U.S. Department of Energy

West Valley Demonstration Project

# Phase 1 Decommissioning Plan for the West Valley Demonstration Project

Revision 2

December 2009

Prepared by

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*The proposed decommissioning approach described in this plan is based on the preferred alternative in the Revised Draft Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center, which is referred to as the Decommissioning EIS. If changes to that document occur during the course of the National Environmental Policy Act process that affect this plan, such as changes to the preferred alternative, or if a different approach is selected in the Record of Decision, this plan will be revised as necessary to reflect the changes.*

*Note that many of the comments received during the public comment period for the Revised Draft Decommissioning EIS stated that the 30-year time period for making the decision on the approach to Phase 2 of the decommissioning was too long. Therefore, as the agencies consider the public comments, DOE is evaluating the potential to reduce this time period. In recognition of this potential change, this plan acknowledges that the Phase 2 decision could be made within 10 years from the issuance of the Record of Decision and Findings Statement if the Phased Decisionmaking Alternative is selected.*

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# Record of Revisions

| No. | Date | Purpose |
|---|---|---|
| 0 | December 2008 | Initial issue for U.S. Nuclear Regulatory Commission review. |
| 1 | March 2009 | (1) Corrected some page numbers in the Contents.<br>(2) Changed the preliminary, order-of-magnitude dose estimate for Waste Management Area 2 on page ES-19 from approximately 0.1 to approximately 0.05 millirem per year.<br>(3) Added report USGS 2007 in the Section 3 reference list.<br>(4) Incorporated radiological data on subsurface soil from the 2008 background and Process Building area north plateau groundwater plume investigations into Section 4.<br>(5) Replaced Figure 5-3 with a modified figure to more accurately show the Lavery till depth where samples were taken.<br>(6) Revised Table 5-1 to reflect the 2008 radiological data.<br>(7) Corrected values in Table 5-10 and Table 5-11 to be consistent with Table C-1.<br>(8) Revised Section 5.4.4 to show a maximum of 1.3 millirem per year for Waste Management Area 1 and 0.04 mrem per year for Waste management Area 2, clarified basis for estimates.<br>(9) Corrected French drain location on Figure 7-10.<br>(10) Corrected soil data reference on Figure 7-11 and modified the figure to more accurately show the Lavery till depth where samples were taken. (This figure is the same as Figure 5-3.)<br>(11) Added WVNSCO 2004 to Section 7 references.<br>(12) Corrected some values in Table 9-1, 9-2, and 9-3.<br>(13) Changed cited page numbers on pages A-12 and A-13.<br>(14) Incorporated radiological data on subsurface soil from the 2008 background and Process Building area north plateau groundwater plume investigations into Appendix B.<br>(15) Revised Table C-4 to add the 2008 data and to clarify the content.<br>(16) Added Appendix C, Attachment 2 to provide another electronic file (Table C-4B Excel spreadsheet for the preliminary, order-of-magnitude dose estimates).<br>(17) Revised Appendix D to describe additional groundwater modeling using revised STOMP model. Corrected French drain location on Figure D-2.<br>Revision 1 changes appear in a blue font. Vertical lines used in the right margin to identify these changes were removed in Revision 2. |

# Record of Revisions

| No. | Date | Purpose |
|---|---|---|
| 2 | December 2009 | Revision 2 incorporates changes made in response to the Requests for Additional Information (RAIs) submitted by the U.S. Nuclear Regulatory Commission (NRC) on May 15, 2009 and includes other changes made in response to comments on the plan submitted by other agencies. DOE specifically identified the changes being incorporated in Revision 2 of the plan in connection with the RAIs in the RAI responses provided to NRC.<br><br>Revision 2 changes appear in a red font and are marked with vertical lines in the right margin with the following exceptions. Two types of prevalent changes are not so marked: changing “would” to “will” to make the plan appropriately prescriptive and deleting the word “proposed” for the same reason. The three appendices added in Revision 2 (E, F, and G) are not marked either because they are entirely new.<br><br>Because Revision 2 changes appear in all parts of the plan, each page is identified as Revision 2 to facilitate a complete reissue of the plan, although the contents of some pages are unchanged from Revision 1.<br><br>Changes of “would” to “will” and deleting the word “proposed” were made in each part of the plan. The following summary identifies other key changes.<br><br>**Executive Summary.** Added information on DOE onsite presence after Phase 1 and movement of vitrified high-level waste canisters. Provided for optional surface soil remediation during Phase 1 in selected areas. Updated derived concentrations guideline levels (DCGLs) and cleanup goals. Clarified text related to NRC review and underground waste tank status.<br><br>**Section 1.** Expanded information on Phase 1 studies, changed period for the Phase 2 decision from 30 years to the possibility that the decision could be made within 10 years, provided for NRC review of certain detailed designs, added Waste Management Plan, made several clarifications.<br><br>**Section 2.** Made minor changes to Table 2-5. Added information to Table 2-17. Made several clarifications.<br><br>**Section 3.** Updated information on groundwater modeling, geologic interpretation, maximum probable flood, underground waste tank status, the permeable reactive barrier, the Supernatant Treatment System, and historical earthquakes. Made several clarifications.<br><br>**Section 4.** Provided clarifying information on uranium radionuclide ratios, underground waste tank status. Made minor changes to Tables 4-9 and 4-10 for consistency with other similar tables and made several clarifications.<br><br>**Section 5.** Revised Table 5-1. Added information on analyses of alternate conceptual models. Renumbered some subsections. Added new Section 5.2.7 on probabilistic uncertainty analysis and new Section 5.2.8 on multi-source analysis. Revised cleanup goals based on results of these analyses. Added information on DOE presence after completion of Phase 1. Added new Figure 5-12 to define where streambed sediment cleanup goals apply. |

## Record of Revisions

| No. | Date | Purpose |
|---|---|---|
| 2 | December 2009 | **Section 6.** Added new Section 6.2 on ALARA good practices, new Section 6.3.6 on intergenerational concerns.<br><br>**Section 7.** Replaced Figures 7-6 and 7-8, added new figures showing filled deep excavations, revised conceptual schedule. Expanded information on mitigative measures. Added information on DOE presence after completion of Phase 1. Provided for optional surface soil remediation during Phase 1 in selected areas. Made other clarifying changes.<br><br>**Section 8.** Clarified acceptance criteria. Updated some references.<br><br>**Section 9.** Added new Section 9.4.4 on applying data quality objectives. Added new tables, other information on scan surveys. Expanded information on in-process surveys. Revised text for consistency with contents of Characterization Sample and Analysis Plan and Final Status Survey Plan, deleting some not-applicable information and rearranging other information. Made reference to new Appendix F and Appendix G.<br><br>**Appendix A.** Changed page numbers to reflect Revision 2 changes.<br><br>**Appendix B.** Only changed “would” to “will” and omitted “proposed.”<br><br>**Appendix C.** Made changes to reflect minor changes in the deterministic conceptual models.<br><br>**Appendix D.** Added information on hydraulic barrier wall design. Provided for providing final designs to NRC for review. Added information on DOE monitoring, maintenance, and security after Phase 1. Incorporated changes in groundwater model.<br><br>**Appendix E.** Added new appendix on probabilistic uncertainty analysis details.<br><br>**Appendix F.** Added new appendix on details of subsurface piping and the associated residual radioactivity.<br><br>**Appendix G.** Added new appendix on the conceptual framework for the Final Status Survey Plan. |

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## List of Effective Pages

| Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| i | 2 | 2-43 | 2 | 3-111 | 2 | 5-1 | 2 | 7-40 | 2 | A-38 | 2 | D-16 | 2 |
| ii | 2 | 2-44 | 2 | 3-112 | 2 | 5-2 | 2 | 7-41 | 2 | A-39 | 2 | D-17 | 2 |
| iii | 2 | 2-45 | 2 | 3-113 | 2 | 5-3 | 2 | 7-42 | 2 | A-40 | 2 | D-18 | 2 |
| iv | 2 | 2-46 | 2 | 3-114 | 2 | 5-4 | 2 | 7-43 | 2 | B-1 | 2 | D-19 | 2 |
| v | 2 | 2-47 | 2 | 3-115 | 2 | 5-5 | 2 | 7-44 | 2 | B-2 | 2 | D-20 | 2 |
| vi | 2 | 2-48 | 2 | 3-116 | 2 | 5-6 | 2 | 7-45 | 2 | B-3 | 2 | D-21 | 2 |
| vii | 2 | 2-49 | 2 | 3-117 | 2 | 5-7 | 2 | 7-46 | 2 | B-4 | 2 | D-22 | 2 |
| viii | 2 | 2-50 | 2 | 3-118 | 2 | 5-8 | 2 | 7-47 | 2 | B-5 | 2 | D-23 | 2 |
| ix | 2 | 2-51 | 2 | 3-119 | 2 | 5-9 | 2 | 7-48 | 2 | B-6 | 2 | D-24 | 2 |
| x | 2 | 2-52 | 2 | 3-120 | 2 | 5-10 | 2 | 7-49 | 2 | B-7 | 2 | D-25 | 2 |
| xi | 2 | 3-1 | 2 | 3-121 | 2 | 5-11 | 2 | 7-50 | 2 | B-8 | 2 | D-26 | 2 |
| xii | 2 | 3-2 | 2 | 3-122 | 2 | 5-12 | 2 | 7-50 | 2 | B-9 | 2 | D-27 | 2 |
| xiii | 2 | 3-3 | 2 | 3-123 | 2 | 5-13 | 2 | 7-51 | 2 | B-10 | 2 | D-28 | 2 |
| xiv | 2 | 3-4 | 2 | 3-124 | 2 | 5-14 | 2 | 7-52 | 2 | B-11 | 2 | D-29 | 2 |
| xv | 2 | 3-5 | 2 | 3-125 | 2 | 5-15 | 2 | 7-53 | 2 | B-12 | 2 | D-30 | 2 |
| xvi | 2 | 3-6 | 2 | 3-126 | 2 | 5-16 | 2 | 7-54 | 2 | B-13 | 2 | D-32 | 2 |
| xvii | 2 | 3-7 | 2 | 3-127 | 2 | 5-17 | 2 | 7-55 | 2 | B-14 | 2 | D-32 | 2 |
| xviii | 2 | 3-8 | 2 | 3-128 | 2 | 5-18 | 2 | 7-56 | 2 | B-15 | 2 | D-33 | 2 |
| xix | 2 | 3-9 | 2 | 3-129 | 2 | 5-19 | 2 | 7-57 | 2 | B-16 | 2 | D-34 | 2 |
| xx | 2 | 3-10 | 2 | 3-130 | 2 | 5-20 | 2 | 7-58 | 2 | B-17 | 2 | E-1 | 2 |
| xxi | 2 | 3-11 | 2 | 3-131 | 2 | 5-21 | 2 | 8-1 | 2 | B-18 | 2 | E-2 | 2 |
| xxii | 2 | 3-12 | 2 | 3-132 | 2 | 5-22 | 2 | 8-2 | 2 | B-19 | 2 | E-3 | 2 |
| xxiii | 2 | 3-13 | 2 | 3-133 | 2 | 5-23 | 2 | 8-3 | 2 | B-20 | 2 | E-4 | 2 |
| xxiv | 2 | 3-14 | 2 | 3-134 | 2 | 5-24 | 2 | 8-4 | 2 | B-21 | 2 | E-5 | 2 |
| xxv | 2 | 3-15 | 2 | 3-135 | 2 | 5-25 | 2 | 8-5 | 2 | B-22 | 2 | E-6 | 2 |
| xxvi | 2 | 3-16 | 2 | 3-136 | 2 | 5-26 | 2 | 8-6 | 2 | B-23 | 2 | E-7 | 2 |
| xxvii | 2 | 3-17 | 2 | 3-137 | 2 | 5-27 | 2 | 8-7 | 2 | B-24 | 2 | E-8 | 2 |
| ES-1 | 2 | 3-18 | 2 | 3-138 | 2 | 5-28 | 2 | 8-8 | 2 | B-25 | 2 | E-9 | 2 |
| ES-2 | 2 | 3-19 | 2 | 3-139 | 2 | 5-29 | 2 | 8-9 | 2 | B-26 | 2 | E-10 | 2 |
| ES-3 | 2 | 3-20 | 2 | 3-140 | 2 | 5-30 | 2 | 8-10 | 2 | B-27 | 2 | E-11 | 2 |
| ES-4 | 2 | 3-21 | 2 | 3-141 | 2 | 5-31 | 2 | 8-11 | 2 | B-28 | 2 | E-12 | 2 |
| ES-5 | 2 | 3-22 | 2 | 3-142 | 2 | 5-32 | 2 | 8-12 | 2 | B-29 | 2 | E-13 | 2 |
| ES-6 | 2 | 3-23 | 2 | 3-143 | 2 | 5-33 | 2 | 8-13 | 2 | B-30 | 2 | E-14 | 2 |
| ES-7 | 2 | 3-24 | 2 | 3-144 | 2 | 5-34 | 2 | 8-14 | 2 | B-31 | 2 | E-15 | 2 |
| ES-8 | 2 | 3-25 | 2 | 3-145 | 2 | 5-35 | 2 | 8-15 | 2 | B-32 | 2 | E-16 | 2 |
| ES-19 | 2 | 3-26 | 2 | 3-146 | 2 | 5-36 | 2 | 9-1 | 2 | B-33 | 2 | E-17 | 2 |
| ES-10 | 2 | 3-27 | 2 | 3-147 | 2 | 5-37 | 2 | 9-2 | 2 | B-34 | 2 | E-18 | 2 |
| ES-11 | 2 | 3-28 | 2 | 3-148 | 2 | 5-38 | 2 | 9-3 | 2 | B-35 | 2 | E-19 | 2 |
| ES-12 | 2 | 3-29 | 2 | 3-149 | 2 | 5-39 | 2 | 9-4 | 2 | B-36 | 2 | E-20 | 2 |
| ES-13 | 2 | 3-30 | 2 | 3-150 | 2 | 5-40 | 2 | 9-5 | 2 | B-37 | 2 | E-21 | 2 |
| ES-14 | 2 | 3-31 | 2 | 3-151 | 2 | 5-41 | 2 | 9-6 | 2 | B-38 | 2 | E-22 | 2 |
| ES-15 | 2 | 3-32 | 2 | 3-152 | 2 | 5-42 | 2 | 9-7 | 2 | B-39 | 2 | E-23 | 2 |

## List of Effective Pages

| Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| ES-16 | 2 | 3-33 | 2 | 3-153 | 2 | 5-43 | 2 | 9-8 | 2 | B-40 | 2 | E-24 | 2 |
| ES-17 | 2 | 3-34 | 2 | 3-154 | 2 | 5-44 | 2 | 9-9 | 2 | B-41 | 2 | E-25 | 2 |
| ES-18 | 2 | 3-35 | 2 | 3-155 | 2 | 5-45 | 2 | 9-10 | 2 | B-42 | 2 | E-26 | 2 |
| ES-19 | 2 | 3-36 | 2 | 4-1 | 2 | 5-46 | 2 | 9-11 | 2 | B-43 | 2 | E-27 | 2 |
| ES-20 | 2 | 3-37 | 2 | 4-2 | 2 | 5-47 | 2 | 9-12 | 2 | B-44 | 2 | E-28 | 2 |
| ES-21 | 2 | 3-38 | 2 | 4-3 | 2 | 5-48 | 2 | 9-13 | 2 | B-45 | 2 | E-29 | 2 |
| ES-22 | 2 | 3-39 | 2 | 4-4 | 2 | 5-49 | 2 | 9-14 | 2 | B-46 | 2 | E-30 | 2 |
| 1-1 | 2 | 3-40 | 2 | 4-5 | 2 | 5-50 | 2 | 9-15 | 2 | B-47 | 2 | E-31 | 2 |
| 1-2 | 2 | 3-41 | 2 | 4-6 | 2 | 5-51 | 2 | 9-16 | 2 | B-48 | 2 | E-32 | 2 |
| 1-3 | 2 | 3-42 | 2 | 4-7 | 2 | 5-52 | 2 | 9-17 | 2 | B-49 | 2 | E-33 | 2 |
| 1-4 | 2 | 3-43 | 2 | 4-8 | 2 | 5-53 | 2 | 9-18 | 2 | B-50 | 2 | E-34 | 2 |
| 1-5 | 2 | 3-44 | 2 | 4-9 | 2 | 5-54 | 2 | 9-19 | 2 | B-51 | 2 | E-35 | 2 |
| 1-6 | 2 | 3-45 | 2 | 4-10 | 2 | 5-55 | 2 | 9-20 | 2 | B-52 | 2 | E-36 | 2 |
| 1-7 | 2 | 3-46 | 2 | 4-11 | 2 | 5-56 | 2 | 9-21 | 2 | B-53 | 2 | E-37 | 2 |
| 1-8 | 2 | 3-47 | 2 | 4-12 | 2 | 5-57 | 2 | 9-22 | 2 | B-54 | 2 | E-38 | 2 |
| 1-9 | 2 | 3-48 | 2 | 4-13 | 2 | 5-58 | 2 | 9-23 | 2 | C-1 | 2 | E-39 | 2 |
| 1-10 | 2 | 3-49 | 2 | 4-14 | 2 | 5-59 | 2 | 9-24 | 2 | C-2 | 2 | E-40 | 2 |
| 1-11 | 2 | 3-50 | 2 | 4-15 | 2 | 5-60 | 2 | 9-25 | 2 | C-3 | 2 | E-41 | 2 |
| 1-12 | 2 | 3-51 | 2 | 4-16 | 2 | 5-61 | 2 | 9-26 | 2 | C-4 | 2 | E-42 | 2 |
| 1-13 | 2 | 3-52 | 2 | 4-17 | 2 | 5-62 | 2 | 9-27 | 2 | C-5 | 2 | E-43 | 2 |
| 1-14 | 2 | 3-53 | 2 | 4-18 | 2 | 5-63 | 2 | 9-28 | 2 | C-6 | 2 | E-44 | 2 |
| 1-15 | 2 | 3-54 | 2 | 4-19 | 2 | 5-64 | 2 | 9-29 | 2 | C-7 | 2 | E-45 | 2 |
| 1-16 | 2 | 3-55 | 2 | 4-20 | 2 | 5-65 | 2 | 9-30 | 2 | C-8 | 2 | E-46 | 2 |
| 1-17 | 2 | 3-56 | 2 | 4-21 | 2 | 5-66 | 2 | 9-31 | 2 | C-9 | 2 | E-47 | 2 |
| 1-18 | 2 | 3-57 | 2 | 4-22 | 2 | 6-1 | 2 | 9-32 | 2 | C-10 | 2 | E-48 | 2 |
| 1-19 | 2 | 3-58 | 2 | 4-23 | 2 | 6-2 | 2 | 9-33 | 2 | C-11 | 2 | E-49 | 2 |
| 1-20 | 2 | 3-59 | 2 | 4-24 | 2 | 6-3 | 2 | 9-34 | 2 | C-12 | 2 | E-50 | 2 |
| 1-21 | 2 | 3-60 | 2 | 4-25 | 2 | 6-4 | 2 | 9-35 | 2 | C-13 | 2 | E-51 | 2 |
| 1-22 | 2 | 3-61 | 2 | 4-26 | 2 | 6-5 | 2 | 9-36 | 2 | C-14 | 2 | E-52 | 2 |
| 1-23 | 2 | 3-62 | 2 | 4-27 | 2 | 6-6 | 2 | 9-37 | 2 | C-15 | 2 | E-53 | 2 |
| 1-24 | 2 | 3-63 | 2 | 4-28 | 2 | 6-7 | 2 | 9-38 | 2 | C-16 | 2 | E-54 | 2 |
| 1-25 | 2 | 3-64 | 2 | 4-29 | 2 | 6-8 | 2 | 9-39 | 2 | C-17 | 2 | F-1 | 2 |
| 1-26 | 2 | 3-65 | 2 | 4-30 | 2 | 6-9 | 2 | 9-40 | 2 | C-18 | 2 | F-2 | 2 |
| 1-27 | 2 | 3-66 | 2 | 4-31 | 2 | 6-10 | 2 | 9-41 | 2 | C-19 | 2 | F-3 | 2 |
| 1-28 | 2 | 3-67 | 2 | 4-32 | 2 | 6-11 | 2 | 9-42 | 2 | C-20 | 2 | F-4 | 2 |
| 1-29 | 2 | 3-68 | 2 | 4-33 | 2 | 6-12 | 2 | 9-43 | 2 | C-21 | 2 | F-5 | 2 |
| 2-1 | 2 | 3-69 | 2 | 4-34 | 2 | 6-13 | 2 | 9-44 | 2 | C-22 | 2 | F-6 | 2 |
| 2-2 | 2 | 3-70 | 2 | 4-35 | 2 | 6-14 | 2 | 9-45 | 2 | C-23 | 2 | F-7 | 2 |
| 2-3 | 2 | 3-71 | 2 | 4-36 | 2 | 7-0 | 2 | 9-46 | 2 | C-24 | 2 | F-8 | 2 |
| 2-4 | 2 | 3-72 | 2 | 4-37 | 2 | 7-1 | 2 | 9-47 | 2 | C-25 | 2 | F-9 | 2 |
| 2-5 | 2 | 3-73 | 2 | 4-38 | 2 | 7-2 | 2 | 9-48 | 2 | C-26 | 2 | F-10 | 2 |
| 2-6 | 2 | 3-74 | 2 | 4-39 | 2 | 7-3 | 2 | A-1 | 2 | C-27 | 2 | F-11 | 2 |

## List of Effective Pages

| Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev | Page | Rev |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 2-7 | 2 | 3-75 | 2 | 4-40 | 2 | 7-4 | 2 | A-2 | 2 | C-28 | 2 | F-12 | 2 |
| 2-8 | 2 | 3-76 | 2 | 4-41 | 2 | 7-5 | 2 | A-3 | 2 | C-29 | 2 | F-13 | 2 |
| 2-9 | 2 | 3-77 | 2 | 4-42 | 2 | 7-6 | 2 | A-4 | 2 | C-30 | 2 | F-14 | 2 |
| 2-10 | 2 | 3-78 | 2 | 4-43 | 2 | 7-7 | 2 | A-5 | 2 | C-31 | 2 | F-15 | 2 |
| 2-11 | 2 | 3-79 | 2 | 4-44 | 2 | 7-8 | 2 | A-6 | 2 | C-32 | 2 | F-16 | 2 |
| 2-12 | 2 | 3-80 | 2 | 4-45 | 2 | 7-9 | 2 | A-7 | 2 | C-33 | 2 | F-17 | 2 |
| 2-13 | 2 | 3-81 | 2 | 4-46 | 2 | 7-10 | 2 | A-8 | 2 | C-34 | 2 | F-18 | 2 |
| 2-14 | 2 | 3-82 | 2 | 4-47 | 2 | 7-11 | 2 | A-9 | 2 | C-35 | 2 | G-1 | 2 |
| 2-15 | 2 | 3-83 | 2 | 4-48 | 2 | 7-12 | 2 | A-10 | 2 | C-36 | 2 | G-2 | 2 |
| 2-16 | 2 | 3-84 | 2 | 4-49 | 2 | 7-13 | 2 | A-11 | 2 | C-37 | 2 | G-3 | 2 |
| 2-17 | 2 | 3-85 | 2 | 4-50 | 2 | 7-14 | 2 | A-12 | 2 | C-38 | 2 | G-4 | 2 |
| 2-18 | 2 | 3-86 | 2 | 4-51 | 2 | 7-15 | 2 | A-13 | 2 | C-39 | 2 | G-5 | 2 |
| 2-19 | 2 | 3-87 | 2 | 4-52 | 2 | 7-16 | 2 | A-14 | 2 | C-40 | 2 | G-6 | 2 |
| 2-20 | 2 | 3-88 | 2 | 4-53 | 2 | 7-17 | 2 | A-15 | 2 | C-41 | 2 | G-7 | 2 |
| 2-21 | 2 | 3-89 | 2 | 4-54 | 2 | 7-18 | 2 | A-16 | 2 | C-42 | 2 | G-8 | 2 |
| 2-22 | 2 | 3-90 | 2 | 4-55 | 2 | 7-19 | 2 | A-17 | 2 | C-43 | 2 | G-9 | 2 |
| 2-23 | 2 | 3-91 | 2 | 4-56 | 2 | 7-20 | 2 | A-18 | 2 | C-44 | 2 | G-10 | 2 |
| 2-24 | 2 | 3-92 | 2 | 4-57 | 2 | 7-21 | 2 | A-19 | 2 | C-45 | 2 | G-11 | 2 |
| 2-25 | 2 | 3-93 | 2 | 4-58 | 2 | 7-22 | 2 | A-20 | 2 | C-46 | 2 | G-12 | 2 |
| 2-26 | 2 | 3-94 | 2 | 4-59 | 2 | 7-23 | 2 | A-21 | 2 | C-47 | 2 | G-13 | 2 |
| 2-27 | 2 | 3-95 | 2 | 4-60 | 2 | 7-24 | 2 | A-22 | 2 | C-48 | 2 | G-14 | 2 |
| 2-28 | 2 | 3-96 | 2 | 4-61 | 2 | 7-25 | 2 | A-23 | 2 | D-1 | 2 | G-15 | 2 |
| 2-29 | 2 | 3-97 | 2 | 4-62 | 2 | 7-26 | 2 | A-24 | 2 | D-2 | 2 | G-16 | 2 |
| 2-30 | 2 | 3-98 | 2 | 4-63 | 2 | 7-27 | 2 | A-25 | 2 | D-3 | 2 | G-17 | 2 |
| 2-31 | 2 | 3-99 | 2 | 4-64 | 2 | 7-28 | 2 | A-26 | 2 | D-4 | 2 | | |
| 2-32 | 2 | 3-100 | 2 | 4-65 | 2 | 7-29 | 2 | A-27 | 2 | D-5 | 2 | | |
| 2-33 | 2 | 3-101 | 2 | 4-66 | 2 | 7-30 | 2 | A-28 | 2 | D-6 | 2 | | |
| 2-34 | 2 | 3-102 | 2 | 4-67 | 2 | 7-31 | 2 | A-29 | 2 | D-7 | 2 | | |
| 2-35 | 2 | 3-103 | 2 | 4-68 | 2 | 7-32 | 2 | A-30 | 2 | D-8 | 2 | | |
| 2-36 | 2 | 3-104 | 2 | 4-69 | 2 | 7-33 | 2 | A-31 | 2 | D-9 | 2 | | |
| 2-37 | 2 | 3-105 | 2 | 4-70 | 2 | 7-34 | 2 | A-32 | 2 | D-10 | 2 | | |
| 2-38 | 2 | 3-106 | 2 | 4-71 | 2 | 7-35 | 2 | A-33 | 2 | D-11 | 2 | | |
| 2-39 | 2 | 3-107 | 2 | 4-72 | 2 | 7-36 | 2 | A-34 | 2 | D-12 | 2 | | |
| 2-40 | 2 | 3-108 | 2 | 4-73 | 2 | 7-37 | 2 | A-35 | 2 | D-13 | 2 | | |
| 2-41 | 2 | 3-109 | 2 | 4-74 | 2 | 7-38 | 2 | A-36 | 2 | D-14 | 2 | | |
| 2-42 | 2 | 3-110 | 2 | 4-75 | 2 | 7-39 | 2 | A-37 | 2 | D-15 | 2 | | |

This page is intentionally blank.

# CONTENTS


**NOTATION (Acronyms, Abbreviations, and Units)** ..... **xxi**

**EXECUTIVE SUMMARY** ..... **ES-1**

**1.0 INTRODUCTION** ..... **1-1**

**1.1 Purpose.** ..... **1-1**

**1.2 Scope** ..... **1-2**

**1.3 Background** ..... **1-5**

**1.4 Environmental Impact Statements** ..... **1-8**

1.4.1 Waste Management EIS ..... 1-8

1.4.2 Decommissioning EIS ..... 1-8

**1.5 Decommissioning Criteria** ..... **1-9**

**1.6 Project Management and Organization** ..... **1-10**

**1.7 Health and Safety Program** ..... **1-12**

**1.8 Environmental Monitoring and Control** ..... **1-12**

**1.9 Radioactive Waste Management** ..... **1-14**

**1.10 Planned End States Before and After Phase 1** ..... **1-15**

1.10.1 The Interim End State ..... 1-15

1.10.2 Facilities and Areas Within Phase 1 Scope ..... 1-19

**1.11 Organizational Responsibilities** ..... **1-23**

1.11.1 DOE ..... 1-24

1.11.2 NRC ..... 1-24

1.11.3 NYSERDA ..... 1-25

**1.12 Organization of This Plan** ..... **1-25**

**1.13 Control of Changes** ..... **1-25**

**1.14 References** ..... **1-26**

**2.0 FACILITY OPERATING HISTORY** ..... **2-1**

**2.1 License History** ..... **2-2**

2.1.1 Nuclear Fuel Services Operations From 1966 to 1982 ..... 2-5

2.1.2 West Valley Demonstration Project From 1982 to 2008 ..... 2-14

**2.2 Site Decommissioning Activities (1966 – 2011)** ..... **2-18**

## CONTENTS


2.2.1 NFS Remediation Activities (1966 – 1981) .... 2-19

2.2.2 WVDP Remediation Activities (1982 – 2011) .... 2-19

**2.3 Spills and Uncontrolled Release of Radioactivity .... 2-32**

2.3.1 North Plateau Groundwater Plume. .... 2-32

2.3.2 Old Sewage Plant Drainage .... 2-35

2.3.3 The Cesium Prong .... 2-36

2.3.4 Summary of Spills During NFS Operations .... 2-36

2.3.5 WVDP Spills .... 2-40

**2.4 Prior Onsite Burials .... 2-42**

2.4.1 Lagoon 1 .... 2-42

2.4.2 The NRC-Licensed Disposal Area .... 2-43

2.4.3 Other Burial Locations .... 2-47

**2.5 References .... 2-47**

**3.0 FACILITY DESCRIPTION .... 3-1**

**3.1 Site Location and Description .... 3-2**

3.1.1 Site Location .... 3-2

3.1.2 Site Description .... 3-2

3.1.3 Facility Description .... 3-3

3.1.4 Surrounding Communities, Businesses, and Transportation System .... 3-27

**3.2 Population Distribution .... 3-29**

3.2.1 Local Population Data .... 3-29

3.2.2 Population Distribution .... 3-32

**3.3 Current and Future Land Use .... 3-35**

3.3.1 Current Land Use. .... 3-35

3.3.2 Summary of Anticipated Land Uses .... 3-38

**3.4 Meteorology and Climatology .... 3-40**

3.4.1 The General Climate of Western New York .... 3-40

3.4.2 Severe Weather Phenomena .... 3-41

3.4.3 Weather-Related Radionuclide Transmission Factors .... 3-41

## CONTENTS


3.4.4 Weather-Related Site Deterioration Parameters .......... 3-42

3.4.5 Site Meteorology and Climatology .......... 3-42

**3.5 Geology and Seismology .......... 3-47**

3.5.1 Regional Physiography .......... 3-47

3.5.2 Site Stratigraphy .......... 3-47

3.5.3 Site Geomorphology .......... 3-52

3.5.4 Regional Structure and Tectonics .......... 3-55

3.5.5 Historical Seismicity .......... 3-61

3.5.6 Evaluation of Seismic Hazard .......... 3-64

**3.6 Surface Hydrology .......... 3-65**

3.6.1 Hydrologic Description .......... 3-65

3.6.2 WVDP Effluents .......... 3-67

3.6.3 Influence of Flooding on Site .......... 3-68

3.6.4 Water Use .......... 3-68

**3.7 Groundwater Hydrology .......... 3-70**

3.7.1 Description of the Saturated Zone .......... 3-70

3.7.2 Monitoring Wells .......... 3-72

3.7.3 Physical Hydrogeologic Parameters in the Saturated Zone .......... 3-73

3.7.4 Unsaturated Zone .......... 3-73

3.7.5 Description of Unsaturated Zone Monitoring Stations .......... 3-75

3.7.6 Physical Parameters .......... 3-75

3.7.7 Numerical Analysis Techniques .......... 3-75

3.7.8 Distribution Coefficients .......... 3-77

3.7.9 Hydraulic Properties .......... 3-81

**3.8 Natural Resources .......... 3-82**

3.8.1 Natural Gas and Oil .......... 3-82

3.8.2 Mineral Resources .......... 3-83

3.8.3 Water Resources .......... 3-84

## CONTENTS

3.8.4 Timber Resources ........ 3-85
3.8.5 Renewable Energy Resources ........ 3-86
**3.9 References ........ 3-86**

**4.0 RADIOLOGICAL STATUS OF FACILITY ........ 4-1**
**4.1 Radiological Status of Facilities, Systems, and Equipment ........ 4-2**
4.1.1 Sources of Available Data ........ 4-2
4.1.2 Impacted Facilities ........ 4-5
4.1.3 Non-Impacted Facilities ........ 4-12
4.1.4 Radionuclide Distributions ........ 4-12
4.1.5 Radiological Status of Facilities ........ 4-16
**4.2 Radiological Status of Environmental Media ........ 4-24**
4.2.1 Data Sources ........ 4-25
4.2.2 Background Levels ........ 4-26
4.2.3 Radiological Status of Surface Soil and Sediment ........ 4-29
4.2.4 Radiological Status of Subsurface Soil ........ 4-31
4.2.5 Radiological Concentrations Exceeding Background in Surface Soil, Sediment, and Subsurface Soil By WMA ........ 4-36
4.2.6 Environmental radiation Levels ........ 4-49
4.2.7 Radiological Status of Onsite Surface Water ........ 4-55
4.2.8 Radiological Status of Groundwater ........ 4-58
**4.3 References ........ 4-68**

**5.0 DOSE MODELING ........ 5-1**
**5.1 Introduction ........ 5-2**
5.1.1 Applicable Requirements and Guidance ........ 5-2
5.1.2 Context for DCGL Development ........ 5-2
5.1.3 Context for the Integrated Dose Assessment ........ 5-9
5.1.4 Potential Impact from Long-Term Erosion ........ 5-13
5.1.5 Potential Changes in Groundwater Flow Fields ........ 5-14
5.1.6 Seepage of Groundwater ........ 5-14

## CONTENTS

5.1.7 Potential Impacts on the Kent Recessional Sequence .......... 5-16
5.1.8 General Dose Modeling Process .......... 5-17
**5.2 DCGL Development .......... 5-19**
5.2.1 Conceptual Models for DCGL Development .......... 5-20
5.2.2 Subsurface Soil Conceptual Models .......... 5-26
5.2.3 Streambed Sediment Conceptual Model .......... 5-34
5.2.4 Mathematic Model .......... 5-38
5.2.5 Summary of Results .......... 5-40
5.2.6 Discussion of Sensitivity Analyses .......... 5-44
5.2.7 Probabilistic Uncertainty Analysis .......... 5-50
5.2.8 Subsurface Soil DCGL Multi-Source Analysis .......... 5-52
5.2.9 Overall Conclusions .......... 5-57
**5.3 Limited Site-Wide Dose Assessment .......... 5-57**
5.3.1 Basis for this Assessment .......... 5-57
5.3.2 Assessment Approach .......... 5-58
5.3.3 Results of the Assessments .......... 5-58
**5.4 Cleanup Goals and Additional Analyses .......... 5-60**
5.4.1 Cleanup Goals .......... 5-60
5.4.2 Refining DCGLs and Cleanup Goals .......... 5-63
5.4.3 Use of a Surrogate Radionuclide DCGL .......... 5-64
5.4.4 Preliminary Dose Assessment .......... 5-64
5.4.5 Final Dose Assessment .......... 5-65
**5.5 Monitoring, Maintenance, and Institutional Controls .......... 5-65**
**5.6 References .......... 5-65**

**6.0 ALARA ANALYSIS .......... 6-2**
**6.1 Introduction .......... 6-2**
6.1.1 Applicable Requirements and Guidance .......... 6-2
6.1.2 Remediation Activities of Interest .......... 6-3
6.1.3 The DCGLs Involved .......... 6-3

## CONTENTS

**6.2 Achieving a Decommissioning Goal Below the Dose Limits .............................. 6-3**

6.2.1 Phase 1 Proposed Decommissioning Strategy Promotes ALARA ................. 6-5

6.2.2 Good Practices that Promote ALARA ........................................................... 6-6

6.2.3 Conservatism in DCGL Development ........................................................... 6-6

6.3.4 Conservatism from the Decontamination and Final Status Survey Processes ........................................................................................................ 6-6

**6.3 DCGL ALARA Analysis ......................................................................................... 6-7**

6.3.1 ALARA Analysis Guidance ........................................................................... 6-7

6.3.2 Calculating Benefits and Costs ..................................................................... 6-8

6.3.3 Surface Soil Preliminary ALARA Analysis ...................................................... 6-9

6.3.4 Subsurface Soil Preliminary ALARA Analysis ............................................. 6-10

6.3.5 Streambed Sediment Preliminary ALARA Analysis ..................................... 6-11

6.3.6 Addressing Intergenerational Concerns ....................................................... 6-11

**6.4 Additional Analyses ............................................................................................. 6-12**

**6.5 References ........................................................................................................... 6-13**

**7.0 PLANNED DECOMMISSIONING ACTIVITIES ............................................................. 7-1**

**7.1 Conditions at the Beginning of the Phase 1 Decommissioning Work .............. 7-2**

**7.2 General Approach and General Requirements .................................................. 7-3**

7.2.1 General Approach ......................................................................................... 7-3

7.2.2 General Requirements ................................................................................... 7-5

**7.3 WMA 1 Decommissioning Activities .................................................................. 7-11**

7.3.1 Characterizing Soil and Streambed Sediment ............................................ 7-12

7.3.2 Relocating the Vitrified HLW Canisters ....................................................... 7-12

7.3.3 Removing the Above-Grade Portion of the Process Building ...................... 7-16

7.3.4 Removing the Above-Grade Portion of the Vitrification Facility .................. 7-18

7.3.5 Removing the 01-14 Building and the Vitrification Off-Gas Line ................. 7-19

7.3.6 Removing the Load-In/Load-Out Facility ..................................................... 7-19

7.3.7 Removing Other WMA 1 Structures ........................................................... 7-20

7.3.8 Removing the Underground Structures and Equipment and the Plume Source Area ........................................................................................ 7-21

## CONTENTS


7.3.9 Site Restoration ........ 7-29

**7.4 WMA 2 Decommissioning Activities ........ 7-30**

7.4.1 Characterizing Soil and Sediment ........ 7-31

7.4.2 Removing Structures ........ 7-31

7.4.3 Decommissioning the Lagoons ........ 7-32

**7.5 WMA 3 Decommissioning Activities ........ 7-38**

7.5.1 Removing Structures ........ 7-38

7.5.2 Removing the Waste Tank Pumps and Pump Support Structures ........ 7-39

7.5.3 Removing the HLW Transfer Trench Piping and Equipment ........ 7-39

7.5.4 Monitoring and Maintenance ........ 7-40

**7.6 WMA 5 Decommissioning Activities ........ 7-40**

7.6.1 Removing Lag Storage Addition 4 and the Shipping Depot ........ 7-40

7.6.2 Removing the Remote-Handled Waste Facility ........ 7-41

7.6.3 Removing Remaining Floor Slabs and Foundations and Gravel Pads ........ 7-41

**7.7 WMA 6 Decommissioning Activities ........ 7-42**

7.7.1 Removing the Sewage Treatment Plant ........ 7-42

7.7.2 Removing the Equalization Basin ........ 7-42

7.7.3 Removing the Equalization Tank ........ 7-43

7.7.4 Removing the Demineralizer Sludge Ponds ........ 7-43

7.7.5 Removing the South Waste Tank Farm Test Tower ........ 7-43

7.7.6 Removing the Remaining Floor Slabs and Foundations ........ 7-43

7.7.7 Establishing that Surface Soil Meets cleanup Goals ........ 7-44

**7.8 WMA 7 Decommissioning Activities ........ 7-44**

**7.9 WMA 9 Decommissioning Activities ........ 7-44**

**7.10 WMA 10 Decommissioning Activities ........ 7-45**

**7.11 Establishing Areas Where Surface Soil Meets Cleanup Goals ........ 7-46**

7.11.1 Areas of Interest ........ 7-46

7.11.2 Process to be Followed ........ 7-46

7.11.3 Additional Areas of Interest ........ 7-47

**7.12 Remedial Technologies ........ 7-47**

## CONTENTS

7.12.1 Pipe Cutting and Other Metal Cutting ........ 7-47
7.12.2 Tool Positioning Technologies ........ 7-49
7.12.3 Concrete Cutting and Demolition ........ 7-50
7.12.4 Concrete Decontamination ........ 7-51
7.12.5 Demolition of Structures ........ 7-53
7.12.6 Excavation and Grading ........ 7-54
**7.13 Schedule** ........ 7-55
**7.14 References** ........ 7-57

**8.0 QUALITY ASSURANCE PROGRAM ........ 8-1**
**8.1 Quality Assurance Organization ........ 8-2**
8.1.1 Quality Assurance Organization Duties and Responsibilities ........ 8-3
8.1.2 Decommissioning Project Quality Assurance Duties and Responsibilities ........ 8-4
**8.2 Assuring Quality in Preliminary Engineering Work ........ 8-4**
8.2.1 Dose Modeling ........ 8-4
8.2.2 Engineered Barrier Design ........ 8-5
8.2.3 Other Engineering Work ........ 8-6
**8.3 Decommissioning Quality Assurance Program ........ 8-6**
8.3.1 General Description of the Program ........ 8-7
8.3.2 Characterization and Final Status Survey Data ........ 8-7
8.3.3 Engineering Design and Data, Calculations, and Modeling ........ 8-10
**8.4 Document Control ........ 8-11**
**8.5 Control of Measuring and Test Equipment ........ 8-12**
**8.6 Control of Purchased Material and Subcontractor Services ........ 8-12**
**8.7 Corrective Action ........ 8-12**
**8.8 Audits and Surveillances ........ 8-12**
**8.9 Quality Assurance Records ........ 8-13**
**8.10 References ........ 8-14**

**9.0 FACILITY RADIATION SURVEYS ........ 9-1**

## CONTENTS

**9.1 Release Criteria .......... 9-2**
**9.2 Types of Surveys and Their Purposes .......... 9-5**
9.2.1 Background Surveys .......... 9-5
9.2.2 Scoping Surveys .......... 9-5
9.2.3 End-of-Task Surveys .......... 9-6
9.2.4 Characterization Surveys .......... 9-6
9.2.5 In-Process Surveys .......... 9-6
9.2.6 Final Status Surveys .......... 9-7
9.2.7 Confirmation Surveys .......... 9-7
**9.3 Background Surveys .......... 9-8**
**9.4 Characterization Surveys .......... 9-8**
9.4.1 Characterization Sample and Analysis Plan Content .......... 9-9
9.4.2 Characterization Data Quality Objectives .......... 9-13
9.4.3 Characterization Quality Requirements .......... 9-15
9.4.4 Applying DQOs for Characterization Surveys .......... 9-16
9.4.5 Characterization Surveys of Structures .......... 9-17
**9.5 In-Process Surveys .......... 9-20**
9.5.1 Measurement Methods and Instrumentation .......... 9-20
9.5.2 Scan Surveys and Direct Measurements .......... 9-20
9.5.3 Documentation .......... 9-23
**9.6 The Phase 1 Final Status Survey .......... 9-23**
9.6.1 Phase 1 Final Status Survey Plan Content .......... 9-24
9.6.2 Data Quality Objectives for the Phase 1 Final Status Survey .......... 9-29
9.6.3 Phase 1 Final Status Survey Quality Requirements .......... 9-30
**9.7 The Survey Process by Waste Management Area .......... 9-30**
9.7.1 WMA 1 Process Building and Vitrification Facility Area .......... 9-30
9.7.2 WMA 2 Low-Level Waste Treatment Facility Area .......... 9-35
9.7.3 WMA 3 Waste Tank Farm Area .......... 9-37
9.7.4 WMA 5 Waste Storage Area .......... 9-38
9.7.5 WMA 6 Central Project Premises .......... 9-40

## CONTENTS

9.7.6 WMA 7 NDA and Associated Facilities ....................................................... 9-41
9.7.7 WMA 8, State-Licensed Disposal Area ....................................................... 9-41
9.7.8 WMA 9, Radwaste Treatment System Drum Cell Area ............................... 9-41
9.7.9 WMA 10, Support and Services Area ......................................................... 9-42
9.7.10 WMA 11, Bulk Storage Warehouse and Hydrofracture Test Well Area ..... 9-43
9.7.11 WMA 12, Balance of the Site ................................................................... 9-43
9.7.12 Environmental Media ................................................................................ 9-43

**9.8 Phase 1 Final Status Survey Report Requirements .......................................... 9-45**

9.8.1 Overview of Results ..................................................................................... 9-45
9.8.2 Discussion of Changes ................................................................................ 9-45
9.8.3 Description of How Numbers of Samples Were Determined ...................... 9-46
9.8.4 Sample Number Determination Values ...................................................... 9-46
9.8.5 Results for Each Survey Unit ....................................................................... 9-46
9.8.6 Survey Unit Changes ................................................................................... 9-46
9.8.7 ALARA Practices ......................................................................................... 9-46
9.8.8 Actions Taken for Failed Survey Units ........................................................ 9-46
9.8.9 Impact of Survey Unit Failures ................................................................... 9-47

**9.9 References ........................................................................................................ 9-47**

**APPENDIX**

A Decommissioning Plan Annotated Checklist .......................................................... A-1
B Environmental Radioactivity Data ............................................................................. B-1
C Details of DCGL Development and the Integrated Dose Assessment ..................... C-1
D Engineered Barriers and Post-Remediation Activities ........................................... D-1
E Dose Modeling Probabilistic Uncertainty Analysis ................................................... E-1
F Estimated Radioactivity in Subsurface Piping ....................................................... F-1
G Phase 1 Final Status Survey Plan Conceptual Framework .................................... G-1

**FIGURES**

ES-1 Location of the Western New York Nuclear Service Center ................................ ES-3

# CONTENTS

## FIGURES


ES-2 The Former Nuclear Fuel Reprocessing Plant and the WVDP in 2006 ES-4

ES-3 Waste Management Areas 1-10 ES-6

ES-4 The Plant During the Early Years ES-11

ES-5 Important Sources of Radioactive Contamination on the Project Premises ES-14

