their current level on the upgradient side of the barrier wall to prevent groundwater mounding against the wall, prevent potential overtopping of the wall, and promote groundwater flow towards the non-source area of the north plateau groundwater plume.

The French drain would be constructed by excavating a trench, approximately four feet wide and 10 feet deep, placing perforated pipe into the bottom of the trench, and backfilling the trench with permeable granular materials. The northwest and southeast portions of the French

[3] As explained in Section 7 of this plan, any soil found to exceed cleanup goals would be removed only within the confines of the planned excavation, that is, within the confines of the downgradient hydraulic barrier wall and the sheet piles.

drain would meet at a concrete manhole located near the mid-point of the barrier wall. The French drain would be sloped to the southeast to discharge by gravity flow to a surface water drainage discharging to Erdman Brook.

1.1.3 Durability of Engineered Barriers

The materials used in the construction of the soil-cement-bentonite slurry walls are common natural geologic construction materials that exhibit long-term durability within the natural environment. The engineered barriers are expected to retain their design effectiveness until the start of Phase 2 of the decommissioning at a minimum. Their continued use would be among the factors evaluated in determining the approach to Phase 2 of the decommissioning.

The low-permeability bentonite used in the slurry wall construction is a natural geologic material exhibiting demonstrated long-term mineralogical and geologic stability (references D-2 and D-3). Chemical contaminants that might degrade the physical characteristics and/or compromise the hydraulic conductivity of soil-bentonite slurry walls include:

- Concentrated solutions of organic fluids (Mille, et al. 1992 and Khera and Tirumala 1992),
- Organic groundwater contaminants (Evans, et al. 1985b and Grube 1992), and
- Acidic or highly alkaline solutions (Evans, et al. 1985a and Fang et al. 1992).

However, these conditions are not present within the project premises.

The backfill to be used for slurry wall construction would be a mixture of soil and commercial sodium bentonite. The soil can be any material that could be classified as CL, CL/ML or ML/CL by the Unified Soil Classification System. The soil backfill would be natural geologic materials similar to the sand and gravel unit in the North Plateau. Uncontaminated sand and gravel from the trench excavation may also be used as soil backfill for the slurry wall. The sodium bentonite would be added at a rate recommended by the vendor to achieve a hydraulic conductivity on the order of 1 E-08 to 1 E-06 cm/s.

The geotechnical stability of the soil-bentonite slurry wall has been evaluated under combined static and seismic loading conditions. The evaluation results indicate that the proposed soil-bentonite slurry wall would provide the necessary strength to withstand damage from static and seismic loads predicted to occur during a hypothetical earthquake generating a horizontal acceleration of 0.20 g in the soil, with an approximate factor of safety of greater than 1.3 to greater than 3.0 (URS 2000).

The French drain would be constructed of natural (stone backfill) and man-made (perforated drain pipe, geotextile) materials. The French drain trench backfill would be designed to minimize silting of the drainpipe. The French drain would be periodically monitored and maintained until the start of Phase 2 decommissioning to ensure it is functioning properly.

1.1.4 Engineered Barriers and Groundwater Flow

Groundwater flow in the sand and gravel unit is currently to the northeast across the north plateau through WMA 1 and parallel to WMA 2 (Figure D-2). The permanent hydraulic barrier wall and French drain installed on the downgradient side of the WMA 1 excavation are nearly perpendicular to the current groundwater flow path in the sand and gravel unit in the north

plateau.

A three-dimensional near-field groundwater model was developed to simulate groundwater flow conditions near the engineered barriers installed at WMA 1 and WMA 2 using the STOMP computer code (Nichols, et al. 1997)[4]. The permanent barrier wall downgradient of the Process Building is oriented parallel to the groundwater elevation contours and perpendicular to groundwater flow in Figure D-2. The segment of barrier wall between the Process Building and the Waste Tank Farm has been modeled parallel to groundwater flow due to the model constraints.

Groundwater modeling suggests groundwater flow patterns upgradient of the barrier wall and French drain are similar to current flow patterns in the sand and gravel unit (Figure D-2). However, the hydraulic gradient becomes steeper at the barrier wall reflecting the effect of this barrier on groundwater flow. Water table elevations are approximately 15 feet higher on the upgradient side of the barrier wall compared to water levels immediately downgradient of the wall. This steep hydraulic gradient suggests that groundwater would preferentially flow from the backfilled WMA 1 excavation across the barrier wall into the non-source area of the North Plateau Plume, rather than from the non-source area of the plume into the backfilled WMA 1 excavation. Higher groundwater elevations are also found on the upgradient side of the barrier wall separating the WMA 1 excavation from the Waste Tank Farm, suggesting potential flow from WMA 1 into the Waste Tank Farm area. Flow contours east of the barrier wall suggest that groundwater flows to the east into the area of the backfilled WMA 2 excavation, as discussed in Section 1.2.4 of this appendix.

Modeling suggests that groundwater flow in the sand and gravel unit downgradient of the permanent barrier wall in WMA 1 continues to the northeast across the North Plateau. However, the upgradient diversion of groundwater flow by the barrier wall system results in an overall reduction in the hydraulic gradient of the non-source area of the north plateau groundwater plume.

[4] STOMP (Subsurface Transport Over Multiple Phases) solves the relevant conservation equations for the flow of both liquid and gas (air with water vapor) phases in a porous matrix confined in a cylindrical shape. This computer code was developed by DOE's Pacific Northwest National Laboratory.