1-1 The Former Nuclear Fuel Reprocessing Plant and the WVDP in 2006 1-6

1-2 The Project Premises Showing WMAs and the Phase 1 Excavations 1-16

1-3 WMA 1 Area in 2007 1-17

1-4 WMA 2 in 2007 1-18

1-5 Facilities Within the Scope of Phase 1 of the Decommissioning, North Plateau 1-21

1-6 Facilities Within the Scope of Phase 1 of the Decommissioning, South Plateau 1-22

1-7 The WVDP in the Interim End State 1-23

1-8 The WVDP After Completion of Phase 1 1-23

2-1 Spent Fuel Reprocessing Diagram (PUREX Process) 2-7

2-2 Simplified HLW Pretreatment Process Diagram 2-15

2-3 Previous and Current Locations of Radionuclides in North Plateau Facilities at the WVDP 2-21

2-4 Previous and Current Locations of Radionuclides in South Plateau Facilities at the WVDP 2-22

2-5 Location of Process Building Remediation Activities for Site facilities Prior to Phase 1 of the Proposed Decommissioning 2-23

2-6 Sr-90 Groundwater Plume on the North Plateau 2-33

2-7 1984 Aerial Radiation Survey Isopleths of the WVDP and Surrounding Area 2-37

2-8 NDA Disposal Area Burials 2-46

3-1 Location of the Center in Western New York 3-93

3-2 The Center, the WVDP, and the Surrounding Area 3-94

3-3 Topography of the Western New York Nuclear Service Center 3-95

3-4 Topography of Project Premises, Showing 100-Year Floodplain 3-96

3-5 Security Fence Around WVDP Premises Boundary 3-97

3-6 North Plateau Geologic Cross Section 3-98

3-7 South Plateau Geologic Cross Section 3-98

## CONTENTS

### FIGURES


| | | |
|---|---|---|
| 3-8 | WMAs 1 Through 10 | 3-99 |
| 3-9 | WMAs 11 and 12 | 3-100 |
| 3-10 | WMA 1 | 3-101 |
| 3-11 | Aerial View of Process Building and Vitrification Facility Area in 2007 | 3-102 |
| 3-12 | Construction of the Process Building | 3-103 |
| 3-13A | Process Building Layout – Below Grade | 3-104 |
| 3-13B | Process Building Layout at 100-Foot Elevation | 3-105 |
| 3-13C | Process Building Layout at 114.5-Foot Elevation | 3-106 |
| 3-13D | Process Building Layout at 131-Foot Elevation | 3-107 |
| 3-13E | Process Building Layout at 144-Foot Elevation | 3-108 |
| 3-13F | Process Building Layout at 160-Foot Elevation | 3-109 |
| 3-14 | West Side of the Process Building | 3-110 |
| 3-15 | Fuel Receiving and Storage Area | 3-110 |
| 3-16 | HLW Canisters Stored in the Chemical Process Cell | 3-111 |
| 3-17 | Conditions in the General Purpose Cell in 1999 | 3-111 |
| 3-18 | Process Mechanical Cell During Deactivation | 3-112 |
| 3-19 | Extraction Cell 3 | 3-112 |
| 3-20 | The Spent Fuel Pool After Deactivation | 3-113 |
| 3-21 | Equipment Decontamination Room Before Cleanup | 3-113 |
| 3-22 | Vitrification Facility General Arrangement | 3-114 |
| 3-23 | Vitrification Cell at Time of Startup | 3-114 |
| 3-24 | WMA 2 | 3-115 |
| 3-25 | The Low-Level Waste Treatment Facility | 3-116 |
| 3-26 | The LLW2 Building that Replaced the 02 Building | 3-117 |
| 3-27 | The Lagoon 1 Area | 3-117 |
| 3-28 | The New Interceptors | 3-118 |
| 3-29 | WMA 3 | 3-119 |
| 3-30 | Aerial View of WMA 3 Area | 3-120 |
| 3-31 | Cutaway View of 750,000-Gallon HLW Tank | 3-120 |

## CONTENTS

### FIGURES


3-32 HLW Transfer and Mobilization Pumps ... 3-121

3-33 HLW Transfer Trench Under Construction ... 3-122

3-34 Typical HLW Pump Pit ... 3-122

3-35 WMA 5 ... 3-123

3-36 The Remote Handled Waste Facility ... 3-124

3-37 The Remote Handled Waste Facility First Floor Layout ... 3-124

3-38 WMA 6 ... 3-125

3-39 The Rail Spur ... 3-126

3-40 The New Cooling Tower ... 3-126

3-41 WMA 7 ... 3-127

3-42 WMA 9 ... 3-128

3-43 WMA 10 ... 3-129

3-44 Population Around the WVDP by Compass Vector ... 3-130

3-45 Land Use in the Vicinity of the Center ... 3-131

3-46 Tornado Events in Western New York (1950 – 2002) ... 3-132

3-47 Thunderstorm Wind Events in Western New York (1950 – 2002) ... 3-133

3-48 Hail Events in Western New York (1950 – 2002) ... 3-134

3-49 Wind Rose Diagram ... 3-135

3-50 Cloud Ceiling Information ... 3-136

3-51 Regional Physiographic map ... 3-137

3-52 Bedrock and Glacial Stratigraphy of the WVDP ... 3-138

3-53 Surface Geology of the Project Premises and the SDA ... 3-139

3-54 Fold and Selected Joint Trends in the Appalachian Plateau of Western and Central New York ... 3-140

3-55 Seismo-Tectonic Map of Western New York Showing Selected Regional Geologic Structures ... 3-141

3-56 Major Northwest Trending Lineaments in New York and Pennsylvania ... 3-142

3-57 Location of Seismic Lines WVN1 and BER 83-2A ... 3-143

3-58 Seismic Hazard Curves for Peak Horizontal Acceleration ... 3-144

3-59 Seismic Hazard Curves for 1.0 Second Horizontal Acceleration ... 3-145

## CONTENTS

### FIGURES


3-60 Seismic Source Contributions to Mean Peak Horizontal Acceleration Hazard ...3-146

3-61 Buttermilk Creek Drainage Basin ...3-147

3-62 Groundwater Elevation Contours of the Sand and Gravel Unit, First Quarter 2008 ...3-148

3-63 Groundwater Elevation Contours of the Weathered Lavery Till, First Quarter 2008 ...3-149

3-64 Groundwater Elevation Contours of the Lavery Till Sand, First Quarter 2008 ...3-150

3-65 Groundwater Elevation Contours of the Kent Recessional Sequence, First Quarter 2008 ...3-151

3-66 Vertical Distribution of Cesium $K_d$ in the Weathered and Unweathered Tills...3-152

3-67 Vertical Distribution of Iodine $K_d$ in the Weathered and Unweathered Tills...3-152

3-68 Vertical Distribution of Strontium $K_d$ in the Weathered and Unweathered Tills ..3-153

3-69 Locations of Natural Gas and Oil Wells in Western New York ...3-154

3-70 Locations of Natural Gas and Oil Wells in the Vicinity of the WVDP ...3-155

4-1 Location of WMAs on the Project Premises...4-7

4-2 Impacted and Non-Impacted Facilities in WMA 1 ...4-8

4-3 Impacted and Non-Impacted Facilities in WMA 2 ...4-9

4-4 Impacted Facilities in WMA 3...4-10

4-5 Impacted and Non-Impacted Facilities in WMA 6 ...4-11

4-6 Surface Soil and Sediment Locations With Radionuclide Concentrations in Excess of Background...4-32

4-7 Subsurface Soil Locations With Radionuclide Concentrations in Excess of Background ...4-33

4-8 Cross Section of Sr-90 Concentrations Versus Depth in Subsurface Soil in WMA 1...4-35

4-9 Onsite Environmental TLD Locations ...4-51

4-10 WVDP Radiological Control Areas...4-54

4-11 Surface Water Locations With Radionuclide Concentrations in Excess of Background ...4-56

4-12 Routine Groundwater Monitoring Locations With Radionuclide Concentrations in Excess of Background ...4-63

4-13 Geoprobe® Groundwater Locations with Radionuclide Concentrations in Excess of Background...4-66

## CONTENTS

### FIGURES


4-14 North Plateau Groundwater Plume ....4-67

5-1 Areas of Interest – Surface Soil, Subsurface Soil, and Streambed Sediment Within the Project Premises ....5-3

5-2 Conceptual Cross Section of WMA 1 Excavation With Representative Data on Sr-90 Concentrations ....5-5

5-3 Conceptual Cross Section of WMA 2 Excavation With Representative Data on Subsurface Soil Contamination....5-6

5-4 Sources at the Conclusion of Phase 1 of the Proposed Decommissioning....5-11

5-5 Locations of Perimeter Seeps on the North Plateau....5-15

5-6 General Dose Modeling Process ....5-18

5-7 Conceptual Model for Surface Soil DCGL Development ....5-21

5-8 Conceptual Model for Subsurface Soil DCGL Development ....5-27

5-9 Recreationist Conceptual Model Cross Section....5-32

5-10 Conceptual Model for Streambed Sediment DCGL Development....5-34

5-11 Franks Creek Looking Upstream ....5-35

5-12 Areas Where Streambed Sediment DCGLs and Cleanup Goals Apply ....5-43

5-13 Modified Conceptual Model for Subsurface Soil DCGL Development....5-53

6-1 Unit Remediation Costs vs. Monetary Discount Rate ....6-12

7-1 WMA 1 in 2007 ....7-11

7-2 Conceptual Arrangement for Transferring Vitrified HLW Canisters ....7-14

7-3 Storage Module Conceptual Design ....7-15

7-4 Process Building General Arrangement ....7-16

7-5 Layout of Process Building Underground Structures ....7-22

7-6 Conceptual Layout of WMA 1 Excavation ....7-23

7-7 Conceptual WMA 1 Excavation Contour, With Selected Subsurface Soil Data ....7-24

7-8 Excavation Cross Sections ....7-25

7-9 Conceptual Cross-Section View of the WMA 1 Excavation....7-30

7-10 WMA 2 in 2007 ....7-30

7-11 Conceptual Arrangement of WMA 2 Excavation, Plan View ....7-33

7-12 Conceptual Arrangement of WMA 2 Excavation, Cross Section ....7-34

## CONTENTS

### FIGURES


7-13 Conceptual Cross Section of the Backfilled WMA 2 Excavation ......................... 7-38

7-14 Typical Demolition Machine ................................................................................ 7-50

7-15 Moose® ................................................................................................................ 7-52

7-16 Universal Demolition Processor ........................................................................... 7-54

7-17 Conceptual Schedule of Phase 1 Decommissioning Activities ............................ 7-56

8-1 Decommissioning Organization Quality Assurance Relationships ........................ 8-2

B-1 Background Sampling Locations More Than 10 Kilometers From the WVDP ......B-3

B-2 Sampling Locations Within 10 Kilometers From the WVDP Used for Background Calculations ........................................................................................B-6

B-3 Onsite Groundwater and Subsurface Soil Locations Used as Background ..........B-8

D-1 Plan View of the WMA 1 Excavation.................................................................... D-4

D-2 Plan View of the WMA 2 Excavation.................................................................... D-7

D-3 North Plateau Groundwater Flow Model Boundary .............................................. D-9

D-4 Cross Section of North Plateau Near-Field Model – Southwest to Northeast, Distance of 0 to 80 Meters ................................................................................... D-10

D-5 Cross Section of North Plateau Near-Field Model – Southwest to Northeast, Distance of 80 to 120 Meters ............................................................................... D-10

D-6 Cross Section of North Plateau Near-Field Model – Southwest to Northeast, Distance of 120 to 250 Meters ............................................................................. D-11

D-7 Cross Section of North Plateau Near-Field Model – Southwest to Northeast, Distance of 250 to 310 Meters ............................................................................. D-11

D-8 Cross Section of North Plateau Near-Field Model – Southwest to Northeast, Distance of 310 to 820 Meters ............................................................................. D-12

D-9 Groundwater Flow Associated with the WMA 1 and WMA 2 Engineered Barriers ........................................................................................... D-16

D-10 Groundwater Monitoring Locations Within the Project Premises During the Phase 1 Institutional Control Period.............................................................. D-21

D-11 Surface Water and Sediment Sampling Locations on the Project Premises During the Phase 1 Institutional Control Period.................................................. D-25

D-12 Offsite Surface Water and Sediment Sampling Locations During the Phase 1 Institutional Control Period..................................................................... D-26

D-13 Storm Water and Sediment Sampling Locations on the Project Premises During the Phase 1 Institutional Control Period.................................................. D-28

## CONTENTS

### FIGURES

D-14 Air Monitoring Locations on the Project Premises During the Phase 1 Institutional Control Period .......... D-29

D-15 Direct Radiation Monitoring Locations on the Project Premises During the Phase 1 Institutional Control Period .......... D-30

D-16 Offsite Direct Radiation Monitoring Locations During the Phase 1 Institutional Control Period .......... D-31

E-1 Probabilistic Uncertainty Analysis Process .......... E-8

Att-1 Probabilistic and Deterministic Dose-to-Source Ratio vs. Time, Sr-90 Surface Soil .......... E-33

Att-2 Cumulative Probability Dose-to-Source Ratio, Sr-90 Surface Soil .......... E-34

Att-3 Probabilistic and Deterministic Dose-Source Ratio, Cs-137 – Surface Soil .......... E-35

Att-4 Cumulative Probability Dose-Source Ratio, Cs-137 Surface Soil .......... E-36

Att-5 Probabilistic and Deterministic Dose-Source Ratio vs. Time, U-232 – Surface Soil .......... E-37

Att-6 Cumulative Probability Dose-Source Ratio, U-232 Surface Soil .......... E-38

Att-7 Probabilistic and Deterministic Dose-to-Source Ratio vs. Time, Sr-90 Subsurface Soil .......... E-39

Att-8 Cumulative Probability Dose-to-Source Ratio, Sr-90 Subsurface Soil .......... E-40

Att-9 Probabilistic and Deterministic Dose-Source Ratio, Cs-137 – Subsurface Soil .......... E-41

Att-10 Cumulative Probability Dose-Source Ratio, Cs-137 Subsurface Soil .......... E-42

Att-11 Probabilistic and Deterministic Dose-Source Ratio vs. Time, U-232 – Subsurface Soil .......... E-43

Att-12 Cumulative Probability Dose-to-Source Ratio, U-232, Subsurface Soil .......... E-44

Att-13 Probabilistic and Deterministic Dose-Source Ratio vs. Time, Sr-90 Streambed Sediment .......... E-45

Att-14 Cumulative Probability Dose-Source Ratio, Sr-90 Streambed Sediment .......... E-46

Att-15 Probabilistic and Deterministic Dose-Source Ratio vs. Time, Cs-137 – Streambed Sediment .......... E-47

Att-16 Cumulative Probability Dose-Source Ratio, Cs-137 Streambed Sediment .......... E-48

F-1 Location of Pipelines Beneath the Process Building .......... F-9

F-2 Leachate Transfer Line Routing from NDA to Lagoon 1 .......... F-16

G-1 Decision Logic for Surface Soil and Streambed Sediment Survey Units .......... G-16

## CONTENTS

### FIGURES


G-2 Decision Logic for WMA 1 and WMA 2 Subsurface Soils .................................. G-17


### TABLES


ES-1 $DCGL_W$ Values for 25 Millirem per Year (pCi/g) ............................................ES-19

ES-2 Cleanup Goals to be Used in Remediation (pCi/g) ..........................................ES-20

1-1 Facilities and Areas Within Phase 1 Scope ..........................................................1-20

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

2-2 Limits for Unirradiated Source Material ..................................................................2-3

2-3 Limits Used for Standards, Test, Measurements, and Calibrations .......................2-3

2-4 Nuclear Fuel Received and Reprocessed ..............................................................2-6

2-5 Estimated Radionuclide Content (in Curies) of Tanks 8D-2 and 8D-4 at the Completion of Reprocessing ..................................................................................2-9

2-6 Chemical Composition of Tank 8D-2 Supernatant at the Completion of Reprocessing ........................................................................................................ 2-10

2-7 Chemical Composition of Tank 8D-2 Sludge at the Completion of Reprocessing ........................................................................................................ 2-12

2-8 Chemical Composition of Tank 8D-4 Waste at the Completion of Reprocessing ........................................................................................................2-13

2-9 Chemical Composition of Glass Waste Form .......................................................2-17

2-10 Typical HLW Canister Radionuclide Content ....................................................... 2-18

2-11 Activity Removed by NFS for the Period 1972 Through 1977 .............................2-19

2-12 Facilities Shown in Figures 2-3 through 2-6 .........................................................2-24

2-13 Facilities Remediated or to be Remediated by the WVDP Before Decommissioning ..........................................................................2-25

2-14 DOE 10 CFR 835 Surface Contamination Guidelines ........................................ 2-30

2-15 Radiological Areas and Radioactive Material Areas ........................................... 2-31

2-16 Released Radionuclide Activity Estimates for the North Plateau Plume ..............2-34

2-17 Principal Radionuclides in Major Spills Occurring During NFS Operations ..........2-38

2-18 WVDP Spills Impacting Environmental Media (1982 – 2007) ...............................2-41

2-19 Estimated Residual Radioactivity in Lagoon 1 .....................................................2-43

2-20 Summary of Wastes in the NRC-Licensed Disposal Area ....................................2-44

2-21 Estimated Radioactivity in the NDA .......................................................................2-45

## CONTENTS

### TABLES


3-1 Businesses, Farms, and Community Centers Within a 3.1 Mile Radius of the Site....3-27

3-2 2002 Resident Population Estimates by Directional Sector Within a 3.1 Mile Radius of the Main Plant Stack....3-29

3-3 Nearest Residences by Sector....3-29

3-4 Number of Residential Wells or Springs Used for Drinking Water By Sector Within a 3.1-Mile Radius of the Main Plant Stack.... 3-31

3-5 Locations and Populations of Towns and Villages Partially or Totally Within 10 Miles of the Site ....3-32

3-6 Populations Of Selected Municipalities, Counties, and States Within 50 Miles of the Site (1960-2000) ....3-33

3-7 Leading Agricultural Products in Cattaraugus County ....3-36

3-8 2002 Consumable Animal Population Estimates by Sector Within a 3.1-Mile Radius of the Main Plant Stack ....3-36

3-9 2002 Crop Estimates in Acres by Sector Within a 3.1-Mile Radius of the Main Plant Stack....3-37

3-10 Wind Speed and Direction Frequency Distribution at 10 Meters ....3-45

3-11 Wind Speed and Direction Frequency Distribution at 60 Meters ....3-46

3-12 Generalized Paleozoic Stratigraphic Section for Southwestern New York ....3-48

3-13 Summary of Erosion Rates Near the WVDP ....3-53

3-14 Summary of Observed Faults on Seismic Lines WVN-1 and BER83-2A ....3-56

3-15 Historic Seismicity Within 320 Kilometers (200 Miles) of Site ....3-61

3-16 Site-Specific Mean Spectral Accelerations on Hard Rock (g’s) ....3-65

3-17 Site-Specific Mean Spectral Accelerations on Soil (g’s) for the North Plateau ....3-65

3-18 Site-Specific Mean Spectral Accelerations on Soil (g’s) for the South Plateau ....3-65

3-19 WVDP Hydraulic Conductivity (K) Testing Summary Table....3-73

3-20 Distribution Coefficients....3-80

3-21 Total Porosity....3-82

3-22 2001 Natural Gas and Oil Production in Ashford, Cattaraugus, and Erie Counties, and the State of New York....3-83

4-1 Scaling Factors for Spent Fuel Reprocessed....4-13

4-2 Batch 10 Sample Data ....4-14

## CONTENTS

### TABLES


4-3 Relative Fractions of Process Building Dominant Radionuclides ......................... 4-15

4-4 Relative Fractions of Vitrification Facility Dominant Radionuclides ...................... 4-16

4-5 Estimated Process Building Residual Activity at Start of Decommissioning ......... 4-16

4-6 Estimated Total Activity in Representative Process Building Areas ..................... 4-17

4-7 Measured Maximum Gamma Radiation Levels in Process Building Areas .......... 4-18

4-8 Estimated Total Activity in the Vitrification Facility .............................................. 4-19

4-9 Estimated Radioactivity in the Underground Waste Tanks .................................. 4-21

4-10 Estimated Radioactivity in the NDA ..................................................................... 4-23

4-11 Median and Maximum Background Concentrations for Environmental Media at the WVDP ........................................................................................ 4-27

4-12 Above-Background Concentrations of Radionuclides in Subsurface Soil at WMA 1 ......................................................................................... 4-37

4-13 Above-Background Concentrations of Radionuclides in Surface Soil From WMA 2 ......................................................................................... 4-37

4-14 Above-Background Concentrations of Radionuclides in Sediment From WMA 2 ......................................................................................... 4-38

4-15 Above-Background Concentrations of Radionuclides in Subsurface Soil From WMA 2 ......................................................................................... 4-40

4-16 Above-Background Concentrations of Radionuclides in Surface Soil at WMA 3 ......................................................................................... 4-41

4-17 Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil From WMA 4 ........................................................ 4-41

4-18 Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 5 ............................................................ 4-43

4-19 Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 6 ............................................................ 4-44

4-20 Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil at WMA 7 ............................................................ 4-46

4-21 Above-Background Concentrations of Radionuclides in Surface Soil and Sediment at WMA 10 ........................................................................................... 4-47

4-22 Above-Background Concentrations of Radionuclides in Surface Soil and Sediment at WMA 12 ....................................................................................... 4-48

4-23 Environmental Radiation Levels on the WVDP Site (1998-2007 Data) ................ 4-52

4-24 Radionuclide Concentrations (pCi/L) in Excess of Background in Surface Water ........................................................................................... 4-57

## CONTENTS

### TABLES


4-25 Routine Groundwater Monitoring Locations With Radionuclide Concentrations (pCi/L) in Excess of Background ....4-59

4-26 Geoprobe® Groundwater Monitoring Points With Radionuclide Concentrations (pCi/L) in Excess of Background ....4-64

5-1 Measured Maximum Lavery Till Radionuclide Concentrations ....5-7

5-2 Exposure Pathways for Surface Soil DCGL Development ....5-23

5-3 Key Input Parameters for Surface Soil DCGL Development ....5-23

5-4 Exposure Pathways for Subsurface Soil DCGL Development ....5-29

5-5 Key Input Parameters for Subsurface Soil DCGL Development ....5-30

5-6 Exposure Pathways for Streambed Sediment DCGL Development ....5-34

5-7 Key Input Parameters for Streambed Sediment DCGL Development ....5-37

5-8 DCGLs For 25 mrem (pCi/g) ....5-40

5-9 Summary of Parameter Sensitivity Analyses – Surface Soil DCGLs ....5-44

5-10 Summary of Parameter Sensitivity Analyses – Subsurface Soil DCGLs ....5-47

5-11 Summary of Parameter Sensitivity Analyses – Streambed Sediment DCGLs ....5-49

5-11a Summary of Results of Probabilistic Uncertainty Analyses ....5-51

5-11b Exposure Pathways for Modified Subsurface Soil DCGL Model ....5-54

5-11c Subsurface Soil DCGL Comparison ....5-56

5-12 Limited Site-Wide Dose Assessment 1 Results (DLGLs in pCi/g) ....5-59

5-13 Limited Site-Wide Dose Assessment 2 Results (DLGLs in pCi/g) ....5-59

5-14 Cleanup Goals to be Used in Remediation in pCi/g ....5-62

6-1 Possible Benefits and Costs Related to Decommissioning ....6-7

7-1 Expected Facility and Area Conditions at the Beginning of Phase 1 ....7-2

9-1 Surface Soil Cleanup Goal Area Factors ....9-3

9-2 Subsurface Soil Cleanup Goal Area Factors ....9-3

9-3 Streambed Sediment Cleanup Goal Area Factors ....9-4

9-4 Radiological Field Instruments ....9-11

9-5 Radionuclide Target Sensitivity For Laboratory Sample Analysis ....9-12

9-6 Radiological Field Instruments for Facility Characterization ....9-19

## CONTENTS

### TABLES


9-7 Estimated Scanning Minimum Detectable Concentrations (MDCs) of Radionuclides in Soil ....9-21

9-8 Radiological Field Instruments for Phase 1 Final Status Survey ....9-26

9-9 Scan Surveys for Different Area Classifications....9-27

9-10 Laboratory Methods....9-31

A-1 NUREG-1757 Checklist – Phase 1 Decommissioning Plan Comparison .... A-2

B-1 Summary of Comparisons of Radionuclide Data from Test Surface Soil Locations vs. SFGRVAL Background .... B-4

B-2 Summary Information for Environmental Medium Background Calculations.... B-9

B-3 Surface Soil Background Radionuclide Concentrations for the WVDP .... B-11

B-4 Sediment Background Radionuclide Concentrations for the WVDP.... B-12

B-5 Subsurface Soil Background Radionuclide Concentrations for the WVDP.... B-13

B-6 Surface Water Background Radionuclide Concentrations for the WVDP.... B-14

B-7 Groundwater Background Radionuclide Concentrations for the WVDP .... B-15

B-8 Background Environmental Radiation Levels the WVDP.... B-16

B-9 Radionuclides in Surface Soil: Ratios to Cs-137.... B-22

B-10 Radionuclides in Sediment: Ratios to Cs-137.... B-23

B-11 Radionuclides in Subsurface Soil: Ratios to Cs-137.... B-25

B-12 Summary of Radionuclide Results from Routine Onsite Sediment Monitoring Locations .... B-27

B-13 Summary of Radionuclide Results from Routine Onsite Surface Water Monitoring Locations .... B-29

B-14 Summary of Radionuclide Results from Routine Onsite Groundwater Monitoring Locations .... B-33

B-15 Groundwater Monitoring Locations: Coordinates, Depth, Screened Interval, and Geologic Unit.... B-42

B-16 Location, Elevation, and Depth of Geoprobe® Groundwater Sampling Points .... B-45

B-17 Groundwater Points Excluded from the Evaluation.... B-50

C-1 RESRAD Input Parameters....C-4

C-2 Soil/Water Distribution Coefficients....C-12

C-3 Scenario Exposure Pathways for WVDP DCGL Development....C-14

## CONTENTS

### TABLES


| | | |
|---|---|---|
| C-4 | Radionuclide Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas | C-15 |
| C-50 | Summary of Surface Soil DCGL Sensitivity Analysis | C-34 |
| C-84 | Summary of Subsurface Soil DCGL Sensitivity Analysis | C-37 |
| C-100 | Summary of Sediment DCGL Sensitivity Analysis | C-40 |
| C-117 | Summary of Primary Dose Pathways | C-42 |
| D-1 | North Plateau Near-Field Model Calibration for Head | D-13 |
| D-2 | Summary of Sand and Gravel Unit Flow Balance | D-13 |
| D-3 | Average Linear Velocity for Flow Path Originating at the Process Building | D-14 |
| D-4 | Summary of Sand and Gravel Unit Flow Balance After Phase 1 | D-15 |
| D-5 | Average Linear Velocity for Flow Path Originating at the Process Building After Phase 1 | D-17 |
| E-1 | Input Parameter Distributions for Surface Soil Model (Other than $K_d$ and Biotransfer Factor Values) | E-10 |
| E-2 | Input Parameter Distributions for Subsurface Soil Model (Other than $K_d$ and Biotransfer Factor Values) | E-11 |
| E-3 | Input Parameter Distributions for Streambed Sediment Model (Other than $K_d$ and Biotransfer Factor Values) | E-12 |
| E-4 | Summary of Data on $K_d$ Parameter (mL/g) for the 10 Elements of Interest | E-13 |
| E-5 | Lognormal Distribution Parameters for $K_d$ Values from Literature | E-14 |
| E-6 | Lognormal Distribution Used for $K_d$ Uncertainty Analyses | E-15 |
| E-7 | Input Correlations for Probabilistic Evaluation | E-16 |
| E-8 | Key Output Dose Statistics (DSRs) – Surface Soil Model (mrem/y per pCi/g) | E-18 |
| E-9 | Surface Soil $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g | E-19 |
| E-10 | Key Output Dose Statistics (DSRs) – Subsurface Soil Model (mrem/y per pCi/g) | E-20 |
| E-11 | Subsurface Soil $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g | E-21 |
| E-12 | Key Output Dose Statistics (DSRs) – Streambed Sediment Model (mrem/y per pCi/g) | E-22 |
| E-13 | Streambed Sediment $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g | E-23 |
| E-14 | Summary of Parameter Rankings – Surface Soil Model | E-25 |
| E-15 | Summary of Parameter Rankings – Subsurface Soil Model | E-26 |

## CONTENTS

**TABLES**


E-16 Summary of Parameter Rankings – Streambed Sediment Model ........................ E-28

Att-1 Estimated WMA 1 Doses from Observed Maximum Radionuclide Concentrations in the Lavery Till ........................ E-49

Att-2 Estimated WMA 2 Doses from Observed Maximum Radionuclide Concentrations in the Lavery Till ........................ E-50

F-1 List of Buried Pipelines ........................ F-2

F-2 Estimated Process Drain Line Activity in Curies ........................ F-10

F-3 Estimated Waste Transfer Line Activity in Curies ........................ F-10

F-4 Estimated Cask Decon Line Activity in Curies ........................ F-11

F-5 Estimated Wastewater Drain Line Activity in Curies ........................ F-12

F-6 Estimated Total Residual Inventory in Lines Under the Process Building ........................ F-12

F-7 Estimated Residual Inventory of Lines West of the Process Building in Curies ... F-14

F-8 Estimated Residual Inventory of Lines East of the Process Building in Curies .... F-15


*Note that other tables appear in the Appendix C, Attachment 1 electronic files.*
*A single Table (C-4B) is included in the Appendix C, Attachment 2 electronic files*
*Additional tables are also included with the Appendix E electronic files.*

# NOTATION

## Acronyms and Abbreviations

| | |
|---|---|
| AEC | U.S. Atomic Energy Commission |
| ALARA | as low as reasonably achievable |
| ASTM | American Society for Testing and Materials |
| CFR | Code of Federal Regulations |
| BH | bore hole |
| CG | cleanup goal |
| DCGL | derived concentration guideline level |
| $DCGL_W$ | derived concentration guideline level, wide |
| $DCGL_{EMC}$ | derived concentration guideline level, elevated measurement concentration |
| $DCGL_{scan}$ | derived concentration guideline level, scan |
| DOE | Department of Energy |
| DQO | data quality objective |
| DSR | dose/source ratio |
| E | east |
| EIS | environmental impact statement |
| EMC | elevated measurement concentration |
| EPA | U.S Environmental Protection Agency |
| F | Fahrenheit |
| FR | Federal Register |
| FUSRAP | Formerly Utilized Sites Remedial Action Program |
| HEPA | high-efficiency particulate air |
| HLW | high-level waste |
| ICORS | Interagency Steering Committee on Radiation Standards |
| K | hydraulic conductivity |
| $K_d$ | distribution coefficient |
| KRS | Kent recessional sequence |
| LLW | low-level waste |
| LTR | License Termination Rule |
| LTS | Lavery till sand |
| MARSSIM | Multi-Agency Radiation Survey and Site Investigation Manual |
| MDC | minimum detectable concentration |
| MMI | Modified Mercalli Intensity |
| N | north |

## NOTATION

| | |
|---|---|
| ND | not detected |
| NDA | NRC-Licensed Disposal Area |
| NESHAP | National Emissions Standards for Hazardous Air Pollutants |
| NPR | New Production Reactor |
| NRC | Nuclear Regulatory Commission |
| NFS | Nuclear Fuel Services, Inc. |
| NYSDEC | New York State Department of Environmental Conservation |
| NYSERDA | New York State Energy Research and Development Authority |
| PUREX | plutonium uranium refining by extraction |
| QA | quality assurance |
| QC | quality control |
| qtr | quarter |
| RCRA | Resource Conservation and Recovery Act |
| RESRAD | Residual radioactivity [computer code] |
| RFI | RCRA facility investigation |
| S&G | sand and gravel |
| SAIC | Science Applications International Corporation |
| SB | subsurface soil |
| SD | stream bank sediment |
| SDA | State-Licensed Disposal Area |
| SPDES | State Pollutant Discharge Elimination System |
| SS | surface soil |
| THOREX | thorium uranium extraction process |
| TLD | thermoluminescent dosimeter |
| ULT | unweathered Lavery till |
| W | west |
| WLT | weathered Lavery till |
| WMA | waste management area |
| WSMS | Washington Safety Management Solutions |
| WVDP | West Valley Demonstration Project |
| WVES | West Valley Environmental Services |
| WVNSCO | West Valley Nuclear Services Company |

# NOTATION

## Units

| | |
|---|---|
| Ci | curie |
| cfm | cubic feet per minute |
| cm | centimeter |
| $cm^2$ | centimeter squared |
| $cm^3$ | centimeter cubed |
| cpm | counts per minute |
| dpm | disintegrations per minute |
| g | gram [mass] |
| g | acceleration due to gravity [in reference to accelerations] |
| h | hour |
| kg | kilogram |
| km | kilometer |
| L | liter |
| m | meter |
| mCi | millicurie |
| millirem | 0.001 Roentgen equivalent man |
| mL | milliliter |
| mrem | millirem |
| mR | milli Roentgen |
| μCi | 0.000001 curie |
| μR | micro Roentgen |
| μrem | micro rem |
| μL | 0.000001 liter |
| pCi | $10^{-12}$ curie |
| R | Roentgen |
| rem | Roentgen equivalent man |
| s | second |
| y | year |

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# EXECUTIVE SUMMARY

**PURPOSE OF THIS EXECUTIVE SUMMARY**

The purpose of this part of the Decommissioning Plan is to provide readers a synopsis of the plan content.

**INFORMATION IN THIS SECTION**

The following matters are addressed in the order given:

- The requirements of the West Valley Demonstration Project Act, the decommissioning requirements, and the decommissioning approach;
- The name and address of the licensee and site owner;
- The location and address of the site;
- A brief description of the site and immediate environs;
- A summary of prior licensed activities and other activities involving radioactivity;
- The nature and extent of radioactive contamination at the site;
- The decommissioning objective;
- Decommissioning controls;
- Derived concentration guideline levels and cleanup goals;
- A summary of ALARA (as low as reasonably achievable) evaluations performed and planned;
- Planned initiation and completion dates for the decommissioning; and
- A summary of post-remediation activities.

**RELATIONSHIP TO OTHER PLAN SECTIONS**

This summary briefly describes the content of key parts of the plan.

The U.S. Department of Energy (DOE) has prepared this plan pursuant to its statutory obligations for decontamination and decommissioning of the West Valley Demonstration Project (WVDP) under the WVDP Act of 1980, Public Law 96-368, and to satisfy commitments made to the U.S. Nuclear Regulatory Commission (NRC) in 1981 and 2003 to prepare a decommissioning plan for the project and submit it to NRC for review.

This plan addresses Phase 1 of the two phases of the WVDP decommissioning. Phase 1 activities are expected to take eight to 10 years to complete. During this eight to 10 year period, a number of activities will be conducted to help determine the best technical approach to complete decommissioning of the remaining facilities. These activities will include further characterization of site contamination and additional scientific studies.

The approach for Phase 2 will be determined later after consideration of the results of the characterization and additional studies. The basis for this approach and the general context for the decommissioning are explained in the sidebar discussion on the next page.

**The WVDP Act and the WVDP**

This decommissioning project is being conducted under the WVDP Act of 1980. The WVDP Act directed DOE to carry out the following activities: (1) solidify the high-level waste (HLW) at the site, (2) develop containers suitable for permanent disposal of the solidified HLW, (3) transport the waste to a federal repository for permanent disposal, (4) dispose of low-level radioactive waste and transuranic waste produced in the solidification of the HLW, and (5) decontaminate and decommission the tanks, facilities, materials, and hardware used in the project in accordance with requirements prescribed by the NRC. The WVDP was initiated to allow DOE to carry out its responsibilities under the WVDP Act. This plan focuses on the fifth activity – decontamination and decommissioning.

**Decommissioning Requirements**

The NRC has prescribed the requirements in its License Termination Rule in Code of Federal Regulations 10 CFR Part 20, Subpart E to WVDP facilities and as the decommissioning goal for the entire NRC-licensed site.

**The Phased Decision-Making Approach**

The environmental impacts of the approach described in this plan are being analyzed in the *Environmental Impact Statement on Decommissioning and/or Long-Term Stewardship of the WVDP and Western New York Nuclear Service Center*, hereafter referred to as the Decommissioning EIS. Decommissioning will not begin until the Record of Decision is issued. The decommissioning is to be accomplished in two phases, with Phase 1 expected to begin in 2011. This phased decision-making approach is the preferred alternative in the Decommissioning EIS. (If changes are made to the Decommissioning EIS during the course of the National Environmental Policy Act process that affect this approach, such as changes to the preferred alternative, the approach will be revised as necessary to reflect those changes.)

Phase 1 of the decommissioning will entail removal of the Main Plant Process Building, the Low-Level Waste Treatment Facility, and certain other facilities within the WVDP area, which is known as the project premises. These activities will clean up much of the project premises to standards that will not prejudice decisions on the approach for Phase 2, which will complete the decommissioning. The Phase 2 decision could be made within 10 years of the Record of Decision and Findings Statement documenting the Phase 1 decisions. Phase 2 actions will complete the decommissioning or long-term management decision-making following the approach determined most appropriate during the additional Phase 1 evaluations for each remaining facility.

**The Phase 1 Decommissioning Scope**

The scope of this plan is limited to certain facilities on the north plateau area of the project premises and to removal of one major facility on the south plateau, the Radwaste Treatment System Drum Cell, a former radioactive waste storage area. This plan also provides for potential remediation of surface soil in selected areas of the project premises.

This plan does not address decommissioning of the underground waste storage tanks, the region of subsurface environmental contamination known as non-source area of the north plateau groundwater plume, or the two inactive radioactive waste disposal facilities on the south plateau, the NRC-Licensed Disposal Area and the State-Licensed Disposal Area, all of which will be considered in Phase 2 of the decommissioning.

**Site Owner and Site Location**

Although DOE will accomplish the decommissioning for the portion of the site used by the WVDP, the entire site remains under the ownership of the New York State Energy Research and Development Authority (NYSERDA), who is the licensee. NYSERDA's main office is in Albany at the following address:

NYSERDA
17 Columbia Circle
Albany, New York 12203-6399

NYSERDA also maintains an office near the site with the following mailing address:

10282 Rock Springs Road
West Valley, New York 14171-9799

The site, which is known as the Western New York Nuclear Service Center (the Center), is located at the latter address in a rural area in Cattaraugus and Erie counties approximately 30 miles south of the city of Buffalo as shown in Figure ES-1.

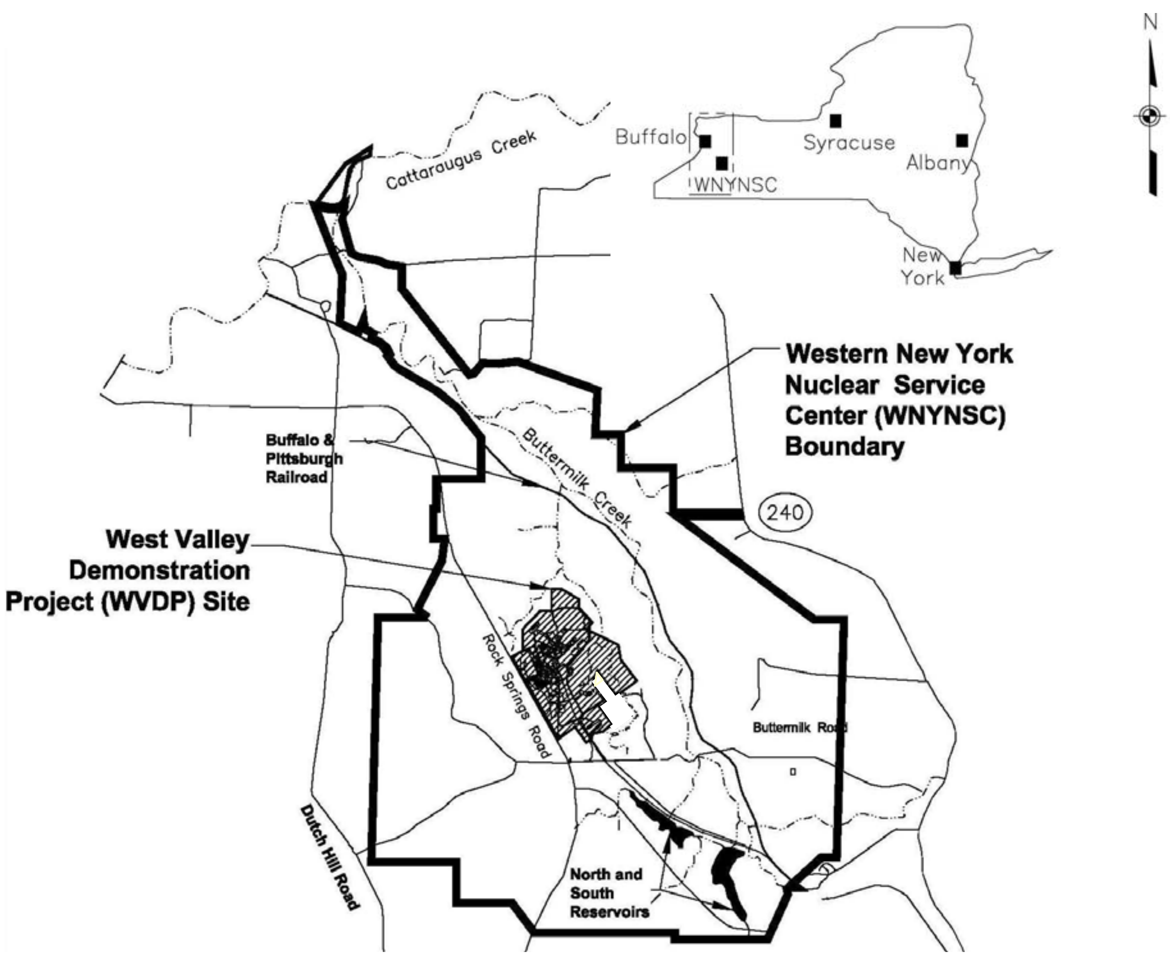


**Figure ES-1. Location of the Western New York Nuclear Service Center**

### Description of the Site and Immediate Environs

The Center property comprises approximately 3,345 acres ranging in elevation from 1,000 to 1,800 feet above mean sea level. The area of the WVDP ranges from 1,300 to 1,445 feet above sea level. The undeveloped part of the Center remains a mixture of forest, wetlands, and abandoned farmland.

The following description of the site and its environs begins with the former irradiated nuclear fuel reprocessing plant and the WVDP facilities and then addresses the remainder of the Center property, known as the retained premises, and the surrounding area. The project premises are shown in Figures ES-1 and ES-2. Note that residual radioactivity associated with the facilities is described later in this summary under the heading "Nature and Extent of Contamination at the Site."

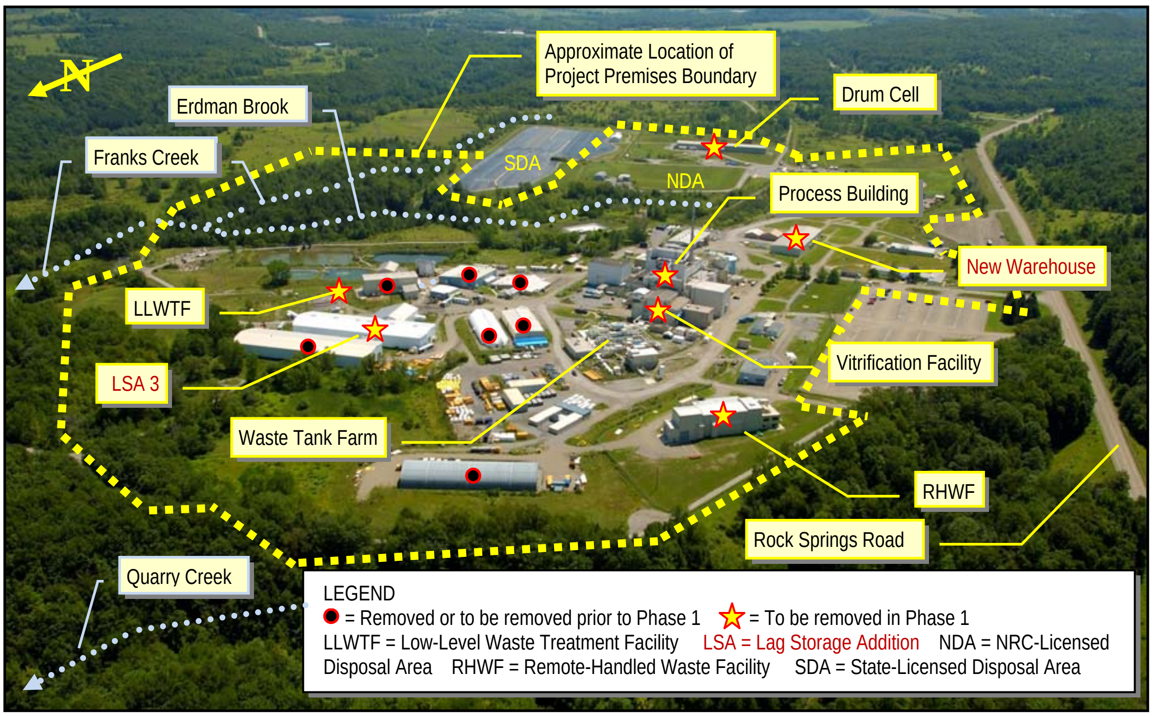


**Figure ES-2. The Former Nuclear Fuel Reprocessing Plant and the WVDP in 2006**

***The Project Premises***. At the approximate middle of the Center property lies the former nuclear fuel reprocessing plant operated by Nuclear Fuel Services, Inc. from 1966 through 1972. In 1982, control of a 156.4-acre parcel of land that included this facility and the NRC-Licensed Disposal Area was transferred to DOE for accomplishment of the WVDP[1].

Figure ES-2 shows part of the Center and the project premises as they appeared in 2006. On the right side of the photograph in Figure ES-2, one can see the Vitrification

[1] Control of two additional small parcels of land was transferred to DOE in 1986, bringing the total to approximately 167 acres. One parcel is located in the area of the Radwaste Treatment System Drum Cell. The other is located on the retained premises, which is that portion of the 3,345 acres outside of the initial 156.4 acres for which control but not ownership was transferred to DOE for accomplishment of the WVDP. The parcel on the retained premises is not within the scope of Phase 1 decommissioning activities.

Facility and the Process Building standing just behind the Waste Tank Farm where the underground waste tanks are located. Dotted lines delineate the approximate location of the perimeter of the project premises and the two streams on the project premises.

At the top of Figure ES-2 are the two shallow-land disposal sites for radioactive waste on the Center, the NRC-Licensed Disposal Area and the State-Licensed Disposal Area. The State-Licensed Disposal Area, which is licensed and permitted by the State and controlled by NYSERDA, lies outside of the project premises.

The approximate locations of the courses of the three named streams in the vicinity – Erdman Brook, Franks Creek, and Quarry Creek – are indicated in Figure ES-2. Erdman Brook divides the project premises into two areas known as the north plateau and the south plateau, with the Process Building standing on the north plateau.

When the Phase 1 decommissioning activities begin, the project premises will be in a condition known as the interim end state. The interim end state will be the condition of the project premises at the conclusion of the waste reduction and material removal campaign currently underway. As part of this work, DOE is partially decontaminating certain facilities and removing other unneeded ancillary buildings. Several buildings shown in Figure ES-2 have been removed since the photograph was taken. These and others to be removed in establishing the interim end state are identified in the figure, along with key structures to be removed during Phase 1 of the decommissioning.

Part of the site has been divided into 12 waste management areas for remediation purposes. Nine of the waste management areas are located on the project premises and one (Waste Management Area 12) is partially within the project premises, as shown in Figure ES-3. The facilities of interest are addressed below as they fall within a particular waste management area.

***Waste Management Area 1, the Process Building and Vitrification Facility Area***. The multi-story Process Building structure is approximately 130 feet by 270 feet in area and rises approximately 79 feet above ground at its highest point (not including the main stack). Most of the structure is reinforced concrete. Parts of the building lie as much as 45 feet below ground.

Within the Process Building are a number of shielded cells where disassembly and chemical reprocessing of nuclear fuel took place. Various rooms housed supporting activities. Aisles provided equipment for remote operations in the shielded cells and access to various plant areas.

On the east side of the building stands the Fuel Receiving and Storage Area. This steel-framed, steel-sheathed structure contains two fuel pools. The floor of the deeper pool lies 45 feet below grade at its lowest point.

The Vitrification Facility, which was constructed by the WVDP, is attached to the north side of the Process Building. The Vitrification Facility is a structural steel frame and sheet metal building housing the reinforced concrete Vitrification Cell, operating aisles, and a control room. It is approximately 91 feet wide and 150 feet long with the peak of the roof standing approximately 50 feet high. The pit in the Vitrification Cell extends 14 feet below grade.

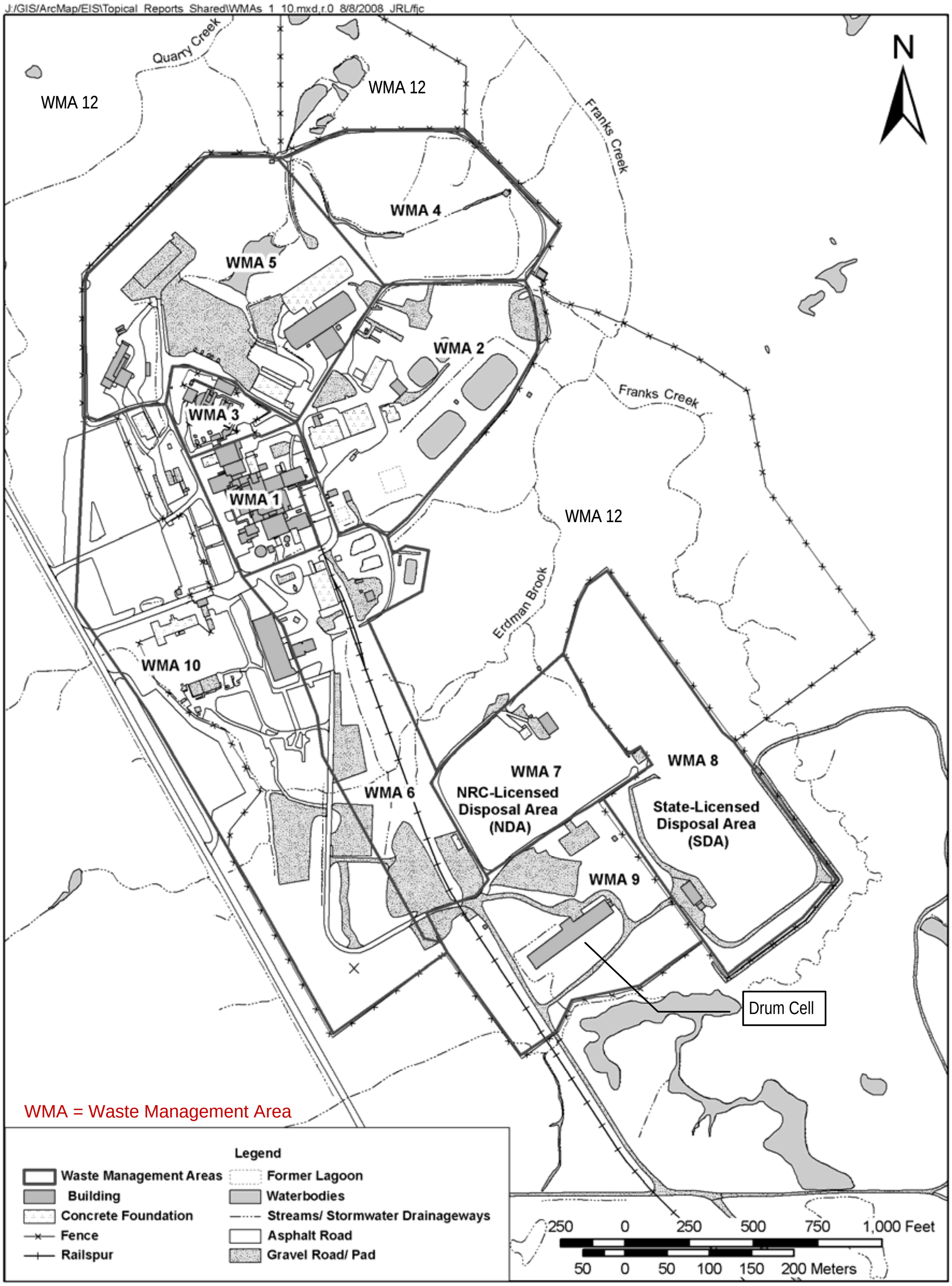


**Figure ES-3. Waste Management Areas 1-10.** (Part of WMA 12 is also shown. The State-Licensed Disposal Area in WMA 8 is not within the project premises or the scope of this plan. WMA 11 is entirely outside of the project premises.)

The steel-framed, steel-sheathed Load-In/Load-Out Facility connects to the west side of the Process Building as does the concrete block Plant Office Building. The 60-foot tall concrete and steel frame 01-14 Building stands at the southwest corner of the Process Building.

On the south side is the concrete-block Utility Room, with an addition known as the Utility Room Expansion, and the Laundry, which will be removed before decommissioning begins. The Fire Pump House and a large water storage tank stand south of the Process Building and an electrical substation is located on the east side.

All of the Waste Management Area 1 facilities are within the scope of this plan.

***Waste Management Area 2, the Low-Level Waste Treatment Facility****.* This facility, located east of the Process Building, includes five lagoons used to manage radioactive wastewater, including Lagoon 1, which was removed from service in 1984. It also includes the LLW2 Building that contains liquid waste treatment equipment, two in-ground concrete interceptor tanks, the small underground concrete Neutralization Pit, and underground pipelines connecting these facilities. All of these facilities are within the scope of this plan, along with several concrete slabs, the Maintenance Shop leach field, and the inactive Solvent Dike.

***Waste Management Area 3, the Waste Tank Farm Area***. Located just north of the Vitrification Facility, this area contains two 750,000-gallon carbon steel underground waste tanks, designated Tanks 8D-1 and 8D-2, and two 15,000-gallon stainless steel underground waste tanks, designated 8D-3 and 8D-4. These tanks are housed in concrete vaults, with Tanks 8D-3 and 8D-4 sharing a common vault. Only Tanks 8D-2 and 8D-4 were used to store HLW during reprocessing operations; Tank 8D-1 was subsequently exposed to HLW during the WVDP. A tank and vault drying system will be used to promote evaporation of the remaining liquid in the tanks, all of which are expected to be completely dry by approximately 2015.

Also in this area are the Supernatant Treatment System Support Building and the Permanent Ventilation System Building, both built by the WVDP, several smaller structures, and the HLW transfer trench that contains piping that was used to transfer waste to the Vitrification Facility.

The following facilities in Waste Management Area 3 are within the scope of this plan: the Equipment Shelter and the associated condensers, the Con-Ed Building, the HLW mobilization and transfer pumps in the underground waste tanks, and the piping and equipment within the HLW transfer trench.

***Waste Management Area 4, the Construction and Demolition Debris Landfill Area***. This 10 acre area contains the 1.5 acre landfill, which was used to dispose of non-radioactive waste, and is located north of the Low-Level Waste Treatment Facility. No facilities in this area are within the scope of Phase 1 of the decommissioning.

***Waste Management Area 5, the Waste Storage Area***. This area, which is located west of Waste Management Area 4, will contain two structures when the interim end state is reached, both of which are within the scope of this plan. One is Lag Storage Addition 4, a clear span, steel frame, metal sheathed building with an attached steel frame, metal sheathed shipping depot. The other is the Remote-Handled Waste Facility. This steel sided building contains concrete cells and rooms and is currently being used by the WVDP for processing and packaging high-activity radioactive waste. Several concrete floor slabs and gravel pads in this area are also within plan scope.

***Waste Management Area 6, the Central Project Premises***. This area is located west of the NRC-Licensed Disposal Area and south of the Process Building. Facilities in this area, all of which are within plan scope, are the Sewage Treatment Plant, the south Waste Tank Farm Test Tower, the equalization basin, the concrete equalization tank, and two demineralizer sludge ponds, along with several asphalt and gravel pads and the concrete Cooling Tower basin.

***Waste Management Area 7, the NRC-Licensed Disposal Area***. In this area lies the 400-foot by 600-foot radioactive waste burial ground, which is no longer used for radioactive waste disposal. Only remaining concrete and gravel pads in this area are within plan scope.

***Waste Management Area 8, the State-Licensed Disposal Area***. This radioactive waste disposal area covers approximately 15 acres. It is no longer used for radioactive waste disposal and is not within the scope of the Phase 1 decommissioning activities.

***Waste Management Area 9, the Radwaste Treatment System Drum Cell Area***. This area, which is located on the south plateau, contains one building, the Drum Cell, a former radioactive waste storage area identified in Figure ES-3. The Drum Cell has a concrete block foundation and concrete shield walls and is enclosed by a pre-engineered metal building 375 feet long, 60 feet wide, and 26 feet high. It is within the scope of this plan, as are several asphalt, concrete, and gravel pads.

***Waste Management Area 10, the Support and Services Area***. The remaining concrete slabs and gravel pads in this area are within the scope of this plan, as is the New Warehouse, which is located south of the Process Building. This area borders Rock Springs Road.

***Waste Management Area 11, the Bulk Storage Warehouse and Hydrofracture Test Well Area***. This area is located on the retained premises south and east of the project premises. There are no facilities in this area within the scope of this plan.

***Waste Management Area 12, the Balance of the Site***. Only the small portion of this area within the project premises is within plan scope and that only for characterization of soil and streambed sediment and possible remediation of surface soil.

***Underground Piping and Equipment***. Fifty-seven lines or portions of lines beneath the Process Building carried radioactive liquid, along with other lines near the Process Building and at the Low-Level Waste Treatment Facility. Three underground stainless steel wastewater tanks near the Process Building contain radioactivity. The three wastewater tanks are within the scope of this plan, as are the underground lines within Waste Management Area 1 and some of the underground lines within Waste Management Area 2.

***Site Geomorphology***. Streams in the area are at a relatively young stage of development and are characterized by steep profiles, V-shaped cross sections, and little or no flood plains. Erosion within the drainage basin has been dominated by slump block formation along the stream valley walls. Gullies tend to form along the stream banks during thaws and after heavy rain.

***Surface Hydrology****.* The WVDP watershed is drained by Quarry Creek, Franks Creek, and Erdman Brook. Most surface water runoff from the project premises funnels into a single stream channel at the confluence of Franks Creek and Erdman Brook located just inside the perimeter of the project premises east of the lagoons as shown in Figure ES-3.

These waters flow into Buttermilk Creek, which runs through the retained premises east and north of the project premises. Buttermilk Creek enters Cattaraugus Creek at the north end of the Center; Cattaraugus Creek eventually flows into Lake Erie. Figure ES-1 shows both creeks.

***Subsurface Conditions***. Underlying the north plateau and the south plateau is more than 500 feet of Pleistocene-age glacial tills. From the surface downward, the following layers are encountered:

- The surficial sand and gravel unit – with an average composition of 55 percent gravel, 20 percent sand, and 25 percent clay – with thickness ranging from 41 feet near the Process Building to a few feet near the northern, eastern, and southern margins of the north plateau. This unit is not present on the south plateau.
- The Lavery till – a silty-clay glacial till that contains lenses of sand, silt, and clay-silt laminations, with an average composition of 50 percent clay, 30 percent silt, 10 percent sand, and 10 percent gravel – with thickness ranging from a few feet at its western margin to more than 130 feet near Buttermilk Creek. On the south plateau, the upper three to 16 feet is weathered, with fractures and root tubes, and is known as the weathered Lavery till. Below the north plateau and the weathered Lavery till on the south plateau, the unit is referred to as the unweathered Lavery till.
- The Lavery till-sand unit – a lenticular-shaped silty, sandy layer – located on the north plateau immediately south of the Process Building. It is up to seven feet thick and lies within the upper 20 feet of the unweathered Lavery till.
- The Kent recessional sequence – with both lacustrine and kame delta deposits – underlies the Lavery till on both the north and south plateaus. It is 30 to 60 feet thick in the WVDP area.

- Shale bedrock underlies the Lavery till and other geological units on both the north and south plateaus.

***Groundwater Hydrology***. The depth of groundwater in the sand and gravel unit on the north plateau ranges from the surface to 16 feet below the surface. The groundwater flows generally northeastward toward Franks Creek. Near the northwestern margin of the sand and gravel until, flow is toward Quarry Creek and, at the southeastern margin, toward Erdman Brook. Groundwater seeps to the surface in places along stream banks and the edges of the north plateau.

***The Surrounding Area***. The nearest incorporated village is Springville, 3.5 miles to the north of the WVDP. The hamlet of West Valley lies 3.4 miles to the southeast. The communities of Riceville and Ashford Hollow also lie within a five-mile radius of the site. The closest major highway is U.S. Route 219, located 2.6 miles to the west.

***Population Distribution***. A 2002 demographic survey showed 1,056 people living within a 3.1-mile radius of the WVDP. The nearest residence was 0.76 miles away. In 2008, the U.S. Census Bureau estimated that 79,699 people lived in Cattaraugus County.

**Summary of Licensed Activities**

Provisional Operating License Number CSF-1 was issued on April 19, 1966 by the U.S. Atomic Energy Commission to Nuclear Fuel Services and the New York State Atomic and Space Development Authority to operate a spent fuel reprocessing and radioactive waste disposal facility at the Center. This Part 50 license provided possession limits for nuclear fuel of 21,000 kilograms (about 46,000 pounds) of U-235, 3,200 kilograms (about 7,055 pounds) of U-233, and 4,000 kilograms (about 8,800 pounds) of plutonium. Possession limits for unirradiated source material were 50,000 pounds of natural uranium, 100,000 pounds of uranium depleted in U-235, and 50,000 pounds of thorium. The license specified typical limits for radioactivity used for standards, measurements, and calibration purposes.

From 1966 to 1972, Nuclear Fuel Services reprocessed under this license more than 600 metric tons (600,000 kilograms or about 1,320,000 pounds) of spent nuclear fuel and generated approximately 600,000 gallons of liquid HLW as a result. Irradiated nuclear fuel reprocessing operations ended in 1972. In 1976, Nuclear Fuel Services informed New York State that it intended to withdraw from the reprocessing business and not renew the lease for the property when the initial term expired at the end of 1980. In February of 1982, Nuclear Fuel Services transferred possession of the reprocessing facilities to DOE so DOE could carry out its responsibilities under the WVDP Act.[2]

Figure ES-4 shows the plant at the beginning of the WVDP.

[2] While Nuclear Fuel Services transferred physical possession of the reprocessing facilities to DOE, NYSERDA granted exclusive use and possession of the project premises to DOE to carry out the WVDP under the provisions of the WVDP Act.

**Figure ES-4. The Plant During the Early Years** (The lagoons appear in the foreground. The Process Building can be seen in the background.)

Fuel received for reprocessing came from the N-Reactor at the Atomic Energy Commission's Hanford site and from nine commercial reactors. Reprocessing took place in the Process Building.

The first step in reprocessing entailed disassembling and sectioning the fuel. The pieces of fuel were dissolved in concentrated nitric acid. The resulting aqueous stream underwent a five-stage solvent extraction process. After further purification, the uranium and plutonium product solutions were concentrated, packaged, and eventually shipped off site. The process utilized is known as the PUREX process for plutonium uranium refining by extraction.

Aqueous waste generated was reduced in volume by evaporation, neutralized, and stored in 750,000-gallon Tank 8D-2. The neutralization process caused most fission products (not including cesium) to precipitate out and form sludge on the tank bottom. The remaining radionuclides were retained in the supernatant liquid.

Fuel received included thorium-enriched uranium, which was reprocessed using the THOREX (thorium reduction extraction) process. The resulting 12,000 gallons of liquid HLW, which was not neutralized to avoid precipitating out the thorium, were stored in 15,000-gallon Tank 8D-4.

The amounts of radioactivity in Tanks 8D-2 and 8D-4 at the completion of reprocessing, with fission and activation products decay-corrected to July 1987, were:

- Tank 8D-2 supernatant – approximately 14,000,000 curies, primarily from Cs-137, and Ba-137m;
- Tank 8D-2 sludge – approximately 15,000,000 curies, primarily from Sr-90 and Y-90; and

- Tank 8D-4 – approximately 2,000,000 curies, primarily from Sr-90, Y-90, Cs-137, and Ba-137m.

During initial plant operations, low-level wastewater from the Process Building was piped underground to an interceptor tank and then held in the lagoon system before being discharged into Erdman Brook. In 1971, a new Low-Level Waste Treatment Facility (the O2 Building) entered service. Since that time, wastewater has been treated prior to discharge from the lagoon system, which can be seen in Figure ES-4.

During the 1970s, after the termination of irradiated fuel reprocessing, Nuclear Fuel Services decontaminated many of the Process Building cells and flushed many of the systems. On February 18, 1982, the facility was formally transferred to DOE for performance of the WVDP.

During plant operations, 30 amendments were made to License CSF-1, most related to technical specifications. License amendment 31 was issued in September 1981 to transfer the project premises to DOE in accordance with the WVDP Act. Amendment 32 was issued in February 1982 to terminate the responsibility and authority of Nuclear Fuel Services. No further amendments have been made, with the license technical specifications effectively being held in abeyance until completion of the WVDP.

**Summary of WVDP Activities**

To solidify the HLW, DOE built the Integrated Radwaste Treatment System and the Vitrification Facility.

The Integrated Radwaste Treatment System included (1) the Supernatant Treatment System that decontaminated HLW tank solutions by ion exchange, (2) the Liquid Waste Treatment System to concentrate Supernatant Treatment System liquid waste by evaporation, (3) the Cement Solidification System to solidify Liquid Waste Treatment System concentrates, and (4) the Drum Cell to store cement solidified waste. By 1995, the Integrated Radwaste Treatment System had produced 19,877 71-gallon drums of solidified waste, which were stored in the Drum Cell. These drums were later shipped offsite for disposal.

Tanks 8D-1 and 8D-2 were modified and used to support the HLW solidification process. Supernatant Treatment System ion exchange columns were installed inside Tank 8D-1.

The Vitrification Facility was used to stabilize HLW sludge, loaded ion exchange resin (zeolite), and acidic THOREX waste from Tank 8D-4 in a borosilicate glass contained in stainless steel canisters. A number of modifications were made to the former reprocessing facilities to accommodate the vitrification system and the related systems. Among these changes were removing equipment from the Chemical Process Cell, decontaminating it, and installing storage racks for the HLW canisters.

Solidification of the HLW was completed in September 2002. A total of 275 canisters of vitrified HLW were produced and placed in interim storage in the former Chemical Process Cell, now known as the HLW Interim Storage Facility. DOE has deactivated portions of the Process Building and several other site facilities. In 2009 deactivation work, which includes removal of unneeded ancillary facilities, remained underway. Additional deactivation work to be completed before activities under this plan begin will result in conditions known as the interim end state.

Before much of the work to remove the Process Building is undertaken, the 275 vitrified HLW canisters will be relocated to a new Canister Interim Storage Facility to be established on the south plateau. The canisters will remain there until a decision is made and implemented with regard to their final disposal.

**Nature and Extent of Contamination at the Site**

Due to problems experienced during reprocessing operations, contamination of the site is extensive. Radionuclides include the fission products Sr-90 and Cs-137, along with uranium radionuclides and actinides such as Pu-238, Pu-239, Pu-241, and Am-241. Substantial contamination levels exist in many of the cells and rooms of the Process Building and some contamination is present inside other facilities. Subsurface soil and groundwater contamination is widespread. Figure ES-5 shows key areas of interest that are discussed below. This figure identifies major sources to be removed during Phase 1 of the decommissioning and others to be considered in Phase 2.

Figure ES-5 shows two major areas of environmental contamination at the site: the cesium prong and the north plateau groundwater plume. The cesium prong is a large area northwest of the Process Building where surface soil became contaminated with Cs-137 as a result of two ventilation system filter failures in the Process Building in 1968[3]. The north plateau groundwater plume originated that same year when releases of radioactive acid leaked into soil under the southwest corner of the Process Building. Since that time, mobile radionuclides such as Sr-90 have gradually migrated more than 40 feet under the building and approximately one-quarter mile northeast of the building.

[3] Note that the cesium prong area delineated on the figure provides only an approximation of the region of surface soil impacted by the ventilation system filter failures. Data to determine the extent of the resulting soil contamination on the project premises are not available. Such data would be collected early in Phase 1 of the decommissioning to establish the extent of residual surface and near surface soil contamination in the impacted area within the project premises. Note that other Process Building main stack releases that occurred in 1968 may have contributed to the cesium prong.

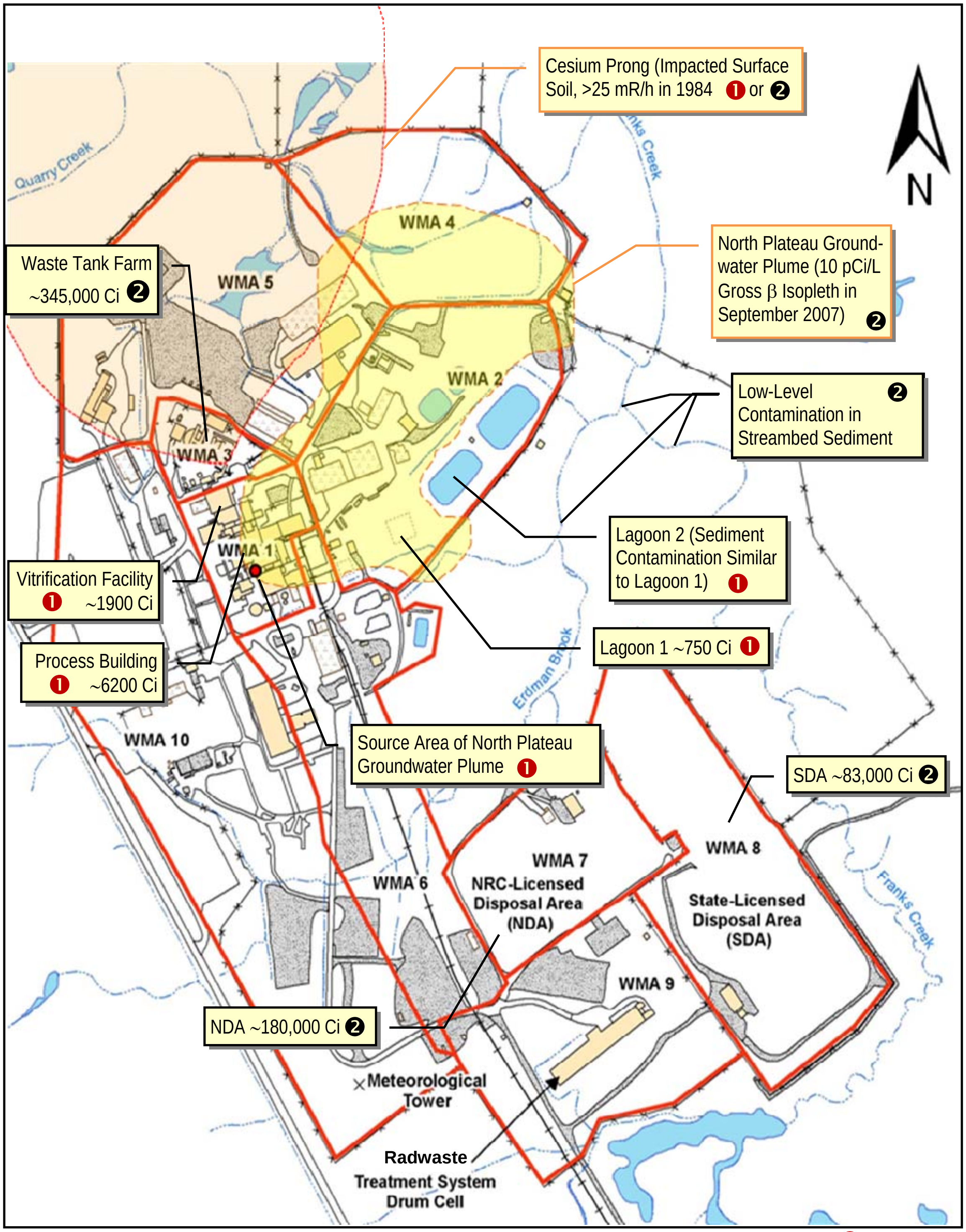


**Figure ES-5. Important Sources of Contamination on the Project Premises** (The ❶ symbol denotes major sources to be removed during Phase 1 of the decommissioning while the ❷ symbol denotes major sources to be considered in Phase 2. The estimates for total residual radioactivity are for 2011.)

The following summary of radioactive contamination addresses the more significant contaminated facilities and areas and is organized by waste management area. DOE will perform additional characterization before Phase 1 begins or early during Phase 1. The estimates of residual radioactivity are as of 2011, when Phase 1 is anticipated to start.

***Waste Management Area 1, Process Building and Vitrification Facility Area.***

- The total residual radioactivity in the Process Building is expected to be approximately 6,200 curies, with Cs-137, Sr-90, and Pu-241 being the predominant radionuclides.[4]
- The total residual radioactivity in the Vitrification Facility is expected to be approximately 1,900 curies, with Cs-137 and Sr-90 being the predominant radionuclides.
- The total residual radioactivity inside the vitrification off-gas line that runs within a concrete trench from the Vitrification Facility to the 01-14 Building is expected to be approximately 340 curies.
- Underground wastewater Tank 7D-13 is expected to contain up to 84 curies of residual radioactivity.
- Some of the underground lines in the area are expected to contain significant residual radioactivity, with one HLW transfer line expected to contain approximately 0.4 curies per linear foot.
- The subsurface soil and groundwater under the Process Building is expected to contain significant levels of residual contamination, from one or more releases of radioactivity that occurred during reprocessing that resulted in the impacted area known as the north plateau groundwater plume.

***Waste Management Area 2, Low-Level Waste Treatment Facility***

- Lagoon 1, which has been deactivated, is expected to contain approximately 750 curies, predominately Cs-137 and Pu-241, with most of this amount associated with sediment.
- The sediment in Lagoon 2, some of which was pumped from Lagoon 1 in 1984, is expected to contain a similar amount of residual radioactivity.
- The other three lagoons are known to contain residual radioactivity in their sediment, with concentrations much lower than concentrations in Lagoons 1 and 2.
- The water in all four active lagoons is expected to contain low levels of radioactivity, with the highest concentrations in Lagoon 2.
- The interceptors and the Neutralization Pit are expected to contain low levels of contamination, with the highest levels in the Old Interceptor.

[4] This estimate does not include radioactivity in the 275 vitrified HLW canisters temporarily stored inside the building, which are estimated to contain an average of approximately 30,000 curies each in 2011.

- Subsurface soil and groundwater in much of this waste management area has been impacted by Sr-90 associated with the north plateau groundwater plume.
- Surface soil near the interceptors contains low levels of contamination, particularly Cs-137.

***Waste Management Area 3, the Waste Tank Farm Area***

- The four underground waste tanks together are expected to contain approximately 345,000 curies of residual radioactivity.
- The waste mobilization and transfer pumps, which will be removed during Phase 1, are expected to contain significant amounts of residual radioactivity, with gamma radiation levels around 50 R/h.
- Some of the piping and equipment in the HLW transfer trench, which also will be removed during Phase 1, is also expected to be highly radioactive.
- The Con-Ed Building and the Equipment Shelter and condensers, which will be removed during Phase 1, are expected to contain low levels of residual radioactivity, mostly inside equipment.

***Waste Management Area 4, Construction and Demolition Debris Landfill Area***

- Although the buried waste in the landfill was not radioactive when it was emplaced, some of it is now expected to be contaminated with low levels of Sr-90 from the north plateau groundwater plume.
- Low levels of radioactivity are present in sediment in drainage ditches and in surface soil in this area.

***Waste Management Area 5, Waste Storage Area***

- The Remote-Handled Waste Facility is expected to have low levels of residual radioactivity.
- The other remaining facility – Lag Storage Addition 4 and the attached Shipping Depot – is expected to have little if any contamination above detection limits.
- Low-level contamination, especially Cs-137 associated with the cesium prong, is expected in surface soil in much of the area.
- Subsurface soil and groundwater in the eastern side of the area is known to have been impacted by the north plateau groundwater plume.

***Waste Management Area 6, Central Project Premises.*** The soil in the two demineralizer sludge ponds is expected to contain low levels of radioactive contamination, as is the Cooling Tower basin, the remaining part of the Cooling Tower, which is being removed in establishing the interim end state.

***Waste Management Area 7, the NRC-Licensed Disposal Area.*** The buried radioactive waste in this inactive waste disposal facility is expected to contain approximately 180,000 curies.

***Waste Management Area 8, the State-Licensed Disposal Area.*** The buried radioactive waste in this inactive waste disposal facility is expected to contain approximately 83,000 curies. The State-Licensed Disposal Area is not within the scope of this plan, as noted previously.

***Waste Management Area 9, the Radwaste Treatment System Drum Cell Area.*** The Drum Cell is expected to have little if any radioactive contamination above detection limits.

***Waste Management Area 10, the Support and Services Area.*** No facilities in this area are expected to have been impacted by radioactivity.

***Waste Management Area 12, Balance of the Site.*** Only the small part of this waste management area within the project premises security fence is within the scope of this plan. The sediment in Erdman Brook and the portion of Franks Creek within the fenced area is expected to contain low levels of contamination, especially Cs-137.

**The Decommissioning Objective**

The overall objective of Phase 1 of the decommissioning is to remove certain facilities and remediate portions of the project premises to criteria for unrestricted release in the License Termination Rule in 10 CFR 20.1402, thereby fulfilling part of DOE's responsibilities under the WVDP Act for decontaminating and decommissioning the tanks, facilities, materials, and hardware used in the WVDP in accordance with requirements prescribed by the NRC. The Phase 1 decommissioning activities are intended to reduce short-term and long-term health and safety risks in a manner that will support any approach that could be selected for Phase 2 of the decommissioning, which will complete decontamination and decommissioning of the Center.

Consistent with the overall objective of Phase 1, surface soil in certain areas of the project premises may be remediated as necessary to ensure that residual radioactivity concentrations satisfy the cleanup goals discussed below. These areas will be identified after completion of the characterization program. They will undergo Phase 1 final status surveys to ensure that the cleanup goals have been achieved, along with any independent confirmatory surveys to be performed by NRC or its contractor.

The objective of the Phase 1 decommissioning is not license termination of any portion of the Center, which would be beyond DOE's purview since NYSERDA is the NRC licensee. However, the Phase 1 decommissioning activities are designed to support license termination for remediated portions of the project premises if license termination for all or part of the Center were to become an objective for Phase 2 of the decommissioning.

**Decommissioning Controls**

The decommissioning will be accomplished by a contractor employed by DOE. DOE will provide appropriate oversight. The decommissioning organization will be structured to ensure that certain functions – radiological controls, health and safety, and quality assurance – are independent of the organizational elements performing the work.

The decommissioning will be accomplished in accordance with applicable DOE and NRC requirements, and in accordance with applicable requirements of other federal agencies and the

State of New York. However, given DOE's authority under the WVDP Act and, and considering that the Department is not the NRC licensee for the site, certain aspects of the decommissioning will be controlled in accordance with DOE procedures, i.e., DOE regulations, directives, and technical standards. These aspects are:

- Project management and organization,
- Radiological safety controls and monitoring of workers,
- Environmental monitoring and control, and
- Radioactive waste management.

**DCGLs and Cleanup Goals**

To support Phase 1 decommissioning activities and later decisions for Phase 2 of the decommissioning, derived concentration guideline levels (DCGLs) were developed for surface soil, subsurface soil in the deep WMA 1 and WMA 2 excavations, and streambed sediment using the RESRAD RESidual RADioactivity computer Code, Version 6.4 and other models. To ensure that the conceptual models initially used in DCGL development were sufficiently conservative, a number of alternative conceptual models were also analyzed. These included evaluation of potential doses to offsite receptors from radioactivity displaced by erosion.

One alternative conceptual model for subsurface soil DCGL development involved consideration of continuing releases of residual radioactivity from the bottom of the remediated deep excavations, as well as radioactivity brought to the surface during installation of a cistern type well. This multi-source conceptual model proved to be more limiting than other conceptual models for most radionuclides of interest.

Sensitivity analyses were performed to identify model input parameters with the most influence on dose. In addition, a probabilistic uncertainty analysis was performed using RESRAD Version 6.4 to help ensure that key model input parameters were sufficiently conservative.

Table ES-1 provides the calculated DCGLs for 18 radionuclides of interest for surface soil, subsurface soil, and streambed sediment. These DCGLs, which take into account the results of the alternate conceptual model analyses and the probabilistic uncertainty analysis, assure that the dose to the average member of the critical group will not exceed 25 millirem per year when considering the dose contribution from each radionuclide individually.[5]

[5] The DCGLs for Sr-90 and Cs-137 apply to the year 2041 and later, that is, they incorporate a 30-year decay period from 2011. The 30-year decay period was selected for these key radionuclides because of their short half-life. As noted previously, the Phase 2 decision could be made within 10 years of issue of the Record of Decision and Findings Statement documenting the Phase 1 decision. If this approach were to involve unrestricted release of the site, achieving this condition would be expected to take more than 20 years due to the large scope of effort to exhume the underground waste tanks and the NDA. It is therefore highly unlikely that conditions for unrestricted release of the project premises could be established before 2041. If Phase 2 were to involve closing radioactive facilities in place, then institutional controls would remain in place after 2041. DOE will be responsible for maintaining institutional control of the project premises and providing for monitoring and maintenance of the project premises until completion of Phase 2 of the decommissioning.

**Table ES-1. $DCGL_W$ Values For 25 Millirem per Year (pCi/g)[1]**

| Nuclide | Surface Soil | Subsurface Soil | Streambed Sediment |
|---|---|---|---|
| Am-241 | 2.9E+01 | 6.3E+03 | 1.0E+04 |
| C-14 | 1.6E+01 | 9.9E+02 | 1.8E+03 |
| Cm-243 | 3.5E+01 | 1.1E+03 | 3.1E+03 |
| Cm-244 | 6.5E+01 | 2.2E+04 | 3.8E+04 |
| Cs-137[2] | 1.5E+01 | 3.0E+02 | 1.0E+03 |
| I-129 | 3.3E-01 | 7.5E+00 | 7.9E+02 |
| Np-237 | 2.6E-01 | 1.0E+00 | 3.2E+02 |
| Pu-238 | 4.0E+01 | 1.3E+04 | 1.2E+04 |
| Pu-239 | 2.5E+01 | 3.1E+03 | 1.2E+04 |
| Pu-240 | 2.6E+01 | 3.4E+03 | 1.2E+04 |
| Pu-241 | 1.2E+03 | 2.4E+05 | 3.4E+05 |
| Sr-90[2] | 4.1E+00 | 2.8E+02 | 4.7E+03 |
| Tc-99 | 2.1E+01 | 5.9E+02 | 6.6E+05 |
| U-232 | 1.5E+00 | 7.4E+01 | 2.2E+02 |
| U-233 | 8.3E+00 | 1.9E+02 | 2.2E+04 |
| U-234 | 8.4E+00 | 2.0E+02 | 2.2E+04 |
| U-235 | 3.5E+00 | 2.1E+02 | 2.3E+03 |
| U-238 | 9.8E+00 | 2.1E+02 | 8.2E+03 |

NOTES: (1) The $DCGL_W$ is the DCGL applicable to the average concentration over a survey unit.
(2) DCGLs for Sr-90 and Cs-137 apply to the year 2041 and later.

Phase 1 decommissioning activities will involve removal of subsurface soil in the bottom and sides of the large excavation for removal of the Waste Management Area 1 facilities and the large excavation in Waste Management Area 2 for removal of Lagoon 1, Lagoon 2, Lagoon 3, the interceptors, and the Neutralization Pit. Phase 1 decommissioning activities may include remediation of surface soil in selected areas, as discussed previously.

The DCGLs in Table ES-1 were developed considering the separate areas of interest and the critical group for exposure to radioactivity in surface soil and subsurface soil is different from the critical group for exposure to radioactivity in streambed sediment. In consideration of this situation, and because only limited portions of the project premises will be remediated during Phase 1 of the decommissioning, two assessments were performed that involved apportioning doses from different portions of the remediated project premises to ensure that DCGLs used for remediation in Phase 1 of the decommissioning will not limit Phase 2 options.

Considering the results of these assessments, and the results of the ALARA analysis discussed below, DOE has established the following cleanup goals, which are lower than the

DCGLs, to ensure that remediation accomplished during Phase 1 of the decommissioning will support any approach that might be used during Phase 2 of the decommissioning.

**Table ES-2. Cleanup Goals to be Used in Remediation in pCi/g(1)**

| Nuclide | Surface Soil(3) | Subsurface Soil(4) | Streambed Sediment |
|---|---|---|---|
| Am-241 | 2.6E+01 | 2.8E+03 | 1.0E+03 |
| C-14 | 1.5E+01 | 4.5E+02 | 1.8E+02 |
| Cm-243 | 3.1E+01 | 5.0E+02 | 3.1E+02 |
| Cm-244 | 5.8E+01 | 9.9E+03 | 3.8E+03 |
| Cs-137(2) | 1.4E+01 | 1.4E+02 | 1.0E+02 |
| I-129 | 2.9E-01 | 3.4E+00 | 7.9E+01 |
| Np-237 | 2.3E-01 | 4.5E-01 | 3.2E+01 |
| Pu-238 | 3.6E+01 | 5.9E+03 | 1.2E+03 |
| Pu-239 | 2.3E+01 | 1.4E+03 | 1.2E+03 |
| Pu-240 | 2.4E+01 | 1.5E+03 | 1.2E+03 |
| Pu-241 | 1.0E+03 | 1.1E+05 | 3.4E+04 |
| Sr-90(2) | 3.7E+00 | 1.3E+02 | 4.7E+02 |
| Tc-99 | 1.9E+01 | 2.7E+02 | 6.6E+04 |
| U-232 | 1.4E+00 | 3.3E+01 | 2.2E+01 |
| U-233 | 7.5E+00 | 8.6E+01 | 2.2E+03 |
| U-234 | 7.6E+00 | 9.0E+01 | 2.2E+03 |
| U-235 | 3.1E+00 | 9.5E+01 | 2.3E+02 |
| U-238 | 8.9E+00 | 9.5E+01 | 8.2E+02 |

NOTES: (1) These cleanup goals, which, like the $DCGL_W$ values in Table ES-1, apply to the average concentration over a survey unit, are to be used as the criteria for the Phase 1 remediation activities.

(2) Cleanup goals for Sr-90 and Cs-137 apply to the year 2041 and later. That is, they incorporate a 30-year decay period from 2011. The 30-year decay period was selected for these key radionuclides because of their short half-life. License termination actions that may take place in Phase 2 of the decommissioning will not likely be fully implemented before 2041.

(3) The surface soil cleanup goals apply to the upper one meter (3.3 feet) of surface soil.

(4) The subsurface soil cleanup goals apply only to the bottoms of the large WMA 1 and WMA 2 excavations and to the sides of these excavations one meter (3.3 feet) or more below the surface.

Since these cleanup goals were developed for individual radionuclides of interest, a sum-of-fractions approach based on radionuclide distributions in different areas will be used to ensure that potential doses from the remediated areas will be no more than the dose from one of the individual radionuclides at the concentration specified in Table ES-2.

Although the subsurface soil cleanup goals in Table ES-2 form the criteria for residual radioactivity in the two large excavations, remediation plans involve excavation at least one foot into the Lavery till and, in Waste Management Area 2, at least one foot below the sediment in the bottoms of Lagoons 2 and 3. This approach is expected to produce residual radioactivity levels well below the cleanup goals, based on limited existing data on residual radioactive contamination in the Lavery till. A preliminary, order-of-magnitude dose analysis using these data suggests that potential futures doses from these excavated areas will be approximately eight millirem per year for Waste Management Area 1 and approximately 0.2 millirem per year for Waste Management Area 2.

After additional characterization data become available, the DCGLs and the cleanup goals will be reevaluated using these data and refined as appropriate. After the Phase 1 decommissioning activities have been completed, another dose analysis using Phase 1 final status survey data will be performed to estimate the potential doses from the remediated subsurface areas.

**Summary of ALARA Evaluations**

DOE has performed a preliminary cost-benefit analysis using NRC methodology to determine whether removal of soil or sediment with radioactivity concentrations below the DCGLs will be consistent with the ALARA principal. These analyses compared the cost of disposal of additional soil or sediment with the reduction in radiation exposure associated with removal of additional soil or sediment below the DCGLs valued at $2000 per person-rem as set forth in NRC guidance. They indicate that removal of soil or sediment with radioactivity concentrations below the DCGLs will not be cost-effective.

DOE will perform another similar analysis when the subsurface soil remediation work is in progress (and when surface soil and streambed sediment remediation is in progress, if that work is done in Phase 1) to confirm the results of the preliminary ALARA evaluation. This second, more-detailed analysis will use updated information and consider other factors such as other societal and socioeconomic considerations and costs related to transportation of additional waste. This second analysis will also consider the impacts of using lower discount rates on the estimated cost of remediation so that intergenerational concerns are taken into account.

**Initiation and Completion Dates**

Subject to the decision in the Record of Decision for the Decommissioning EIS, expected to be issued in early 2010, DOE will begin Phase 1 of the decommissioning in 2011 and it is expected to last approximately eight to 10 years.

**Post-Remediation Activities**

The post remediation activities fall into two categories: (1) a monitoring and maintenance program and (2) an institutional control program, both of which focus on the project premises.

The monitoring and maintenance program will continue until Phase 2 of the decommissioning starts, when it will be reevaluated. It will include an environmental monitoring program tailored to conditions that will exist at the conclusion of the Phase 1 decommissioning activities. This program will monitor onsite groundwater, storm water, and air, along with onsite and offsite surface water, sediment, and radiation. Groundwater monitoring will be accomplished using approximately 36 monitoring wells.

The monitoring and maintenance program will also ensure that important facilities and systems serve their intended purposes. Facilities and systems within the scope of this program include:

- The subsurface hydraulic barrier wall and French drain to be installed during Phase 1 on the north and east sides of the excavation for removal of the Waste Management Area 1 facilities,
- The subsurface hydraulic barrier wall to be installed during Phase 1 on the northwest and northeast sides of the excavation for removal of key Waste Management Area 2 facilities,
- The tank and vault drying system for the underground waste tanks that is to be installed before Phase 1 of the decommissioning,
- The dewatering well used to minimize in-leakage into the underground waste tank vaults,
- The hydraulic barrier wall and geomembrane cover for the NRC-Licensed Disposal Area, and
- The security features and monitoring systems installed for the new Canister Interim Storage Facility to be established on the south plateau.

Performance of the hydraulic barrier walls will be assessed with hydraulic monitoring piezometers.

Insofar as institutional controls are concerned, DOE will continue control of the project premises and provide for monitoring and maintenance of the project premises until completion of DOE's Phase 2 decommissioning requirements. Institutional controls will include security fences and signs along the perimeter of the project premises, a full-time security force, provisions for controlled access through designated gateways, and appropriate security measures for the new Canister Interim Storage Facility on the south plateau, which will be established during Phase 1 of the decommissioning. DOE will be responsible for institutional controls for the new Canister Interim Storage Facility until the HLW canisters are shipped offsite.

# 1.0 INTRODUCTION

**PURPOSE OF THIS SECTION**

The purpose of this section is to provide introductory information to help readers understand this plan, which is particularly complex for several reasons.

**INFORMATION IN THIS SECTION**

This section explains the purpose of this plan and describes its scope. It briefly summarizes the background related to the decommissioning.

It then discusses the two environmental impact statements that pertain to the decommissioning, along with the decommissioning criteria. It briefly describes four programs pertaining to the decommissioning that will be carried out in accordance with Department of Energy directives and technical standards: (1) project management and organization, (2) the health and safety program, (3) the environmental monitoring and control program, and (4) the radioactive waste management program.

It describes the interim end state for the site that will be reached at the conclusion of deactivation work scheduled to end in 2011, which will form the starting conditions for the Phase 1 decommissioning work. It then briefly summarizes the Phase 1 decommissioning work.

Finally, this introduction briefly describes the responsibilities of the organizations involved, explains how the plan is organized, and describes the process to be used to control changes to the plan after initial issue.

**RELATIONSHIP TO OTHER PLAN SECTIONS**

The information in this section establishes the context for the other parts of this plan.