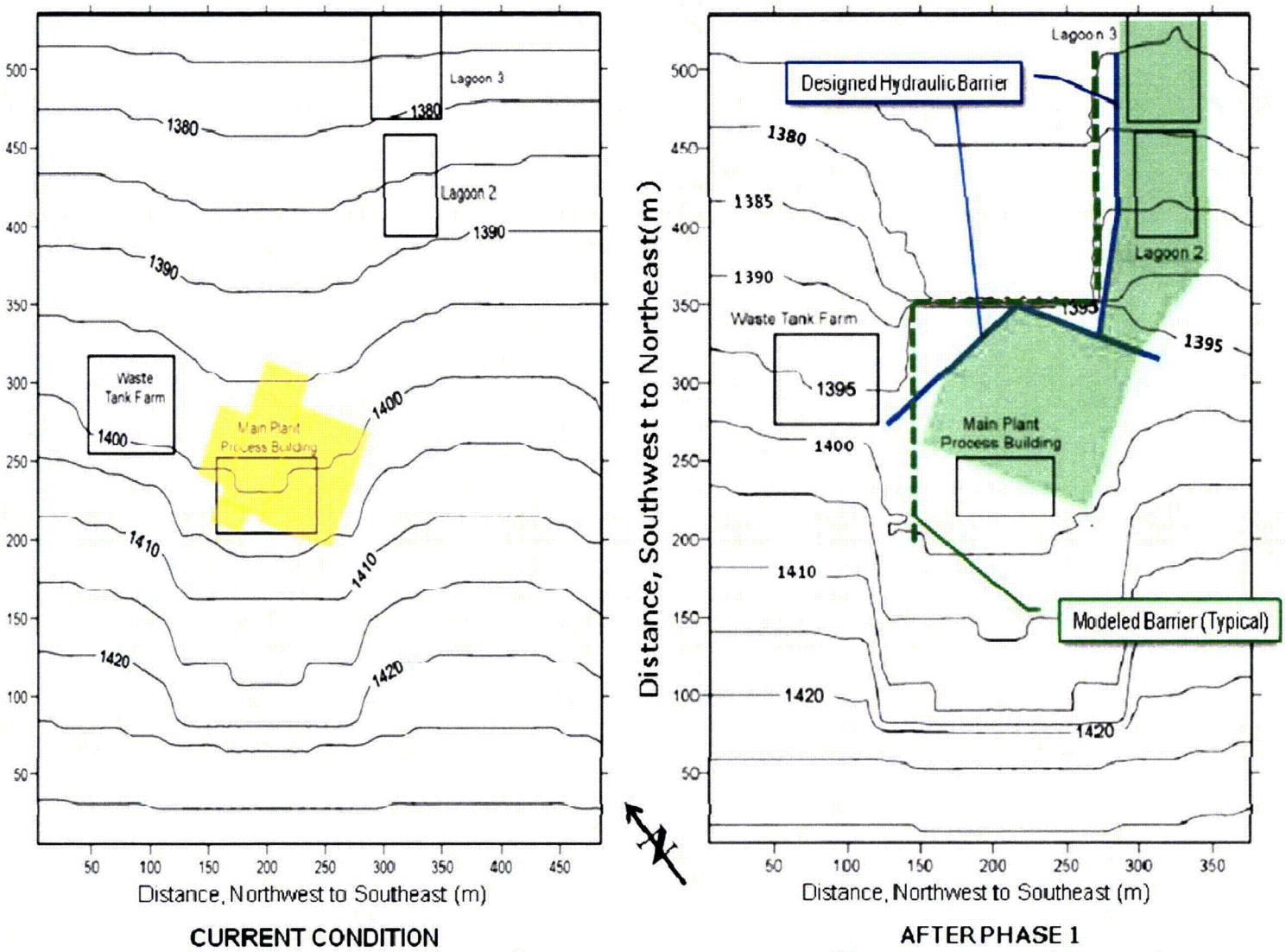

Figure D-2. Groundwater Flow Associated with the WMA 1 and WMA 2 Engineered Barriers

1.2 Waste Management Area 2

The Phase 1 proposed decommissioning activities in WMA 2 would include the removal of Lagoons 1 through 3, the Neutralization Pit, Interceptors, Solvent Dike, and surrounding contaminated soils within a single excavation down into the underlying Lavery till. Most of this excavation is cross gradient to the non-source area of the North Plateau Plume (Figure D-3). The removal of the lagoons, sub-grade structures, and surrounding soils would require the installation of a permanent subsurface hydraulic barrier wall prior to excavation to facilitate removal activities and to prevent potential recontamination of the area from the non-source area of the north plateau groundwater plume as described in Section 7. The barrier wall for WMA 2 is described in greater detail below.

1.2.1 Need for Subsurface Engineered Barriers

Lagoons 1 through 3, sub-grade structures, and surrounding contaminated vadose and saturated soils would be removed to a depth of approximately 14 feet to meet the unrestricted release criteria in 10 CFR 20.1402. Most of the WMA 2 excavation may be impacted by migration of Sr-90 contaminated groundwater from the adjacent non-source area of the north plateau groundwater plume. The need for a subsurface hydraulic barrier wall for the 4.2-acre excavation area across WMA 2 is the same as the rationale described earlier in Section 1.1.1 of this Appendix for the excavation of WMA 1.

A permanent hydraulic barrier of slurry wall type construction would be installed on the northwest side of the WMA 2 excavation to act as an intrusion barrier to prevent the migration of Sr-90 contaminated groundwater from the non-source area of the north plateau groundwater plume into the WMA 2 excavation. This permanent downgradient hydraulic barrier would prevent recontamination of the remediated and backfilled WMA 2 excavation from the non-source area of the north plateau plume until a Phase 2 decommissioning decision is made.

1.2.2 Hydraulic Barrier Wall

Before excavation activities begin in WMA 2 a permanent subsurface hydraulic barrier wall would be installed on the northwest side of the WMA 2 excavation as shown on Figure D-3.

Permanent Hydraulic Barrier Wall

The permanent hydraulic barrier wall constructed on the northwest side of the WMA 2 excavation would be a vertical soil-bentonite slurry wall installed using slurry wall trenching technology. This hydraulic barrier technology was selected because of its long history of successful usage. This wall would prevent migration of Sr-90 contaminated groundwater from the non-source area of the North Plateau Plume into the WMA 2 excavation both during excavation and after the excavation has been backfilled with clean fill.

The hydraulic barrier wall installed northwest of the WMA 2 excavation would be installed under a carefully planned and rigorous quality control-quality assurance program as described in Section 8. It would be a mixture of 90 percent soil and 10 percent bentonite and it would have an initial design hydraulic conductivity of 1.0 E-7 cm/s. The barrier wall would be approximately 1,100 feet long, sufficiently wide to provide the stability necessary to permit excavation close to the edge of the excavation, and up to 20 feet deep, with an average depth of 16 feet. The wall would extend through the sand and gravel unit and a minimum of two feet into the Lavery till to minimize groundwater flow beneath the bottom of the wall.

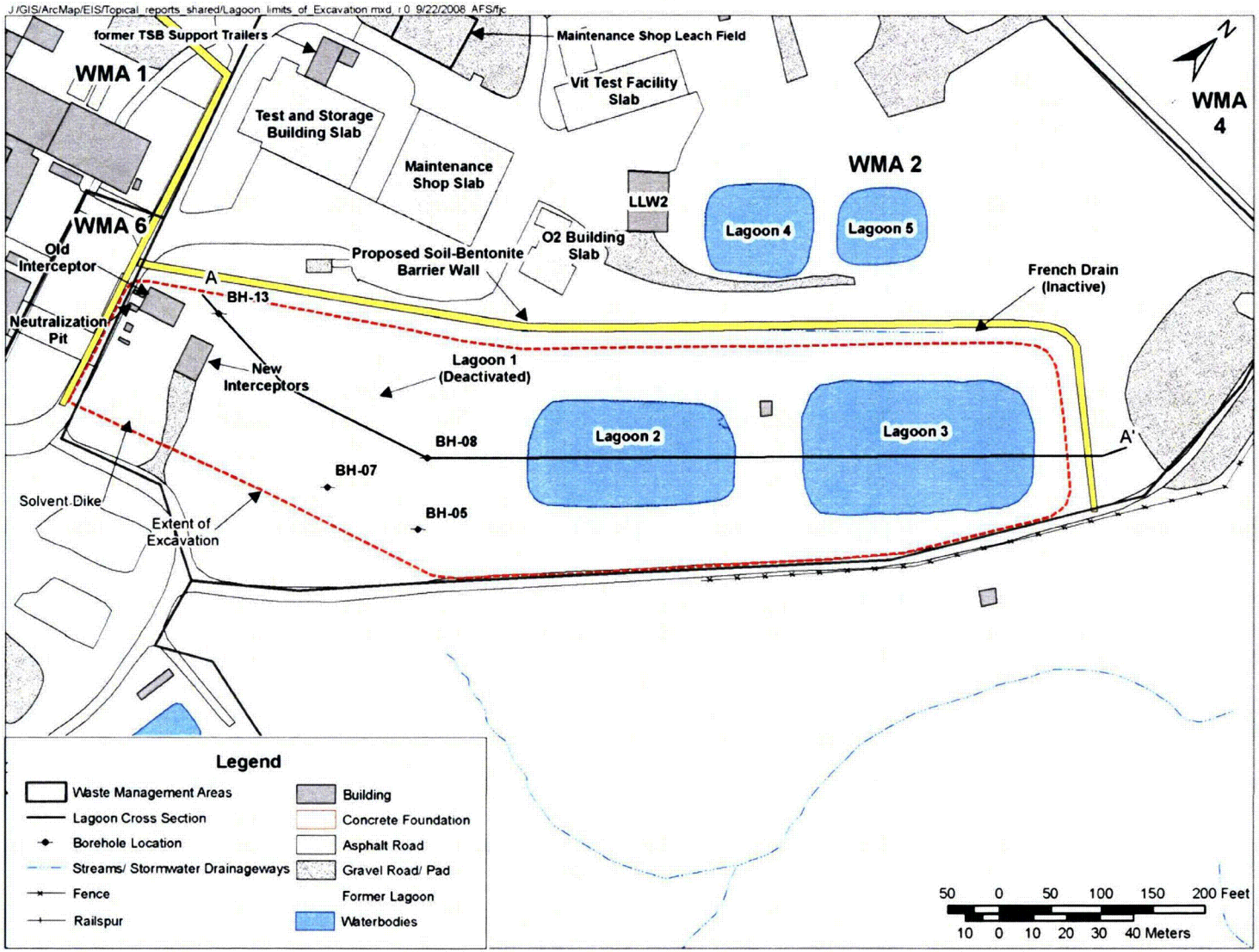

Figure D-3. Plan View of the WMA 2 Excavation

The upper three feet of the barrier wall would be constructed of clean backfill similar to the surrounding sand and gravel unit. This material would allow vehicular traffic over the barrier wall without damaging the underlying barrier wall.