## 1.1 Purpose

This plan is being issued by the U.S. Department of Energy (DOE) to fulfill part of its statutory obligations under Public Law 96-368, the West Valley Demonstration Project (WVDP) Act of 1980, which holds DOE responsible for decontamination and decommissioning of facilities used in solidification of high-level radioactive waste (HLW) and material and hardware used in connection with this project.[1]

The decommissioning is being accomplished in two phases following a “phased decision-making” approach. This plan addresses Phase 1, describing:

(1) The activities that will take place during this phase of the decommissioning;

[1] The WVDP Act states that “The Secretary [of Energy] shall decontaminate and decommission (A) the tanks and other facilities of the Center in which the high level waste solidified under the project was stored, (B) the facilities used in the solidification of the waste, and (C) any material and hardware used in connection with the project, in accordance with such requirements that the [Nuclear Regulatory] Commission may prescribe.”

(2) The site conditions that will exist at the conclusion of Phase 1; and

(3) The methods that will be used to organize and manage the project, to protect the health and safety of workers and the public, to protect the environment, and to ensure quality in the decommissioning work.

Phase 2 of the decommissioning will be accomplished using an approach determined to be the most appropriate after completion of additional Phase 1 studies and evaluations for remaining facilities.

This plan also provides information to the U.S. Nuclear Regulatory Commission (NRC) on the first of the two phases of the WVDP decommissioning, consistent with the related 1981 Memorandum of Understanding between DOE and NRC (DOE and NRC 1981), which calls for DOE to submit a decommissioning plan to NRC for review. On February 3, 2003, NRC specifically requested that DOE submit a decommissioning plan for the WVDP portion of the site (NRC 2003a). DOE agreed to do so in its response of February 28, 2003 (DOE 2003a).

## 1.2 Scope

Under the provisions of the WVDP Act, DOE exercises control over a portion of the Western New York Nuclear Service Center (the Center) for the purpose of carrying out the WVDP. The Center is owned by the New York State Energy Research and Development Authority (NYSERDA), who is the NRC licensee.

The area controlled by DOE comprises approximately 168 acres, lies in the approximate middle of the Center, and contains the facilities used by Nuclear Fuel Services, Inc. (NFS) from 1966 through 1972 to reprocess spent nuclear fuel. This area is known as the project premises.

A small stream divides the project premises into two regions known as the north plateau and the south plateau. The facilities used by NFS are located on the north plateau, with the exception of two shallow land radioactive waste disposal facilities known as the NRC-Licensed Disposal Area (NDA) and the State-Licensed Disposal Area (SDA)[2], which are located on the south plateau.

The facilities of interest in Phase 1 of the decommissioning are located on the north plateau, with one exception: the WVDP Radwaste Treatment System Drum Cell on the south plateau, which was used for radioactive waste storage. Phase 1 of the WVDP decommissioning will entail removal of the Radwaste Treatment System Drum Cell and all of the north plateau facilities with the exceptions of the Waste Tank Farm with its four underground waste storage tanks, the Waste Tank Farm supporting facilities, and the Construction and Demolition Debris Landfill.

Phase 1 activities include remediation of the “source area” portion of the impacted area known as the north plateau groundwater plume, where groundwater and subsurface soil is

[2] The SDA, which is not part of the project premises, is managed by NYSERDA, licensed by the New York State Department of Health, and permitted by the New York State Department of Environmental Conservation (NYSDEC).

contaminated with radioactivity from spent fuel reprocessing. The source area lies underneath the Main Plant Process Building (Process Building). The non-source area of the plume, which is downgradient of the building, will be considered during Phase 2 of the decommissioning.

Phase 1 includes removal of impacted soil in excavations dug to remove the facilities in the Process Building and Vitrification area and in a portion of the Low-Level Waste Treatment Facility area. Phase 1 also includes characterization of soil and stream sediment within the project premises, especially in the Phase 1 areas.[3]

Phase 2, which this plan does not address, would complete the decommissioning for the Waste Tank Farm, the Construction and Demolition Debris Landfill area, the NDA, the non-source area of the north plateau groundwater plume, and the other remaining impacted areas of the project premises. Phase 2 actions will complete the decommissioning according to the approach determined most appropriate during the additional Phase 1 evaluations for each remaining facility, as previously noted. The studies to be performed during Phase 1 are currently being evaluated by DOE and NYSERDA, the joint lead agencies developing the Environmental Impact Statement for *Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center* (the Decommissioning EIS, which is discussed in Section 1.4.2). Such studies could include:

- Characterization studies, which would include sampling of surface soil, subsurface soil, and stream sediments along with characterization of selected underground piping that would be exposed during other removal activities; and
- Data collection and studies, such as monitoring and evaluating technology developments regarding disposal facilities for orphan waste, underground waste tank cleaning and exhumation, along with research related to long-term performance of engineered barriers and work to enhance site erosion and hydrology models and exhuming buried radioactive waste.

DOE and NYSERDA will evaluate and consider several factors during Phase 1, including:

- The results of analyses to estimate the impacts of residual radioactivity that will remain after completion of the Phase 1 decommissioning activities;
- The additional information developed in the studies to be carried out in Phase 1; and
- The availability of new technologies that might be applied in Phase 2.

[3] The project premises is the portion of the site controlled by DOE as shown in Figure 1-1. The Phase 1 areas are those within the scope of this plan. The Phase 2 areas are the Waste Tank Farm area, the Construction and Demolition Debris Landfill, the non-source area of the north plateau groundwater plume, and the NDA. Although the Waste Tank Farm area is considered to be a Phase 2 area, limited work will be performed in this area during Phase 1, as shown in Table 1-1. Characterization of soil and sediment in the Phase 2 source areas will be limited and will not include the NDA.

The evaluations will take into account the status of the underground waste tanks and the two waste disposal areas, which will be reviewed at approximately five-year intervals, along with the various decommissioning or long-term management approaches. The final decision on the Phase 2 decommissioning and long-term management approach could be made within 10 years of the date of issue of the Phase 1 Record of Decision.

These studies and evaluations are beyond the scope of this plan, except for the soil and sediment characterization within the project premises to be accomplished early in Phase 1, which is discussed in Section 1.10.2.

The Phase 1 activities are designed to be conservative with respect to the extent of remediation in the areas of interest to avoid prejudicing the decision on the Phase 2 approach. More information on the facilities within the scope of this Phase 1 plan appears in Section 1.10.2.

While this plan provides for removal of certain radioactive facilities and remediation of surface and subsurface soil on portions of the project premises, it does not address license termination of any portion of the site. Licensing matters are not within DOE's purview since DOE is neither the licensee nor the property owner. However, the work accomplished under this plan will result in data that can potentially be used by NYSERDA in support of license termination for portions of the Center.

This plan focuses primarily on radioactivity. Hazardous and toxic materials are addressed in some instances and activities specified in this plan will be in compliance with the Resource Conservation and Recovery Act. However, closure of facilities under the provisions of the Resource Conservation and Recovery Act is being addressed separately in coordination with appropriate state and federal agencies and is not within the scope of this plan.

The approach described in this plan represents DOE's preferred alternative among those alternatives evaluated in the *Environmental Impact Statement on Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center*, hereafter referred to as the Decommissioning EIS.[4] Under this alternative, the decommissioning will be performed in two phases, as indicated above.

The organization and content of this plan are based on NRC guidance in Volume 1 of NUREG-1757, *Consolidated Decommissioning Guidance, Decommissioning Process for Materials Licensees* (NRC 2006) and agreements made between NRC and DOE on the applicability of this guidance to the Phase 1 plan (NRC 2008). This plan will be supplemented by more detailed plans for demolition of major facilities that will be completed prior to the start of the decommissioning.

[4] When this plan was completed, the Decommissioning EIS existed in the form of the *Revised Draft Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center*. If changes are made to the Decommissioning EIS during the course of the National Environmental Policy Act process that affect this plan, such as changes to the preferred alternative, this plan will be revised as necessary to reflect those changes.

**The Unique Nature of the Phase 1 Decommissioning**

Among the atypical elements of this decommissioning are (1) the radiological complexity of the site; (2) the project being carried out under the WVDP Act; (3) the project being carried out by a department of the federal government when the property is owned by a New York State Agency that is the NRC licensee; and (4) the purpose of the Phase 1 decommissioning work being limited to removing certain facilities and remediating impacted soil in certain areas, rather than terminating the NRC license.

## 1.3 Background

Situated approximately 30 miles south of Buffalo on 3,345 acres of property owned by the State of New York, the Center is the location of the only NRC-licensed commercial spent nuclear fuel reprocessing facility to operate in the United States. NFS reprocessed irradiated nuclear fuel to recover uranium and plutonium until 1972. Figure 1-1 shows a portion of the Center and the WVDP as they appeared in 2006.

The reprocessing operations produced approximately 600,000 gallons of HLW, which were stored in two underground waste tanks. These operations were conducted under License CSF-1, which was issued by the U.S. Atomic Energy Commission in 1966. After NFS withdrew from the reprocessing business, NYSERDA became the sole licensee.[5]

Reprocessing work resulted in extensive radioactive contamination of site facilities, especially the Process Building where the chemical processes that separated uranium and plutonium from fission products in the spent fuel were carried out. The Low-Level Waste Treatment Facility – which included five lagoons – also became contaminated with licensed radioactivity.

Environmental contamination also resulted from site operations. The contaminated areas of most significance are known today as the north plateau groundwater plume and the cesium prong. The approximate lateral extent of both impacted areas is shown in Figure 1-1.[6]

The north plateau groundwater plume impacts a subsurface area of more than 15 acres under and northeast of the Process Building. This contamination likely resulted from multiple leaks of nitric acid solution containing licensed radioactive material that occurred during fuel reprocessing. Groundwater movement has carried mobile radionuclides such as strontium 90

[5] In 1976, NFS informed New York State that it intended to withdraw from the reprocessing business and not renew the lease for the property when the initial term expired at the end of 1980. In February of 1982, NFS transferred possession of the reprocessing facilities to DOE so DOE could carry out its responsibilities under the WVDP Act.

[6] Note that the cesium prong area delineated on the figure provides only an approximation of the region of surface soil impacted by the ventilation system filter failure. Data to determine the extent of the resulting soil contamination on the project premises are not available. Such data will be collected early in Phase 1 of the decommissioning to establish the extent of residual surface and near surface soil contamination in the impacted area within the project premises.

approximately one-quarter mile northeast of the Process Building. Contamination beneath the Process Building is known to extend at least 40 feet below the ground.

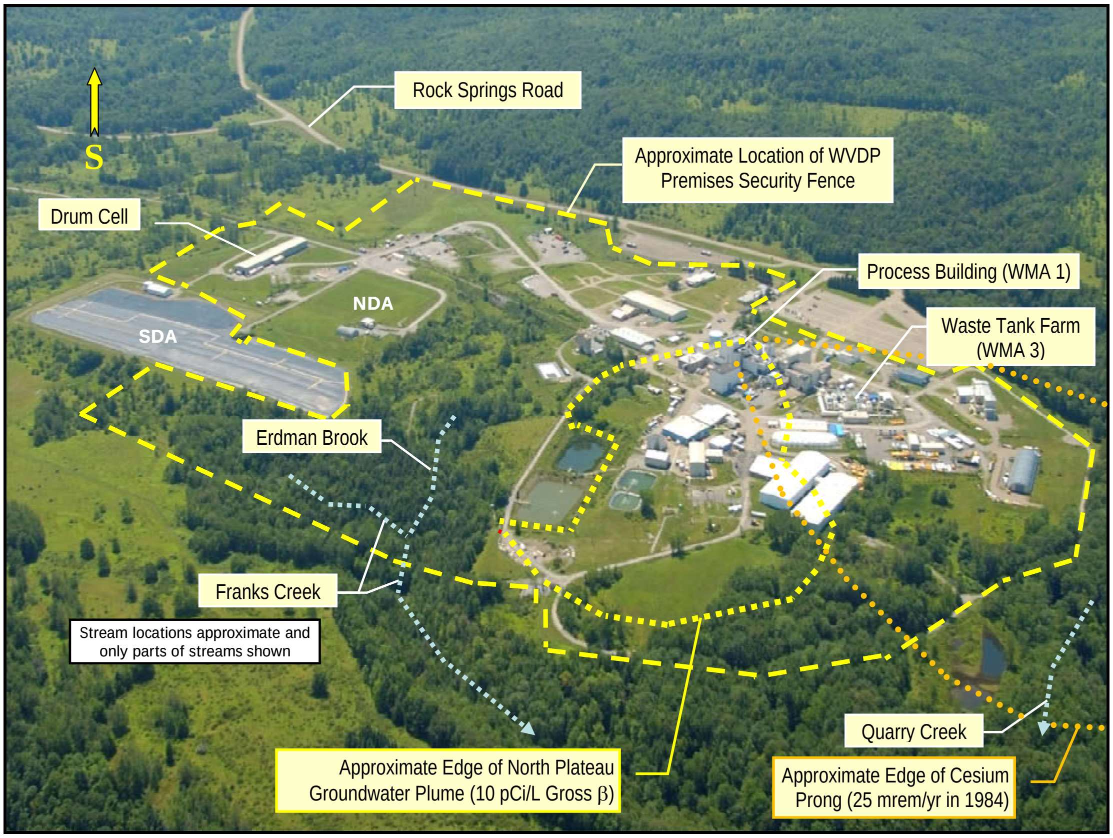


**Figure 1-1. The Former Nuclear Fuel Reprocessing Plant and the WVDP in 2006**

The cesium prong is an impacted area that extends northwest of the Process Building as a result of at least two ventilation system accidents that occurred in 1968. A series of investigations that included aerial monitoring surveys has shown that cesium 137 released from the Process Building main stack contaminated surface soil in the northwest part of the Center and offsite.

Streams in the vicinity of the project premises were also impacted with radioactivity from regulated discharges of treated wastewater, surface water runoff, and contaminated groundwater that seeps to the surface at several points on the project premises.

There are also other places on the Center where environmental media have been impacted by unplanned releases of radioactivity. These include low levels of contamination in a drainage channel near a sewage outfall that resulted from a 1974 underground sewer line failure and low levels of contamination in drainage ditches resulting from a 1985 spill of radioactive condensate in the area of the underground waste tanks. Low levels of radioactive contamination have also been identified in surface and subsurface soil in other areas.

In 1980, Congress enacted the WVDP Act to establish the WVDP as a research and development project to demonstrate solidification techniques for HLW. The WVDP Act assigned the primary responsibility for the project to DOE, although it did not authorize the federal government to acquire title to the HLW.[7] Since 1981, portions of NYSERDA's NRC Part 50 license for the Center, including the technical specifications, have been effectively suspended by NRC to facilitate execution of the provisions of the WVDP Act.

In 2002, DOE completed solidification of the HLW using a vitrification process. The solidified HLW is contained within 275 stainless steel canisters that are presently stored in the Process Building. This material will have to remain on site until disposition decisions are made and implemented, which is one factor in DOE's decision to pursue a two-phase decommissioning approach.

DOE in recent years has been partially decontaminating portions of the Process Building and other facilities and removing unneeded ancillary facilities in preparation for the WVDP decommissioning. This effort is expected to culminate in 2011, achieving site conditions known as the interim end state, which are described in Section 1.10.1.

The amounts of residual radioactivity at the site are now substantially less than when the facility was shutdown in 1972 owing to radioactive decay and NFS and WVDP decontamination efforts. However, a significant amount of radioactivity will remain on site when the Phase 1 decommissioning activities are scheduled to begin in 2011. The estimated amounts in key areas in 2011, exclusive of radioactivity in the HLW waste canisters, include:

- The Process Building, approximately 6200 curies;
- The Vitrification Facility, approximately 1900 curies;
- Lagoon 1, approximately 750 curies;
- The four underground waste tanks, approximately 345,000 curies;
- The NDA, approximately, 180,000 curies; and
- The SDA, approximately, 83,000 curies.

The Process Building, the Vitrification Facility, and the Low-Level Waste Treatment Facility lagoons are addressed in Phase 1 of the decommissioning, as explained below. The other facilities – commonly referred to, along with the radioactivity in the non-source area of the north plateau groundwater plume, as Phase 2 sources – will be addressed in Phase 2 of the decommissioning.

[7] The WVDP Act states in pertinent part: "The Secretary [of DOE] shall carry out, in accordance with this Act, a high level radioactive waste demonstration project at the Western New York Nuclear Service Center in West Valley, New York, for the purpose of demonstrating solidification techniques which can be used for preparing high level radioactive waste for disposal. . . . The State will make available to the Secretary the facilities of the Center and the high level radioactive waste at the Center which are necessary for completion of the project. The facilities and the waste shall be made available without transfer of title and for such period as may be required for completion of the project."

## 1.4 Environmental Impact Statements

In 1996, DOE prepared a Draft EIS covering the remaining actions to be completed under the WVDP Act and evaluating different alternatives for closure and long-term stewardship of the facilities at the Center. Based upon comments received, ongoing discussions between DOE and NYSERDA, and various other factors, DOE decided not to move forward with the 1996 Draft EIS in its immediate form. Instead, DOE decided to revise its strategy to address the remaining activities required under the WVDP Act in two phases (and two EISs) – the first covering short-term, offsite waste disposal activities and the second covering longer-term closure and stewardship activities.

### 1.4.1 Waste Management EIS

The Final West Valley Demonstration Project Waste Management EIS (DOE 2003b) on short-term, offsite waste disposal activities was issued by DOE on January 12, 2004. It addresses, as DOE's preferred alternative:

- Continued onsite management of HLW until disposition decisions are made and implemented,
- Shipping low-level radioactive waste (LLW) and mixed (radioactive and hazardous) LLW offsite for disposal,
- Shipping transuranic waste to the Waste Isolation Pilot Plant near Carlsbad, New Mexico, and
- Actively managing the underground waste tanks, including ventilating them to minimize moisture and associated corrosion.

The Waste Management EIS Record of Decision was issued in the Federal Register on June 16, 2005 (70 FR 115). It partially implemented the preferred alternative, deferring the decision on transuranic waste shipment pending a determination that this waste meets all statutory and regulatory requirements for disposal at the Waste Isolation Pilot Plant.

### 1.4.2 Decommissioning EIS

The Decommissioning EIS addresses DOE's remaining activities under the WVDP Act, any waste management activities that could arise as a result of decommissioning activities, and activities related to decommissioning or long-term stewardship of the balance of the Center. DOE and NYSERDA are jointly preparing this EIS.

The Decommissioning EIS also evaluates potential management and disposition actions for those facilities and areas, including the SDA, for which NYSERDA is responsible. The NRC is participating in the Decommissioning EIS as a cooperating agency, as are the U.S. Environmental Protection Agency (EPA), NYSDEC, and the New York State Department of Health (NYSDOH). A Notice of Intent to prepare the Decommissioning EIS appeared in the Federal Register on March 13, 2003 (68 FR 49).

As noted previously, the decommissioning approach described in this plan is DOE's preferred alternative in the Decommissioning EIS. If changes to that document occur during the National Environmental Policy Act process that affect this plan, such as changes to the preferred alternative, this plan will be revised as necessary to reflect the changes. The activities under the Decommissioning Plan will begin only after issuance of the Decommissioning EIS Record of Decision.

### 1.5 Decommissioning Criteria

Under the authority of the WVDP Act, the NRC in 2002 issued its Final Policy Statement on the decommissioning criteria for the WVDP (67 FR 22) specifying the application of its License Termination Rule (10 CFR 20, Subpart E) to the decommissioning. This policy statement indicated that the final end-state may involve a long-term or even perpetual license for parts of the site where cleanup to License Termination Rule requirements would be prohibitively expensive or technically impractical. The policy statement also indicated that closure of the underground waste tanks (if the tanks were to be closed in place) must meet specified criteria for incidental waste as set forth in NRC's Final Policy Statement.

The criteria of the License Termination Rule are being applied to the decommissioning of: (1) underground waste tanks and other facilities in which HLW, solidified under the project, was stored; (2) facilities used in the solidification of the waste; and (3) any material and hardware used in connection with the WVDP.

Requirements in 10 CFR 20.1402 address license termination without restrictions. Requirements in 10 CFR 20.1403 address license termination under restricted conditions.

The unrestricted release criteria 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 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.

The restricted release criteria of 10 CFR 20.1403 involve addressing matters such as the following:

- That residual radioactivity levels are ALARA;
- Provisions for legally enforceable institutional controls that provide reasonable assurance that the total effective dose equivalent to the average member of the critical group will not exceed 25 mrem per year;
- Financial assurance;
- Considering the advice of individuals and institutions in the community who may be affected by the decommissioning or planned institutional controls; and

- That residual radioactivity at the site has been reduced so that if the institutional controls were no longer in effect, there is reasonable assurance that the total effective dose equivalent from residual radioactivity to the average member of the critical group is ALARA and would not exceed either (1) 100 mrem per year or (2) 500 mrem per year provided certain conditions are met.

In 2003, NRC issued an Implementation Plan for its Final Policy Statement on the Decommissioning Criteria for the WVDP (NRC 2003b).

Although Phase 1 of the WVDP decommissioning will not result in license termination under either restricted or unrestricted conditions, this plan does include derived concentration guideline levels (DCGLs) and associated cleanup goals to be used for remediation of surface and subsurface soil in the excavated areas on the project premises described previously that are based on the unrestricted release criteria of 10 CFR 20.1402.[8] The cleanup goals take into account the results of a limited, site-wide integrated dose assessment. This assessment was performed to ensure that conditions in the excavations for the Process Building-Vitrification Facility and Low-Level Waste Treatment Facility lagoon areas at the conclusion of Phase 1 will not limit potential approaches that may be considered for Phase 2 of the decommissioning.

## 1.6 Project Management and Organization

The project will be managed in accordance with DOE requirements in a manner similar to deactivation work currently underway at the WVDP. Necessary tasks will be defined and scheduled. Appropriate schedules will be used for this purpose, such as a long-range schedule, short-range schedules, and plans-of-the-week. NRC will be provided copies of these schedules for information.

Implementing plans will be prepared as necessary in support of the work. Examples of these plans include:

- A Health and Safety Plan to implement requirements outlined in Section 1.7;
- Decommissioning Work Plans for demolition of major facilities, which are discussed in Section 7;
- A Quality Assurance Project Plan, which is described in Section 8;
- A Characterization Sample and Analysis Plan, which is described in Section 9,
- A Waste Management Plan to implement requirements outlined in Section 1.9; and
- A Final Status Survey Plan, which is also described in Section 9.

NRC will be provided copies of these plans for information.

[8] The DCGLs and cleanup goals for Sr-90 and Cs-137 incorporate a 30-year decay period from 2011. That is, achieving residual radioactivity levels less than the cleanup goals for these radionuclides will ensure that dose criteria of 10 CFR 20.1402 will be met in 2041 and any time thereafter, around the time when the vitrified HLW canisters were expected to be shipped to the federal geologic repository when Revision 0 to this plan was issued. The year for shipping the vitrified HLW canisters offsite is uncertain until disposition decisions are made and implemented; however, shipment is not expected to take place before 2041.

Detailed engineering designs for the decommissioning will be developed based on the conceptual designs outlined in this plan. Detailed design information on the following engineered features will be provided to NRC to provide an opportunity for NRC to review and comment on the safety aspects of the designs: the designs for the large excavations in WMA 1 and WMA 2, including the hydraulic barriers, the French drain, the groundwater control provisions, and the groundwater monitoring system as discussed in Section 7 and Appendix D.

DOE will also provide information on the detailed design of the new Canister Interim Storage Facility to NRC and consult with NRC on the related documented safety analysis.

Written procedures will be prepared as necessary to support the project activities. Work packages would be used for individual procedures or groups of procedures. After completion of work activities, the work packages would be formally closed out to ensure that all required work was accomplished.

Radiological work permits will be prepared as necessary and approved by the Radiological Control Manager or his or her designee in accordance with applicable DOE procedures. Persons working in areas covered by radiological work permits will be briefed before starting work in accordance with DOE procedures.

Training of project personnel will be commensurate with their experience, their responsibilities and the potential hazards to which they could be exposed. Records will be maintained showing the employee's name, training date, type of training received and other relevant information.  This training will include, as applicable:

- General Employee Training, which will consist of a general orientation on site requirements and policies;
- Radiation worker training, with formal written and practical examinations to certify that the individuals are qualified as radiation workers;
- Radiological control technician training, also with formal written and practical examinations to certify individual qualification;
- Job-specific training, which will be performed as appropriate for individual jobs; and
- Pre-shift briefings, which will be conducted as appropriate at the beginning of each work shift.

DOE will employ a contractor to accomplish the Phase 1 decommissioning activities[9]. The decommissioning contractor organization will provide the necessary functions to this end, such as operations, engineering, radiological controls, health and safety, quality assurance, and training.

The decommissioning contractor senior executive will be responsible to the Director of the WVDP for carrying out the decommissioning work in accordance with applicable DOE requirements and guidance as specified in the contract. The requirements will include this plan and all of its provisions, such as those associated with the health and safety program,

[9] DOE may employ more than one contractor.

environmental monitoring and control, and radioactive waste management as specified in the subsections that follow. Additional contractual provisions may also be invoked by DOE, such as compliance with DOE-STD-1107-97, *Knowledge, Skills, and Abilities for Key Radiation Protection Positions at DOE Facilities*, and with DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for Nuclear Facilities*.

## 1.7 Health and Safety Program

The health and safety program for Phase 1 of the decommissioning will be based on DOE procedures. This approach is consistent with DOE's authority and responsibilities to protect human health and safety under applicable laws and the provisions of the WVDP Act.

The DOE procedures that address radiological safety controls during decommissioning appear in the form of regulations, directives (orders, policies, guides, and manuals), and supplemental technical standards, and in contract conditions with its site or decommissioning contractors. DOE and its decommissioning contractor will follow these procedures for radiation safety controls and monitoring for workers during Phase 1 of the decommissioning, along with other applicable requirements and guidance.

Among the applicable DOE procedures is a policy statement that expresses the Department's position to ensure that radiation exposures to its workers and the public and releases of radioactivity to the environment are maintained below regulatory limits, and that deliberate efforts are taken to further reduce exposures and releases to ALARA. This statement appears in DOE Policy 441.1.

Applicable requirements include the following:

- 10 CFR 830, *Nuclear Safety Management*
- 10 CFR 835, *Occupational Radiation Protection*
- 29 CFR 1910.134, *Respiratory Protection*
- DOE Policy 450.4, *Safety Management System Policy*
- DOE Order 420.1B, *Facility Safety*
- DOE Order 430.1B, *Real Property Asset Management*
- DOE Order 5400.5, *Radiation Protection of the Public and the Environment*
- DOE Manual 231.1-1A, *Environment, Safety, and Health Reporting Manual*

The Department's supplemental technical standards associated with these requirements will also be followed.

## 1.8 Environmental Monitoring and Control

DOE has maintained an extensive environmental monitoring and control program at the site since 1982 to satisfy the environmental monitoring requirements of federal and state laws and regulations and of DOE Orders and technical standards, and to comply with environmental permits that have been issued to the WVDP by NYSDEC and the EPA. Annual environmental monitoring reports (WVES and URS 2008) describe the results of this program.

The environmental monitoring and control program that will be implemented during Phase 1 of the decommissioning will be based on the program currently in place at the WVDP. It will continue to comply with federal and state laws, federal and state environmental permits, DOE Orders and technical standards, and other applicable requirements and guidance under which the WVDP operates, which are consistent with the applicable NRC requirements of 10 CFR 20.

Three major elements of this program are: (1) the ALARA evaluation program, (2) the effluent monitoring program, and (3) the effluent control program. The program will be modified as necessary during decommissioning to ensure compliance with applicable requirements. As noted in Section 1.7, it is DOE policy to ensure that releases of radioactivity to the environment are maintained below regulatory limits, and that deliberate efforts are taken to further reduce releases to ALARA (DOE Policy 441.1).

The decommissioning environmental program will meet the following monitoring and control requirements:

- Clean Air Act of 1970, as amended
- Clean Water Act of 1977
- Resource Conservation and Recovery Act of 1976, as amended
- Executive Order 11988, *Floodplain Management* (42 FR 26951)
- Executive Order 11990, *Protection of Wetlands* (42 FR 26961)
- Executive Order 12856, *Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements* (58 FR 150)
- Executive Order 13101, *Greening the Government through Waste Prevention, Recycling, and Federal Acquisition* (63 FR 179)
- Executive Order 13148, *Greening the Government through Leadership in Environmental Management* (65 FR 81)
- 10 CFR 830.122, *Quality Assurance Criteria*
- 40 CFR 61, *National Emission Standards for Hazardous Air Pollutants*
- 40 CFR 141, *National Primary Drinking Water Regulations*
- 40 CFR 143, *National Secondary Drinking Water Regulations*
- DOE Manual 231.1-1A, *Environment, Safety, and Health Reporting Manual*
- DOE Order 414.1C, *Quality Assurance*
- DOE Order 435.1, *Radioactive Waste Management*
- DOE Order 440.1B, *Worker Protection Management for DOE Federal Employees*
- DOE Order 450.1, *Environmental Protection Program*

- DOE Order 451.1B, *National Environmental Policy Act Compliance Program*
- DOE Order 5400.5, *Radiation Protection of the Public and the Environment*

DOE and the decommissioning contractor will also comply with applicable DOE technical standards, active site environmental permits, and active administrative orders of consent associated with the Resource Conservation and Recovery Act.

Note that information specified in NUREG-1748, *Environmental Review Guidance for Licensing Actions Associated with NMSS Programs* (NRC 2003c), that is normally provided in decommissioning plans, can be found in Section 3 of this plan, in the Decommissioning EIS, or both.

## 1.9 Radioactive Waste Management

The radioactive waste management program for Phase 1 of the decommissioning will also be based on DOE procedures, consistent with the provisions of the WVDP Act. The WVDP Act states that DOE shall, in accordance with applicable license requirements, dispose of LLW and transuranic waste produced by the solidification of the HLW under the project.[10]

The DOE procedures that address waste management appear in the form of requirements contained in the Code of Federal Regulations, in DOE Orders, and in guidance contained in supplemental technical standards. DOE and its decommissioning contractor will follow these procedures for management of radioactive waste during Phase 1 of the decommissioning, along with other applicable requirements and guidance.

The principal requirements for management of DOE radioactive waste appear in DOE Order 435.1, *Radioactive Waste Management*. This order applies to HLW, transuranic waste, and LLW, and to the radioactive component of mixed waste. Additional detailed requirements appear in DOE Manual 435.1-1, *Radioactive Waste Management Manual*. Detailed guidance for implementation of these requirements is given in DOE Guide 435.1, *Implementation Guide for Use with DOE M 435.1*.

Other applicable requirements include the following:

- 10 CFR 830.120, *Quality Assurance Requirements*
- 10 CFR 835, *Occupational Radiation Protection*
- DOE Order 414.1C, *Quality Assurance*
- DOE Order 460.1B, *Packaging and Transportation Safety*

The Phase 1 decommissioning waste management activities will also be consistent with applicable federal laws such as the Resource Conservation and Recovery Act of 1976, as

[10] The WVDP Act also states that DOE "shall, as soon as feasible, transport in accordance with applicable provisions of law, the waste solidified at the Center [the vitrified HLW canisters] to an appropriate Federal repository for permanent disposal." This activity will take place in Phase 2 of the decommissioning once disposition decisions related to the HLW are made and implemented.

amended, and the Toxic Substances Control Act of 1976, as amended, and with applicable permits and consent orders. These activities will also be consistent with other applicable DOE guidance, such as that contained in DOE Guide 460.1-1, *Implementation Guide for Use with DOE Order 460.1A.*

All radioactive waste produced during the decommissioning will be disposed of offsite at appropriate government-owned or commercial facilities. In some cases, waste produced will be temporarily stored onsite for later shipment. Note that at the time this plan was completed, there was no approved disposal path for transuranic waste that will be generated during Phase 1 of the decommissioning. Transuranic waste generated will therefore be temporarily stored onsite until such time that it can be shipped to an approved disposal facility.

## 1.10 Planned End States Before and After Phase 1

Site deactivation activities will produce conditions known as the interim end state that will be the conditions in effect at the start of the Phase 1 decommissioning work.

### 1.10.1 The Interim End State

The map of the project premises shown in Figure 1-2 depicts the facilities that will still be in place at the start of Phase 1 decommissioning activities. It shows the waste management areas (WMAs) into which the project premises has been divided for remediation purposes. It also shows the two large excavations for removal of facilities in WMA 1 and WMA 2 during the Phase 1 decommissioning work, as explained in Section 1.10.2 below.

The deactivation activities required to achieve the interim end state will include removal of other ancillary facilities not shown in Figure 1-2. Certain facilities will be partially decontaminated to facilitate demolition during Phase 1 without the use of radiological containment. Section 3 of this plan describes the facilities in detail.

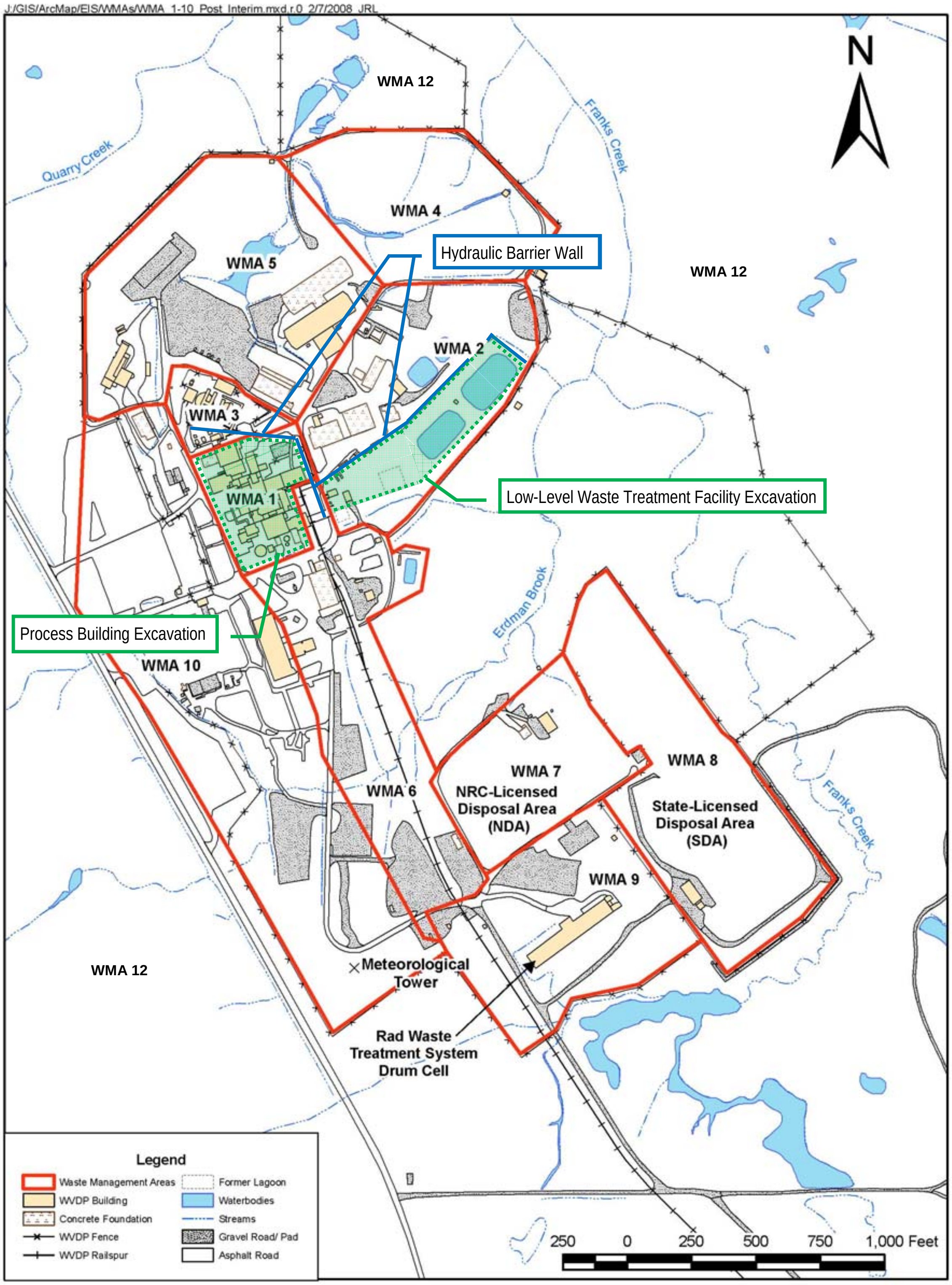


**Figure 1-2. The Project Premises Showing WMAs and the Phase 1 Excavations**

**WMA 1**

The partially decontaminated facilities in WMA 1 are the Process Building, the Vitrification Facility, and the 01-14 Building. The other facilities that will remain within WMA 1 when the interim end state is reached are the Utility Room, the Utility Room Expansion, the Plant Office Building, the Load-in/Load-out Facility, the Electrical Substation, the Fire Pumphouse, and the Water Storage Tank. Figure 1-3 shows these facilities, along with the Laundry Room, which will be removed in achieving the interim end state.[11]

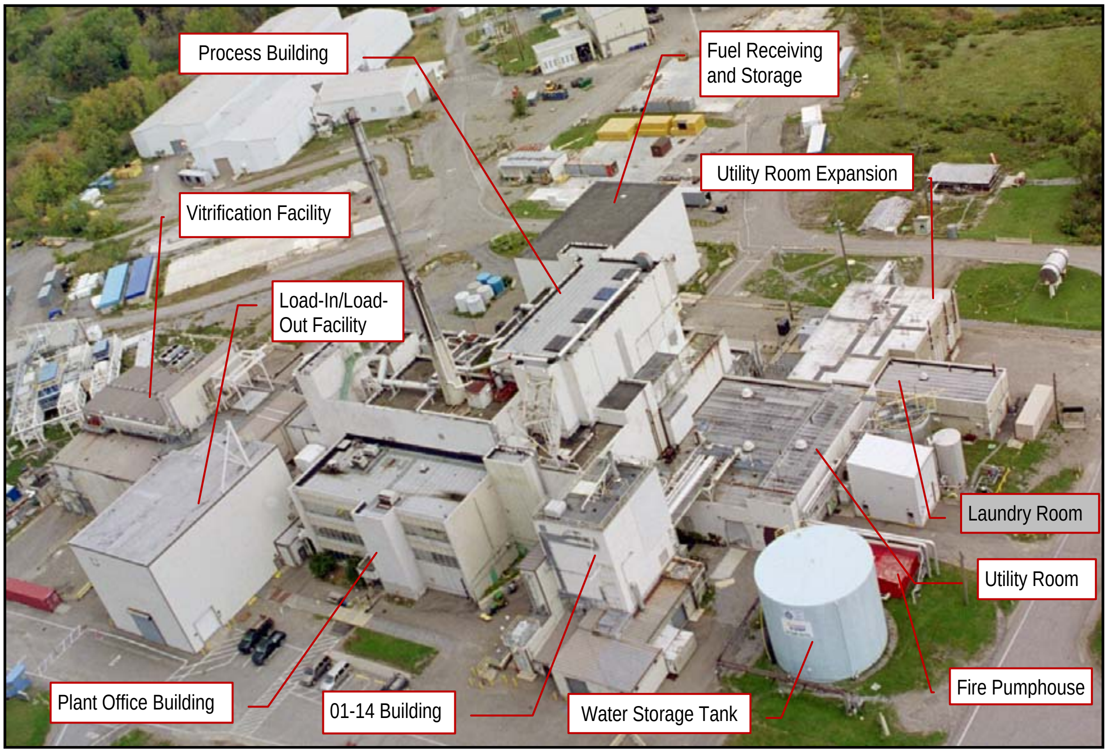


**Figure 1-3. WMA 1 Area in 2007**

**WMA 2**

The facilities that will remain in WMA 2, the Low-Level Waste Treatment Facility area, when the interim end state is reached include the five lagoons, with Lagoon 1 having been backfilled in 1984; the LLW2 Facility; the two New Interceptors; the Old Interceptor; the Neutralization Pit; the inactive Solvent Dike, the pilot permeable treatment wall; and the Maintenance Shop Leach Field. Concrete floor slabs and foundations for removed facilities such as the Maintenance Shop will also remain in place. Figure 1-4 shows this area.

One additional facility will be installed in WMA 2 as part of the work to achieve the interim end state: a full-scale permeable treatment wall to control the leading edge of the north plateau groundwater plume.

[11] The Electrical Substation, which is located behind the Process Building, cannot be seen in the photograph.

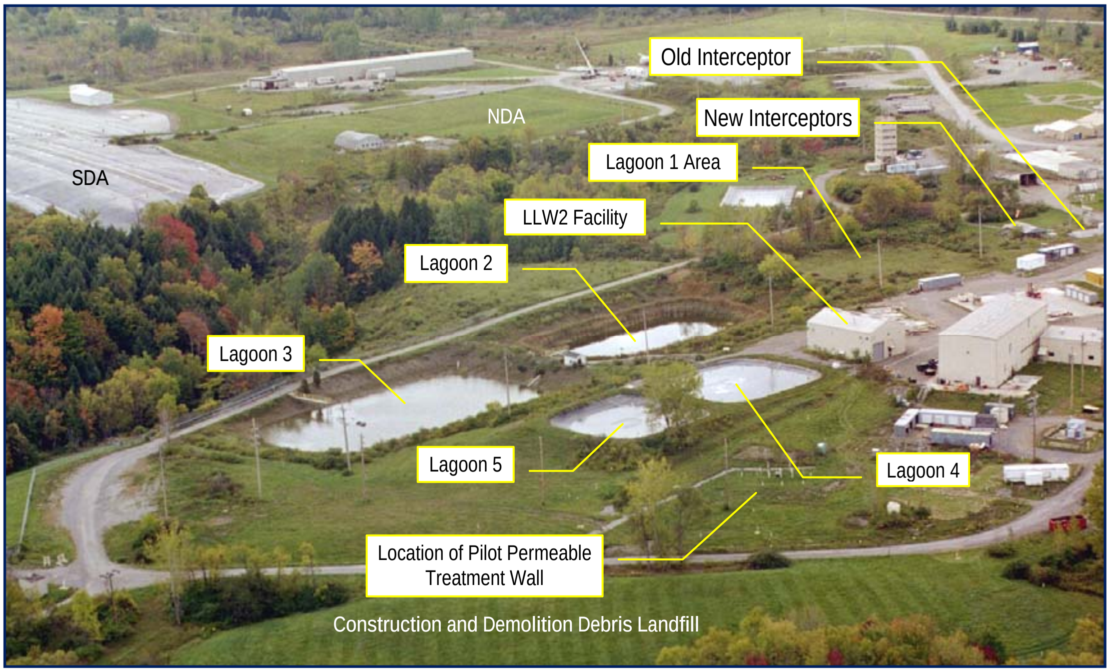


**Figure 1-4. WMA 2 in 2007**

**WMA 3**

In WMA 3, the four underground waste tanks will remain in place, along with the Permanent Ventilation System Building, the Supernatant Treatment System Support Building, the Equipment Shelter and condensers, the Con-Ed Building, and the HLW transfer trench. The tank drying system used to dry up liquid in the waste tanks may be operational[12]. The tank mobilization and transfer pumps and their support structures will remain in place.

**Other WMAs**

The closed Construction and Demolition Debris Landfill will remain in WMA 4. A full-scale passive permeable treatment wall is expected to be installed before Phase 1 of the decommissioning to mitigate the off-site migration of Sr-90 contaminated groundwater in the sand and gravel unit in the north plateau. It is expected to be composed of granular zeolite to reduce Sr-90 concentrations in groundwater through ion-exchange and be located in WMA 2 immediately south of the Construction Demolition and Debris Landfill in WMA 4.

Two buildings will remain in WMA 5, Lag Storage Addition 4 and its associated shipping depot and the Remote-Handled Waste Facility. Two structures will remain in WMA 6 along with the Equalization Basin, the Equalization Tank, and the two demineralizer sludge ponds. The Old Sewage Treatment Plant will have been completely removed.

[12] The tank and vault drying system may be operational when Phase 1 of the decommissioning begins, but some liquid is expected to still be present in the tanks at that time.

The NDA will remain in place in WMA 7, with the Interim Waste Storage Area removed and a new geomembrane cover and upgradient hydraulic barrier wall installed to control infiltration. The Radwaste Treatment System Drum Cell will remain in place in WMA 9. The New Warehouse, the Meteorological Tower, and the Security Gatehouse will remain in place in WMA 10, along with the security fence that surrounds the project premises.

### 1.10.2 Facilities and Areas Within Phase 1 Scope

Table 1-1 lists the facilities that are within the scope of Phase 1 of the decommissioning. These facilities are described in Section 3 of this plan. Figures 1-5 and 1-6 show their locations on the project premises. Remediation of surface soil and sediment on the project premises will be accomplished as indicated in the table.

The new Canister Interim Storage Facility for the vitrified HLW canisters will be constructed on the south plateau near the rail spur early in Phase 1 and the canisters moved to this location. The HLW canisters will be stored at this facility inside shielded canisters[13].

The soil and sediment characterization program described in Section 9 will be accomplished to better define the nature and extent of radioactive contamination in soil and stream sediment on the project premises. However, removal of contaminated soil excess of the cleanup goals will be limited to the areas of the major excavations in WMA 1 and WMA 2 unless DOE elects to provide for additional surface soil remediation after evaluation of the characterization data.

Before the large excavations for removal of the Process Building and the Low-Level Waste Treatment Facility shown in Figure 1-2 are filled in, Phase 1 final status surveys[14] of the excavated areas will be performed and arrangements made for regulator confirmatory surveys. The same process will be used for excavations associated with removal of concrete floor slabs, foundations, and gravel pads, which will be up to two feet deep.

Mitigative measures will be taken as described in Section 7 to eliminate or reduce potential impacts to human health and the environment during the decommissioning work and to prevent recontamination of remediated areas.

[13]Section 7 of this plan describes the general conceptual design of the new Canister Interim Storage Facility, which may be changed somewhat as the design is finalized.

[14] These surveys will be performed following guidance in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000) and the provisions of NUREG-1575, Volume 2 (NRC 2006).