1.2.3 Durability of Engineered Barriers

Refer to Section 1.1.3 of this Appendix for a discussion on the assumed durability of the soil-bentonite slurry wall installed at WMA 2.

1.2.4 Engineered Barriers and Groundwater Flow

Groundwater flow in the sand and gravel unit is currently to the northeast across the north plateau through WMA 1 and parallel to WMA 2 (Figure D-2). The permanent hydraulic barrier wall installed on the northwest side of the WMA 2 excavation nearly parallels the current groundwater flow path in the sand and gravel unit in the north plateau.

Groundwater modeling suggests groundwater flow patterns in the non-source area of the north plateau groundwater plume north of the WMA 2 barrier wall are similar to current flow patterns in the sand and gravel unit (Figure D-2). However, the overall hydraulic gradient of the non-source area of the north plateau plume is shallower than the current gradient due to the reduction of groundwater flow contribution attributed to the WMA 1 barrier wall system.

Groundwater modeling suggests the potential for higher groundwater levels within the backfilled WMA 2 excavation and the potential for groundwater flow from the excavation towards Erdman Brook and across the WMA 2 barrier wall towards the non-source area of the North Plateau Plume. The modeled groundwater levels in the backfilled WMA 2 excavation reflect contributions of groundwater flow from the WMA 1 excavation around the southeast end of the WMA 1 barrier wall.

2.0 Post-Remediation Site Monitoring and Maintenance

This section describes the post-remediation site monitoring and maintenance program to be implemented by the DOE at the project premises following the completion of Phase 1 of the proposed decommissioning. The program would include monitoring and maintenance associated with engineered barriers installed within the project premises and monitoring of environmental media within and outside the project premises. This monitoring and maintenance program would continue until the start of Phase 2 of the decommissioning, when the program requirements would be re-evaluated. DOE concludes that this program would be adequate to control and maintain the project premises because it is similar to the successful program currently in use and because it appropriately addresses all facilities of importance.

2.1 Monitoring and Maintenance of Engineered Barriers and Systems

The performance of the engineered barriers installed at WMA 1 and WMA 2 during Phase 1 proposed decommissioning would be routinely monitored up to the start of Phase 2 of the decommissioning to ensure they function as designed. Systems and engineered barriers installed during work leading to the interim end state, such the Tank and Vault Drying System at WMA 3 and the geomembrane cover and slurry wall at WMA 7, would also be routinely monitored and maintained as part of the DOE monitoring and maintenance program. Corrective actions would be implemented to correct any observed defects or irregularities with these engineered barrier and systems.

2.1.1 North Plateau Subsurface Barrier Walls and French Drain

The monitoring and maintenance program would monitor the performance and condition of the subsurface hydraulic barriers installed at WMA 1 and WMA 2, and the French drain at WMA 1. This program would include routine inspections of these systems for signs of degradation or loss of performance.

Hydraulic Barrier Walls

Piezometers would be installed upgradient and downgradient of the permanent hydraulic barrier walls installed downgradient of the WMA 1 and northwest of the WMA 2 excavations (Figure D-4). These piezometers would be spaced at intervals at least equal to the maximum lateral spacing recommended by the U.S. Environmental Protection Agency (EPA 1998). Water levels in these piezometers would be routinely monitored to evaluate the performance of these hydraulic barriers. Groundwater would be sampled and analyzed semi-annually for the radiological indicator parameters (gross alpha, gross beta, tritium) and for Sr-90 to evaluate the effectiveness of the barrier walls in preventing recontamination of WMA 1 and WMA 2.

If groundwater monitoring suggests repairs to the walls are required, these repairs would be accomplished through grouting, consistent with past industry experience and practice (e.g., EPA 1998).

French Drain

Monitoring and maintenance activities associated with the French drain installed upgradient of the WMA 1 hydraulic barrier wall would include monitoring of groundwater levels in piezometers installed on the upgradient and downgradient sides of the French drain following installation.

The need for and extent of repairs to the French drain, if any, would be determined based on analysis of the groundwater level data, which would be evaluated to identify evidence for any localized defect(s) in the French drain.

2.1.2 Waste Tank Farm Tank and Vault Drying System

The Tank and Vault Drying System installed in WMA 3 during the work to establish the interim end state would be routinely monitored and maintained during the Phase 1 period to ensure its continued operation as designed. The major components of the system – such as the blowers, heaters, and dehumidifier units – would be inspected and repaired or replaced as necessary to ensure continued operation of the system.

2.1.3 Waste Tank Farm Dewatering Well

As specified in Section 7 of this plan, the existing dewatering well would continue to be used to artificially lower the water table to minimize in-leakage of groundwater into the tank vaults. The water from this well would be collected, sampled, treated if necessary using a portable wastewater treatment system, and released to Erdman Brook through a State Pollutant Discharge Elimination System-permitted outfall.

2.1.4 NRC-licensed Disposal Area Engineered Barriers

The geomembrane cover and the hydraulic barrier wall installed at the NDA during work to establish the interim end state would be routinely monitored and maintained throughout Phase

Geomembrane Cover

The geomembrane cover would be routinely inspected for signs of deterioration or damage to the membrane. The seams connecting the geomembrane panels would be inspected to evaluate their condition. The geomembrane cover would be repaired to remedy any defects or irregularities identified during these inspections.

Hydraulic Barrier Wall

A monitoring and maintenance program similar to that described for the barrier walls installed at WMA 1 and WMA 2 would be implemented for the hydraulic barrier wall installed upgradient of the NDA. Twenty-one piezometers were installed upgradient and downgradient of the barrier wall during its construction. Water levels in these piezometers would be routinely monitored during Phase 1 to evaluate the performance of the barrier wall in limiting groundwater flow into the NDA.

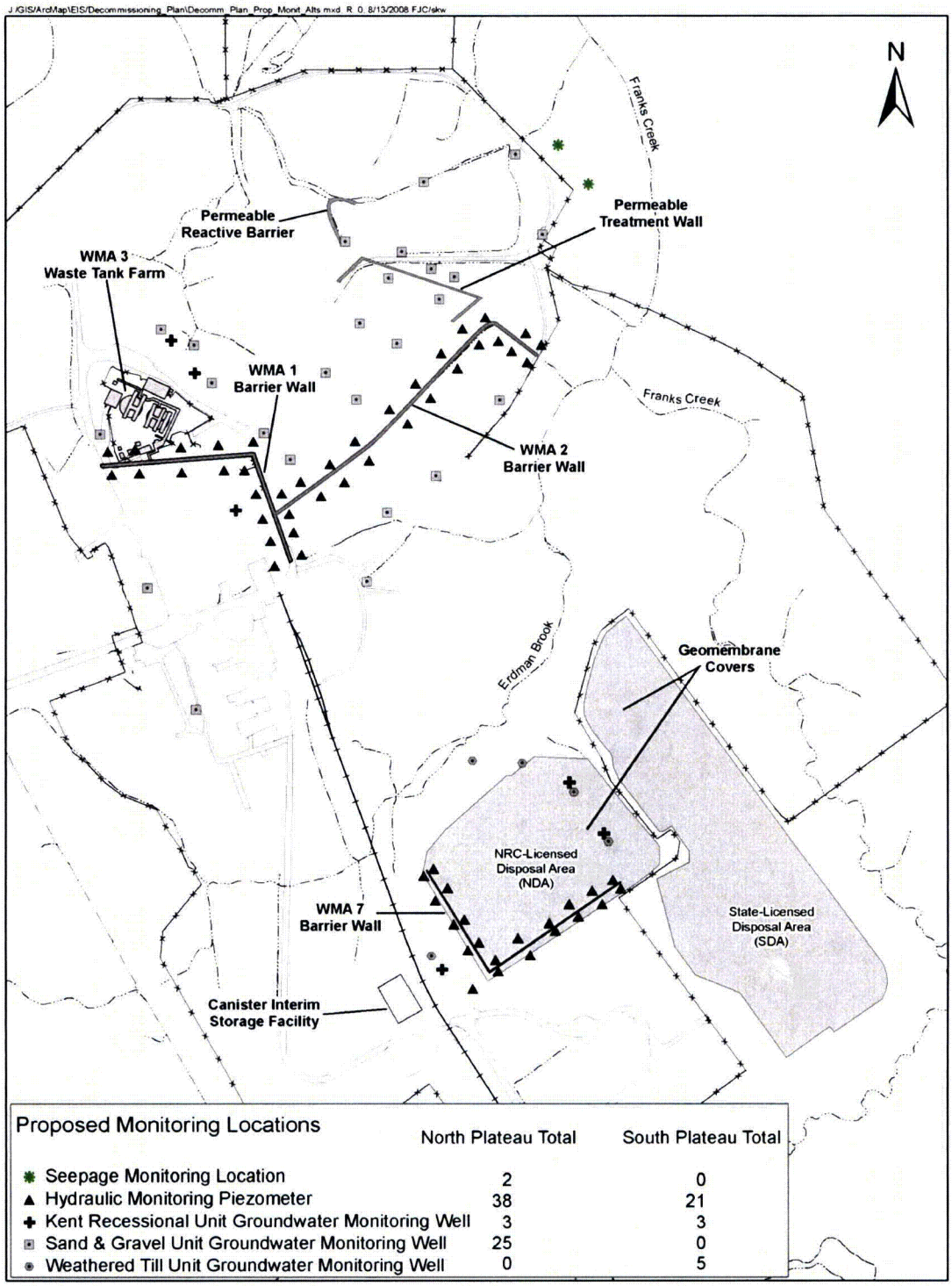

Figure D-4. Groundwater Monitoring Locations within the Project Premises during the Phase 1 Institutional Control Period

2.1.4 Security Features

The features important to security on the project premises and to security of the new Canister Interim Storage Facility during the period before Phase 2 of the decommissioning would be periodically inspected and maintained in good repair. These features include the security fences, signs, and security lighting described in Section 3.2 of this appendix.

2.2 Environmental Monitoring

The Phase 1 proposed decommissioning activities would include the removal of the following facilities:

- Above-ground and below-grade facilities in WMA 1 and the underlying source area of the north plateau groundwater plume within a single excavation down into the underlying Lavery till;
- Lagoons 1, 2, and 3, the Neutralization Pit, Interceptors, Solvent Dike, and surrounding contaminated soils in WMA 2 within a single excavation down into the underlying Lavery till; and
- Most remaining facilities and concrete slabs down to a maximum depth of two feet.

The following facilities and contamination areas within the project premises would not be considered during Phase 1 of the proposed decommissioning but would be addressed during Phase 2:

- The Waste Tank Farm in WMA 3, including the Permanent Ventilation System Building and the Supernatant Treatment System Support Building;
- The Construction Demolition Debris Landfill in WMA 4;
- The NDA in WMA 7; and
- The non-source area of the north plateau groundwater plume.

The DOE would implement an environmental monitoring program to monitor closed and remaining facilities and the non-source area of the north plateau groundwater plume as part of its management of the project premises during the Phase 1 institutional control period. Environmental monitoring would include onsite groundwater, storm water, and air monitoring, and both onsite and offsite surface water, sediment, and radiation monitoring as described below. Annual reports would be issued summarizing the monitoring results. These reports would include analyses of the data collected, along with conclusions about trends and compliance with regulatory limits.

2.2.1 Groundwater Monitoring Within the Project Premises

Groundwater within the project premises would be monitored during the Phase 1 institutional control period in accordance with the DOE WVDP Groundwater Monitoring Plan in effect at the time. Offsite groundwater monitoring would not be performed as this monitoring program was discontinued in 2007. The onsite grounding monitoring program for the project premises is described below and shown on Figure D-4. A total of 36 groundwater wells would be routinely monitored along with 59 piezometers.

WMA 1 - Process Building and Vitrification Facility Area

Groundwater in the sand and gravel unit in the backfilled WMA 1 excavation would be

monitored using the network of piezometers installed to monitor the effectiveness of the hydraulic barrier wall and French drain described in Section 2.1.1 of this Appendix. A monitoring well screened in the sand and gravel unit would also be installed in the upgradient portion of the WMA 1 excavation to provide information on groundwater quality flowing into the backfilled excavation.

An additional monitoring well screened in the Kent Recessional Sequence would be installed immediately upgradient of the WMA 1 hydraulic barrier wall to monitor groundwater in this unit and to evaluate potential migration of groundwater from the source area of the north plateau groundwater plume that was removed during Phase 1 of the proposed decommissioning.

Groundwater from these piezometers and monitoring wells would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium) and for Sr-90 during the Phase 1 institutional control period.

WMA 2 - Low-Level Waste Treatment Facility Area

Groundwater in the sand and gravel unit in the backfilled WMA 2 excavation would be monitored using the network of piezometers installed to monitor the effectiveness of the hydraulic barrier wall and French drain described in Section 2.1.1 of this Appendix. Three monitoring wells screened in the sand and gravel unit would also be installed on the southeastern boundary of the WMA 2 excavation to provide information on groundwater flow and quality in this area.

Groundwater from these piezometers and monitoring wells would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium) and for Sr-90 during the Phase 1 institutional control period.

WMA 3 - Waste Tank Farm Area

Groundwater in the sand and gravel unit and the Kent Recessional Sequence would be routinely monitored at WMA 3 during the Phase 1 institutional control period. Four wells would be screened in the sand and gravel unit with one well upgradient and three wells downgradient of the Waste Tank Farm. Two wells screened in the Kent Recessional Sequence would be installed downgradient of the Waste Tank Farm.

Groundwater from these wells would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium) and for Sr-90 during the Phase 1 institutional control period.

WMA 4 - Construction Demolition Debris Landfill Area

Groundwater in the sand and gravel unit at WMA 4 would be routinely monitored at six locations, including four monitoring wells around the Construction and Demolition Debris Landfill, and at two groundwater seep locations along the edge of the north plateau outside of the WVDP fence line.

Groundwater at WMA 4 would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium) and for Sr-90.

WMA 6 - Central Project Premises

Groundwater in the sand and gravel unit at WMA 6 would be routinely monitored at two well

locations, including one well upgradient of the rail spur and the other well downgradient of the rail spur and the removed Demineralizer Sludge Ponds and Equalization Basin.

Groundwater at these locations would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium).