**Table 1-1. Facilities and Areas Within Phase 1 Decommissioning Scope[1]**

| WMA | Facility or Area to be Removed or Remediated | Remarks |
|---|---|---|
| 1 | Process Building<br>Utility Room<br>Utility Room Expansion<br>Plant Office Building<br>Vitrification Facility<br>01-14 Building<br>Load-in/Load-out Facility<br>Fire Pumphouse<br>Water Storage Tank<br>Electrical Substation<br>Off-Gas Trench<br>Underground piping and wastewater tanks (3)<br>Other remaining concrete slabs<br>Source area of North Plateau Groundwater Plume | The HLW canisters will be moved to a new Canister Interim Storage Facility located on the south plateau.<br>All listed facilities will be removed along with the source area of the north plateau groundwater plume. A single large excavation will be dug for this purpose. A vertical hydraulic barrier wall will be installed on the north and east sides of the excavation as shown in Figure 1-2.<br>The soil in the excavated area will be removed to cleanup goals for unrestricted release.<br>The vertical hydraulic barrier wall installed on the north and east side of the excavation will remain in place. The south and west hydraulic barrier walls (sheet pilings) will be removed after the excavation is backfilled. |
| 2 | Low-Level Waste Treatment Facility Building<br>Lagoons 1 – 5<br>New Interceptors (2)<br>Old Interceptor<br>Neutralization Pit<br>Solvent Dike<br>Maintenance Shop Leach Field<br>Remaining concrete floor slabs and foundations | A single excavation will be made to remove Lagoons 1, 2, and 3, the Interceptors, the Neutralization Pit, and the Solvent Dike. Underlying soil and sediment in this excavation will be removed to cleanup goals that support unrestricted release.<br>The vertical hydraulic barrier wall shown in Figure 1-2 will remain in place. |
| 3 | Mobilization and Transfer Pumps<br>Piping and equipment in HLW Transfer Trench<br>Con-Ed Building<br>Equipment Shelter and Condensers | The support structures for the mobilization and transfer pumps will be removed as well as the pumps themselves. |
| 5 | Lag Storage Addition 4 and Shipping Depot<br>Remote-Handled Waste Facility<br>Remaining concrete floor slabs, hardstands, and gravel pads | |
| 6 | Sewage Treatment Plant<br>South Waste Tank Farm Test Tower<br>Remaining concrete floor slabs and foundations<br>Asphalt, concrete, and gravel pads[2]<br>Equalization Basin<br>Equalization Tank<br>Demineralizer Sludge Ponds (2)<br>Cooling Tower basin | The rail spur will remain operational. |
| 7 | NDA hardstand | |
| 9 | Radwaste Treatment System Drum Cell | |
| | Trench soil container area, other pads | |
| 10 | New Warehouse<br>Former Waste Management Storage Area<br>Remaining concrete floor slabs and foundations | |
| | Surface soil and sediment within the project premises | To be remediated only in the Process Building-Vitrification Facility and Low-Level Waste Treatment Facility excavation areas. Surface soil in other selected areas may also be remediated in Phase 1. |

NOTES: (1) See Section 3 of this plan for facility descriptions. (2) Including the LLW Rail Packaging and Staging Area.

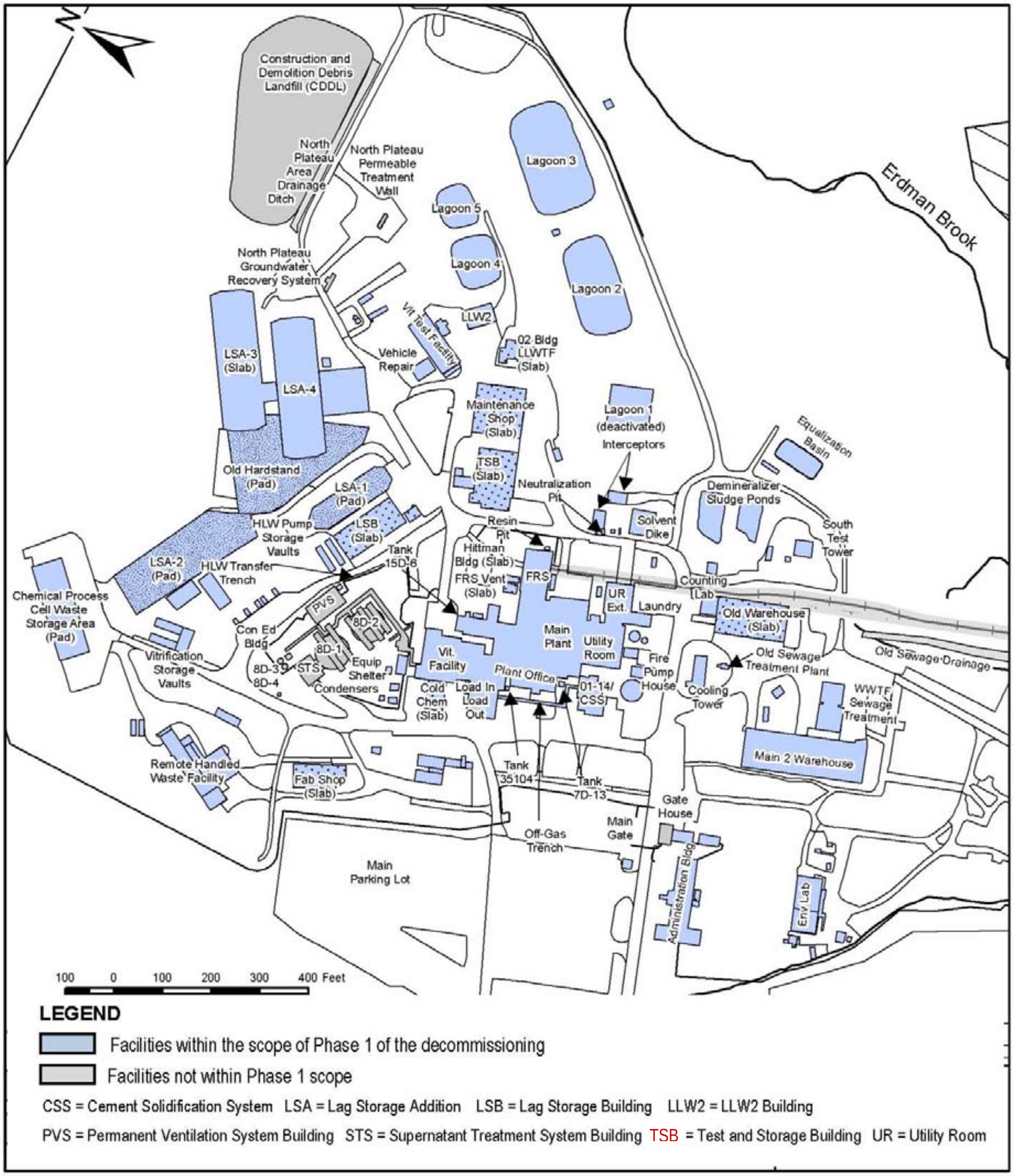


**Figure 1-5. Facilities Within the Scope of Phase 1 of the Decommissioning, North Plateau**

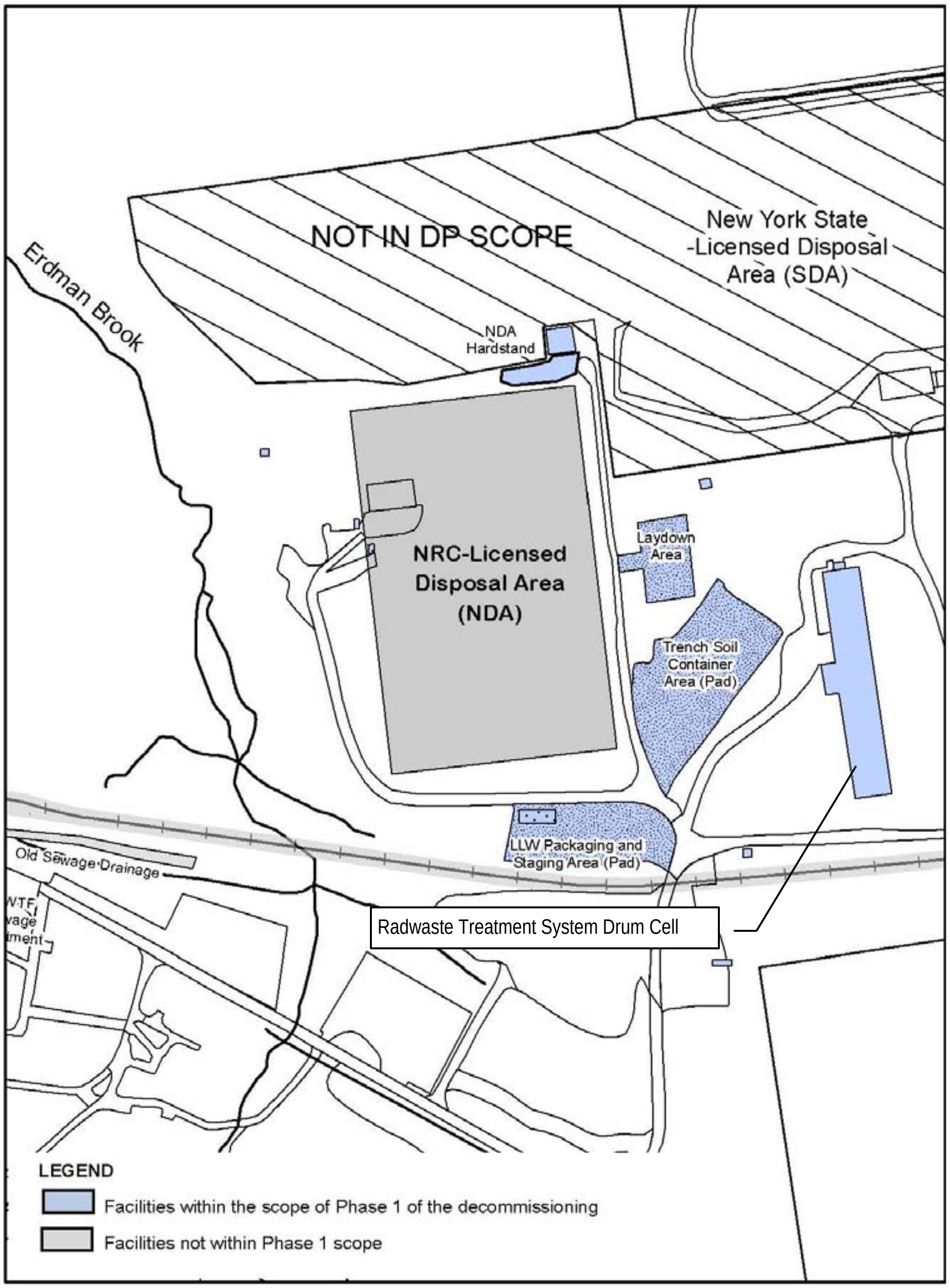


**Figure 1-6. Facilities Within the Scope of Phase 1 of the Decommissioning, South Plateau**

Figure 1-7 shows the expected appearance of the project premises in the interim end state, when Phase 1 decommissioning activities will begin.

**Figure 1-7. The WVDP in the Interim End State**

Figure 1-8 shows the planned general appearance of the project premises after completion of the Phase 1 decommissioning activities. The interim storage area for the HLW canisters will be located on the south plateau near the rail spur. Note that surface soil, subsurface soil, and stream sediment contamination will remain in some areas after Phase 1.


**Figure 1-8. The WVDP After Completion of Phase 1**

## 1.11 Organizational Responsibilities

Because the WVDP decommissioning is being carried out under the authority of the WVDP Act, organizational responsibilities are different from decommissioning of a typical

NRC-licensed site. The organizational responsibilities prescribed by the WVDP Act for decontamination and decommissioning of the WVDP are summarized below.

### 1.11.1 DOE

The Act directed the DOE to carry out the following activities: (1) Solidify the HLW, (2) develop containers suitable for permanent disposal of the solidified HLW waste, (3) transport the waste to a federal repository for permanent disposal, (4) dispose of LLW and transuranic waste produced in the solidification of the HLW, and (5) decontaminate and decommission the tanks, facilities, materials, and hardware used in the project in accordance with requirements prescribed by the NRC.

The Act also directed DOE to enter into a cooperative agreement with the State for the State to make available to DOE the facilities and HLW necessary to carry out the project, without transfer of title, with DOE providing technical assistance in securing required license amendments. The Act directed DOE to enter into an agreement with the NRC for review and consultation on the project by NRC and to afford NRC access to the site to monitor activities under the project for the purposes of health and safety. Both of these agreements were formalized in 1981 (DOE and NYSERDA 1981, DOE and NRC 1981).

The Act further directed DOE to consult with the EPA in carrying out the project. Under the WVDP Act, DOE is responsible for the activities outlined above and for determining the manner in which facilities, materials, and hardware for which DOE is responsible are managed or decommissioned, in accordance with applicable federal and state requirements. To this end, DOE will determine what, if any, material or structures for which DOE is responsible will remain on site and what, if any, institutional controls, engineered barriers, or stewardship provisions will be needed.

The Act also set up a cost sharing arrangement for the WVDP, with DOE paying 90 percent of the total project costs and the State paying 10 percent of these costs.

DOE is responsible as noted previously for certain matters associated with the decommissioning: (1) project management and the decommissioning organization, (2) safety and health, (3) waste management, and (4) environmental protection.

### 1.11.2 NRC

The WVDP Act gave NRC the authority to prescribe requirements for decontamination and decommissioning and to review and consult with DOE, not to include formal procedures or actions pursuant to the Atomic Energy Act or any other law. It also gave NRC monitoring responsibilities for the purpose of assuring public health and safety. Pursuant to these responsibilities, NRC will issue public reports during decommissioning to document its position with respect to DOE compliance with NRC decommissioning criteria. The WVDP Act does not give NRC licensing authority over DOE.

Consistent with its role in the project, NRC submitted requests for additional information in connection with its review of Revision 0 and Revision 1 of this plan. DOE evaluated these requests for additional information, provided written responses to NRC, and has incorporated the related changes in Revision 2 to this plan.

NRC is also a cooperating agency in development of the Decommissioning EIS, as mentioned previously.

### 1.11.3 NYSERDA

As explained in the NRC Implementation Plan (NRC 2003b), NYSERDA will determine the manner in which facilities and property for which NYSERDA is responsible are managed and decommissioned, in accordance with applicable federal and state requirements. To this end, NYSERDA will determine what, if any, material or structures for which it is responsible will remain on the site and what, if any, institutional controls, engineered barriers, or stewardship provisions will be needed.

The NRC Implementation Plan also indicates that if NYSERDA decides to terminate the license after DOE completes decommissioning activities for the project facilities, NYSERDA will be required to submit a decommissioning plan. As noted previously, NYSERDA is jointly preparing the Decommissioning EIS with DOE.

## 1.12 Organization of this Plan

The organization and content of this plan are generally consistent with Volume 1 of NUREG-1757 (NRC 2006). Differences are described in Appendix A, which consists of an annotated version of the decommissioning plan evaluation checklist found in Appendix D to NUREG-1757, Volume 1 (DOE 2006). NRC has concurred with certain topics not being applicable to this decommissioning as shown in the Appendix A checklist (NRC 2008).

The contents of the plan are described in the Table of Contents. To aid readability, certain details appear in appendices.

## 1.13 Control of Changes

DOE plans to treat this plan as a “living document,” revising it when circumstances warrant. DOE may issue revisions to make significant changes that could affect the project end conditions. Such revisions will be provided to NRC for review and comment prior to issue. After NRC comments are incorporated or otherwise formally resolved, DOE will issue the revised plan.

DOE may make changes to the plan that could not affect the project end conditions without providing them to NRC for review and comment. DOE will informally consult with NRC on such changes prior to issue to ensure that NRC concurs that the changes could not affect project end conditions. NRC will be provided copies of such changes when they are issued. Examples of such changes could include:

- A change to reflect actual conditions of a particular facility at the end of deactivation work planned for the 2008 – 2011 period,
- A change in decontamination methods, or

- A change to include information on additional ALARA analyses performed after decommissioning activities began that did not result in a change to the decommissioning approach.

As indicated previously, DOE may elect to remediate surface soil in certain areas of the project premises during Phase 1 after evaluation of characterization data. DOE would notify NRC in advance of such remediation, but no change to the Decommissioning Plan would be required for this activity.

## 1.14 References


**Federal Statutes**

Clean Air Act of 1970, as amended.

Clean Water Act (Federal Water Pollution Control Act) of 1977.

Toxic Substances Control Act of 1976, as amended.

Resource Conservation and Recovery Act of 1976, as amended.

West Valley Demonstration Project Act, Public Law 96-368 (S. 2443), of October 1, 1980 (and related legislative history).

**Code of Federal Regulations and Federal Register Notices**

10 CFR 20, *Standards for Protection Against Radiation*

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

10 CFR 830, *Nuclear Safety Management.*

10 CFR 835, *Occupational Radiation Protection*

29 CFR 1910.134, *Respiratory Protection.*

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

40 CFR 141, *National Primary Drinking Water Regulations*.

40 CFR 143, *National Secondary Drinking Water Regulations*.

42 FR 26951, Executive Order 11988, *Floodplain Management*. Federal Register, May 24, 1977.

42 FR 26961, Executive Order 11990, *Protection of Wetlands*. Federal Register, May 24, 1977.

58 FR 150, Executive Order 12856, *Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements*. Federal Register, August 6, 1993.

63 FR 179, Executive Order 13101, *Greening the Government through Waste Prevention, Recycling, and Federal Acquisition*. Federal Register, September 16, 1998.

65 FR 81, Executive Order 13148, *Greening the Government through Leadership in Environmental Management*. Federal Register, April 26, 2000.

67 FR 22, *Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site; Final Policy Statement*. Federal Register, February 1, 2002.

68 FR 49, *Notice of Intent to Prepare an Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center*. Federal Register, March 13, 2003.

70 FR 115*, West Valley Demonstration Project Waste Management Environmental Impact Statement, Record of Decision.* Federal Register, June 16, 2005.

**DOE Orders, Policies, Manuals, Standards, and Guides**

DOE Order 414.1C, Change 1, *Quality Assurance*. U.S. Department of Energy, Washington, D.C., June 17, 2005.

DOE Order 420.1B, *Facility Safety*. U.S. Department of Energy, Washington, D.C., December 22, 2005.

DOE Order 430.1B, *Real Property Asset Management*. U.S. Department of Energy, Washington, D.C., February 8, 2008.

DOE Order 435.1, Change 1, *Radioactive Waste Management*. U.S. Department of Energy, Washington, D.C., August 28, 2001.

DOE Order 440.1B, *Worker Protection Management for DOE Federal and Contractor Employees.* U.S. Department of Energy, Washington, D.C., May 17, 2007.

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

DOE Order 451.1B, Change 1, *National Environmental Policy Act Compliance Program*. U.S. Department of Energy, Washington, D.C., September 28, 2001.

DOE Order 460.1B, *Packaging and Transportation Safety*. U.S. Department of Energy, Washington, DC, April 4, 2003.

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.

DOE Order 5480.20A, *Personnel Selection, Qualification, and Training Requirements for Nuclear Facilities*. U.S. Department of Energy, Washington, D.C., November 15, 1994.

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

DOE Policy 450.4, *Safety Management System Policy*. U.S. Department of Energy, Washington, D.C., October 15, 1996.

DOE Manual 231.1-1A, Change 1, *Environment, Safety, and Health Reporting Manual*. U.S. Department of Energy, Washington, D.C., June 3, 2004.

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

DOE Guide 435.1-1, *Implementation Guide for Use with DOE M 435.1*. U.S. Department of Energy, Washington, D.C., July 9, 1999.

DOE Guide 460.1-1, *Implementation Guide for Use with DOE Order 460.1A.* U.S. Department of Energy, Washington, D.C., June 5, 1997.

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# 2.0 FACILITY OPERATING HISTORY

**PURPOSE OF THIS SECTION**

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

**INFORMATION IN THIS SECTION**

This section provides the following information:

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

**RELATIONSHIP TO OTHER PLAN SECTIONS**

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

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

## 2.1 License History

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

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

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

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

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

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

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

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

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

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

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

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

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

From 1966 to 1972, NFS reprocessed under the license more than 600 metric tons of spent fuel in the Process Building (Table 2-4) and generated approximately 600,000 gallons of liquid high-level waste. The facility shut down in 1972 for modifications to increase reliability and to expand capacity. In 1976, without restarting the operation, NFS informed New York State that it intended to withdraw from the reprocessing business and not renew the lease when the initial term expired at the end of 1980. The WVDP Act was enacted in 1980 providing for solidification of the HLW from reprocessing, then decontamination and decommissioning of the facilities used in the solidification effort. In February of 1982, NFS transferred possession of the reprocessing facilities to DOE for that purpose.

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

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

On February 11, 1982, the NRC issued License Amendment 32, as requested by NFS, to terminate the authority and responsibility of NFS under the license effective upon DOE assumption of exclusive possession of the project premises. Control of the project premises was formally transferred to DOE effective February 26, 1982 (WVNSCO 1983a). Section 2.1.1 describes NFS activities under the license in more detail. As noted in Section

1, portions of NYSERDA's NRC Part 50 license for the Center, including the technical specifications, have been effectively suspended by NRC since 1981 to facilitate execution of the provisions of the WVDP Act.

### 2.1.1 Nuclear Fuel Services Operations From 1966 to 1982

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

**Receiving Fuel for Reprocessing**

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

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

**Process Building Arrangements**

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

**Sectioning and Dissolving the Fuel**

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

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

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

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

**Separating Uranium, Plutonium, and Fission Products**

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

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

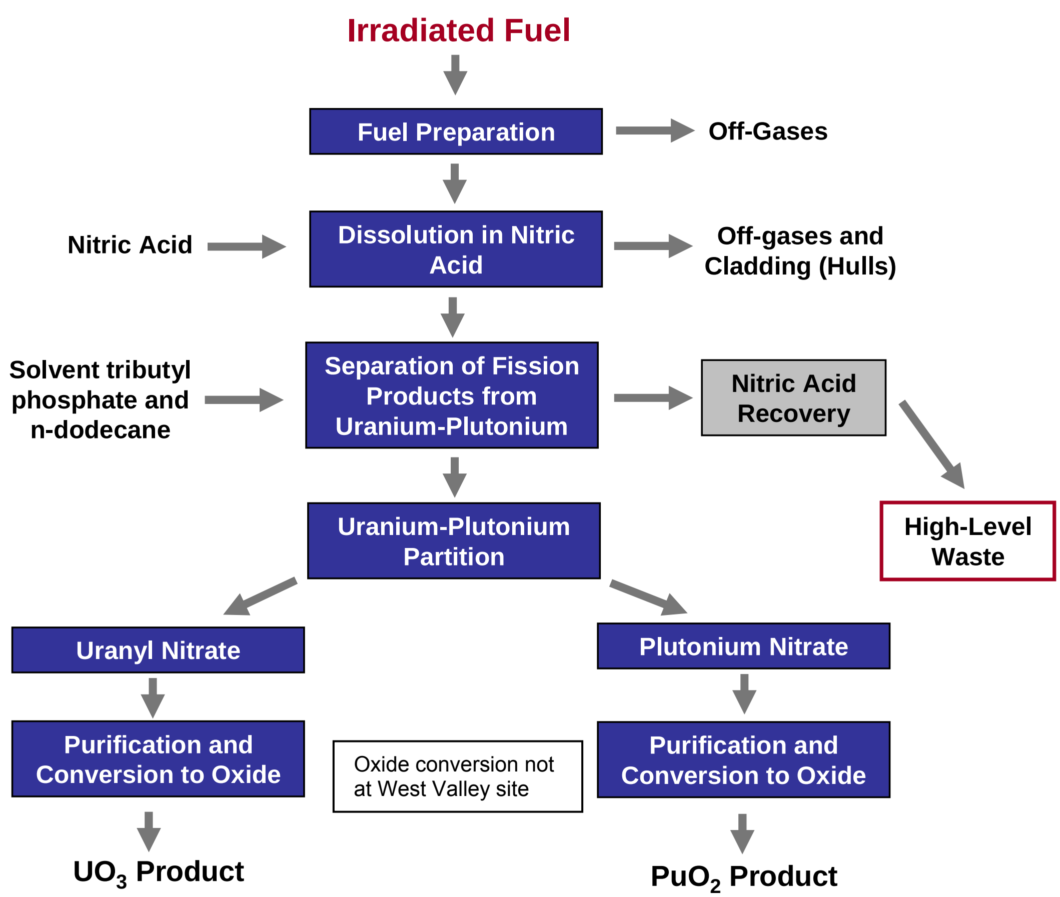


**Figure 2-1. Spent Fuel Reprocessing Diagram (PUREX Process)**

[1] Plutonium uranium refining by extraction.

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

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

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

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

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

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

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

The Solvent Waste Catch Tank received the spent sodium carbonate and dilute nitric acid wash solutions that were used in the solvent cleanup system. The sodium carbonate and nitric acid washes used in the solvent cleanup were also collected in the Waste Catch Tank and then transferred to the Solvent Waste Hold Tank where they were sampled and

[2] Actinides were also precipitated out into the sludge. Table 4-9 shows estimates of residual radioactivity in the underground waste tanks as of 2011.

[3] Thorium reduction extraction.

subsequently sent through normal plant waste processing (Tank 8D-2 or LLW treatment) depending on their radioactivity concentration.

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

**Use of the Low-Level Waste Treatment Facility**

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

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

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

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

| Radionuclide | Half-Life (Year)(2) | Tank 8D-2 Supernatant(3) | Tank 8D-2 Sludge(4) | Tank 8D-4(5) | Total |
|---|---|---|---|---|---|
| C-14 | 5.7E+03 | 1.4E+02 | 9.9E-02 | 1.3E-01 | 1.4E+02 |
| Sr-90 | 2.9E+01 | 3.0E+03 | 6.9E+06 | 4.5E+05 | 7.4E+06 |
| Tc-99 | 2.1E+05 | 1.6E+03 | Note (6) | 1.0E+02 | 1.7E+03 |
| I-129 | 1.6E+07 | 2.1E-01 | Note (6) | <1.8E-01 | 3.9E-01 |
| Cs-137 | 3.0E+01 | 7.4E+06 | Note (6) | 4.8E+05 | 7.9E+06 |
| U-232 | 7.2E+01 | Note (6) | 4.4E+00 | 2.7E+00 | 7.1E+00 |
| U-233 | 1.6E+05 | 5.0E-01 | 6.9E+00 | 2.1E+00 | 9.5E+00 |
| U-234 | 2.5E+05 | 3.0E-01 | 4.0E+00 | 2.2E-01 | 4.5E+00 |

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

| Radionuclide | Half-Life (Year)(2) | Tank 8D-2 Supernatant(3) | Tank 8D-2 Sludge(4) | Tank 8D-4(5) | Total |
|---|---|---|---|---|---|
| U-235 | 7.0E+08 | 6.5E-03 | 8.9E-02 | 5.2E-03 | 1.0E-01 |
| U-238 | 4.5E+09 | 5.8E-02 | 7.9E-01 | 7.1E-05 | 8.5E-01 |
| Np-237 | 2.1E+06 | Note (6) | 2.6E+01 | 3.0E-01 | 2.6E+01 |
| Pu-238 | 8.8E+01 | 1.3E+02 | 6.5E+03 | 4.8E+02 | 7.1E+03 |
| Pu-239 | 2.4E+04 | 2.5E+01 | 1.7E+03 | 1.5E+01 | 1.7E+03 |
| Pu-240 | 6.6E+03 | 1.9E+01 | 1.3E+03 | 8.1E+00 | 1.3E+03 |
| Pu-241 | 1.4E+01 | 1.6E+03 | 8.5E+04 | 8.5E+02 | 8.7E+04 |
| Am-241 | 4.3E+02 | Note (6) | 6.9E+04 | 2.4E+02 | 6.9E+04 |
| Cm-243 | 2.9E+01 | Note (6) | 3.1E+01 | 2.3E-01 | 3.1E+01 |
| Cm-244 | 1.8E+01 | Note (6) | 2.0E+04 | 1.4E+01 | 2.0E+04 |

NOTES:
(1) From Rykken 1986. This report provides estimates for numerous other radionuclides as well the key radionuclides included in this table.
(2) Half-life values from Grove Engineering 2003.
(3) From Rykken 1986, Table 6, as of 7/1/86.
(4) From Rykken 1986, Table 22 as of 7/1/87.
(5) From Rykken 1986, Table 12 as of 7/1/87.
(6) Not reported.

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

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

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

| Compound | % (weight of compound/total weight of supernatant) Wet Basis | % (weight of compound/total weight of compounds) Dry Basis | Total Weight of compounds in Supernatant (Kg) |
|---|---|---|---|
| $Na_3PO_4$ | 0.133 | 0.34 | 3,799 |
| $Na_2MoO_4$ | 0.0242 | 0.06 | 691 |
| $Na_3BO_3$ | 0.0209 | 0.05 | 597 |
| $CsNO_3$ | 0.0187 | 0.05 | 534 |
| NaF | 0.0176 | 0.04 | 503 |
| $Sn(NO_3)_4$ | 0.00859 | 0.02 | 245 |
| $Na_2U_2O_7$ | 0.00808 | 0.02 | 231 |
| $Si(NO_3)_4$ | 0.00806 | 0.02 | 230 |
| $NaTcO_4$ | 0.00620 | 0.02 | 177 |
| $RbNO_3$ | 0.00416 | 0.01 | 119 |
| $Na_2TeO_4$ | 0.00287 | 0.007 | 82 |
| $AlF_3$ | 0.00271 | 0.007 | 77 |
| $Fe(NO_3)_3$ | 0.00152 | 0.004 | 43 |
| $Na_2SeO_4$ | 0.00054 | 0.001 | 15 |
| $LiNO_3$ | 0.00048 | 0.001 | 14 |
| $H_2CO_3$ | 0.00032 | 0.0008 | 9 |
| $Cu(NO_3)_3$ | 0.00022 | 0.0005 | 6 |
| $Sr(NO_3)_2$ | 0.00013 | 0.0004 | 4 |
| $Mg(NO_3)_2$ | 0.0008 | 0.0002 | 2 |
| **Compound Totals** | **39.53 %** | **100.00 %** | **1,129,038** |
| **Total $H_2O$ (100% - 39.53%)** | **60.47 %** | **NA** | **1,727,164** |

NOTE: (1) From Eisenstatt 1986.

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

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

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

**Table 2-8. Chemical Composition of Tank 8D-4 Waste at the Completion of Reprocessing[1]**

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

NOTE: (1) From Eisenstatt 1986. LEGEND: X = Am-241, Am-243, Cm-242, Cm-243, Cm-244, Cm-245.

### 2.1.2 West Valley Demonstration Project From 1982 to 2008

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

#### Integrated Radwaste Treatment System

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

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

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

#### Vitrification Facility

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

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

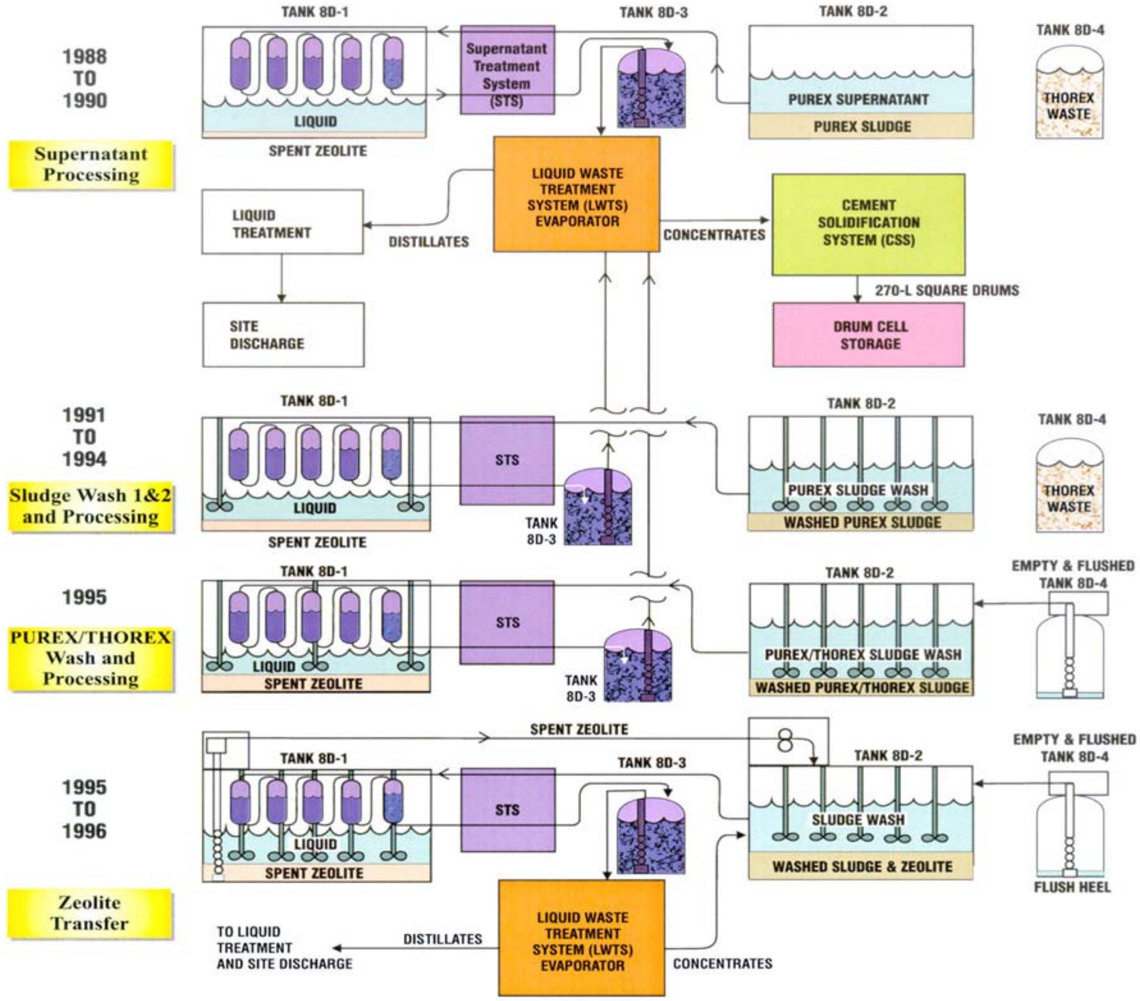


**Figure 2-2. Simplified HLW Pretreatment Process Diagram**

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

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

**Solidification Activities**

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

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

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

A total of 275 canisters of HLW were produced. Two additional canisters were filled with materials which remained in the melter. The solidification of the liquid HLW waste was completed in September 2002 and the Vitrification Facility was radiologically characterized in November 2002 (Lachapelle 2003)[4].

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

**Sodium-Bearing Waste**

As a component of tank management over time, sodium salts were added to the Tanks 8D-1 and 8D-2 to limit corrosion of the carbon steel tanks. More recently, clean utility water used to cool the in-tank mobilization pumps added excess fluids to the tanks before and during vitrification. Since sodium is a limiting ingredient in a qualified glass recipe, the high-sodium water was segregated from the HLW feed mixture. A process was developed to volume-reduce the waste water containing high levels of sodium and solidify the 11,500 gallons of concentrate into a form suitable for LLW land disposal. The solidification was completed within the 01-14 building in 2004, and the sodium-bearing waste was shipped for disposal in 2007. (Rowell 2001, WVNSCO and URS 2005, Bower 2008)

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

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

**Liquid LLW Streams**

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

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

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

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

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

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

NOTE: (1) From Eisenstatt 1986.

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

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

NOTE: (1) From WVNSCO 2007a

## 2.2 Site Decontamination Activities (1966 – 2011)

This section summarizes remediation activities[5] performed by NFS, those that have been performed by the WVDP, and those that will be performed by the WVDP to establish the interim end state before the beginning of activities under this plan. Although the WVDP remediation activities have generally been performed in connection with cleanup, modifications, or deactivation work, they are relevant to the starting point for the decommissioning.

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

### 2.2.1 NFS Remediation Activities (1966 – 1981)

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

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

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

**Table 2-11. Activity Removed by NFS for the Period 1972 Through 1977**[1]

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

NOTES: (1) From Riethmiller 1981.

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

### 2.2.2 WVDP Remediation Activities (1982 – 2011)

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

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

**List of Facilities Remediated or to be Remediated**

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

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

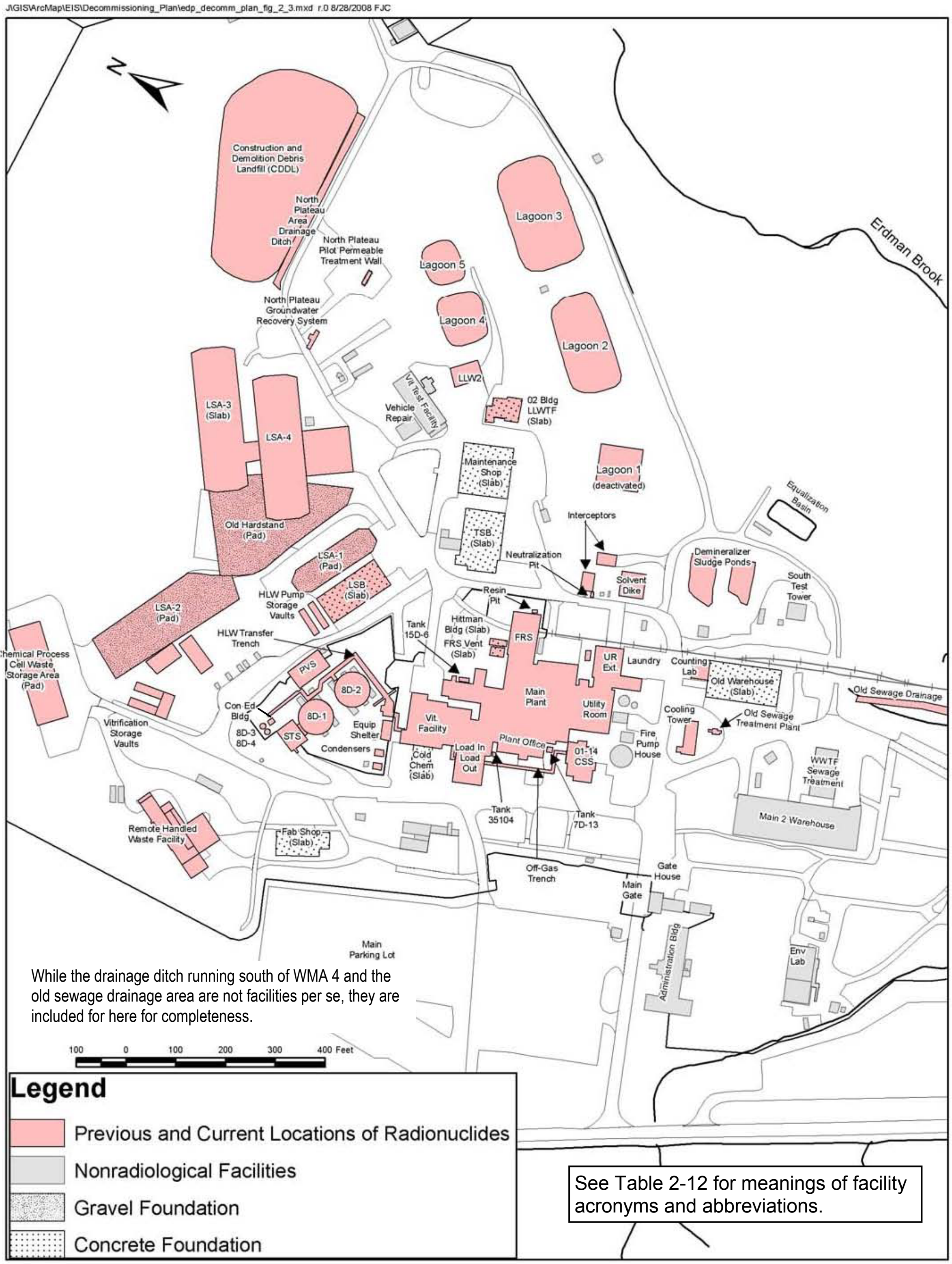


**Figure 2-3. Previous and Current Locations of Radionuclides in North Plateau Facilities at the WVDP**

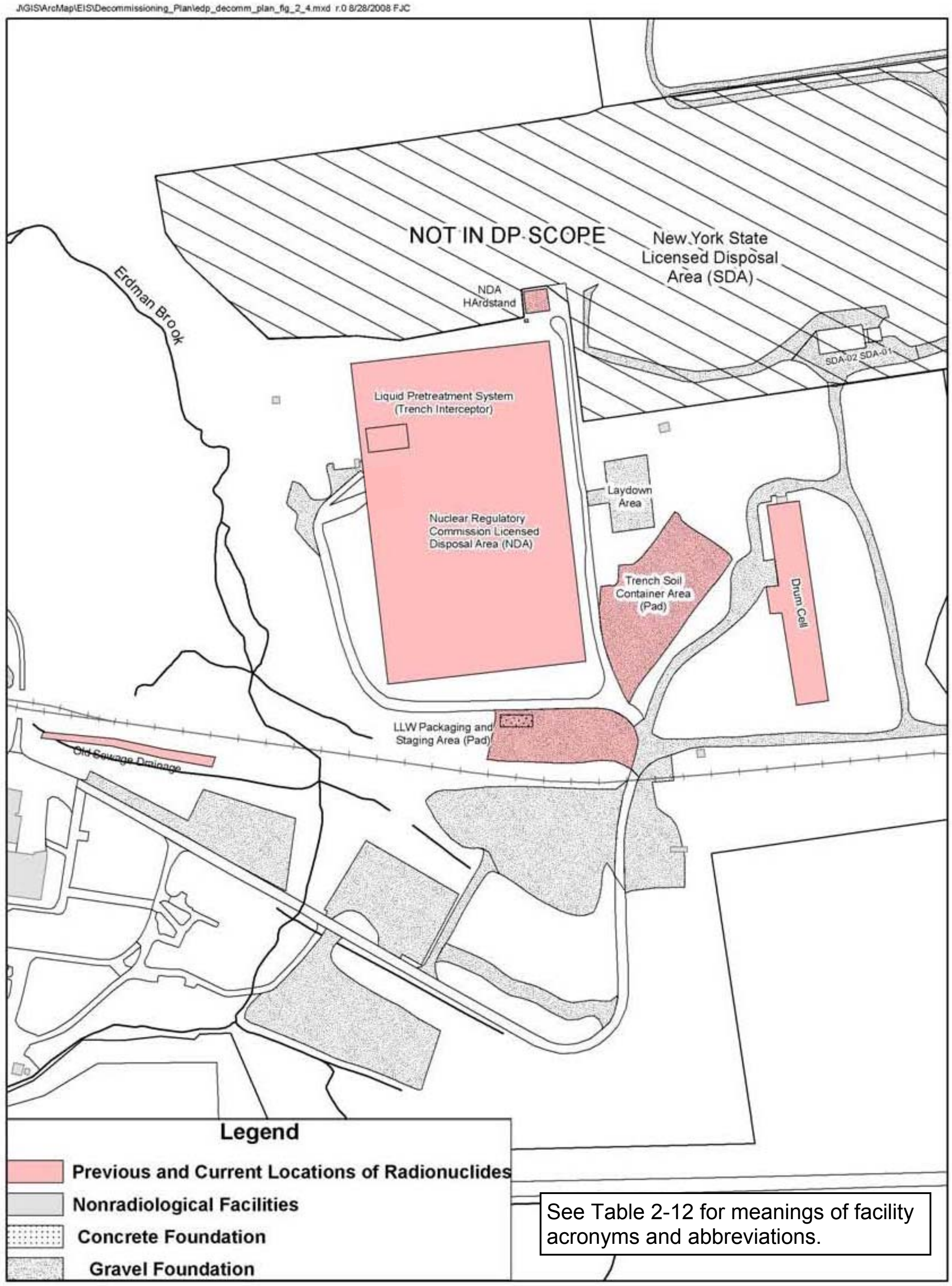


**Figure 2-4. Previous and Current Locations of Radionuclides in South Plateau Facilities at the WVDP**

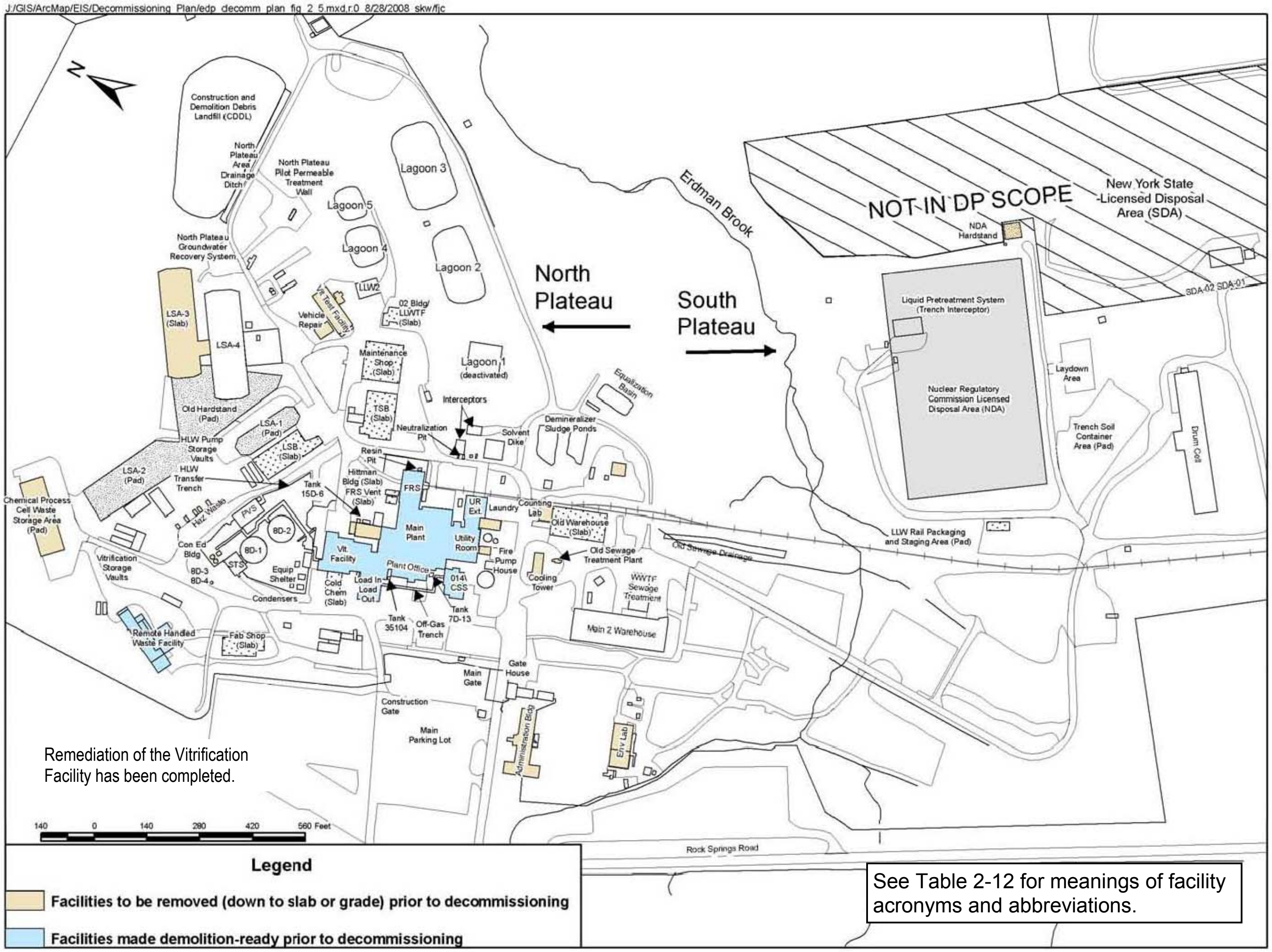


**Figure 2-5. Locations of Planned Remediation Activities for Site Facilities Prior to Phase 1 of the Decommissioning**

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

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

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

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

**Table 2-13. Facilities Remediated or to be Remediated by the WVDP Before Decommissioning(1)**

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

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

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

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

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

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

**Information in Table 2-13**

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

**Discussion of WVDP Remediation Efforts**

Historical remediation activities are summarized in Section 2.2. Areas in which initial deactivation work was completed in late 2004 include three cells in the Process Building: the General Purpose Cell, the Process Mechanical Cell, and Extraction Cell 2. Remediation of WVDP facilities continued in 2009.

Work accomplished in 2009 included removal of equipment from the Acid Recovery Cell, Extraction Cell 3, and the Hot Acid Cell. A two-inch layer of grout was applied to the floor of the Acid Recovery Cell. Some pump niches in the Upper Warm Aisle were decontaminated and deactivated. The Product Purification Cell was partially decontaminated. In late 2009, decontamination of the Process Mechanical Cell Crane Room was underway. Additional decontamination is planned for the floors and walls of the General Purpose Cell and the Process Mechanical Cell.

Deactivation of the Vitrification Cell in the Vitrification Facility was completed in 2005. In 2009, the cell was being used for remote sorting and packaging of radioactive waste so conditions in this area are subject to change and additional decontamination may be performed before Phase 1 of the decommissioning.

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

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

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

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

NOTES:
(1) The values in this table, with the exception noted in note (6) below, apply to radioactive contamination deposited on, but not incorporated into the interior or matrix of, the contaminated item. Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides apply independently.

(2) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

(3) The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 $cm^2$ is less than three times the value specified. For purposes of averaging, any square meter of surface shall be considered to be above the surface contamination value if: (1) from measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 $cm^2$ area exceeds three times the applicable value.

(4) The amount of removable radioactive material per 100 $cm^2$ of surface area should be determined by swiping the area with dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note - The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 $cm^2$ is determined, the activity per unit area shall be based on the actual area and the entire surface shall be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

(5) This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

(6) Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply. In certain cases, a “Total” value of 10,000 dpm/100 $cm^2$ may be applicable either to metals of the types from which insoluble special tritium compounds (STCs) are formed, that have been exposed to tritium, or to bulk materials to which insoluble special tritium compound particles are fixed to a surface.

(7) These limits apply only to the alpha emitters within the respective decay series.

**Table 2-15. Radiological Areas and Radioactive Material Areas**[1]

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

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

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

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

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

## 2.3 Spills and Uncontrolled Release of Radioactivity

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

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

### 2.3.1 North Plateau Groundwater Plume

The north plateau groundwater plume is a zone of groundwater contamination. Based on characterization data collected in 2008, the plume is approximately 600 feet wide by 1,400 feet long extending northeastward from the Process Building in WMA 1 to the Construction and Demolition Debris Landfill in WMA 4, where it splits into three distinct lobes as shown in Figure 2-6. The 2008 data, which are discussed in Section 4, indicate that the center leading edge lobe extends to the central portion of the Construction and Demolition Debris Landfill (WVES 2009).

Lagoon 1 is also a possible contributor of gross beta activity in part of the plume, at least in this lagoon's immediate vicinity (WVES and URS 2009).

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

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

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

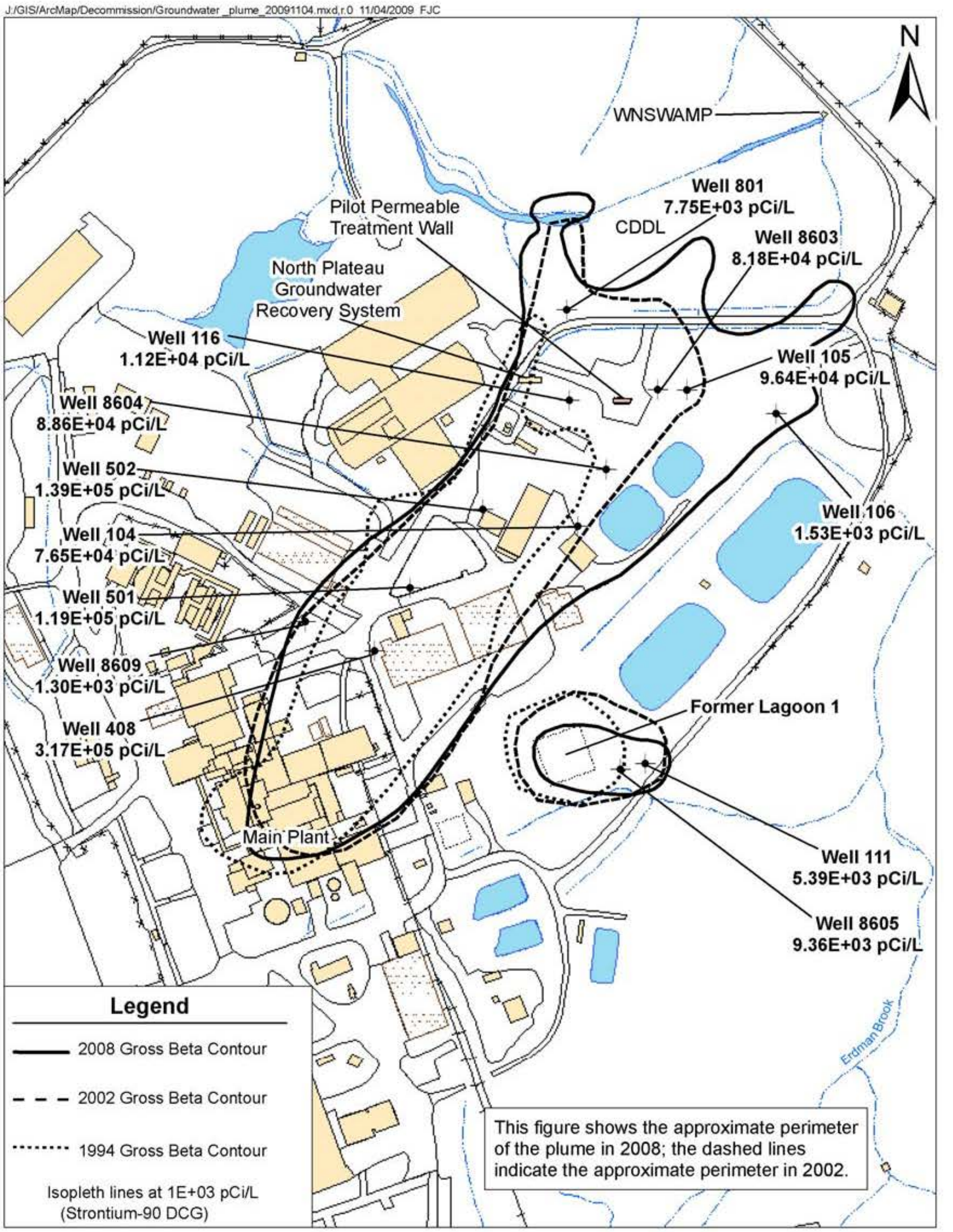


**Figure 2-6. Sr-90 Groundwater Plume on the North Plateau**

Mobile radionuclides such as H-3, Sr-90, and Tc-99 have migrated with groundwater along the northeast groundwater flow path in the north plateau. The Lagoon 1 design (to allow liquid to seep from the impoundment while retaining sediment and non-aqueous contaminants inside the basin) allowed tritiated water, originally containing about 6,000 curies of tritium in leachate pumped from the SDA for treatment, to infiltrate areas of the north plateau groundwater in the mid-1970s (Smokowski 1977). These conditions were an unintended consequence of the lagoon design, and resulted in an extensive investigation by NFS, extending through the transfer of operational control to DOE in the early 1980s (Marchetti 1982).

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

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

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

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

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

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

NOTE: (1) From Westcott 1998. Note that the values in Table 2-16 are based on a 1998 estimate of radioactivity in soil and groundwater beneath and downgradient of the Process Building that did not take into account radioactivity in groundwater that may have seeped to the surface and entered ditches or streams.

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

In 2008 and 2009, data were collected to support design of a full-scale permeable treatment wall to be installed to mitigate the leading edge of the plume (WVES 2009). These data included concentrations of Sr-90 in subsurface soil and groundwater from samples collected in WMA 2, WMA 4, and WMA 5 as discussed in Section 4.

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

### 2.3.2 Old Sewage Plant Drainage

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

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

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

In order to prevent further contaminant transport downstream, a new drainage channel was excavated to the west of the contaminated drain, and the spoil was placed over the old channel. At least three feet of soil covers the old drainage channel sediment.[6] Some drainage near the old outfall exhibits residual surface contamination. (See Section 4.)

### 2.3.3 The Cesium Prong

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

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

[6] Section 5 describes cleanup goals for surface soil (within one meter, or approximately 3 feet of the surface) and for subsurface soil in the deep WMA 1 and WMA 2 excavations. Section 5 does not provide cleanup goals for near surface soil contamination below 3 feet from the surface such as that expected to be present in the old drainage channel. Remediation of this contamination is not within the scope of Phase 1 decommissioning activities.

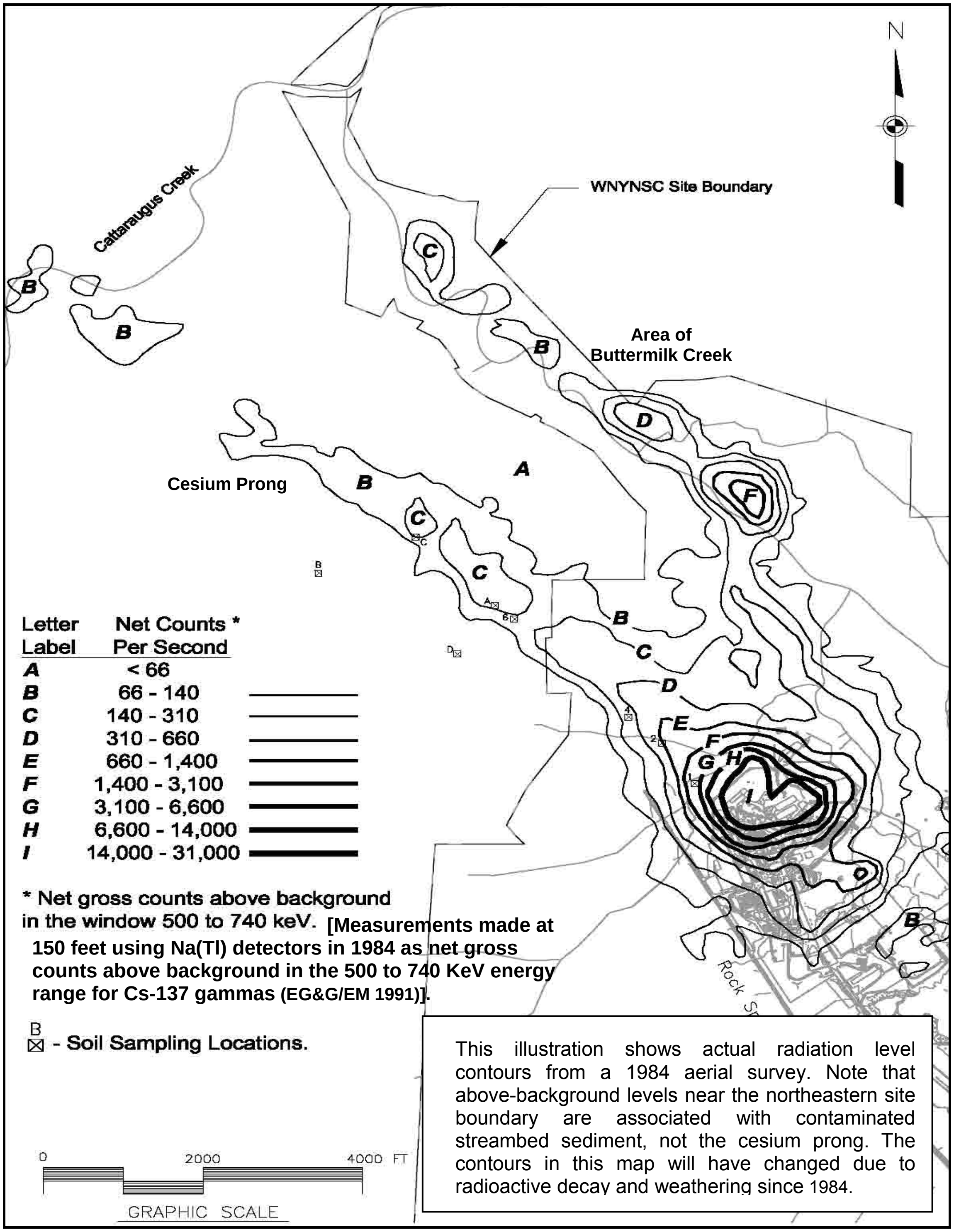


**Figure 2-7. 1984 Aerial Radiation Survey Isopleths of the WVDP and Surrounding Area**

### 2.3.4 Summary of Spills During NFS Operations

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

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

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

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

| Release Event and Origin Location | Principal Radionuclides | | | |
|---|---|---|---|---|
| | Type | Form | Activity or Concentration | Documentation Notes |
| In 1967, a line from the in-cell LLW Evaporator to acid recovery failed in-cell during waste transfer to Tank 8D-2. WMA 1: ARPR in southwest corner of Process Building.<br>This release may have impacted soil and groundwater under this part of the Process Building. | Fission products and transuranics from spent fuel | Liquid to soil, groundwater | Leakage resulted in 555 gallons of liquid waste entering the ARPR sump and draining to the Old Interceptor (sufficient to read >~ 100 mR/hr at the interceptor), and requiring pumpout back to the Process Building for treatment. This event led to installation of 12 inches of concrete shielding on the Interceptor floor. A radiation level of 408 mR/h was measured in the Interceptor in 2003. | Line 7P-170-2A failed in-cell on 2/14/67. Reported by NFS [7], [5] |
| Sanitary sewer line leak near Process Building allowed contaminated groundwater to affect Sewage Treatment Plant. WMA 1: in-leakage near southwest side of Process Building. | Cs-137, Sr-90, I-129 | Liquid to soil, groundwater, eventual release to Erdman Brook impacting water, sediment. | Estimated 0.052 Ci Sr-90 released: sewage treatment outfall area soil contaminated to 1 mR/h. | Sewage Treatment Plant and outfall drainage were contaminated to low levels, effluent concentrations subsiding after leak was repaired.<br>Reported by NFS [8], [14] |
| Overflow of Lagoons 4 and 5: treated water released to local soil and groundwater. WMA 2: northeast of the O2 Building. | Cs-137, Sr-90 | Liquid to soil, groundwater | Unknown amount and activity: probably close to free release level of < 3E-7 µCi /mL. | Temporary loss of Lagoon 3 capacity allowed overflow of releasable treated effluent to occur at an unplanned location.<br>Reported by NFS [9] |

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

| Release Event and Origin Location | Principal Radionuclides | | | |
|---|---|---|---|---|
| | Type | Form | Activity or Concentration | Documentation Notes |
| Leakage from waste containers or fuel casks contaminated asphalt "Old Hardstand" north of the Process Building. WMA 5: footprint located west of LSA 3 and LSA 4. | Fission products and transuranics from spent fuel | Liquid to soil, groundwater | Unknown amount and activity of leaks: maximum surface reading was 100 mR/hr on localized surfaces. Material was removed and placed in Lagoon 1 in 1984. Approximately 1,700 cubic yards of removed material, <10,000 dpm/g beta-gamma, <2,000 dpm alpha. (11) | Leakage from waste transport trailers parked on the hardstand contaminated the asphalt surface. Runoff contaminated the adjacent soil and drainage ditch. Noted, but not detailed during 1982 environmental characterization.(8) Significant contamination was noted in 1983.(10) |
| Cesium prong created by particulate deposition following 1968 dissolver off-gas HEPA filter failure. WMA 1, 3, 4, 5, 10: general deposits to the north-northwest of the Process Building. Detectable deposits extend several miles (outside the scope of this plan). | Cs-137 | Airborne particulate to exposed surfaces, soil | Approximately 0.33 Ci particulate gross beta radioactivity released. Offsite- 44 pCi/g localized; 21pCi/g averaged over 2,500 $m^2$ (26,900 $ft^2$). Offsite data from Luckett.(13) | Several events contributed to the deposits. A DOG filter failure in March, and a main plant filter failure in September appear to have been the main sources of the observed depositions. Reported by NFS (12),(8) |

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

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

### 2.3.5 WVDP Spills

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

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

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

| Release Event and Origin Location | Principal Radionuclides | | | |
|---|---|---|---|---|
| | Type | Form | Activity or Concentration | Documentation Notes |
| 1985 spill of radioactive water at the Waste Tank Farm. WMA3: from valve pit northwest of 8D-2, between 8D-2 and 8D-1. | Cs-137, H-3 | Liquid to groundwater, soil | ~400 gal at 4.6 E-02 µCi/mL gross beta, ~4E-03 µCi/mL H-3. | Spill of radioactive water March 1985 at the Waste Tank Farm from a condensate line running from Tank 8D-1 to Tank 8D-2 due to failure of flanged valve bolts. Some water (4.6E-02 µCi/mL gross beta) flowed out of valve pit. Contaminated soil was removed. Documented by Unusual Occurrence Report(1) |
| In 1987, condensate from a ventilation unit spilled on top of Tank 8D-2. WMA3: upon disassembling the unit, condensate leaked out onto the gravel surface. | Radionuclide mix typical of feed and waste contamination | Liquid to groundwater, soil | Less than 10 gallons spilled, water probably ~2E-5 µCi/mL gross beta. | A portable ventilation unit was disassembled after operations on March 2, 1987 near Tank 8D-2. Condensate from the housing spilled onto the gravel surface of Tank 8D-2 top. No soil or water contamination noted in samples collected.(2) |
| In 1987, the Neutralization Pit overflowed during transfer of liquid waste to the interceptor. WMA2: the overflow went to the ground near the interceptors and Lagoon 1. | Radionuclide mix typical of feed and waste contamination | Liquid to groundwater, soil | Approximately 5,000 gallons of waste water was spilled, ~5E-05 uCi/mL gross beta. | The Neutralization Pit overflowed on February 25, 1987 due to a malfunctioning drain valve. The overflow went to the ground near the interceptors and Lagoon 1. The flow was stopped when noted by an operator. Documented by Unusual Occurrence Report (3) |
| In 1987, water from a 55-gallon drum containing spent resin leaked. WMA 5: water spilled on the ground before or during transfer of the drum to a processing station. | Radionuclide mix typical of feed and waste contamination | Liquid to soil, potentially to groundwater | <15 gallons likely spilled, wetted soil was <100 dpm/g gross beta. | Drum was being transferred from the Lag Storage Building hardstand to a waste solidification area in the Process Building when leakage was noted.(4) |

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

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

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

## 2.4 Prior Onsite Burials

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

### 2.4.1 Lagoon 1

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

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

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

**Table 2-19. Estimated Residual Radioactivity in Lagoon 1**[1]

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

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

### 2.4.2 The NRC-Licensed Disposal Area

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

Prior to 1972, the NDA was used exclusively for the disposal of highly radioactive solid wastes generated by the reprocessing plant. Wastes routinely buried in the area included spent fuel hulls, fuel assembly hardware, failed process vessels and large equipment, degraded process solvent absorbed on suitable solid medium, and miscellaneous packaged trash including laboratory wastes, small equipment, ventilation filters, and other process-related debris.

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

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

**Table 2-20 Summary of Wastes in the NRC-licensed Disposal Area**[1]

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

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

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

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

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

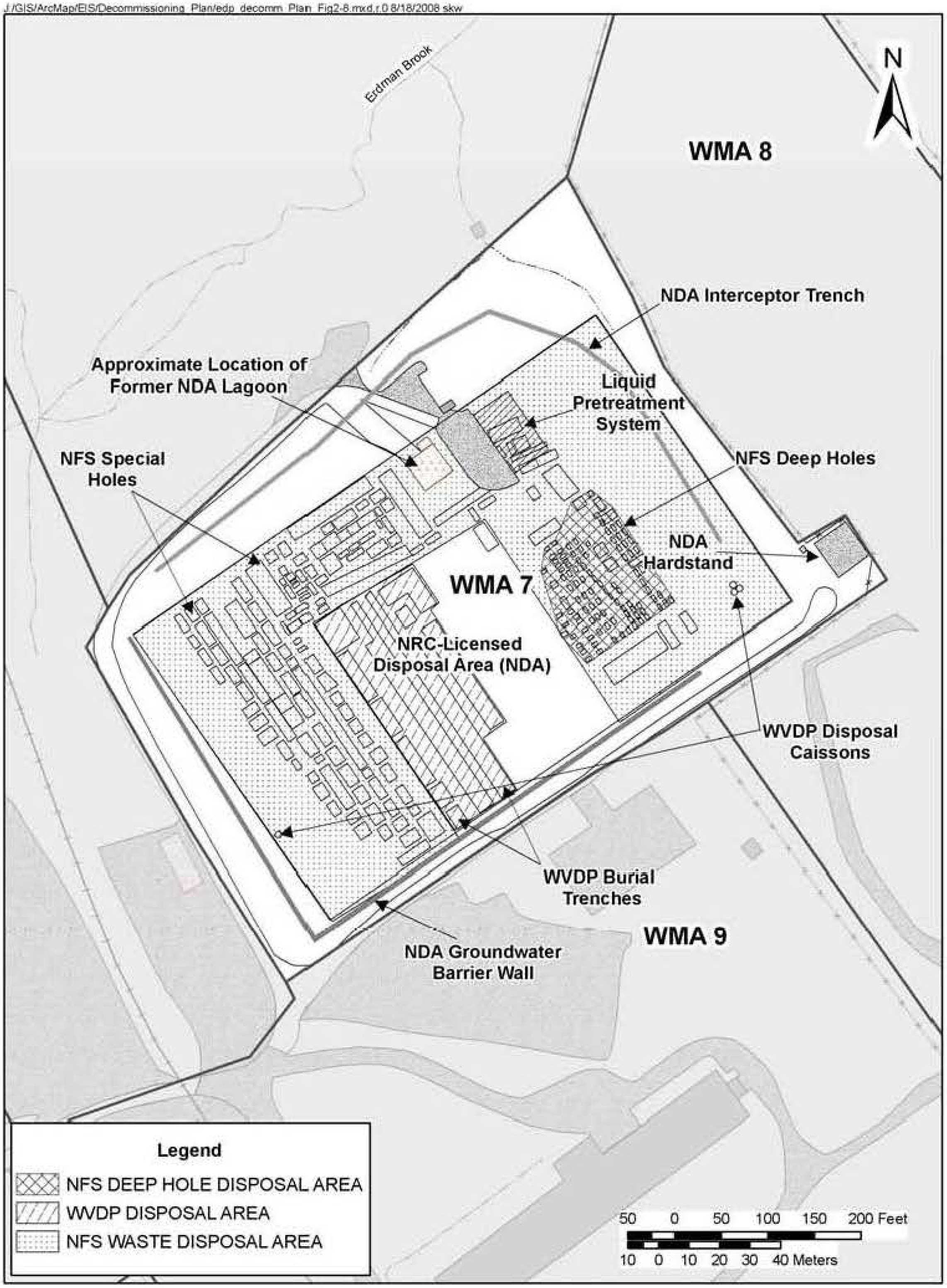


**Figure 2-8. NDA Disposal Area Burials** (The geomembrane cover is not shown in this view.)

### 2.4.3 Other Burial Locations

Two other areas on the Center contain buried radioactive material, although neither is within the scope of this plan[7]. One, the SDA, is not on the project premises. The other, the Construction and Demolition Debris Landfill in WMA 4, is briefly described here for completeness because it is located within the project premises.

**Construction and Demolition Debris Landfill**

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

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

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

Because this landfill is located in the path of the north plateau groundwater plume, radioactively contaminated groundwater in the plume is assumed to have come in contact with waste in the landfill as discussed in Section 2.3.1. Portions of the buried waste are therefore expected to be radioactive.

## 2.5 References

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Atomic Energy Act of 1954

Energy Reorganization Act of 1974

West Valley Demonstration Project Act of 1980

**Code of Federal Regulations**

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

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# 3.0 FACILITY DESCRIPTION

**PURPOSE OF THIS SECTION**

The purpose of this section is to describe the facility and its environs. This information provides a foundation for understanding the rest of the plan. Section 3 is also intended to provide information to allow NRC staff to evaluate DOE's estimation of (1) the impacts of the decommissioning activities on the site and its surrounding areas, and (2) the impacts of the environment on the site in the event of natural phenomena such as floods, tornados, and earthquakes.

**INFORMATION IN THIS SECTION**

This section begins with the location and description of the site, including subsurface conditions. Facilities associated with the WVDP are addressed, including those that existed in 2008 and are to be removed before activities under this plan begin. As with other sections of the plan, these facilities are organized by waste management area (WMA), with the focus on facilities located on the project premises.

The following matters are also addressed: (1) population distribution, (2) current land use and plans for future land use, (3) meteorology and climatology, (4) geology and seismology, (5) surface water hydrology, (6) groundwater hydrology, and (7) natural resources in the area.

All figures referred to in the text, which include photographs, are grouped at the end of the section.

**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 Phase 1 decommissioning. Consideration of the information in Section 2 on site history, processes, and spills will also help place information in Section 3 into context. The information in this section serves as the foundation for later sections, such as radiological status in Section 4, the dose modeling in Section 5, and the decommissioning activities in Section 7.

## 3.1 Site Location and Description

### 3.1.1 Site Location

The WVDP is located about 30 miles south of Buffalo, in the Town of Ashford, Cattaraugus County, New York at approximately 42.450° north latitude and 78.654° west longitude. The site location with respect to major natural and man-made features in the region is shown in Figure 3-1.

The facility (i.e., the project premises) lies 2.4 miles southeast of Cattaraugus Creek at its nearest approach. Cattaraugus Creek forms the boundary between Cattaraugus and Erie counties. Buttermilk Creek, a tributary to Cattaraugus Creek, is 0.5 mile east of the project premises. Lake Erie lies approximately 30 miles west.

### 3.1.2 Site Description

The WVDP site consists of approximately 167 acres within the 3,345-acre Center. Figure 3-2 delineates the boundaries of the Center and the WVDP. The brief description here focuses on the Center, the WVDP, subsurface conditions on the site, and site groundwater.

**The Center**

The Center is located within the glaciated northern portion of the Appalachian Plateau Province of Western New York which is characterized by deep valleys which dissect rather flat-topped plateaus and range in elevation from 1,100 to 1,850 feet above mean sea level (Figure 3-3). The average elevation across the Center is 1,300 feet above mean sea level.

Slopes range from less than five percent to greater than 25 percent, with five to 15 percent slopes predominant. The Center is drained by Buttermilk Creek, which flows into Cattaraugus Creek.

Prior to 1961, much of the Center was cleared for agriculture. As a result, the Center now consists of a mixture of abandoned agricultural areas in various stages of ecological succession, forested tracts, and wetlands, along with transitional ecotones between these areas. The area of the WVDP would be classified as an industrial land use.

**The WVDP Site**

The WVDP lies on a plateau that ranges in elevation from 1,300 to 1,445 feet above mean sea level, 1929 datum. The plateau margins are defined by Franks Creek, Erdman Brook, and Quarry Creek which drain the WVDP and empty into Buttermilk Creek. This plateau is subdivided by Erdman Brook into the north plateau and south plateau areas. The topography on and around the WVDP site is shown on Figure 3-4.

A posted, barbed-wire fence surrounds the Center. An inner, eight feet high chain-link fence surrounds the WVDP site, with access controlled through one gate. The inner fence defining the WVDP boundary, i.e., the project premises, is shown in Figure 3-5.

Most major activities related to the WVDP, including all involving radioactivity, are performed within the WVDP site boundary. Although the State-Licensed Disposal Area

(SDA) is located within the WVDP security fence, as shown in Figure 3-5, it is not considered part of the project premises.

**Subsurface Conditions and Groundwater**

The subsurface conditions underlying the north plateau are different from the subsurface conditions underlying the south plateau, as shown in Figures 3-6 and 3-7. The thickness of the unsaturated zone in the weathered till on the south plateau fluctuates annually, averaging approximately 10 feet below ground surface. Groundwater flow in the weathered Lavery till on the south plateau is generally controlled by surface topography and flow is eastward (WVNSCO 1995).

More detailed information on subsurface conditions and groundwater can be found below in Sections 3.5, 3.6, and 3.7.

### 3.1.3 Facility Description

The following descriptions focus on the WVDP facilities as they are expected to appear at the conclusion of the interim end state in 2011. The facilities to be removed before 2011 are also briefly described.

**Major Facilities**

The principal facilities at the site include the former irradiated nuclear fuel reprocessing facility, known as the Main Plant Process Building; the Waste Tank Farm; and the Low-Level Waste Treatment Facility. These facilities are located on the north plateau. The two radioactive waste burial areas, the NRC-Licensed Disposal Area (NDA) and the SDA, are located on the south plateau. Figure 3-8 shows the locations of these facilities.

**Waste Management Areas**

For administrative purposes, the Center has been divided into 12 WMAs as listed below. The locations of WMA 1 through WMA 10 are shown in Figure 3-8. WMAs 11 and 12 are shown in Figure 3-9.

- WMA 1 Main Plant Process Building and Vitrification Facility area,
- WMA 2 Low Level Waste Treatment Facility area,
- WMA 3 Waste Tank Farm area,
- WMA 4 Construction and Demolition Debris Landfill,
- WMA 5 Waste Storage Area,
- WMA 6 Central Project Premises,
- WMA 7 NDA and associated facilities,
- WMA 8 SDA and associated facilities,
- WMA 9 Radwaste Treatment System Drum Cell Area,
- WMA 10 Support and Services Area,

- WMA 11 Bulk Storage Warehouse and Hydrofracture Test Well Area, and
- WMA 12 Balance of the Site.

| Project Premises Facilities Removed Before Decommissioning Activities Begin | |
|---|---|
| **WMA 1** | **WMA 5** (continued) |
| Cold Chemical Facility | Lag Storage Addition 2 |
| Contact Size Reduction Facility | Lag Storage Addition 3 |
| Emergency Vehicle Shelter | Hazardous Waste Storage Lockers |
| Laundry Room | **WMA 6** |
| Master-Slave Manipulator Repair Shop | Old Warehouse |
| Radwaste Process (Hittman) Building | Old Sewage Treatment Facility |
| Recirculation Ventilation System Building | New Cooling Tower (except basin) |
| **WMA 2** | North Waste Tank Farm Training Platform |
| O2 Building | Road-Salt and Sand Shed |
| Test and Storage Building | **WMA 7** |
| Maintenance Shop | Interim Waste Storage Facility |
| Maintenance Storage Area | NDA Hardstand |
| Vehicle Repair Shop | **WMA 10** |
| Vitrification Test Facility | Administration Building |
| **WMA 5** | Expanded (Environmental) Laboratory |
| Chemical Process Cell Waste Storage Area | Fabrication Shop |
| Lag Storage Building | Vitrification Diesel Fuel Oil Building |
| Lag Storage Addition 1 | |

**WMA 1: Main Plant Process Building and Vitrification Facility Area**

Figure 3-10 shows the layout of WMA 1. Figure 3-11 is an aerial photograph of the Main Plant Process Building and Vitrification Facility area. A description of each facility in WMA 1 follows:

WMA 1 facilities within the scope of this plan are:

- Main Plant Process Building;
- Vitrification Facility;
- Load-In/Load-Out Facility;
- Utility Room and Utility Room Expansion;
- Fire Pumphouse and Water Storage Tank;

- Plant Office Building;
- Electrical Substation;
- 01-14 Building;
- Vitrification Off-Gas Trench;
- Source Area of the North Plateau Plume; and
- Concrete Floor Slabs for the Laundry Room, Fuel Receiving and Storage Ventilation Building, Radwaste Process Building, Cold Chemical Facility, and other removed facilities.

***Main Plant Process Building.*** The Main Plant Process Building (Process Building) was built between 1963 and 1966, and was used by Nuclear Fuel Services (NFS) from 1966 to 1971 to recover uranium and plutonium from spent nuclear fuel. This multi-storied building is approximately 130 feet wide and 270 feet long, and rises approximately 79 feet above the ground surface at its highest point. Figures 3-12 through 3-21 show the building exterior, interior layouts, and representative areas.

The major Process Building structure rests on approximately 480 driven steel H-piles. The building is composed of a series of cells, aisles, and rooms that are constructed of reinforced concrete and concrete block. The reinforced concrete walls, floors and ceilings range from one to six feet thick. The reinforced concrete walls are typically surrounded by walls of lighter concrete and masonry construction and metal deck flooring. Six floor layout plans of different levels of the Process Building appear in Figures 3-13A through 3-13F.

Most of the facility was constructed above grade, with some of the cells extending below ground (i.e., below the ground surface reference elevation of 100 feet). The deepest cell, the General Purpose Cell, extends approximately 27 feet below-grade. The Cask Unloading Pool and the Fuel Storage Pool, located in the Fuel Receiving and Storage Area on the east side of the building, were used to receive and store spent fuel received for reprocessing, and extend approximately 49 and 34 feet below grade, respectively.

Cells such as the Process Mechanical Cell, the Chemical Process Cell, and Extraction Cells 1, 2, and 3 were constructed of reinforced high-density concrete three to five feet thick. Such thicknesses were needed to provide radiation shielding.

The operations performed in the cells were remotely controlled by individuals working in the various aisles of the Process Building, which were formed by adjacent walls of the cells. The aisles contained the manipulator controls and valves needed to support operations in the cells. Rooms not expected to contain radioactivity were typically constructed with concrete block and structural steel framing.

Wastewater generated during reprocessing was managed in one of two ways, depending on activity. High-level waste was transferred from the Process Building to the Waste Tank Farm via two underground transfer lines (7P-113 and 7P-120) to Tank 8D-2 and Tank 8D-4. Low-level wastewater was transferred to the Low Level Waste Treatment Facility via below-grade transfer lines associated with the interceptor system.

The WVDP modified portions of the Process Building to support its primary mission of solidifying HLW. Equipment in the Chemical Process Cell was removed to allow its use for storage of canisters of vitrified HLW. Extraction Cell 3 and the Product Purification Cell were emptied of equipment which was replaced with equipment used to support the Liquid Waste Treatment System. This system was used to manage supernatant and sludge wash solutions from Tank 8D-2 which contained HLW.

***Vitrification Facility.*** Shown in Figures 3-22 and 3-23, this structural steel frame and sheet metal building houses the Vitrification Cell, operating aisles, and a control room. The Vitrification Cell is 34 feet wide, 65 feet long and 42 feet high. Figure 3-23 shows how it looked when it went into service.

At the north end of the Vitrification Cell is the melter pit. The pit is 34 feet wide by 25 feet long with its bottom about 14 feet below grade. The Vitrification Cell is lined with 0.125-inch-thick stainless steel up to 22 feet above grade.

As explained in Section 2, HLW transferred from HLW Tank 8D-2 was mixed with glass formers and vitrified into borosilicate glass within the Vitrification Cell. Vitrification operations were performed remotely by operators in the operating aisles or in the control room. The Vitrification Cell contained the Concentrator Feed Makeup Tank, Melter Feed Hold Tank, the slurry-fed ceramic melter, turntable, off-gas treatment equipment, canister welding station, and the canister decontamination station. All equipment was removed from the Vitrification Cell during the deactivation of this facility in 2003 and 2004.

***Load-In/Load-Out Facility.*** The Load-In/Load-Out Facility is located adjacent to the west wall of the Equipment Decontamination Room of the Process Building in WMA 1. It is a structural steel and steel sided building that is approximately 80 feet long, 55 feet wide, and 54 feet tall. The floor is poured concrete, and the roof is metal sheeting with insulation.

This facility was used to move empty canisters and equipment into and out of the Vitrification Cell. It has a truck bay and a 15-ton overhead crane that is used to move canisters and equipment. After the new Canister Storage Facility is constructed, the Load-In/Load-Out facility will be used to transfer the vitrified HLW canisters from the Process Building to that facility.

***Utility Room and Utility Room Expansion.*** The Utility Room and the Utility Room Expansion can be seen in Figures 3-10 and 3-11. The Utility Room is a concrete block and steel framed building located on the south end of the Process Building. It consists of two adjoining buildings that were built at different times, the original Utility Room and the Utility Room Expansion.

The original Utility Room, which was built during the construction of the Process Building, makes up the western portion of the facility and is 80 feet wide, 88 feet long, and

20 feet high. It contains equipment that supplies steam, compressed air, and various types of water to the Process Building.

The Utility Room Expansion was built in the early 1990s immediately adjacent to the original Utility Room. The Utility Room Expansion is approximately 85 feet long, 56 feet wide, and 25 feet high. It contains equipment similar to that in the Utility Room.

***Fire Pump House and Water Storage Tank***. The Fire Pump House was constructed in 1963 and is 20 feet wide, 24 feet long, and 10 feet high at the peak. The structure is of steel frame and sheet metal construction on a four-inch concrete slab floor, which is supported on a concrete foundation wall. Its location is shown in Figure 3-10.

The Pump House contains two pumps on concrete foundations. An adjacent small metal storage shed is used to store fire hoses and fire extinguishers. The 475,800-gallon water storage tank (Tank 32D-1) is located outside the Utility Room, as shown in Figure 3-11.

***Plant Office Building***. The Plant Office Building is a three-story concrete block and structural steel framed structure located adjacent to the west side of the Process Building. It is approximately 40 feet wide, 95 feet long, and 44 feet high and contains offices and men's and women's locker rooms. Figures 3-11 and 3-14 show the building.

***Electrical Substation***. The electrical substation is located adjacent to the southeast corner of the Process Building. A 34.5 kilovolt/480 volt transformer rests on a concrete foundation behind a steel framed structure. Its location is shown in Figure 3-10.

***01-14 Building***. The 01-14 Building is a four-story, 64 feet tall concrete and steel frame building located next to the southwest corner of the Process Building, as shown in Figures 3-10 and 3-11. This building was built in 1971 to house an NFS off-gas system and acid recovery system, but it was never used to support NFS operations. The 01-14 Building was modified to house the Vitrification Off-Gas System and the Cement Solidification System.

The off-gas system was used to treat off-gases generated in the melter in the Vitrification Facility. The Cement Solidification System was used to stabilize radioactive waste generated from the Liquid Waste Treatment System in a cement matrix and to package this mixture in drums that were stored in the Radwaste Treatment System Drum Cell in WMA 9.

***Laundry Room***. The Laundry Room is located southeast of the Utility Room as shown in Figure 3-10. It is a concrete block structure 26 feet by 56 feet by 20 feet high with metal decking and asphalt roofing. The floor is a concrete slab six inches thick, which contains a sump.

The Laundry Room contains a commercial size washer and dryer, along with sorting tables and racks for laundering contaminated protective clothing. It is separated into a radiologically “hot” side and a “clean” side. It will be removed down to its concrete floor slab at grade before the start of Phase 1 decommissioning activities.

***Cold Chemical Facility Slab***. The Cold Chemical Facility was a structural steel frame and sheet metal building that was approximately 34 feet wide, 57 feet long, and 36 feet tall. It was located immediately west of, and adjacent to, the Vitrification Facility, as shown in Figure 3-27. It was used to prepare non-radioactive feed materials, such as nitric acid and glass formers, which were used in the vitrification process. The Cold Chemical Facility was demolished to its concrete floor slab at grade in November 2006.

***Fuel Receiving and Storage Ventilation Building Slab***. This steel-framed and sheet metal sided structure was located adjacent to the Radwaste Process Building. It was 30 feet by 35 feet by 12.2 feet high and rested on a six-inch-thick concrete slab. It contained equipment that provided the majority of the heating, ventilation, and air conditioning systems for the Fuel Receiving and Storage Building. It was removed down to its concrete floor slab at grade in October 2006.

***Radwaste Process Building Slab***. This 15 feet wide by 46 feet long by 12 feet high steel structure, also known as the Hittman Building, was located north of the Fuel Receiving and Storage Building. It was used to manage shielded casks for high-integrity containers used to store loaded resins from the Fuel Pool Submerged Water Filtration System. This building was removed down to its concrete floor slab at grade in October 2006.

**WMA 2: Low-Level Waste Treatment Facility Area**

WMA 2, the Low Level Waste Treatment Facility area as it existed in 2008 is shown in Figure 3-24. Figure 3-25 shows the area before the advent of the WVDP.

This facility was used by NFS and then by the WVDP to process low-level radioactive wastewater generated on-site. The current Low Level Waste Treatment Facility includes the Neutralization Pit, interceptors, Lagoons 2-5, and the LLW2 Building. It is expected to still be in use when Phase 1 decommissioning activities begin.

WMA 2 facilities within the scope of this plan are:

- The LLW2 Building;
- Closed Lagoon 1;
- Active lagoons 2, 3, 4, and 5;
- The two New Interceptors;
- The Old Interceptor;
- The Neutralization Pit;
- The Maintenance Shop Leach Field;
- The Solvent Dike; and
- Concrete floor slabs such as those for the 02 Building, Maintenance Shop, Test and Storage Building, and Vitrification Test Facility.

A description of the WMA 2 facilities follows:

***The LLW2 Building***. Located southwest of Lagoon 4, this pre-engineered, single-story, metal-sided building rests on a concrete wall foundation, measuring 40 feet by 60 feet. The building houses two skid-mounted process equipment modules that are used to treat wastewater from WMA 1, WMA 3, and radiologically contaminated groundwater from the WMA 7 NDA Interceptor Trench and the north plateau groundwater plume. Figure 3-26 shows the building. The LLW2 Building was built in 1998 to replace the 02 Building, the original low-level wastewater treatment facility that was built by NFS in 1971.

The building is divided into three work areas and an office. The processing area contains the process modules (including ion exchangers, valves, piping, pumps, filters, instrumentation, and controllers), two surge tanks, and a sand filter. The packaging room contains a four feet by four feet by nine-feet-deep stainless steel lined catch basin. A portable ventilation unit located outside of the packaging area contains a high-efficiency particulate air (HEPA) filter and a short stack on the roof of the building.

***Lagoon 1***. Lagoon 1 was an unlined pit excavated into the sand and gravel unit that was approximately 80 feet long on each side and 5 feet deep. It was fed directly from the Old Interceptor and the New Interceptors, and had a storage capacity of more than 200,000 gallons. As explained in Section 2, it was removed from service in 1984. Most of the contaminated sediment was transferred to Lagoon 2 and Lagoon 1 was filled with contaminated debris from the NFS hardstand and then capped with clay and topsoil.

Figure 3-27 shows the area of Lagoon 1. Section 2.4.1 discusses the radioactivity in the closed lagoon.

***Lagoon 2***. Lagoon 2 is an unlined 17-foot deep basin excavated in the unweathered Lavery till. This lagoon has a storage capacity of 2.4 million gallons and is used to store wastewater discharged from the New Interceptors before its transfer to the LLW2 for treatment.

From 1965 to 1971, before the installation of the Low Level Waste Treatment Facility system – which initially consisted of the O2 Building and Lagoons 4 and 5 – wastewater was routed through Lagoons 1, 2, and 3 in series before discharge to Erdman Brook. Between 1971 and 1982, low-level wastewater was routed sequentially through Lagoon 1, Lagoon 2, and the O2 Building for treatment, then to Lagoons 4 or 5, and finally to Lagoon 3 before discharge to Erdman Brook. From 1982 following the closure of Lagoon 1 to the present, low-level wastewater has been routed sequentially through Lagoon 2, the O2 Building or LLW2 for treatment, Lagoons 4 or 5, and then to Lagoon 3 before discharge to Erdman Brook.

A French drain was installed on the northwest sides of Lagoons 2 and 3 and the northeast side of Lagoon 3 to prevent groundwater from flowing into Lagoons 2 and 3. The French drain was capped in 2001 and no longer discharges into Erdman Brook.

***Lagoon 3***. Lagoon 3 is a 24-foot deep unlined basin excavated in the unweathered Lavery till. It has a storage capacity of 3.3 million gallons. Lagoon 3 receives treated water from Lagoons 4 and 5. Lagoon 3 is periodically batch discharged to Erdman Brook through a State Pollutant Discharge Elimination System (SPDES) permitted discharge.

***Lagoon 4***. Lagoon 4 is a basin constructed in the sand and gravel unit on the North Plateau with a capacity of 204,000 gallons. It receives only treated water from LLW2 and discharges to Lagoon 3.

Lagoon 4 was originally excavated into the sand and gravel unit on the North Plateau and lined with reworked glacial tills. In 1974 a synthetic membrane liner was installed after NFS identified that Lagoons 4 and 5 were potential sources of tritium to groundwater in the sand and gravel unit (WVNSCO 1997). In the late 1990's, the synthetic membrane liners were removed and replaced with concrete grout and a XR-5 liner, an ethylene inter-polymer alloy membrane.

***Lagoon 5***. Lagoon 5 is a basin constructed in the sand and gravel unit on the North Plateau with a capacity of 166,000 gallons. It receives only treated water from the LLW2 facility and discharges to Lagoon 3.