WMA 7 – NDA

Groundwater in the weathered Lavery till and Kent recessional unit at WMA 7 would be routinely monitored by five wells screened in the weathered Lavery till and three wells screened in the Kent Recessional Sequence. One well cluster would be located upgradient of the NDA and would include a well screened in the weathered Lavery till and one screened in the Kent Recessional Sequence. Two well clusters, each with a well screened in the weathered Lavery till and Kent Recessional Sequence, would be located downgradient of the burial area. The two remaining wells screened in the weathered Lavery till would be located downgradient of the burial area.

Groundwater at WMA 7 would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium) and annually for specific radionuclides (Cs-137, Sr-90, Am-241, and Pu isotopes).

Non-Source Area of the North Plateau Plume

Groundwater in the sand and gravel unit would be routinely monitored at 11 well locations within the non-source area of the north plateau groundwater plume. These wells are located along the length of the plume from the WMA 1 barrier wall to the Construction and Demolition Debris Landfill in WMA 4. Three wells are located downgradient of the Permeable Treatment Wall to evaluate its effectiveness in reducing Sr-90 concentrations in groundwater from the sand and gravel unit.

Groundwater in the non-source area of the north plateau groundwater plume would be sampled semiannually for radiological indicator parameters (gross alpha, gross beta, and tritium) and for Sr-90.

2.2.2 Surface Water, Sediment, and Storm Water Monitoring

Surface water and associated stream sediments would be routinely monitored both within and outside the project premises during the Phase 1 institutional control period. The proposed monitoring locations are currently part of the DOE WVDP annual environmental monitoring program. These locations have been uniquely sited to monitor surface water releases from the WVDP and the Center. Several of the locations have been actively monitored since the implementation of the program in 1982 providing a significant historical record of surface waters leaving the WVDP and the Center.

Eight surface water-sampling locations within the project premises would be routinely monitored during the Phase 1 institutional control period (Figure D-5). These locations monitor streams both within (WNDNKEL, WNSP005, WNNDADR, WNFRC67, WNERB53) and leaving the project premises (WNSW74A, WNSWAMP, and WNSP006). Sediment samples would be collected from three locations where surface waters leave the project premises (SNSW74A, SNSWAMP, and SNSP006).

Surface water would be routinely collected and analyzed from three sampling locations

outside of the project premises (Figure D-6). These locations would monitor surface water quality in Buttermilk Creek and Cattaraugus Creek where these streams leave the Center (WFFELBR, WFBCTCB) and where Buttermilk Creek enters the Center (WFBCBKG). Sediment samples would be collected from all three off-site locations (SFBCSED, SFTCSED, SFCCSED).

Surface water and sediment samples would be collected from these locations semi-annually and would be analyzed for radiological indicator parameters (gross alpha, gross beta, and tritium).

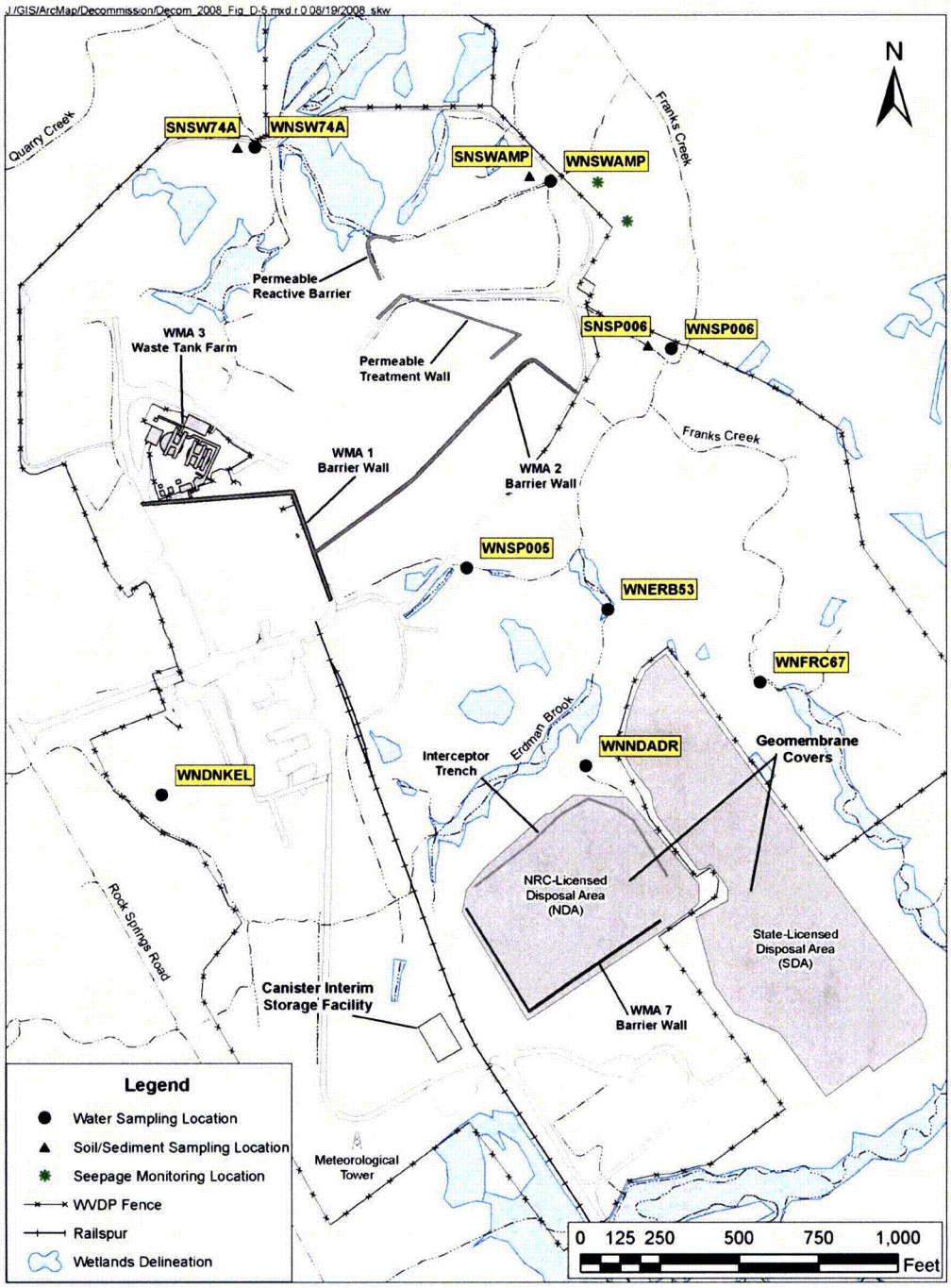

Figure D-5. Surface Water and Sediment Sampling Locations on the Project Premises during the Phase 1 Institutional Control Period