Lagoon 5 was originally excavated into the sand and gravel unit on the north plateau and lined with reworked glacial tills. In 1974 a synthetic membrane liner was installed after NFS identified that Lagoons 4 and 5 were potential sources of tritium to groundwater in the sand and gravel unit (WVNSCO 1997). In the late 1990's, the synthetic membrane liners were removed and replaced with concrete grout and a XR-5 liner, an ethylene inter-polymer alloy membrane.

***Neutralization Pit***. The Neutralization Pit is a nine feet by seven feet by 5.5 feet deep concrete tank constructed with six-inch thick concrete walls and floor that are lined with stainless steel. The pit receives low-level radioactive wastewater from WVDP process areas. This liquid is subsequently transferred to the interceptors.

***Old Interceptor***. The Old Interceptor is a 40 feet by 25 feet by 11.5 feet deep unlined concrete liquid waste storage tank located below-grade. The floor is 24-inches thick and the walls 12 inches thick[1]. The roof is made of steel.

The Old Interceptor received low-level liquid waste generated at the Process Building from the time of initial plant operation until the new interceptors were constructed. The Old Interceptor is currently used for temporarily storing radiologically contaminated liquids that exceed the effluent standard of 0.005 µCi/mL gross beta activity. After verification of acceptable radiological contamination concentrations, the contents are transferred by steam jet to the New Interceptors.

[1] The floor of the Old Interceptor was initially 12 inches thick. An additional 12 inches of concrete was poured on the floor during NSF operations to provide radiation shielding.

***New Interceptors.*** The New Interceptors are twin open-top concrete storage tanks, each 22 feet by 20 feet by 11.5 feet deep, located below grade. The walls and floor are 14 inches thick, and are lined with stainless steel. The roof is steel. The New Interceptors were built in 1967 to replace the Old Interceptor, which had high levels of radioactivity (WVNSCO 1997). The New Interceptors are used to collect and sample wastewater before it is transferred to Lagoon 2.

***Solvent Dike.*** The Solvent Dike is located about 300 feet east of the Process Building. It was an 30 foot by 30 foot unlined basin excavated in the sand and gravel layer. The Solvent Dike received rainwater runoff from the Solvent Storage Terrace, which formerly housed an acid storage tank and three storage tanks containing a mixture of used n-dodecane and tributyl phosphate. The sediment has been removed and the area has been backfilled, but the Solvent Dike still contains radiologically contaminated soil.

***Maintenance Shop Leach Field.*** The Maintenance Shop Leach Field is located just northeast of where the Maintenance Shop stood and consists of three septic tanks, a distribution box, a tile drain field, and associated piping. The leach field, which occupies an area of approximately 1,500 square feet, was used until1988; all three tanks are out of service and filled with sand. Because it is located within the area of the north plateau groundwater plume, low levels of contamination may be present.

***Groundwater Pump and Treat System.*** Installed in 1995, this system is located in the northwest corner of WMA 2 and draws water from two recovery wells at the western lobe of the north plateau groundwater plume, which is discussed in Section 2 and in Section 4.2. Groundwater is pumped to the Low Level Waste Treatment Facility for treatment by ion exchange to remove Sr-90 contaminants. The treated groundwater is pumped to Lagoon 4 or Lagoon 5, and then to Lagoon 3, and, eventually, discharged into Erdman Brook through the permitted outfall.

***Pilot Scale Permeable Treatment Wall.*** Installed in 1999 and located northwest of Lagoon 5, this treatment wall is about 30 feet wide, seven feet thick, and 25 feet deep, extending down to the Lavery till. It is filled with clinoptilolite, a natural zeolite material, and covered with soil. Its purpose was to evaluate the effectiveness of such systems in treating groundwater contaminated with Sr-90.

***O2 Building Slab.*** The O2 Building was a two-story, steel-framed concrete block structure 27 feet wide, 39 feet long, and 30 feet high. It contains a 16 feet deep stainless steel lined sump. Figure 3-25 shows the building when it was in service.

The O2 Building once housed filters, ion exchangers and other equipment used by NFS and the WVDP to treat radioactive wastewater before transfer to Lagoon 3. It was replaced with the LLW2 Building, It was demolished down to its concrete floor slab at grade in October 2006.

***Test and Storage Building Slab.*** The Test and Storage Building was an 80 feet by 120 feet by 22 feet high timber frame and metal sided building located northeast of the

Process Building. It contained office spaces, a tool crib, and garage space. An 18 feet by 26 feet by 12 feet concrete block addition housed radiation and safety operations. It was demolished down to its concrete floor slab at grade in June 2007.

***Vitrification Test Facility****.* This 40 feet wide and 120 feet long and 36 feet high metal building with a concrete floor contains a scale vitrification facility and a bulk chemical storage tank. It will be removed down to its concrete floor slab at grade before Phase 1 of the decommissioning.

***Maintenance Shop Slab****.* The Maintenance Shop was a 60 feet by 100 feet by 28 feet high metal building with steel supports. It housed locker rooms, lavatories, instrument shops, work areas, and a finished office area. The Maintenance Shop was demolished down to its concrete floor slab at grade in June 2007.

***Permeable Treatment Wall***. A full-scale passive permeable treatment wall is expected to be installed before Phase 1 of the decommissioning to mitigate the off-site migration of Sr-90 contaminated groundwater in the sand and gravel unit in the north plateau.

The permeable treatment wall is planned to be located in WMA 2 immediately south of the Construction Demolition and Debris Landfill in WMA 4 approximately perpendicular to the flow path of the north plateau groundwater plume. It will be approximately 750 feet long in a northwest-southeast direction. The permeable treatment wall will be two to four feet thick, extend down into the underlying unweathered Lavery till, and be composed of granular zeolite to reduce Sr-90 concentrations in groundwater through ion-exchange.

Alternatives for potential mitigation of Sr-90 in surface water in the swamp ditch west of the Construction Demolition and Debris Landfill and downgradient of the permeable treatment wall will be considered after installation of the permeable treatment wall.

**WMA 3: Waste Tank Farm Area**

Shown in Figures 3-29 and 3-30, WMA 3 includes the waste storage tanks (8D-1, 8D-2, 8D-3, and 8D-4), and their associated tank vaults, the HLW transfer trench, the Permanent Ventilation System Building, the Equipment Shelter and condensers, the Con-Ed Building, and the Supernatant Treatment System Support Building.

WMA 3 facilities and equipment within the scope of this plan are:

- Tanks 8D-1, 8D-2, 8D-3, 8D-4, and the associated vaults[2];
- The HLW mobilization and transfer pumps;
- The HLW transfer trench piping;
- The Equipment Shelter and Condensers; and
- The Con-Ed Building.

Descriptions of the WMA 3 facilities follow.

[2] Only removal of the pumps from the tanks is within the scope of Phase 1 decommissioning activities.

***Waste Storage Tanks****.* The waste storage tanks were built to store the liquid HLW generated during the spent nuclear fuel reprocessing operations. The WVDP subsequently modified these tanks to support treatment and vitrification of the HLW. Modifications included constructing a fabricated steel truss system over tanks 8D-1 and 8D-2 to carry the weight of sludge mobilization and transfer pumps and installation of the Supernatant Treatment System equipment in Tank 8D-1.

***Tank 8D-1, Tank 8D-2, and Vaults***. Tanks 8D-1 and 8D-2 are identical in size and construction, with each tank housed within its own cylindrical concrete vault. Each tank is 27 feet high by 70 feet in diameter, with a storage capacity of 750,000 gallons. Figure 3-31 shows a cutaway view of a tank.

The tanks were constructed with reinforced carbon steel plate ranging in thickness from 0.4375 inch for the roofs and walls to 0.656 inch for the floors. The roof of each tank is supported internally by forty-five eight-inch diameter vertical pipe columns that rest on a horizontal gridwork of wide flange beams and cross members in the bottom two feet of each tank. Each tank rests on two six-inch-thick layers of perlite blocks that rest on a three-inch layer of pea gravel. The tank, perlite blocks, and pea gravel are contained within a carbon steel pan which rests on a three-inch layer of pea gravel that separates the pan from the floor of the vault.

Each tank and its associated pan are housed within a cylindrical reinforced concrete vault that has an outside diameter of 78.6 feet. The walls of each vault are 18 inches thick and extend nearly 36 feet above the floor of the vaults.

The floor of each vault is 27 inches thick, except under the six 30-inch diameter vertical concrete columns that support the vault roof. These columns pass upward from the floor of the vault through the tanks and are encased in steel pipes 48 inches in diameter that are welded to the top and bottom of each tank. The columns are located approximately 16 feet from the center of the tank. The floor of each vault is underlain by a four feet thick bed of gravel. The concrete vault roof is two feet thick and is supported by the six concrete columns. The top of the vaults are six to eight feet below grade.

Despite their robust construction, the tank vaults have not proven to be watertight. Groundwater seeps into both vaults and has to be regularly pumped out. A tank and vault drying system will be installed during deactivation work to achieve the interim end state to dry Tanks 8D-1, 8D-2, 8D-3, 8D-4 and their associated vaults. The tanks and vaults are expected to be in a dry condition several years after the start of Phase 1 of the decommissioning. The Tank and Vault Drying System will then maintain the tanks and vaults in a dry state.

The current conceptual design of the Tank and Vault Drying System includes a pre-cooling condensing unit and a desiccant wheel with a heater. Outside air will be pre-cooled as needed to lower the relative humidity entering the drying unit. The air will then flow through the desiccant unit for further drying and heating before being distributed to the bottom of the tanks and vaults.

The dry air supplied to the bottom of the tanks will displace moist air which will follow the tank ventilation flow path from the top of the tanks through the tank ventilation lines to the Permanent Ventilation System Building for treatment. At the Permanent Ventilation System Building, the moist air flow from the tanks will flow through a moisture separator, a heater, pre-filters, and two sets of HEPA filters before being discharged through the Permanent Ventilation System Building stack.

The dry air supplied to the bottom of the vaults will be a recirculation loop displacing moist vault air which will be removed at the top of the vaults. Moist exhaust air from the vaults will be drawn back through the desiccant wheel along with the necessary make up air. Make up air will be necessary since the dry air that goes in to the tanks is not returned to the desiccant unit.

The desiccant in the desiccant wheel will need to be regenerated periodically. Moisture in the desiccant unit will be removed with filtered heated air passing through the reactivation sector of the desiccant drying unit. The moist air exiting the unit will be vented to the Permanent Ventilation System Building where it will join the air flow from the Supernatant Treatment System Support Building and the tanks before flowing through the moisture separator, heater, pre-filters and two sets of HEPA filters before discharge through the Permanent Ventilation System Building stack.

The HLW transfer pumps and the mobilization pumps in Tanks 8D-1 and 8D-2 will be removed during Phase 1 of the decommissioning. These pumps are illustrated in Figure 3-32.

Tanks 8D-1 and 8D-2 each contain a single HLW transfer pump. Each centrifugal multi-stage turbine type pump is more than 55 feet long and is driven by a 150 horse power motor. Tanks 8D-1 and 8D-2 also contain a total of nine mobilization pumps. These pumps are approximately the same size as the HLW transfer pumps.

Tanks 8D-1 and 8D-2 each contain an additional suction pump used in waste pretreatment and processing. The Tank 8D-1 pump is a vertical turbine pump mounted on a pipe column with an overall length of approximately 31 feet. The Tank 8D-2 pump is a submersible pump mounted on a three inch pipe column with an overall length of approximately 33 feet. All of the pumps in the underground waste tanks are expected to be highly contaminated as explained in Section 4.1.

Tank 8D-1 was modified by the WVDP to support operation of the Supernatant Treatment System and it contains the following Supernatant Treatment System equipment:

- Supernatant pre-filter
- Supernatant feed tank (1,726 gal)
- Supernatant cooler
- Four zeolite columns (1,900 gal each)
- Supernatant sand filter

- Sluice lift tank (2,142 gal)
- Associated transfer piping.

The operation of the Supernatant Treatment System is described below.

***Tank 8D-3, Tank 8D-4 and Vault.*** Tanks 8D-3 and 8D-4 are identical in size and construction, and both are housed within a single reinforced concrete vault. Each tank is 12 feet in diameter and 15.67 feet high, with a nominal volume of 15,000 gallons. The shell of each tank is 0.313 to 0.375 inch thick; both the tanks and their associated piping were constructed from 304L stainless steel.

The concrete vault that houses the tanks is approximately 32-feet long, 19-feet wide, and 25-feet tall. The walls, floor, and roof of the vault are 21-inches thick. The bottom of the vault is lined with stainless steel to a height of 18 inches above the floor. The floor contains a stainless-steel-lined sump. The top of the vault is six to eight feet below grade.

The HLW transfer pumps in tanks 8D-3 and 8D-4 will be removed to facilitate removal of liquids in these tanks during deactivation work to achieve the interim end state. The transfer pumps will be replaced with submersible pumps equipped with chemical resistant transfer lines. The submersible pumps and transfer lines will be removed during Phase 1 of the decommissioning.

***High-Level Waste Transfer Trench.*** The HLW transfer trench is a long concrete vault containing piping that conveyed waste between the Waste Tank Farm and the Vitrification Facility. Approximately 500 feet long, the trench extends from the Tank 8D-3/Tank 8D-4 vault along the north side of Tank 8D-1 and Tank 8D-2, before turning to the southwest and entering the north side of the Vitrification Facility. It is six to 20 feet wide and its height ranges from six to nine feet. Figure 3-33 shows the trench under construction.

The trench was constructed with reinforced concrete walls and floors, with pre-cast concrete covers. Stainless steel-lined concrete pump pits that house the upper sections of HLW transfer pumps are located on top of each of the tank vaults. The walls and floors of the pump pits are reinforced concrete, with pre-cast concrete covers forming the roof. Figure 3-34 shows a typical pump pit.

There are six piping runs in the trench, two of which are unused spares, comprising approximately 3000 linear feet of double-walled stainless steel pipe.[3] The trench also contains associated valves and jumpers. The pump pits each contain the upper part of the HLW transfer pump and flow monitoring equipment. Pump Pit 8Q-2 over Tank 8D-2 also contains grinding equipment used to size reduce zeolite.

The piping and related equipment will be removed during Phase 1 of the decommissioning.

[3] Portions of the trench contain only two piping runs; the section connecting to the Vitrification Facility contains all six runs.

***Permanent Ventilation System Building.*** The Permanent Ventilation System Building is located approximately 50 feet north of Tank 8D-2, as shown in Figure 3-30. This steel framed and sided building is 40 feet wide, 75 feet long, and 16 feet tall and is attached to a 12 inch thick concrete floor slab supported by concrete footings. The building has a sheet metal roof which supports the Permanent Ventilation System discharge stack.

The Permanent Ventilation System was designed to provide ventilation to the Supernatant Treatment System Support Building, the Supernatant Treatment System valve aisle, the Supernatant Treatment System pipeway, and the HLW tanks. A skid-mounted, Permanent Ventilation System Stack Monitoring Building is located near the east end of the building.

***Equipment Shelter and Condensers***. The Equipment Shelter is a one-story concrete block building lies immediately north of the Vitrification Facility, as shown in Figures 3-29 and 3-30. It is 40 feet long, 18 feet wide, and 12 feet high and has a concrete floor six inches thick, with a small extension on the west side.

This structure houses the Waste Tank Farm ventilation system that was formerly used to ventilate the four waste storage tanks and the Supernatant Treatment System vessels in HLW Tank 8D-1.

The condensers are located immediately west of the Equipment Shelter. They were designed to condense the overheads from Tanks 8D-1 and 8D-2, which were originally designed to be in a self-boiling condition during NFS operations. The Equipment Shelter and condensers will be removed during Phase 1 of the decommissioning.

***Con-Ed Building***. The Con-Ed Building is a concrete block building located on top of the concrete vault containing Tank 8D-3 and Tank 8D-4, as shown in Figures 3-29 and 3-30. This building, which is 10 feet wide, 13 feet long, and 11 feet high, houses the instrumentation and valves used to monitor and control the operation of Tanks 8D-3 and 8D-4. This building will be removed during Phase 1 of the decommissioning.

***Supernatant Treatment System Support Building***. This building is located adjacent to and above Tank 8D-1. It is a two-story structure that contains equipment and auxiliary support systems needed to operate the Supernatant Treatment System.

The Supernatant Treatment System is a zeolite ion-exchange system that was designed to primarily remove radioactive cesium from the high-level PUREX supernatant and sludge wash solutions from Tank 8D-2. The majority of the Supernatant Treatment System equipment is located in Tank 8D-1. This system was also capable of removing strontium and plutonium from these wastes. The high-level supernatant was pumped from Tank 8D-2 and was treated in the Supernatant Treatment System between May 1988 and January 1991.

The Supernatant Treatment System was also used from 1991 to 1995 to remove radioactive cesium from sludge washes generated from the sludge mobilization and wash system which was designed to remove sulfate salts from the sludge in Tank 8D-2 using a dilute caustic wash solution to dissolve the salts. Once a wash cycle was completed, the wash water was treated in the Supernatant Treatment System. Two sludge-wash cycles were completed between 1992 and 1994, and a third sludge wash was completed in 1995. During this third sludge wash campaign, THOREX waste in Tank 8D-4 was transferred to Tank 8D-2, where the combined PUREX/THOREX mixture was washed.

The upper level of the Supernatant Treatment System Support Building is a steel framework structure covered with steel siding. The lower level of the building was constructed with reinforced concrete walls, floors, and ceilings.

This building contains a control room; heating, ventilation and air conditioning equipment; utilities; and storage tanks for fresh water and fresh zeolite to support Supernatant Treatment System operations. A shielded valve aisle is located on the lower level of the support building, adjacent to Tank 8D-1.

The Supernatant Treatment System pipeway is located on top of the Tank 8D-1 vault. This concrete and steel structure contains the Supernatant Treatment System piping and structural members that support the Supernatant Treatment System equipment located in Tank 8D-1.

**WMA 4: Construction and Demolition Debris Landfill Area**

WMA 4, which includes the Construction and Demolition Debris Landfill, is a 10-acre area in the northeast portion on the north plateau of the WVDP as shown in Figure 3-8. The landfill, which was utilized as described in Section 2, is the only waste management unit in WMA 4. It will be monitored and maintained during Phase 1 decommissioning.

**WMA 5: Waste Storage Area**

The facilities in WMA 5 are shown in Figure 3-35 and are described below. WMA 5 facilities within the scope of this plan are:

- Lag Storage Addition 4 and its associated Shipping Depot;
- The Remote-Handled Waste Facility;
- Concrete slabs and foundations for the Lag Storage Building, Lag Storage Additions 1, 2, and 3, Chemical Process Cell Waste Storage Area; and
- Several hardstands consisting of compacted gravel pads.

***Lag Storage Addition 4***. Lag Storage Addition 4 is a clear-span structure, with a pre-engineered steel frame and steel sheathing. Approximately 291 feet long, 88 feet wide and 40 feet high, it rests on a seven-inch concrete slab. It is similar to Lag Storage Addition 3, except that it includes a shipping depot, a container sorting and packaging facility, and a covered passageway between the two storage areas. The shipping depot is connected to Lag Storage Addition 4 and is a 91 feet by 85 feet metal frame structure. This facility and its concrete floor slab will be removed during Phase 1 of the decommissioning.

***Remote-Handled Waste Facility***. The Remote-Handled Waste Facility is located in the western portion of WMA 5 as shown in Figure 3-35. It is a metal-sided, steel-frame building that includes a Receiving Area, a Buffer Cell, a Work Cell, a Waste Packaging Area, an

Operating Aisle, and a load-out /truck bay. Figure 3-36 shows the facility under construction and Figure 3-37 shows the layout of the first floor.

The Receiving Area includes a 20-ton bridge crane that also provides access into the adjacent Buffer Cell. The Buffer Cell is an air lock between the Receiving Area and the contaminated Work Cell. The Work Cell is the primary work area, with provisions for fully remote handling, surveying, segmenting, decontaminating, and repackaging operations. This shielded space is 55 feet by 22 feet by 26 feet high, and is served by a 30-ton bridge crane.

Any spent decontamination solutions generated during operations are transferred to below-grade wastewater storage tanks located in a vault below the building for management before treatment. These tanks and vault will be removed during Phase 1 of the decommissioning.

The Waste Packaging Area includes capability to load both waste drums and boxes. The Operating Aisle houses two waste processing and packaging work stations and one waste sampling transfer work station. Each work station includes a shield window in the shield wall, and controllers for remotely operating facility equipment.

This facility and its concrete floor slab will be removed during Phase 1 of the decommissioning.

***Lag Storage Building Slab***. The Lag Storage Building was a sheet metal structure built in 1984 to store LLW. It was supported by a clear span frame and anchored to a 140 feet long by 60 feet wide concrete slab foundation. The slab surface was coated with an acid-resistant, two-coat, application of epoxy sealer. It was demolished down to its concrete floor slab in October 2006.

***Lag Storage Addition 1 Slab****.* Lag Storage Addition 1 was a pre-engineered steel frame and fabric structure built in 1987 to store containerized LLW. It was 191 feet long by 55 feet wide by 23 feet high. It was removed down to its grade level floor in October 2006.

***Lag Storage Addition 2 Foundation***. Lag Storage Addition 2 was a tent structure that was built in 1988 and dismantled in 1993 after it was damaged by high winds. The foundation consists of eight inches of crushed stone covering an area 65 feet by 200 feet.

***Lag Storage Addition 3****.* Lag Storage Addition 3, like Lag Storage Addition 4, is a clear-span structure, with a pre-engineered steel frame and steel sheathing, about 291 feet long, 88 feet wide and 40 feet high, on a seven-inch concrete slab. It is scheduled to be removed down to its concrete floor slab during the work to achieve the interim end state.

***Hardstands*.** Several compacted gravel pads or hardstands are located within WMA 5:

- The Lag hardstand, also known as the old/new hardstand storage area, is located southwest of Lag Storage Additions 3 and 4 and is used to store packaged equipment and containers of LLW;

- The cold hardstand area, which is located west of the Construction and Demolition Debris Landfill, has been used as a nonradioactive material staging and storage area;
- The vitrification vault and empty container hardstand is located north and west of the hazardous waste storage lockers; and
- The HLW tank pump storage vault area.

***Chemical Process Cell Waste Storage Area.*** This waste storage area is a structure used to store equipment removed from the Chemical Process Cell. It is a 200 feet by 70 feet by 30 feet high galvanized steel-panel enclosure with a gravel pad floor. It will be removed down to its gravel pad during the work to achieve the interim end state.

***Hazardous Waste Storage Lockers***. Four steel hazardous waste storage lockers are located east of the Waste Tank Farm. Each locker measures eight feet by 16 feet by eight feet high and is used for short-term storage of hazardous waste. The lockers will be removed during the work to achieve the interim end state.

**WMA 6: Central Project Premises**

Facilities in WMA 6, the Central Project Premises shown in Figure 3-38, include the rail spur, the above ground petroleum storage tank, the Sewage Treatment Plant, the New Cooling Tower, the two Demineralizer Sludge Ponds, the Equalization Basin, the Equalization Tank, the South Waste Tank Farm Test Tower, the Road-Salt and Sand Shed, and the LLW Rail Packaging and Staging Area.

WMA 6 facilities within the scope of this plan are the:

- Sewage Treatment Plant,
- Equalization Basin and Tank,
- Demineralizer Sludge Ponds,
- South Waste Tank Farm Test Tower,
- Concrete slab for the Old Warehouse, and
- Cooling Tower basin.

***Rail Spur***. The rail spur runs about 8,000 feet from the south side of the Process Building to where it connects to the main line of the railroad. Figure 3-39 shows the tracks near the Process Building. The rails are cast iron and the ties are creosote pressure-treated wood. Low-level radioactive contamination identified in soil along a section of dual track east of the Old Warehouse is discussed in Section 4.2.

***Sewage Treatment Plant.*** The Sewage Treatment Plant is a wood frame structure 41 feet by 44 feet by 15 feet high, with metal siding and roofing. The base of the facility is concrete and crushed stone. The Sewage Treatment Plant is used to treat sanitary waste and it contains six in-ground concrete tanks, one above-ground polyethylene tank, and one above-ground stainless steel tank.

***Equalization Basin***. The Equalization Basin is a lined 75 feet wide, 125 feet long, by 10 feet deep basin excavated into the sand and gravel layer. It has been used for non-radioactive discharges.

***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.

***Demineralizer Sludge Ponds***. The north and south demineralizer sludge ponds are separate, unlined basins excavated in the sand and gravel layer. They are approximately 100 feet long, 50 feet wide, and five feet deep. They were used to receive water softener regeneration waste, clarifier overflow and blow-down, boiler blow-down, sand filter backwash, and demineralizer regeneration waste from the Utility Room.

The north pond is nearly filled with sediment. Both ponds are radiologically contaminated. As of 2004, the ponds were no longer in service.

***Old Warehouse Slab***. The Old Warehouse was a pre-engineered steel building with three sections. The main warehouse section was 80 feet by 144 feet by approximately 21 feet high at the roof peak. A 38 feet by 42 feet by 15 feet high room was attached to the north end of the building that housed a radiological counting facility. A double-wide office trailer was located on a concrete foundation wall at the south end of the building. The Old Warehouse was removed down to its concrete floor slab at grade in May 2007.

***New Cooling Tower***. The new cooling tower, shown in Figure 3-40, is 20 feet by 20 feet by 11 feet high and it stands on a concrete basin. The floor of the basin is an eight-inch-thick concrete slab. The facility will be removed, leaving the basin in place, during work to achieve the interim end state.

***Waste Tank Farm Test Towers***. The Waste Tank Farm Test Towers are pre-engineered structures erected as a stack of modules including ladders, handrails, and grating. The exterior “skin” is fabric. The north Tower was 16 feet by 16 feet by 57 feet high. The south Tower is 16 feet by 16 feet by 48 feet high. The north tower was removed to its foundation in October 2006. The south tower will be removed during Phase 1 of the decommissioning.

***Road-Salt and Sand Shed***. The Road-Salt and Sand Shed is a storage bin and a sand stall resting on asphalt pavement. It is constructed with a wooden frame covered with galvanized steel siding. This facility will be removed during work to achieve the interim end state.

***LLW Rail Packaging and Staging Area***. The LLW Rail Packaging and Staging Area covers approximately 27,000 square feet east of and adjacent to the railroad tracks at the south end of WMA 6. The area contains two eight-inch-thick reinforced concrete pads and another section covered with crushed limestone.

**WMA 7: NDA and Associated Facilities**

WMA 7, shown in Figures 3-8 and Figure 3-41, includes the NDA and ancillary structures. The NDA is a near-surface radioactive waste disposal facility about 400 feet wide and 600 feet long. The only WMA 7 facility within the scope of this plan is the NDA Hardstand gravel pad.

The NDA is divisible into three distinct areas: (1) the NFS waste disposal area containing shallow special holes and deep burial holes, (2) the WVDP disposal trenches and caissons, and (3) the area occupied by the Interceptor Trench Project. Other structures and facilities include the Liquid Pretreatment System, the NDA Hardstand, an inactive plant water line, a leachate transfer line, and a former lagoon located beneath the former Interim Waste Storage Facility floor slab. This floor slab was removed in May 2008 as required for the planned installation of the geomembrane cover over the NDA.

The NDA was operated by NFS under license from the NRC for disposal of solid radioactive waste exceeding 200 mrem/h from fuel reprocessing operations. Section 2.4.2 describes the contents of the NDA and the estimated amount of radioactivity it contains.

Descriptions of the various features of the NDA follow:

***NFS Deep Holes***. About 6,600 cubic feet of leached cladding from reprocessed fuel, also known as hulls, are buried in approximately 100 deep disposal holes located in the eastern portion of the U-shaped area. Most of these holes are 2.7 feet by 6.5 feet by 50 to 70 feet deep.

The hulls were contained in 30-gallon steel drums stacked three abreast in the deep holes. Three of these drums contain irradiated, unreprocessed fuel with damaged cladding from the N-Reactor at the Hanford Site. The deep holes also contain LLW generated during fuel reprocessing.

***NFS Special Holes***. Approximately 230 NFS Special Holes are located in the northern and western portions of the U-shaped NFS burial area. The special holes are typically about 20 feet deep, with various lengths and widths; most are about 12 feet wide and 20 to 30 feet long.

The length and width of each special hole were varied according to the quantity of waste requiring disposal at each disposal event, and the dimensions of large waste items such as failed equipment. Miscellaneous wastes, other than leached hulls or related spent fuel debris, were packaged in several types of containers, including steel drums, wooden crates, and cardboard boxes.

At least 22 1,000-gallon tanks containing a mixture of spent n-dodecane and tributyl phosphate in absorbent material were disposed in several special holes during the late 1960s and the early 1970s (Blickwedehl et al. 1987). Eight of these tanks in special holes 10 and 11 were believed to be the source of n-dodecane and tributyl phosphate detected in a nearby monitoring well in the NDA on November 1983.

The following actions were taken by the WVDP between October 1985 and May 1987 to mitigate the migration of the n-dodecane and tributyl phosphate from special holes 10 and 11 (Blickwedehl et al. 1987):

- The eight 1,000-gallon tanks containing the n-dodecane/tributyl phosphate contaminated absorbents were removed.
- The tanks were size-reduced, contaminated absorbents and soils removed, and all waste packaged for disposal.
- Liquid n-dodecane and tributyl phosphate was removed and solidified into a qualified waste form suitable for disposal.
- Special holes 10 and 11 were backfilled.

Approximately 9,700 cubic feet of packaged contaminated soil, contaminated absorbents, size-reduced tanks, and solidified n-dodecane and tributyl phosphate were generated during this removal activity. Low level waste generated during this removal was either disposed of at the Nevada Test Site or the EnergySolutions Clive, Utah disposal site[4], or remains in storage at the WVDP awaiting disposal. Transuranic waste remains in storage at the WVDP awaiting a path for disposal as WVDP transuranic waste is currently not approved for disposal at the Waste Isolation Pilot Plant.

***WVDP Trenches.*** The twelve WVDP trenches contain approximately 200,000 cubic feet of LLW resulting from decontamination activities performed between 1982 and 1986. Most of these wastes are in the parcel of land located inside the U-shaped disposal area used by NFS.

The WVDP Trenches are typically about 30 feet deep and about 15 feet wide. The lengths vary from 30 feet to 250 feet. Trenches 9 and 11 have composite liners and caps. All other WVDP Trenches are capped with clay.

***WVDP Caissons.*** Four steel-lined concrete caissons – cylindrical concrete vaults seven feet in diameter and 60 feet deep – were constructed by the WVDP near the eastern and southern corners of the NDA. WVDP disposal records indicate approximately 823 cubic feet of waste in drums was placed in Caisson 1. The WVDP disposal records do not indicate that any waste was placed in the other three caissons. The caissons are plugged with concrete for shielding and covered with a plastic shield to prevent rainwater infiltration.

***Interceptor Trench and Liquid Pretreatment System.*** The Interceptor Trench and associated Liquid Pretreatment System were installed after groundwater contaminated with tributyl phosphate, n-dodecane, and several radionuclides was detected in a well in the NDA. The purpose of the project was to intercept potentially contaminated groundwater migrating from the NDA.

[4] Which was the Envirocare Clive, Utah site at the time.

The trench is located on the northeast and northwest boundaries of the disposal area. The base of the trench extends to a minimum of one foot below the interface of the weathered Lavery till with the unweathered Lavery till.

The trench is drained by a drainpipe that directs accumulated water to a collection sump. The collection sump has a submersible pump to transfer groundwater to the Liquid Pretreatment System. As of 2008, no groundwater has ever been transferred to the Liquid Pretreatment System.

Liquid that collects in the sump is routinely sampled, analyzed, and transferred to the Low Level Waste Treatment Facility in WMA 2 for treatment and release. Treated wastewater is discharged from Lagoon 3 in WMA 2 to Erdman Brook through the SPDES permitted outfall.

The liquid pretreatment system consists of seven tanks made of carbon steel: one 5,000-gallon holding tank, two 1,000 gallon pre-filtration holding tanks, two 700-gallon tanks containing granular activated carbon, and two 1,000-gallon post-filtration holding tanks. The granular activated carbon tanks are housed in a wooden shed 12 feet long by 10 feet wide. The other five tanks are located in a Quonset-style building.

***Groundwater Barrier Wall.*** In July 2008, a subsurface groundwater barrier wall was installed on the southwest and southeast sides of the NDA to minimize groundwater migration into the disposal area (Figure 3-41). This barrier wall is a soil-bentonite slurry wall with a maximum hydraulic conductivity of 1E-07 cm/s that is keyed at least five feet into the underlying unweathered Lavery till. The slurry wall is approximately 850 feet long, three feet wide, and is 15 to 20 feet deep.

***Geomembrane Cover.*** In the fall of 2008, the NDA was covered with XR-5, an ethylene inter-polymer alloy geomembrane, to limit infiltration of precipitation into the disposal area. Prior to the installation of the XR-5 geomembrane, imported backfill was placed on the surface of the NDA and the surface was graded to form a suitable foundation for the installation of the XR-5 geomembrane.

***NDA Hardstand.*** The NDA Hardstand, located near the southeast corner of the NDA, was an interim storage area where radioactive waste was staged before being disposed. The NDA Hardstand originally was a three-sided structure with cinder block walls, located on a sloped pad of crushed rock 20 feet wide and 20 feet long. The NDA Hardstand is radiologically contaminated. The block walls were removed down to crushed rock pad in September 2006. The crushed rock pad will be removed during Phase 1 of the decommissioning.

***Inactive Plant Water Line.*** An eight-inch diameter cast iron water line from the plant runs along the southwestern border of the NDA. It was formerly used to supply clean water from the reservoirs to the Process Building, but was taken out of service in 1986 and capped with cement.

***Leachate Transfer Line***. The leachate transfer line is a two-inch diameter polyvinylchloride pipeline that runs along the northeast and northwest sides of the NDA, and continues northward across WMA 6 and terminates at Lagoon 2 in WMA 2. It was originally used to transfer liquids from the SDA lagoons via a pumphouse next to the NDA hardstand, to Lagoon 1

The total length of the line is 4,000 feet. The section of the transfer line from the SDA to the interceptor trench sump is inactive and the two ends are capped. The section of the line from the northeast corner of the NDA to Lagoon 2 is currently used to transfer groundwater from the NDA interceptor trench sump.

***Former Lagoon****.* This lagoon, formerly used by NFS for collecting surface water runoff, was located in the northeastern portion of the NDA. Around 1972 it was filled with radiologically contaminated soil from cleanup after a HEPA filter was dropped at the NDA during disposal operations.

**WMA 8: SDA**

The SDA, which is shown on Figure 3-8, is not within the scope of this plan.

**WMA 9: Radwaste Treatment System Drum Cell**

WMA 9 is located south of WMA 7 and it contains the Radwaste Treatment System Drum Cell (Figure 3-42).

***Drum Cell****.* The Drum Cell was built in 1987 to store radioactive waste solidified in cement and packaged in square 71-gallon drums. It is a pre-engineered metal building 375 feet long, 60 feet wide, and 26 feet high. The facility consists of a base pad, concrete shield walls, remote waste handling equipment, container storage areas, and a control room within the weather structure. The base pad consists of concrete blocks set on a layer of compacted crushed stone, underlain by geotextile fabric and compacted clay. Concrete curbs to support the drum stacks lie on top of the base pad.

All of the drums stored in the Drum Cell were removed in 2007 and disposed of at off-site LLW disposal facilities. The Drum Cell will be removed during Phase 1 of the decommissioning.

***Subcontractor Maintenance Area****.* The Subcontractor Maintenance Area is a compacted gravel pad measuring approximately 20 feet by 30 feet located in the northwest corner of WMA 9. Prior to 1991, it was used by construction subcontractors to clean asphalt paving equipment with diesel fuel. In November 1991, the area was remediated by removing the upper six inches of soil and replacing it with clean gravel. The removed soil was tested for toxicity characteristic leaching procedure parameters and found to be nonhazardous solid waste. Since 1991, the area has been used as a staging area for heavy equipment and construction materials (stone, gravel). The gravel pad will be removed during Phase 1 of the decommissioning.

***NDA Trench Soil Container Area***. The NDA Trench Soil Container Area is a gravel pad storage area located on the north side of WMA 9. It was used to store roll-off containers containing soil excavated during the installation of the NDA Interceptor Trench which was completed in 1990. The containers were covered with tarps to prevent infiltration of precipitation and the rear gate was equipped with a rubber gasket to prevent the discharge of any soil or liquid. The roll-off containers and their contained soil have been removed and disposed of offsite. The gravel pad will be removed during Phase 1 of the decommissioning.

**WMA 10: Support and Services Area**

WMA 10, shown in Figure 3-43, covers approximately 30 acres on the north plateau and south plateau, and includes: (1) the Administration Building, (2) the Expanded Laboratory, (3) the New Warehouse, (4) the security gate house, (5) the Meteorological Tower, (6) the main parking lot, and (7) the south parking lot. In addition, concrete slabs and foundations from several removed structures remain in place, along with the former Waste Management Storage Area.

The WMA 10 facilities within the scope of this plan are the New Warehouse, the former Waste Management Storage Area, and the remaining concrete floor slabs and foundations.

***Administration Building***. The administration building is a single-story structure 130 feet long and 40 feet wide, 10 feet high at the eaves, and 11.7 feet high at the peak. The concrete base is nine inches thick. Construction materials include the concrete foundation, wood frame, metal siding, and metal roofing.

The administration building was built during the 1960s. The trailers were added beginning in 1982, and an addition to the west side of the building was added during the early 1980s. The trailers were removed in 2005. The addition to the administration building is approximately 94 feet long and 30 feet wide with a concrete base six inches thick. This facility will be removed to grade during the work to achieve the interim end state.

***Meteorological Tower***. The meteorological tower is located south of the administration building. Constructed of steel, it stands approximately 200 feet high on a concrete foundation. It has three main support columns with interior trusses and is anchored with five support cables. A stand-by generator and electrical boxes rest on a concrete pad.

***Security Gatehouse and Fences***. The main security gatehouse is located adjacent to the Administration Building. It was constructed in 1963. The gatehouse is 34 feet long, 20 feet wide, and nine feet high at the edge of the roof. Construction materials include a concrete foundation, concrete block walls, a concrete slab floor, and a built-up roof with metal deck.

A barbed wire security fence runs along the perimeter of the Center property line and the public roads running through it. The fencing has a total running length of approximately 24 miles.

A steel security fence surrounds the WVDP, the SDA, and miscellaneous other locations. It is made of galvanized chain link with galvanized steel pipe posts, with a spacing of 10 feet. The fence is seven feet high with a total length of 4.7 miles. Three strands of barbed wire are stretched across the top of the fence. Figure 3-5 shows the location of the fence around the project premises.

***Expanded Lab****.* The Expanded Laboratory is located south of the Administration Building. It was constructed during the early 1990s. The laboratory is 92 feet long and 50 feet wide, and consists of eight one-story modular units supported by 72 concrete piers. It was manufactured from light wood framing, metal roofing, and siding. An addition, 20 feet wide and 50 feet long on a concrete foundation wall, was built on the east side of the laboratory. This facility will be removed to grade during the work to achieve the interim end state.

***New Warehouse****.* The New Warehouse was built during the 1980s and is located east of the administration building. It is a pre-engineered steel building, 80 feet wide, 250 feet long, and 21.5 feet high at the roof peak, resting on about 40 concrete piers and a poured concrete foundation wall. The concrete floor is underlain with a gravel base.

***Former Waste Management Storage Area****.* This area is a lay-down area associated with the New Warehouse.

***Parking Lots and Roadways***. Two parking lots are located off Rock Springs Road: the Main Parking Lot and the South Parking Lot.

The Main Parking Lot has a total paved surface area of 180,000 square feet and is covered with asphalt underlain with gravel. The South Parking Lot with approximately 80,000 square feet of parking area is also paved with asphalt. A guardrail approximately 1,200 feet long borders the lot along its southern, eastern and western sides.

Roadways are constructed of a stone sub-base approximately eight-inches thick, covered with asphalt approximately four-inches thick. The total area of pavement is approximately 1,296,000 square feet.

**WMA 11: Bulk Storage Warehouse and Hydrofracture Test Well Area**

The facilities within WMA 11, as shown in Figure 3-9 are not within the scope of this plan. The Bulk Storage Warehouse was formerly called the Plutonium Storage Facility and it was used by NFS in the late 1960s and early 1970s to store plutonium nitrate solution recovered from its nuclear fuel reprocessing operation. The plutonium nitrate solution was contained in 10-liter doubly sealed polyethylene bottles that were stored in containers consisting of two 55-gallon stainless steel drums welded end-to-end and filled with concrete except for a void formed by an embedded 7-inch pipe. In 1974, the Plutonium Storage Facility was deactivated and all stored plutonium nitrate was removed. The building became known as the Bulk Storage Warehouse as it was used by the WVDP as a warehouse to store files and office equipment and was also used as a primary emergency assembly area for the WVDP.

**WMA 12: Balance of the Site**

The facilities within WMA 12, as shown in Figure 3-9, are not within the scope of this plan.

### 3.1.4 Surrounding Communities, Businesses, and Transportation System

The Center is located in a rural area with few population centers (Figures 3-1 and 3-2). The nearest incorporated village is Springville, 3.5 miles north of the WVDP. The hamlet of West Valley and the communities of Riceville and Ashford Hollow also lie within a five-mile radius of WVDP.

Businesses, farms, and community centers within a 3.1-mile radius of the WVDP site in 2004 are listed in Table 3-1. Additional businesses, community centers and manufacturing facilities between 3.1-and 5 mile radii in 2008 included several retail stores, small manufacturing facilities, a concrete supplier, a nursery, a hospital, and two nursing homes.

**Table 3-1. Businesses, Farms, and Community Centers within a 3.1- Mile Radius of the WVDP Site**

| Sector Direction | Facilities | Distance from Stack (miles) |
|---|---|---|
| | **Businesses** | - |
| NE | Split Rail Farm – Horse boarding and breeding | 1.42 |
| W | Storage Warehouse | 2.36 |
| W | NORCO Propane Co./Pioneer Propane | 2.34 |
| W | Countryside Car Center | 2.37 |
| WSW | Country Gifts and Storage | 2.35 |
| WSW | Starcrest Homes (Home Business) & U-Haul | 2.34 |
| WSW | Heritage Pipe Organ | 2.43 |
| WSW | (Riefler Inc.) | 2.78 |
| ESE | Harrigan Realty – Attorney at Law | 2.13 |
| NW | Springville Country Club | 3.04 |
| WSW | M&M Holland Propane | 2.40 |
| W | L. A. Hazard | 2.27 |
| SE | Gerwitz and McNeil Electric | 2.01 |
| W | Ashford Auto and Marine Repair | 2.31 |
| SE | Fox Valley Greenhouse | 1.83 |
| NW | Jack R. Preston's AutoBarn | 0.94 |
| SW | Phillip's Christmas Tree and Wreath | 3.01 |
| N | Codd's Flower Shop | 1.57 |

**Table 3-1. Businesses, Farms, and Community Centers within a 3.1- Mile Radius of the WVDP Site**

| Sector Direction | Facilities | Distance from Stack (miles) |
|---|---|---|
| NNW | Model Shop | 1.28 |
| W | House of Steel | 2.26 |
| N | Schichtel's Nursery – Bond Rd | 1.56 |
| WNW | Schichtel's Nursery – Peters Rd | 2.62 |
| | **Farms** | - |
| S | Tom Stuebchen - Fruit Trees | 2.28 |
| S | Charles Schichtel – Dairy Farm | 2.32 |
| N | Clemence and Claudia Wolniewicz - Grain and Hay | 2.45 |
| NNW | David Reed – Dairy Farm | 2.33 |
| SE | Wayne Widrig – Dairy Farm | 1.80 |
| SE | Gary Feldman – Dairy Farm | 3.11 |
| WNW | Willard and Ann Miller – Dairy Farm | 2.55 |
| SE | Kevin Hebdon – Dairy Farm | 2.95 |
| WNW | David Cobo – Farm | 1.15 |
| WSW | Timothy Klahn – Dairy Farm | 2.51 |
| | **Community Centers** | - |
| SE | American Legion | 3.00 |
| E | Islamic Academy | 2.91 |
| N | Springfield Field and Stream | 3.09 |
| WNW | Trinity Lutheran | 1.19 |
| ENE | Cattaraugus County Houndsmen and Conservation Club | 1.62 |
| E | Riceville Community Church | 2.83 |
| SE | Ashford Municipal Building | 1.71 |

A small military research installation is located in Cattaraugus County approximately 3.1 miles northeast of the WVDP. This facility was used to conduct research for the U.S. Department of Defense Air Force Automatic Liquid Agent Detector Program.

**Transportation System**

Transportation facilities near the Center include highways, transport repair and refueling services, rail lines, and aviation facilities.

The primary method of transportation near the site is motor vehicle traffic on the highway system shown in Figure 3-2. In Cattaraugus County, all roads with the exception of those within the cities of Olean and Salamanca are considered rural roads.

Rural principal arterial highways connect population and industrial centers. These include U.S. Route 219, located 2.6 miles west of the site, Interstate 86, located approximately 21.7 miles south of the site, and the New York State Thruway (I-90), approximately 21.7 miles north of the site. Traffic volume along the section of U.S. 219 west of the site between New York Route 39 and the Cattaraugus County Line averaged 9966 vehicles per day in 2002 (NYDOT 2005). Construction of a 4.2 mile extension of U.S. Route 219 began in 2007.

Collectors are roads from smaller communities and industrial centers to the rural principal arterial highways. They frequently are intra-county in nature and serve short hauls and cross-county traffic. There are three county collector roads within 1.2 miles of the site. Schwartz Road and Rock Springs Road serve as the principal site access roads. State Route 240, also identified as County Route 32, is 1.2 miles northeast of the site. The average annual daily traffic volume on State Route 240 near the site was 978 vehicles in 2002 (NYDOT 2003).

Dutch Hill Road, approximately one mile west of the WVDP, is an oil and stone chip surface on a gravel base designed to accommodate local, lightweight vehicles. Edies Road is of similar construction. Mill Street is asphalt paved over a gravel base located on unstable soils.

Railroad service in a north-south direction is provided to the central part of Cattaraugus County. The Buffalo and Pittsburgh Railroad transects the Center approximately 0.5 mile east of the project premises at its nearest point. This rail line is now abandoned north of the Center. The Center is served by a railroad siding from this line, often referred to as the rail spur.

There are no commercial airports in the site vicinity. The only major aviation facility in Cattaraugus County is the Olean Municipal Airport, located in the Town of Ischua, 21 miles southeast of the site, which does not offer regularly scheduled commercial air service. The nearest major airport is Buffalo Niagara International Airport, 34 miles north of the site.