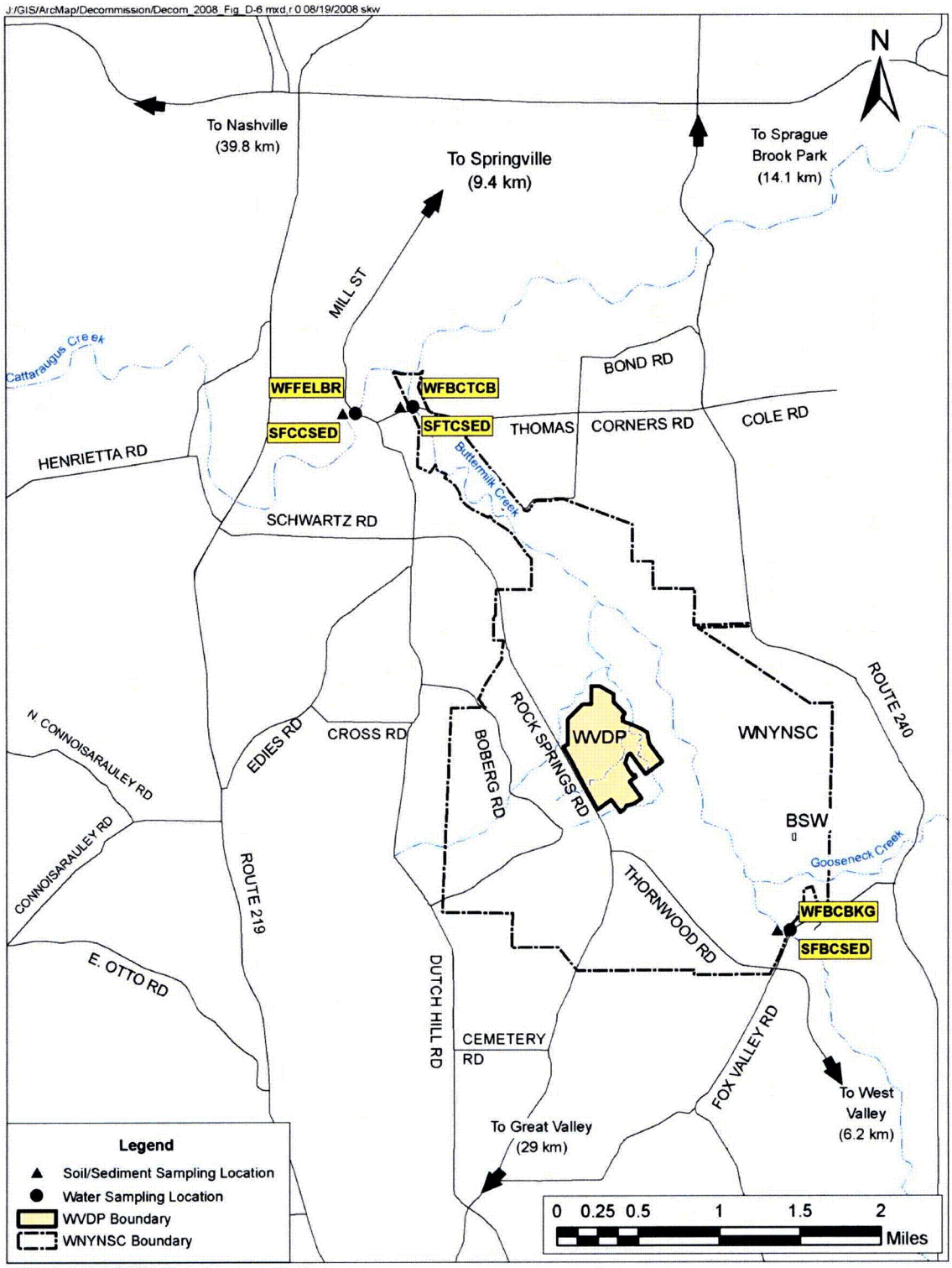

Figure D-6 – Offsite Surface Water and Sediment Sampling Locations during the Phase 1 Institutional Control Period

The New York State Pollutant Discharge Elimination System permit issued to the DOE WVDP requires periodic sampling from storm water outfalls located within the project premises. Sampling from these outfalls during storm events is designed to assess specific chemicals in storm water discharges that may originate from industrial or construction activity runoff from locations within the project premises. The planned storm water sampling locations are identified on Figure D-7. Sampling would be performed semi-annually for the non-radiological parameters specified in the New York State Pollutant Discharge Elimination System permit.

2.2.3 Air Monitoring

The stack discharge from the Permanent Ventilation System Building in the Waste Tank Farm in WMA 3 would be the only air monitoring location to be routinely monitored within and outside of the project premises during the Phase 1 institutional control period (Figure D-8).

The Permanent Ventilation System ventilates the Supernatant Treatment System Valve Aisle and Tanks 8D-1, 8D-2, 8D-3, and 8D-4 in WMA 3. The air discharged from these facilities passes though high-efficiency particulate air filters before discharge through the Permanent Ventilation System Building stack. Air discharged from the Tank and Vault Drying System would also be treated in the Permanent Ventilation System Building.

Air discharges from this location would be analyzed for radiological indicator parameters (gross alpha, gross beta, and tritium) and specific radionuclides (Cs-137, Sr-90, I-129, Am-241, and U and Pu isotopes).

2.2.4 Direct Radiation Monitoring

Direct radiation monitoring using thermoluminescent dosimeters would be performed at 19 locations within and outside of the project premises. These monitoring locations are currently part of the DOE WVDP annual environmental monitoring program and were sited to monitor both on-site and off-site radiation exposure from facilities within the project premises and the State-Licensed Disposal Area. Several of these locations have been actively monitored since 1982.

Eight monitoring locations would be within the project premises (Figure D-9) and eleven stations would be located on the perimeter of the Center (Figure D-10). All locations would be routinely monitored for gamma radiation exposure on a quarterly monitoring schedule.

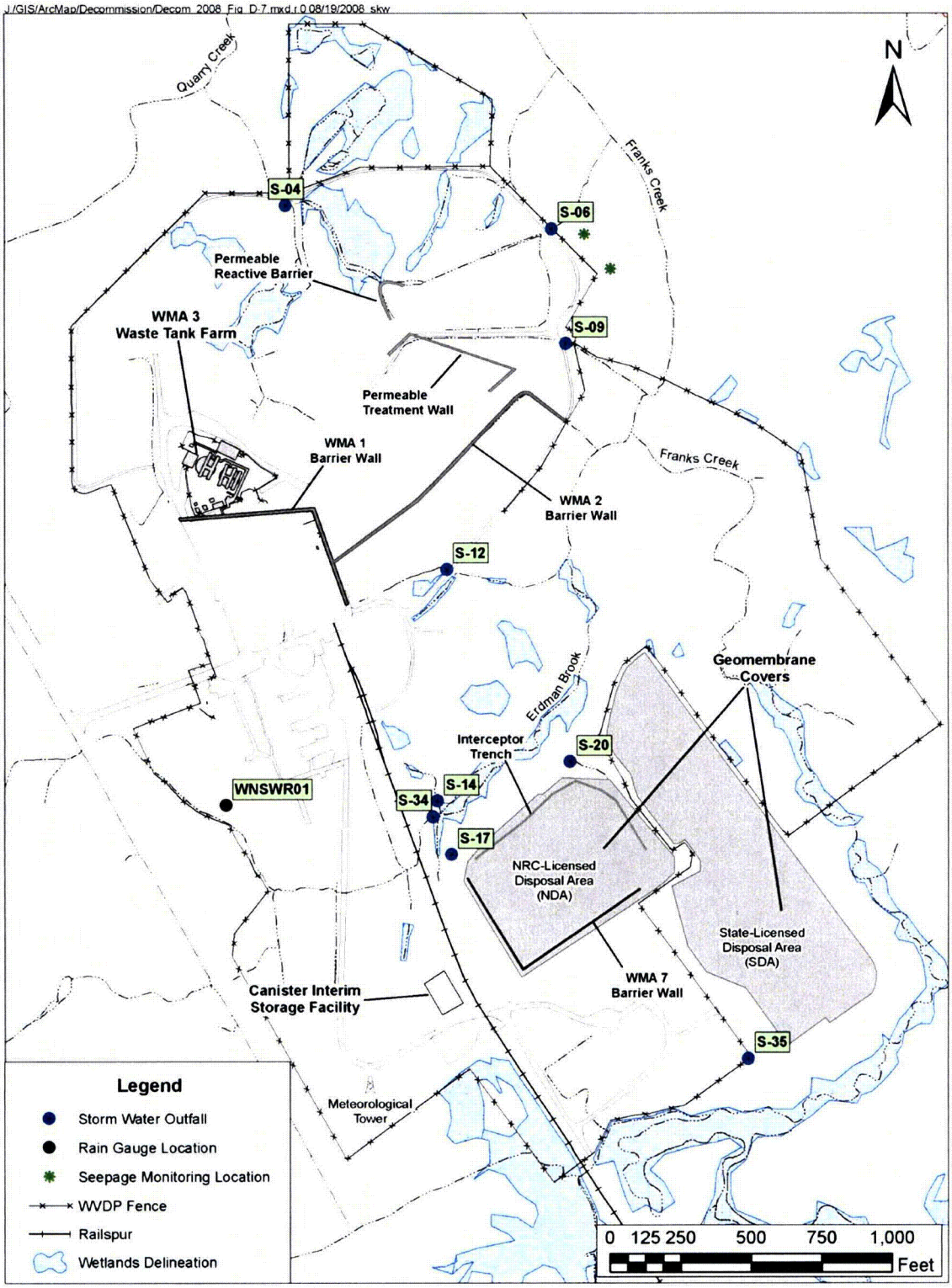

Figure D-7. Storm Water Sampling Locations on the Project Premises during the Phase 1 Institutional Control Period

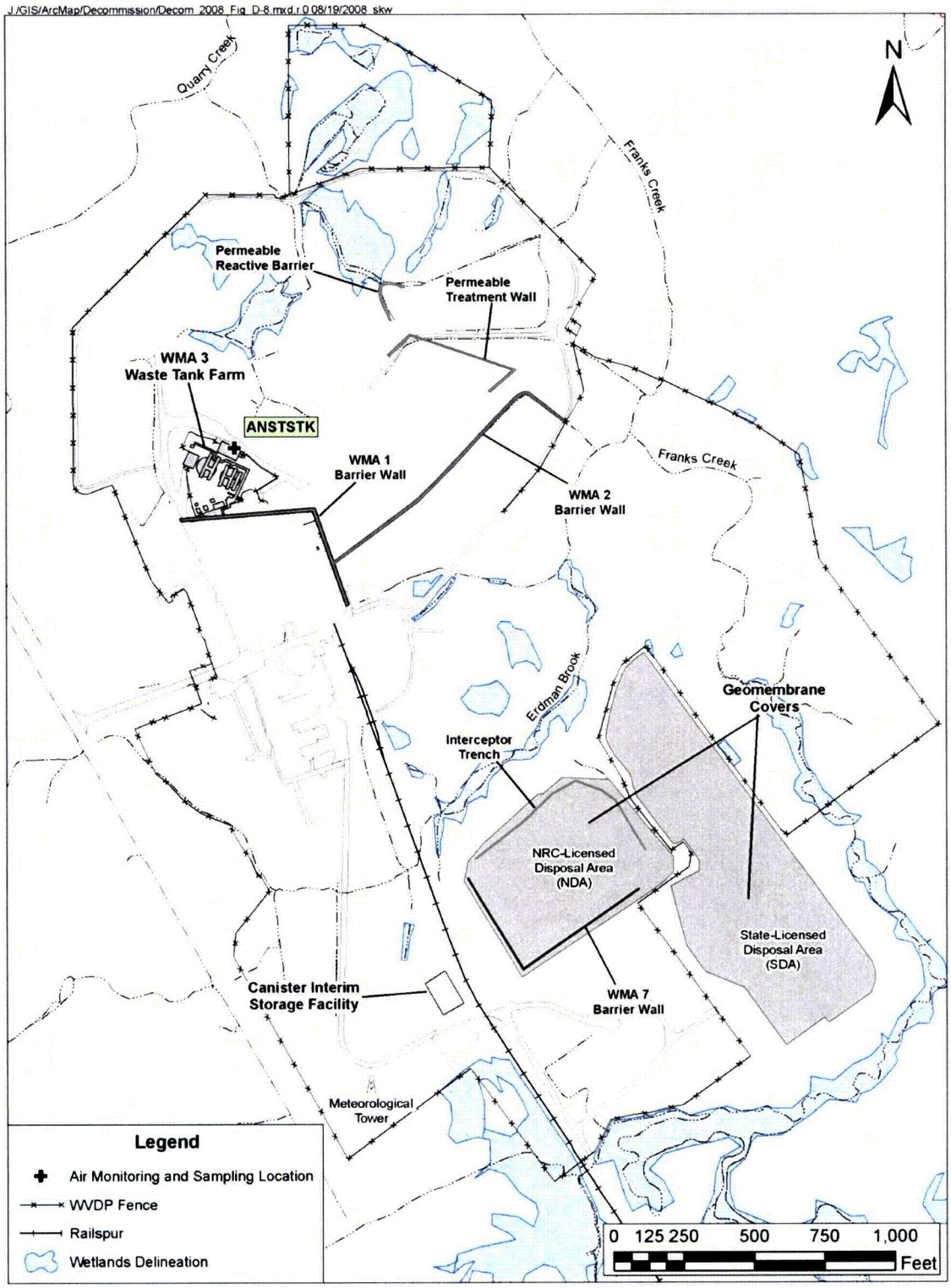

Figure D-8. Air Monitoring Locations on the Project Premises during the Phase 1 Institutional Control Period

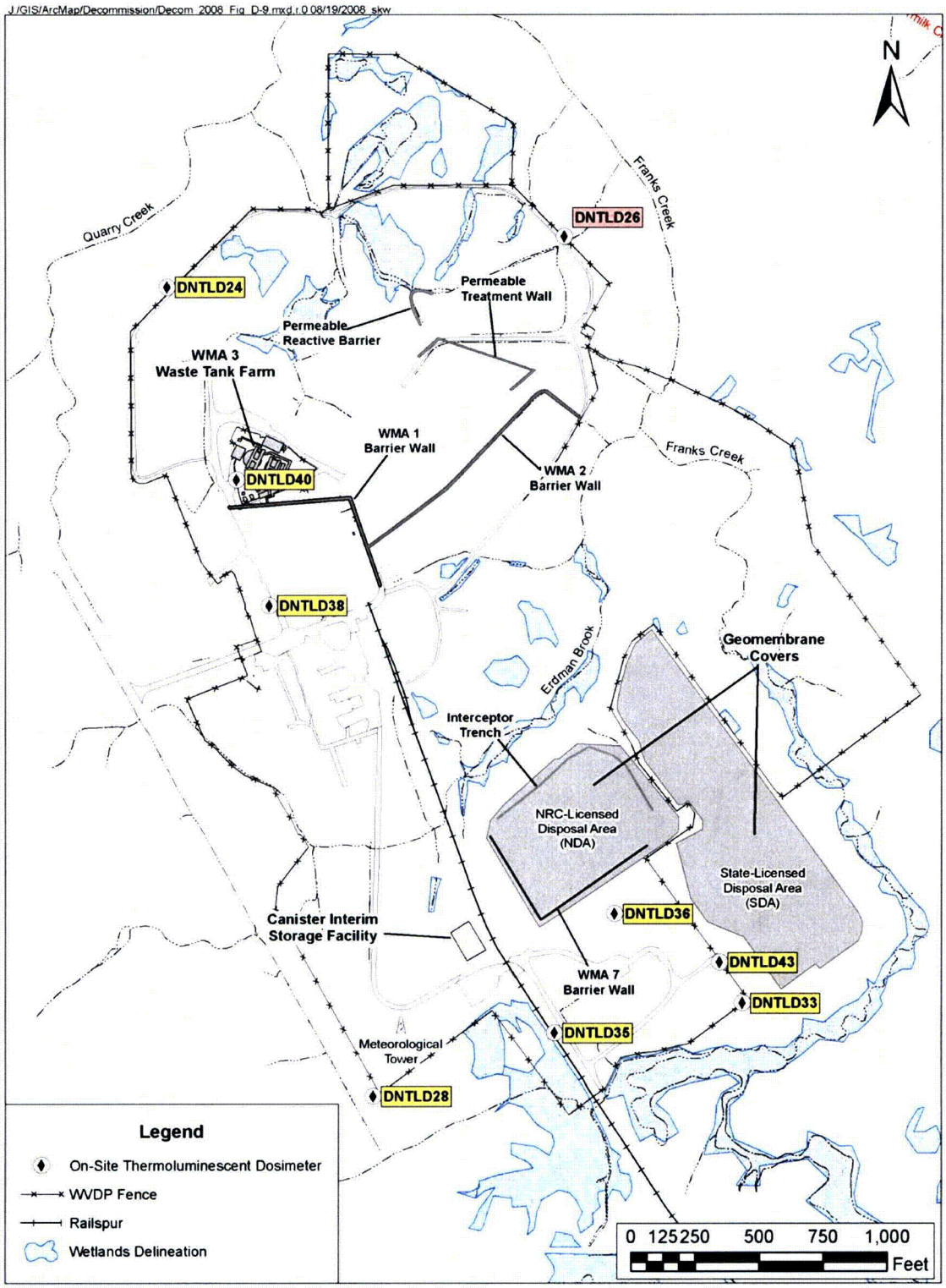

Figure D-9 – Direct Radiation Monitoring Locations on the Project Premises during the Phase 1 Institutional Control Period

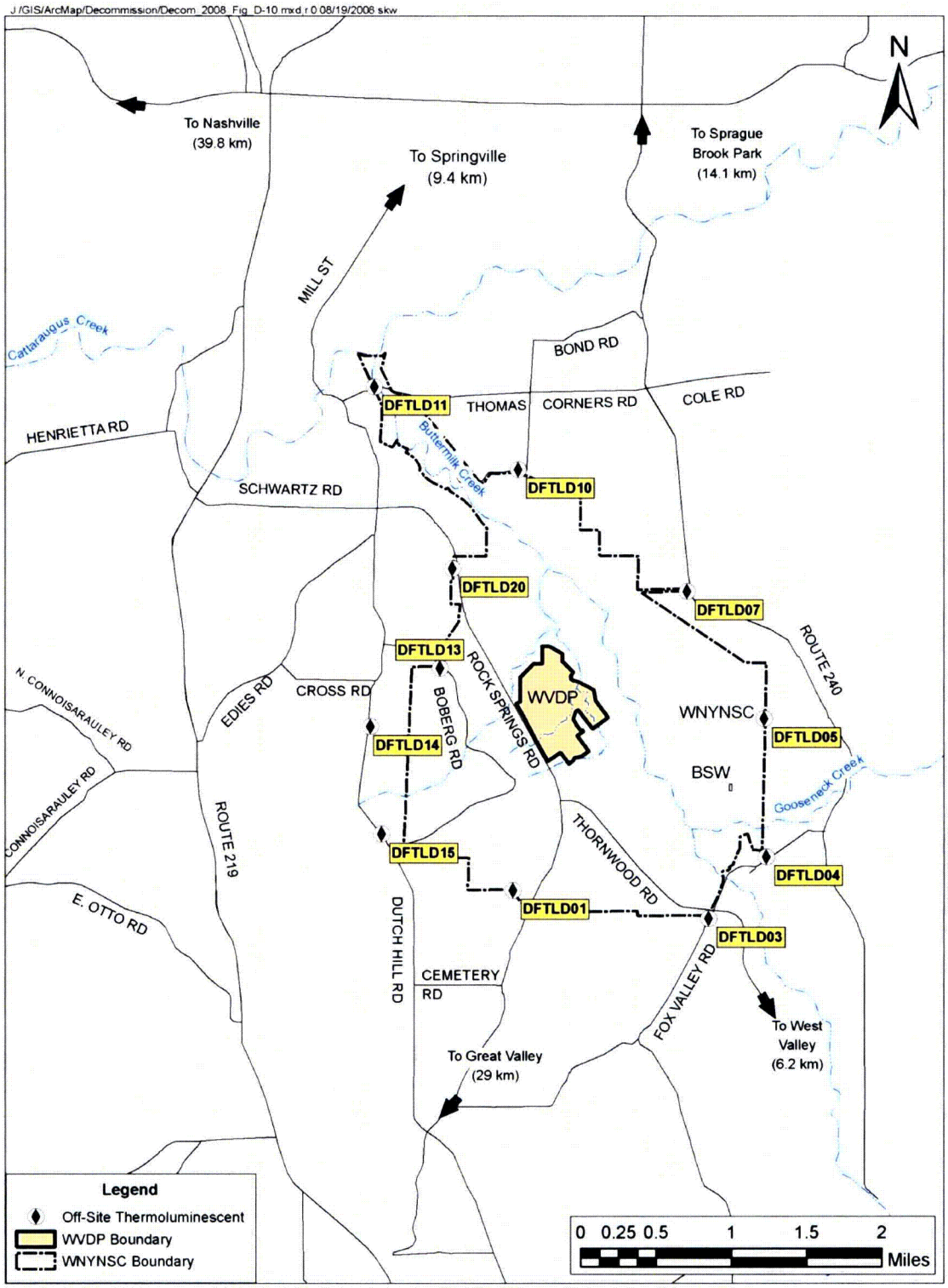

Figure D-10. Offsite Direct Radiation Monitoring Locations during the Phase 1 Institutional Control Period

3.0 Phase 1 Institutional Control Program

This section describes the institutional control program that would be implemented for the project premises following the completion of the Phase 1 remedial activities.

3.1 Government Control of the Project Premises

NYSERDA is the current owner of the project premises property and would remain owner following Phase 1 activities. As stipulated in the Cooperative Agreement with NYSERDA, DOE shall remain in exclusive use and possession of the project premises and project facilities throughout the remainder of the project term (DOE and NYSERDA 1981). DOE would therefore continue control of the project premises during the implementation of the Phase 1 proposed decommissioning activities and during the Phase 1 institutional control period. In this capacity, DOE carries the full authority of the federal government in enforcing institutional controls over the project premises.

DOE would be responsible for operating and maintaining facilities within the project premises such as the Waste Tank Farm, the NDA, and the non-source area of the north plateau groundwater plume in a safe manner. DOE would continue to implement the environmental radiation protection program for the project premises as required by DOE Order 5400.5, *Radiation Protection of the Public and the Environment*. NRC would also be involved in a regulatory oversight capacity over the project premises, which would remain under NRC license.

3.2 Institutional Control Design Features

The institutional control program for the project premises would prevent its unacceptable use and protect against inadvertent intrusion into the site. DOE in its capacity as the steward of the site would ensure that institutional controls are maintained at the project premises during Phase 1 proposed decommissioning and during the Phase 1 institutional control period. These institutional controls would include:

- Security fencing and signage along the perimeter of the project premises to prevent inadvertent intrusion into the site and to notify individuals that access is forbidden without permission from the DOE,
- A full time security force to prevent unauthorized access into the project premises,
- Authorized personnel and vehicle access into the project premises would be limited to designated gateways through the perimeter security fence
- The environmental monitoring program implemented at the project premises during the Phase 1 institutional control period would ensure that operations at the site protect members of the public and the environment from radiation risk.

Additional institutional controls would be provided for the new Canister Interim Storage Facility on the south plateau. These would include measures such as security fencing around the area and appropriate security lighting.

4.0 References

Code of Federal Regulations and Federal Register Notices

10 CFR 20 Subpart E, *Radiological Criteria for License Termination.*

67 FR 22, *Decommissioning Criteria for the West valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement*, U.S. Nuclear Regulatory Commission, Washington, D.C., February 1, 2002.

DOE Orders

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Other References

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URS 2000, *Evaluation of Stability of Proposed WMA1&3/WMA7 Slurry Walls Under Hypothetical Seismically-induced Horizontal Acceleration of 0.2g*, Calculation BUF-2000-069, Rev. 0. URS Corp., Orchard Park, New York, July 17, 2000.