## 3.2 Population Distribution

Local population information was obtained from a demographic survey performed in the area of the WVDP in 2002 (URS 2002) and regional population information from the 2000 U.S. census (Census Bureau 2003). This demographic survey referenced in Sections 3.2 and 3.3 has not been updated as of 2008. For analysis purposes, the area surrounding the WVDP is divided into 16 compass-direction sectors, with the WVDP main stack as the reference point.

### 3.2.1 Local Population Data

The 2002 demographic survey was performed out to a 3.1-mile radius from the WVDP Main Plant stack and included all permanent structures that may be inhabited in that area.

Results of this survey appear in Tables 3-2 and 3-3.

In 2002, approximately 1,050 people lived within a 3.1-mile radius of the site. The largest numbers of individuals were located east of the site. Figure 3-44 shows the results of the demographic survey by compass vectors.

**Table 3-2. 2002 Resident Population Estimates by Directional Sector Within a 3.1-Mile Radius of the Main Plant Stack** (URS 2002)

| Sector | | Radius (miles) | | | | | |
|---|---|---|---|---|---|---|---|
| | | 0.3-0.6 | 0.6-1.2 | 1.2-1.9 | 1.9-2.5 | 2.5-3.1 | TOTAL |
| A | N | 0 | 0 | 19 | 17 | 18 | 54 |
| B | NNE | 0 | 0 | 19 | 52 | 34 | 105 |
| C | NE | 0 | 3 | 17 | 0 | 21 | 41 |
| D | ENE | 0 | 2 | 27 | 0 | 19 | 48 |
| E | E | 0 | 0 | 38 | 55 | 81 | 174 |
| F | ESE | 0 | 0 | 4 | 48 | 15 | 67 |
| G | SE | 0 | 0 | 6 | 29 | 30 | 65 |
| H | SSE | 0 | 0 | 0 | 26 | 24 | 50 |
| I | S | 0 | 0 | 6 | 12 | 8 | 26 |
| J | SSW | 0 | 0 | 2 | 10 | 19 | 31 |
| K | SW | 0 | 0 | 9 | 0 | 43 | 52 |
| L | WSW | 0 | 0 | 9 | 14 | 4 | 27 |
| M | W | 0 | 8 | 35 | 21 | 15 | 79 |
| N | WNW | 0 | 29 | 41 | 4 | 24 | 98 |
| O | NW | 0 | 9 | 65 | 13 | 2 | 89 |
| P | NNW | 0 | 6 | 14 | 19 | 11 | 50 |
| **TOTALS** | | **0** | **57** | **311** | **320** | **368** | **1,056** |

The nearest residences are located 0.76 to 1.94 miles from the WVDP site as shown in Table 3-3. The numbers of wells or springs used as drinking water within 3.1 miles of the WVDP are listed in Table 3-4. The information in the table is not inclusive of every well used for water consumption because the survey was subject to residential participation.

**Table 3-3. Nearest Residences by Sector** (URS 2002)

| Compass Direction | Distance (mi) | Residence Location |
|---|---|---|
| WNW | 0.76 | 6491 Boberg Rd. |
| NW | 0.83 | 10493 Rock Springs Road |
| W | 1.09 | 10314 Dutch Hill Rd. |
| NNW | 1.17 | 10596 Rock Springs Rd. |
| NE | 1.20 | 10653 Rte. 240 |

**Table 3-3. Nearest Residences by Sector** (URS 2002)

| Compass Direction | Distance (mi) | Residence Location |
|---|---|---|
| ENE | 1.22 | 10625 Rte. 240 |
| SW | 1.33 | 10086 Dutch Hill Rd. |
| WSW | 1.33 | 10122 Dutch Hill Rd. |
| S | 1.42 | 9911 Rock Springs Rd. |
| E | 1.53 | 5761 Heinz Rd. |
| N | 1.53 | 10927 Bond Road |
| NNE | 1.63 | 10845 Rte. 240 |
| ESE | 1.63 | 5579 Buttermilk Rd |
| SSW | 1.76 | 10043 Dutch Hill Rd. |
| SE | 1.80 | 5768 Fox Valley Rd. |
| SSE | 1.94 | 5872 Fox Valley Rd. |

**Table 3-4. Number of Residential Wells or Springs used for Drinking Water by Sector within a 3.1-Mile Radius of the Main Plant Stack**

| Sector | Direction | Number of Wells or Springs(1) |
|---|---|---|
| A | N | 14 |
| B | NNE | 23 |
| C | NE | 5 |
| D | ENE | 10 |
| E | E | 36 |
| F | ESE | 20 |
| G | SE | 8 |
| H | SSE | 12 |
| I | S | 7 |
| J | SSW | 11 |
| K | SW | 20 |
| L | WSW | 9 |
| M | W | 22 |
| N | WNW | 24 |
| O | NW | 27 |
| P | NNW | 11 |

**Table 3-4. Number of Residential Wells or Springs used for Drinking Water by Sector within a 3.1-Mile Radius of the Main Plant Stack**

| Sector | Direction | Number of Wells or Springs(1) |
|---|---|---|
| TOTAL | | 259 |

NOTE: (1) Numbers of wells and springs estimated based upon resident interviews in URS 2002.

### 3.2.2 Population Distribution

The Center lies within Cattaraugus and Erie counties. Regional population data within a 50-mile radius of the WVDP was obtained from the 2000 U.S. Census.

**Summary of Current Population In and Around the Site**

The 1960 through 2000 resident populations of towns and villages within 10 miles of the WVDP are presented in Table 3-5[5]. The populations of New York and Pennsylvania counties within 50 miles of the WVDP are presented in Table 3-6.

Erie County had a population of 950,265 in 2000, which is a 10.7 percent decline from 1960. Although both Erie County and the City of Buffalo have experienced a population decline, populations in the rural townships south of Buffalo – such as Orchard Park, Hamburg, East Aurora, and West Falls – have increased. The population of southern Erie County near the WVDP site is concentrated primarily in small villages and along roadways, much like in Cattaraugus County. The majority of people residing in these areas work in agriculture or nearby small industries.

**Table 3-5. Locations and Populations of Towns and Villages Partially or Totally Within 10 Miles of the Site** (from 2000 census)

| TOWN/ VILLAGE(1) | DISTANCE/ DIRECTION (Miles) | POPULATION | | | | | POP. DENSITY per sq.mi. | 1960-1990 % CHG. | 1990-2000 % CHG. |
|---|---|---|---|---|---|---|---|---|---|
| | | 1960 | 1970 | 1980 | 1990 | 2000 | | | |
| Ashford (T) | Note (4) | 1,490 | 1,577 | 1,922 | 2,162 | 2,223 | 43 | 45.1 | 2.82 |
| Concord (T) | 3.0N | 6,452 | 7,573 | 8,171 | 8,387 | 8,526 | 122 | 30.0 | 1.66 |
| Springville (V)(2) | 3.5N | 3,852 | 4,350 | 4,285 | 4,310 | 4,252 | N/A | 11.9 | -1.35 |
| Sardinia (T) | 4.0 NNE | 2,145 | 2,505 | 2,792 | 2,667 | 2,692 | 54 | 24.3 | 0.94 |
| Yorkshire (T) | 3.5 NNE | 2,012 | 2,627 | 3,620 | 3,905 | 4,210 | 114 | 94.1 | 7.81 |
| Delevan (V)(3) | 8.9 ENE | 777 | 994 | 1,113 | 1,214 | 2,321 | N/A | 56.2 | 91.2 |
| Machias (T) | 4.0 ESE | 1,390 | 1,749 | 2,058 | 2,338 | 2,482 | 61 | 68.2 | 6.16 |
| Franklinville (T) | 7.8 SSE | 3,090 | 2,847 | 3,102 | 2,968 | 3,128 | 60 | -3.9 | 5.39 |
| Ellicottville (T) | 12.0 S | 1,968 | 1,779 | 1,677 | 1,607 | 1,738 | 39 | -18.3 | 8.15 |
| Mansfield (T) | 7.5 SSW | 632 | 605 | 784 | 724 | 800 | 20 | 14.6 | 10.50 |

[5] In New York state, a town is the major subdivision of each county and a village is an incorporated area, usually within a town.

**Table 3-5. Locations and Populations of Towns and Villages Partially or Totally Within 10 Miles of the Site** (from 2000 census)

| TOWN/ VILLAGE(1) | DISTANCE/ DIRECTION (Miles) | POPULATION | | | | | POP. DENSITY per sq.mi. | 1960-1990 % CHG. | 1990-2000 % CHG. |
|---|---|---|---|---|---|---|---|---|---|
| | | 1960 | 1970 | 1980 | 1990 | 2000 | | | |
| East Otto (T) | 3.0 SW | 701 | 910 | 942 | 1,003 | 1,105 | 27 | 43.1 | 10.17 |
| Otto (T) | 7.5 WSW | 715 | 731 | 828 | 777 | 831 | 26 | 8.7 | 6.95 |
| Collins (T) | 7.5 WNW | 6,984 | 6,400 | 5,037 | 6,020 | 8,307 | 173 | -13.8 | 37.99 |
| North Collins(T) | 8.9 NW | 3,805 | 4,090 | 3,791 | 3,502 | 3,376 | 79 | -8.0 | -3.60 |
| TOTAL (OR AVERAGE) | | 31,384 | 33,393 | 34,724 | 36,060 | 39,418 | --- | 14.9 | 14.9 |

NOTES: (1) (T) indicates town and (V) indicates village.

(2) Springville village population is included in the town of Concord.

(3) Delevan village population is included in the town of Yorkshire.

(4) The WVDP is located within the geographical boundary of the Town of Ashford.

**Population Density**

Using the 2000 census data, the maximum population density of 448 persons per square mile occurs between 20 and 30 miles from the site. Table 3-5 includes the population densities of towns within 10 miles of the WVDP site.

**Table 3-6. Populations of Selected Municipalities, Counties, and States within 50 Miles of the Site (1960-2000)** (from U.S. Census, years cited)

| MUNICIPALITY/ COUNTY/STATE(1) | POPULATION | | | | | % Change 1960-2000 |
|---|---|---|---|---|---|---|
| | 1960 | 1970 | 1980 | 1990 | 2000 | |
| NEW YORK (S) | 16,782,304 | 18,241,391 | 17,558,072 | 17,990,455 | 18,976,457 | 13.1 |
| Cattaraugus (C) | 80,187 | 81,666 | 85,697 | 84,234 | 83,955 | 4.7 |
| Erie (C) | 1,064,688 | 1,113,491 | 1,015,472 | 968,532 | 950,265 | -10.7 |
| Hamburg (M) | 41,288 | 47,644 | 53,270 | 53,735 | 56,259 | 36.3 |
| Orchard Park (M) | 15,876 | 19,978 | 24,359 | 24,632 | 27,637 | 74.1 |
| Buffalo (M) | 532,759 | 462,768 | 357,870 | 328,123 | 292,648 | -45.1 |
| Allegany (C) | 43,978 | 46,458 | 51,742 | 50,470 | 49,927 | 13.5 |
| Wyoming (C) | 34,793 | 37,688 | 39,895 | 42,507 | 43,424 | 24.8 |
| Chautauqua (C) | 145,377 | 147,305 | 146,925 | 141,895 | 139,750 | -3.9 |
| Livingston (C) | 44,053 | 54,041 | 57,006 | 62,372 | 64,328 | 46.0 |
| Genesee (C) | 53,994 | 58,722 | 59,400 | 60,060 | 60,370 | 11.8 |
| Niagara (C) | 242,269 | 235,720 | 227,101 | 220,756 | 219,846 | -9.3 |

**Table 3-6. Populations of Selected Municipalities, Counties, and States within 50 Miles of the Site (1960-2000)** (from U.S. Census, years cited)

| MUNICIPALITY/ COUNTY/STATE(1) | POPULATION | | | | | % Change 1960-2000 |
|---|---|---|---|---|---|---|
| | 1960 | 1970 | 1980 | 1990 | 2000 | |
| Steuben (C) | 97,691 | 99,546 | 99,135 | 99,088 | 98,726 | -1.1 |
| PENNSYLVANIA (S) | 11,319,366 | 11,800,766 | 11,866,728 | 11,881,643 | 12,281,054 | 8.5 |
| Warren (C) | 45,582 | 47,682 | 47,449 | 45,050 | 43,863 | -3.8 |
| McKean (C) | 54,517 | 51,915 | 50,635 | 47,131 | 45,936 | -15.7 |
| Potter (C) | 16,483 | 16,395 | 17,726 | 16,717 | 18,080 | 9.7 |

NOTE: (1) (M) indicates municipality, (C) indicates county, and (S) indicates state.

#### Transient Population

The transient population around the site includes daily and seasonal transients including the workforce at the WVDP. In 2008, an average of 300 employees was working at the site during daytime hours.

This transient population is projected to vary in future years according to the activities on site. The seasonal transient population is associated with the area's numerous small recreation sites. Where significant, this transient population is included in the distribution and projection figures.

#### Future Projected Population

According to the Greater Buffalo-Niagara Regional Transportation Council, the total Concord/Springville population is expected to reach 10,000 by the year 2020, a gain of almost 10 percent per decade. It is projected that the present 50/50 population split will continue, with Springville having 5070 people and the unincorporated areas of the town 4930 in 2020 (ECPD 1999). Population projections for Cattaraugus County were prepared by Cornell University in September of 2002 and are available for public viewing on the New York State Information System website (http://www.nysis.cornell.edu/cattaraugus.pdf). Projected population changes for Cattaraugus County were as follows:

| | | |
|---|---|---|
| 2005 - 83,881 | 2010 - 83,674 | 2015 - 83,359 |
| 2020 - 82,815 | 2025 - 81,989 | 2030 - 80,886 |

Population trends may be influenced by the expansion of Route 219 through Cattaraugus County. The baseline population projections are projections illustrating the impact of recent rates of population change. Census 2000 county populations have been projected using current life expectancy and survival rates, age specific fertility rates, and rates of net migration. The rates of net migration have the greatest impact on changes in population size. These net migration rates are based on an analysis of total population change between the 1990 census and the 2000 census. In 2008, the U.S. Census Bureau estimated that the population of Cattaraugus County was 79,688.

## 3.3 Current and Future Land Use

This section describes current land use on the site and in the vicinity in detail, and future land use on site and in the vicinity within the limitations of available information.

### 3.3.1 Current Land Use

Detailed information on current land use is available from a number of sources.

**Onsite Land Use**

The project premises have served only industrial uses since the reprocessing plant was built in the 1960s. The balance of the Center, often referred to as the retained premises, has served only as a buffer area for the plant since that time. In 2008, no definitive information on plans for future use of the Center was available.

**Land Use in Vicinity of the WVDP**

Land use within five miles of the WVDP site is predominantly associated with agriculture, arboriculture, and forestry. The major exception is the Village of Springville, in which many areas are devoted to residential, commercial, and industrial land uses. Other major non-agricultural land uses within five miles of the site are:

- Hamlet of West Valley – residential/commercial/land use, 3.4 miles to the southeast;
- Cattaraugus County Forest – forestry/recreation, 3.7 miles to the south;
- Campground – five miles to the southwest;
- Machine shop – industrial land use, four miles to the northwest;
- Two retail shopping complexes - commercial land use - four miles to the north northwest; and
- Warehouse – commercial land use, 3.8 miles to the north-northwest in the village of Springville.

Cattaraugus County ranks fifth in the state for number of farms and eleventh in the state for the amount of land in farming. Approximately 24 percent of the county’s total acreage is farmland (NYASS 2005). Production and sale of important agricultural commodities in Cattaraugus County are shown in Table 3-7. The dairy industry is the dominant agricultural activity, with meat production occurring on a smaller scale.

**Table 3-7. Leading Agricultural Products in Cattaraugus County[1]**

| Product | 2002 Sales in $1000s | Percent of Total Sales | County Rank in New York |
|---|---|---|---|
| Dairy Products | 36,486 | 63 | 19 |
| Nursery and Greenhouse | 9,676 | 17 | 5 |
| Cattle and Calves | 4,832 | 8 | 22 |
| Hay & Silage | 1,976 | 3 | 28 |
| Grains and Dry Beans | 1,628 | 3 | 22 |
| Other Products | 3754 | 6 | |
| Total Sales | 58,352 | - | 22 |

NOTE: (1) From NYASS 2005.

**Farming Statistics**

In 2002, a livestock and crop production survey within a 3.1-mile radius of the WVDP was taken in conjunction with the population survey. The results of this survey are shown in Tables 3-8 and 3-9.

**Table 3-8. 2002 Consumable Animal Population Estimates[1] by Sector within a 3.1-Mile Radius of the Main Plant Stack (URS 2002)**

| Sector | Direction | Dairy Cattle | Beef Cattle | Goats | Sheep | Pigs | Fowl[2] |
|---|---|---|---|---|---|---|---|
| A | N | 0 | 0 | 0 | 0 | 0 | 0 |
| B | NNE | 0 | 11 | 0 | 0 | 0 | 0 |
| C | NE | 0 | 23 | 0 | 0 | 0 | 0 |
| D | ENE | 12 | 11 | 15 | 12 | 5 | 20 |
| E | E | 17 | 31 | 0 | 7 | 0 | 0 |
| F | ESE | 0 | 0 | 0 | 0 | 0 | 6 |
| G | SE | 135 | 0 | 0 | 15 | 0 | 32 |
| H | SSE | 0 | 0 | 0 | 0 | 0 | 0 |
| I | S | 100 | 12 | 0 | 0 | 0 | 0 |
| J | SSW | 60 | 45 | 0 | 0 | 2 | 4 |
| K | SW | 3 | 0 | 0 | 0 | 2 | 17 |
| L | WSW | 0 | 5 | 0 | 0 | 0 | 0 |
| M | W | 0 | 36 | 5 | 0 | 2 | 21 |
| N | WNW | 70 | 0 | 0 | 0 | 0 | 9 |
| O | NW | 5 | 0 | 0 | 0 | 1 | 13 |
| P | NNW | 60 | 0 | 0 | 30 | 0 | 20 |
| **TOTALS** | | **462** | **174** | **20** | **64** | **12** | **142** |

NOTES: (1) Numbers of animals are estimated based upon resident interviews and site reconnaissance.
(2) Fowl includes: Chickens, Ducks, Geese, Turkey, Ostrich (4) and Emu (1).

Dairy and beef cattle farming dominate within 3.1 miles of the WVDP. The majority of livestock production occurs northwest and southeast of the WVDP. Farming within 3.1 miles of the site typically occurs northwest and south and east of the site. The principal use of farmland is hay and pasture land. Hay and pasture lands account for approximately 57 percent of land used for agricultural purposes. The production of corn and oats accounts for 45 percent of agricultural land use.

Land-use surrounding the Center property – based on county land-use maps and tax parcel information – is shown in Figure 3-45.

**Table 3-9. 2002 Crop Estimates in Acres by Sector within a 3.1-Mile Radius of the Main Plant Stack** (from URS 2002)

| Sector | Direction | Corn | Oats | Hay & Pasture | Ground Fruit(1) | Fruit Trees(2) | Garden Vegetables(3) |
|---|---|---|---|---|---|---|---|
| A | N | 60 | 0 | 0 | 1 | 0 | 0.4 |
| B | NNE | 0 | 0 | 0 | 0 | 0 | 1.8 |
| C | NE | 0 | 0 | 0 | 0 | 0 | 0.5 |
| D | ENE | 0 | 0 | 0 | 0 | 0.2 | 1.1 |
| E | E | 0 | 0 | 0 | 0 | 0 | 1.3 |
| F | ESE | 0 | 0 | 100 | 0 | 0 | 0.2 |
| G | SE | 83 | 34 | 250 | 0 | 0 | 1.7 |
| H | SSE | 0 | 0 | 30 | 0 | 0 | 0.4 |
| I | S | 50 | 50 | 100 | 1 | 0 | 1.2 |
| J | SSW | 30 | 30 | 50 | 0 | 0 | 0.8 |
| K | SW | 0 | 0 | 0 | 0 | 0 | 1.0 |
| L | WSW | 0 | 0 | 0 | 0 | 0 | 0.0 |
| M | W | 0 | 0 | 80 | 0 | 0 | 0.8 |
| N | WNW | 230 | 0 | 100 | 0 | 0 | 0.7 |
| O | NW | 0 | 0 | 0 | 0 | 0 | 1.0 |
| P | NNW | 0 | 0 | 0 | 0 | 0 | 0.8 |
| **TOTALS** | | **453** | **114** | **710** | **2** | **0.2** | **13.7** |

NOTES: (1) Ground Fruit includes: blueberries, raspberries, strawberries, and grapes.
(2) Fruit Trees includes: apples and pears
(3) Garden vegetables included: beans, cabbage, corn, cucumbers, peas, potatoes, pumpkins, tomatoes, squash, and zucchini.

Agricultural lands cultivated to produce fruits and vegetables represent less than one percent of the total agricultural acreage within 3.1 miles of the site. Fruit and vegetable fields tend to be smaller than dairy fields, and are not distributed in proportion to the occurrence of farmland. In general a few towns contain a disproportionately large share of these lands. Crops include lettuce, cabbage, broccoli, spinach, snap beans, tomatoes, sweet corn, potatoes, grapes, and apples. Total land area devoted to such production in Erie and Cattaraugus counties is estimated at 10,189 acres and 2,319 acres, respectively.

### 3.3.2 Summary of Anticipated Land Uses

The project premises will be available for only limited future uses in the coming decades. The ability to anticipate land use in the vicinity in future years is limited by the limited available information from planning boards.

**Future Use of Project Premises and the Center**

Future use of the retained premises will depend upon the wishes of NYSERDA as the property owner and will need to be consistent with institutional controls, where applicable. As of 2008, no definitive information on NYSERDA plans for future use of the Center was available. However, the Southern Tier West Regional Planning and Development Board has an ongoing West Valley Redevelopment Strategy Project in response to the ongoing decommissioning of the WVDP.

**Future Use of Land in the Vicinity**

It is expected that future land uses in the vicinity of the Center will be similar to the historical land uses summarized in Section 3.3.1. Information from local, regional, and State planning boards is limited. On June 9, 1999 the Town of Concord and the Village of Springville held a public hearing to review a draft of the joint comprehensive plan (ECPD 1999). The vision of the plan was expressed as follows:

> "The Concord/Springville community values and wishes to preserve the scenic beauty, farmland, hamlets, and unique natural environment of the Town of Concord. It also wishes to enhance and strengthen the Village of Springville as the civic, cultural and economic center of Concord and the surrounding non-town area, and maximize its location at the southern gateway to Erie County."

Proposed developments related to this vision included:

- A 50-acre planned business park adjacent to US Route 219;
- Revitalization of downtown Springville;
- A new planned residential area in the northeastern section of the Village,
- Upgrading of the Town and Village Hall facilities; and
- Park and recreation improvements, which included a new park at Scoby Hill Dam and a new greenway along Spring Brook.

The greenway development would include a four-mile-long park area bordering Spring Brook from Middle Road to Cattaraugus Creek at Felton Bridge on Mill Street. This park would include nature trails, bicycle paths, canoe landings, and picnic areas.

The new park at Scoby Hill Dam would include a canoe landing, fishing access, and recreational use. Further recreational development is proposed to encourage the development of hiking/biking trails, golf, snowmobiling, and skiing.

Additional proposals utilized the abandoned Buffalo-Pittsburgh Railroad line from Springville to Salamanca to be developed either as a tourism train, connected with a railroad museum in Salamanca, or as a extensive bike trail as part of the "rails to trails" program.

Industrial and business development would be encouraged at or near current locations (along Cascade Drive and near the railroad tracks), with the exception of a planned new business park located near the Zoar Valley Road, with a connector road intended to the future Route 219. If Route 219 were to be extended down to Salamanca, certain land adjacent the route would be developed for business and/or industrial use (Ashford 1994).

Sand and gravel mining is a growing industry within the area with nine areas now designated for mining. Future intentions are to develop this industry to promote economic development in the area (Bishop, et al. 2004).

**Cattaraugus County**

The 1994 Comprehensive Master Plan anticipated much of its land use based on the extension of Route 219 and the development of the nuclear fuel industry through the WVDP. Given these assumptions, industrial and business development was planned to occur near the Route 219 extension and on some Center property.

Parcels reserved for industry in the future land use plan are located near the following roads: Henrietta Road (300 acres), Schwartz Road (50 acres), Route 219 (80 acres), Thomas Corners (350 acres), and within the Town of Ashford (265 acres). The closest business development complex to the WVDP property would be the Ashford Business and Education Park at the location of the Ashford Office Complex. The intersection of Route 219 and Schwartz Road, and Thomas Corners have been intended for residential development (Ashford 1994).

The Record of Decision on the Route 219 expansion was published in April 2003. The New York Department of Transportation selected the freeway alternative, which proposes a four-lane freeway from Springville to Salamanca. Construction of the Route 219 expansion began in 2007.

Since the Comprehensive Master Plan was published, gravel mining has expanded rapidly. In 1993, 53 parcels of land totaling 3,455 acres were assessed for mining and quarrying in the Route 16 corridor of Cattaraugus County. This number increased to 76 parcels totaling 4,502 acres in 1999. In 2000, there were 49 active mining permits covering 1,030 acres.

Issues raised by concerned citizens have resulted in the Town of Yorkshire adapting zoning plans to remediate gravel mining activities. As of October 2002, the Town of Ashford had not adapted any zoning regulations.

## 3.4 Meteorology and Climatology

This section begins with a description of the general climate in the region, followed by a discussion of severe weather phenomena. Weather-related radionuclide transmission factors and site deterioration factors are then described. Finally, site meteorology is discussed, along with air quality in the area.

### 3.4.1 The General Climate of Western New York

Western New York is exposed to a variety of air masses that create a moist continental climate. Cold dry air masses that form over Canada reach the area from the northwest. Prevailing winds from the southwest and south bring warm, humid air masses from the Gulf of Mexico and neighboring waters of the subtropical Atlantic Ocean. On occasion, cool, cloudy, and damp weather affects Western New York through air flow from the east and northeast.

Western New York is affected by a variety of cyclonic and anti-cyclonic pressure systems as they move across the continent. Continental storms and frontal systems move frequently across or near this region. In addition, Western New York usually feels the effects of well-developed storms moving up the Atlantic Coast.

#### Temperature

The coldest winter temperature normally varies between -10 °F to -20 °F in the southwestern highlands (WVNSCO 2007). Extreme winter temperatures as cold as -40 °F have been recorded in the higher elevations of Cattaraugus County (WVNSCO 2007). Severe winter cold with below-zero minimums and/or lengthy periods of continuous temperatures below freezing occur between early December and mid-March. Winter thaws typically result in temperatures in the 40s to low 50s for a few days at a time, with rare maximums in the 60s.

The summer seasons are cool with the temperature typically ranging from 60 °F at night to the low 80s in the afternoon (WVNSCO 2007). On the average, temperatures of 90 °F or higher are recorded on five days or less per year at the higher elevations and along the shore of the Great Lakes (WVNSCO 2007). Such temperatures occur between early June and early September. Readings of 100 °F or higher are rare. It is sunny for 65 percent of the total daylight hours on the average during the summer (WVNSCO 2007).

Temperatures from mid-September to mid-October frequently rise to the 60s and 70s in the daytime and cool to the 30s and low 40s at night. The comparatively warm waters of the Great Lakes reduce cooling at night to the extent that freezing temperatures in lakeside counties are normally delayed until mid-October or later.

#### Precipitation

Lake Erie and Lake Ontario exert a major controlling influence on the climate of the region. In winter, cold air crossing unfrozen lake water picks up moisture and releases it as snow as the air stream moves inland over higher terrain. Heavy snow squalls frequently occur, producing from one to two feet of snow and occasionally as much as four to seven feet. Cattaraugus County and Erie County are generally subject to lake-effect snows in

November and December, but as the lake gradually freezes lake-effect snow becomes less frequent. The snow season normally begins in mid-November and extends into mid- or late-April.

Winter precipitation is heaviest east of Lake Erie, where the average total snowfall is in excess of 120 inches (WVNSCO 2007). Summer season precipitation ranges from 10 to 12 inches with the rainfall distribution pattern reflecting the influences of the cool Lake Ontario waters to the north and the hilly terrain in the Southern Tier (WVNSCO 2007). Rains resulting from warm fronts are usually light but last for several days; cold fronts often cause heavier rainfall in shorter periods.

### 3.4.2 Severe Weather Phenomena

Figures 3-46 through 3-48, provided by the National Weather Service observing station in Buffalo, show the distribution patterns of tornadoes (1950-2002), thunderstorm winds (1955-2002), and hail events (1955-2002) for western and north central New York. The National Weather Service has not updated these figures as of 2008. Corresponding charts depict distribution of events by month, time, and rating of severity.

Severe weather phenomena occurred during the 1993-2002 period as follows:

- Six tornadoes;
- Seventy-five thunderstorm wind or hail events (where thunderstorm winds measured 58 mph or greater or produced damage, or where hail measuring 0.75-inch or larger fell);
- Seven injuries due to lightning strikes;
- Forty-nine flood or flash flood events (about one-third due to ice jams);
- Twenty-eight high wind events (high winds caused by large-scale, synoptic low pressure systems);
- Three ice storms (with ice accumulations of one-half inch or greater);
- One blizzard in March 1993 (with winds or frequent gusts of 35 mph or greater and visibilities of less than one-fourth mile sustained for three hours or more); and
- Sixty-six snowstorms (with seven inches or more of snow within a 12- hour period, or nine inches or more of snow within 24 hours, about two-thirds due to lake-effect snows.)

Additional historical meteorological data is provided in WVNSCO 1993b, which summarizes regional meteorological information, analyzes trends, and correlates meteorological data collected by the National Weather Service with data collected at the site's regional and primary monitoring stations.

### 3.4.3 Weather-Related Radionuclide Transmission Factors

Winds at the site are generally from the west and south at about 10.3 miles per hour (4.6 m/s) and 9.6 miles per hour (4.3 m/s) respectively, based on data from 1991-2002. Figure 3-49 depicts the average wind vectors on site.

The strongest winds occur from November through March and are generally southwesterly to west-southwesterly. The weakest winds occur from May to October and are generally southwesterly to southerly (WVNSCO 1993).

Average and extreme duration of precipitation events are not measured at the WVDP. Only annual, monthly, or daily precipitation data are available, recorded as inches fallen in a 24-hour period.

### 3.4.4 Weather-Related Site Deterioration Parameters

Routine and extreme weather-related site deterioration parameters are considered in this section.

#### Routine Parameters

Note that precipitation intensity is indicated by information provided in Section 3.4.5. The hourly average maximum recorded wind speed in the area was 35.3 miles per hour in December of 1987 (WVNSCO 1993).

Wind vectors were addressed in Section 3.4.3. Temperature gradients were discussed in Section 3.4.1. Limited data are available on pressure gradient variation: reported barometric pressure measurements in 1991 and 1992 have ranged from lows of 29.51 in March of 1991 and 28.17 in May of 1992 to highs of 30.67 in December of 1991 and 30.43 in January of 1992 (WVNSCO 1993b).

#### Extreme Parameters

Most extreme weather-related deterioration events that occurred during the 1993 – 2002 period were summarized in Section 3.4.2. Regarding extreme air pollution, the WVDP and Cattaraugus County are considered “in attainment” or “unclassifiable” with respect to the National Ambient Air Quality Standards for criteria pollutants. As of 2002, no extreme air pollution violations have been identified within Cattaraugus County.

### 3.4.5 Site Meteorology and Climatology

Site topographic features previously discussed produce locally significant variations in climate. Meteorological data are collected both on site and at a nearby meteorological station on Dutch Hill Road. Wind speed and direction, barometric pressure, temperature, dewpoint, and rainfall are measured on site. Wind speed and direction are measured at the regional location.

#### Temperature

The average monthly temperatures recorded at site from 1984 – 2002 are listed below:

| | | |
|---|---|---|
| January: 24.26 °F | May: 55.22 °F | September: 58.82 °F |
| February: 25.34 °F | June: 63.86 °F | October: 48.74 °F |
| March: 32.36 °F | July: 67.46 °F | November: 38.66°F |
| April: 44.6 °F | August: 66.02 °F | December: 28.22°F |

Extreme temperatures have been as high as 98.6 °F and as low as -43.6 °F.

**Precipitation and Wind Vectors**

Average annual precipitation for the site is 39.4 inches, including an average 120 inches of snow, based on 1985 – 2002 data, and is evenly distributed throughout the year. Winds are generally from the west and south at about 10.3 miles per hour (4.6 m/s) and 9.6 miles per hour (4.3 m/s) respectively, as previously noted.

**Severe Weather Phenomena**

According to U.S. Weather Bureau meteorological analysis, the theoretically greatest precipitation (probable maximum precipitation) that could be expected over the applicable drainage area in a 24-hour period is 24.9 inches. Factors figuring into this estimate include the size of the 1,200-acre drainage area, its topography, and seasonal effects. The highest measured 24-hour total as of 2003 was five inches.

**Atmospheric Water Vapor**

There are diurnal and seasonal variations in relative humidity, according to measurements made at the Buffalo National Weather Station office. Humidity during predawn hours ranges from 35 to 83 percent throughout the year. Afternoon humidity varies from 55 to 60 percent during the summer (June-August) months and from 18 to 25 percent during winter (December - February).

Figure 3-50 illustrates the percent frequency of occurrence of ceilings (defined as cloud cover of 5/8 or greater) less than 3,000 feet and/or visibility less than three miles at Buffalo and Niagara Falls, the closest locations with this data. The cycle of maximum and minimum occurrence should be approximately the same at West Valley. (WVNSCO 1993)

The normal annual number of hours of sunshine is approximately 2,100. In summer the daily value is approximately nine hours and in winter the normal is 3.5 hours.

**Fog**

Fog has a well-defined seasonal cycle with annual maximums occurring during the winter months. Buffalo has a normal expectation of ten days per year of dense fog; light fog occurs much more frequently.

**Atmospheric Stability**

Measurements of temperature, wind speed, and wind direction made at the 10-meter and 60-meter heights at the on-site meteorological tower are used for determining wind patterns and for determining atmospheric stability characteristics at the site. Seven Pasquill-Gifford atmospheric stability categories (A through F) have been determined for the site based on vertical temperature differences (temperature lapse rates, $\Delta T$) calculated from temperatures measured at the 197 feet (60-meter) and 33 feet (10-meter) heights at the onsite meteorological tower.

These stability class conditions determine how a parcel of air will react when it is displaced adiabatically ($\Delta T/\Delta Z$ method), i.e., without exchanging heat. Stability classifications were determined in accordance with the methodology described in NRC Regulatory Guide 1.23 (NRC 2007) on onsite meteorological programs and Regulatory

Guide 1.145 (NRC 1982) on atmospheric dispersion models. Hourly-averaged values of temperature obtained at the 197 feet (10-meter) and 33 feet (60-meter heights) at the tower were used in the calculations. The temperature differences were derived from temperature data collected at the on-site meteorological tower, from January 1, 1994, through December 31, 1998 (Spector and Grant 2003).

Joint frequency distributions of wind speed and direction for each stability class are tabulated in Table 3-10 for measurements at a height of 33 feet (10 meters) and Table 3-11 for measurements at a height of 197 feet (60 meters) (Spector and Grant 2003). These joint frequency distributions were derived from data collected at the on-site meteorological tower from January 1, 1994, through December 31, 1998. Wind directions are grouped into 16 principal directions (22.5-degree sectors centered on true north, northeast, and so on). Wind speeds are classified into seven wind speed categories. Calms are distributed, in the form of hourly-averaged wind speeds, into the first wind speed category representing the 0-0.5 m/s speed bin (Spector and Grant 2003).

**Table 3-10. Wind Speed and Direction Frequency Distributions at 10 Meters** (January 1, 1994 through December 31, 1998, based on Spector and Grant 2003, Attachment G)

| Stability Class | Wind Speed (m/s) | Direction From | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | N | NNE | NE | ENE | E | ESE | SE | SSE | S | SSW | SW | WSW | W | WNW | NW | NNW |
| A | 0.0-1.5 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.005 | 0.01 | 0.005 | 0.002 | 0.005 | 0.02 | 0 | 0.002 | 0 |
| | 1.5-3.0 | 0.051 | 0.044 | 0.032 | 0.027 | 0.039 | 0.017 | 0.022 | 0.015 | 0.022 | 0.027 | 0.039 | 0.024 | 0.027 | 0.054 | 0.113 | 0.047 |
| | 3.0-6.0 | 0.049 | 0.029 | 0.024 | 0.029 | 0.022 | 0.015 | 0.024 | 0.024 | 0.051 | 0.039 | 0.034 | 0.007 | 0.007 | 0.098 | 0.592 | 0.164 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0 | 0 | 0 | 0 | 0 | 0.005 | 0.015 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| B | 0.0-1.5 | 0 | 0.005 | 0.007 | 0.005 | 0 | 0 | 0.002 | 0.005 | 0 | 0.005 | 0.002 | 0 | 0.002 | 0.002 | 0 | 0 |
| | 1.5-3.0 | 0.059 | 0.069 | 0.054 | 0.032 | 0.037 | 0.024 | 0.037 | 0.047 | 0.056 | 0.083 | 0.122 | 0.064 | 0.083 | 0.164 | 0.291 | 0.083 |
| | 3.0-6.0 | 0.044 | 0.037 | 0.024 | 0.01 | 0.017 | 0.01 | 0.039 | 0.098 | 0.103 | 0.064 | 0.066 | 0.024 | 0.034 | 0.149 | 0.59 | 0.233 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.005 | 0 | 0.007 | 0 | 0 | 0 | 0 | 0.002 | 0.002 | 0.005 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| C | 0.0-1.5 | 0.002 | 0.022 | 0.012 | 0.007 | 0.005 | 0.007 | 0.012 | 0.005 | 0.012 | 0.007 | 0.007 | 0.007 | 0.005 | 0.02 | 0.017 | 0.01 |
| | 1.5-3.0 | 0.174 | 0.095 | 0.081 | 0.044 | 0.042 | 0.054 | 0.095 | 0.095 | 0.166 | 0.181 | 0.25 | 0.118 | 0.174 | 0.35 | 0.497 | 0.233 |
| | 3.0-6.0 | 0.073 | 0.027 | 0.027 | 0.015 | 0.049 | 0.034 | 0.108 | 0.103 | 0.181 | 0.071 | 0.073 | 0.047 | 0.051 | 0.176 | 0.835 | 0.289 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0.01 | 0 | 0.005 | 0.022 | 0 | 0 | 0 | 0 | 0 | 0.005 | 0.01 | 0.012 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| D | 0.0-1.5 | 0.321 | 0.34 | 0.223 | 0.22 | 0.252 | 0.343 | 0.468 | 0.441 | 0.695 | 0.72 | 0.629 | 0.615 | 0.832 | 1.05 | 0.906 | 0.36 |
| | 1.5-3.0 | 1.031 | 0.639 | 0.416 | 0.348 | 0.394 | 0.769 | 1.616 | 1.307 | 2.274 | 2.296 | 1.785 | 1.227 | 2.025 | 3.529 | 6.305 | 1.542 |
| | 3.0-6.0 | 0.308 | 0.113 | 0.071 | 0.286 | 0.313 | 0.495 | 1.709 | 1.951 | 1.506 | 0.693 | 0.443 | 0.235 | 0.524 | 1.809 | 4.447 | 1.205 |
| | 6.0-9.0 | 0 | 0 | 0 | 0.02 | 0.002 | 0.005 | 0.279 | 0.661 | 0.061 | 0.002 | 0.002 | 0 | 0 | 0.002 | 0.02 | 0.01 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.01 | 0.071 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.007 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| E | 0.0-1.5 | 0.093 | 0.093 | 0.078 | 0.132 | 0.233 | 0.279 | 0.673 | 1.408 | 1.983 | 1.092 | 0.686 | 0.654 | 0.71 | 0.776 | 0.428 | 0.147 |
| | 1.5-3.0 | 0.02 | 0.02 | 0.022 | 0.02 | 0.037 | 0.179 | 1.06 | 1.694 | 2.191 | 0.705 | 0.144 | 0.1 | 0.162 | 0.448 | 0.654 | 0.083 |
| | 3.0-6.0 | 0.002 | 0 | 0 | 0 | 0.01 | 0.017 | 0.487 | 1.165 | 0.771 | 0.095 | 0.007 | 0.007 | 0.007 | 0.005 | 0.069 | 0.007 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.007 | 0.23 | 0.024 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.027 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| F | 0.0-1.5 | 0.039 | 0.024 | 0.049 | 0.042 | 0.103 | 0.235 | 0.546 | 1.741 | 1.547 | 0.676 | 0.406 | 0.272 | 0.166 | 0.069 | 0.049 | 0.056 |
| | 1.5-3.0 | 0 | 0.002 | 0 | 0 | 0.002 | 0.034 | 0.176 | 0.333 | 0.24 | 0.022 | 0.002 | 0.01 | 0.017 | 0.005 | 0.015 | 0.01 |
| | 3.0-6.0 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0.007 | 0.024 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| G | 0.0-1.5 | 0.012 | 0.04 | 0.015 | 0.029 | 0.039 | 0.13 | 0.637 | 2.931 | 1.704 | 0.411 | 0.218 | 0.125 | 0.039 | 0.01 | 0.02 | 0.022 |
| | 1.5-3.0 | 0 | 0 | 0 | 0 | 0.002 | 0.007 | 0.066 | 0.208 | 0.054 | 0 | 0 | 0.002 | 0.002 | 0 | 0 | 0 |
| | 3.0-6.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |

**Table 3-11. Wind Speed and Direction Frequency Distributions at 60 Meters** (January 1, 1994 through December 31, 1998, based on Spector and Grant 2003, Attachment H)

| Stability Class | Wind Speed (m/s) | Direction From | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | N | NNE | NE | ENE | E | ESE | SE | SSE | S | SSW | SW | WSW | W | WNW | NW | NNW |
| A | 0.0-1.5 | 0 | 0 | 0 | 0 | 0.002 | 0 | 0.002 | 0.002 | 0 | 0 | 0 | 0 | 0.002 | 0.002 | 0 | 0 |
| | 1.5-3.0 | 0.017 | 0.007 | 0.007 | 0.015 | 0.022 | 0.01 | 0.005 | 0.007 | 0.005 | 0.005 | 0.012 | 0.012 | 0.01 | 0.017 | 0.019 | 0.022 |
| | 3.0-6.0 | 0.005 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0.002 | 0.017 | 0.053 | 0.051 | 0.027 | 0.039 | 0.211 | 0.296 | 0.099 |
| | 6.0-9.0 | 0.005 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0.002 | 0.017 | 0.012 | 0.029 | 0.012 | 0.01 | 0.17 | 0.143 | 0.051 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0 | 0 | 0 | 0.002 | 0.005 | 0.007 | 0.002 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| B | 0.0-1.5 | 0.007 | 0 | 0.002 | 0 | 0 | 0.005 | 0 | 0.005 | 0 | 0.002 | 0.002 | 0 | 0 | 0 | 0 | 0 |
| | 1.5-3.0 | 0.034 | 0.051 | 0.046 | 0.019 | 0.017 | 0.022 | 0.017 | 0.015 | 0.019 | 0.07 | 0.012 | 0.022 | 0.039 | 0.075 | 0.075 | 0.056 |
| | 3.0-6.0 | 0.053 | 0.051 | 0.039 | 0.024 | 0.034 | 0.01 | 0.036 | 0.07 | 0.083 | 0.109 | 0.175 | 0.102 | 0.092 | 0.386 | 0.408 | 0.175 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0.012 | 0.029 | 0.017 | 0.036 | 0.029 | 0.024 | 0.046 | 0.133 | 0.124 | 0.017 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.005 | 0.002 | 0 | 0.002 | 0 | 0.015 | 0.002 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| C | 0.0-1.5 | 0.005 | 0.002 | 0.01 | 0.002 | 0.002 | 0.007 | 0.002 | 0 | 0.01 | 0.005 | 0.005 | 0.002 | 0 | 0.002 | 0.007 | 0.01 |
| | 1.5-3.0 | 0.126 | 0.067 | 0.068 | 0.034 | 0.034 | 0.034 | 0.066 | 0.309 | 0.036 | 0.068 | 0.073 | 0.07 | 0.085 | 0.116 | 0.129 | 0.129 |
| | 3.0-6.0 | 0.109 | 0.053 | 0.041 | 0.034 | 0.051 | 0.036 | 0.097 | 0.092 | 0.148 | 0.26 | 0.294 | 0.172 | 0.279 | 0.645 | 0.631 | 0.238 |
| | 6.0-9.0 | 0 | 0 | 0 | 0.002 | 0.017 | 0.01 | 0.01 | 0.034 | 0.027 | 0.022 | 0.041 | 0.032 | 0.034 | 0.192 | 0.099 | 0.036 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0.007 | 0 | 0.002 | 0.015 | 0 | 0 | 0 | 0 | 0.005 | 0.029 | 0.002 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| D | 0.0-1.5 | 0.199 | 0.204 | 0.18 | 0.184 | 0.15 | 0.206 | 0.209 | 0.092 | 0.102 | 0.058 | 0.07 | 0.112 | 0.119 | 0.119 | 0.17 | 0.163 |
| | 1.5-3.0 | 0.757 | 0.568 | 0.468 | 0.255 | 0.306 | 0.531 | 0.9 | 0.551 | 0.393 | 0.587 | 0.99 | 1.063 | 1.281 | 1.42 | 1.272 | 0.755 |
| | 3.0-6.0 | 0.636 | 0.405 | 0.24 | 0.473 | 0.519 | 0.682 | 1.628 | 1.662 | 1.153 | 2.203 | 3.237 | 2.587 | 4.215 | 5.63 | 3.458 | 1.138 |
| | 6.0-9.0 | 0.034 | 0.002 | 0.15 | 0.024 | 0.029 | 0.08 | 0.548 | 0.784 | 0.675 | 0.495 | 0.718 | 0.439 | 1.228 | 1.815 | 0.781 | 0.112 |
| | 9.0-12.0 | 0 | 0 | 0 | 0.007 | 0.002 | 0 | 0.129 | 0.495 | 0.131 | 0.015 | 0.005 | 0.005 | 0.058 | 0.078 | 0.019 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.015 | 0.109 | 0.012 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| E | 0.0-1.5 | 0.113 | 0.104 | 0.087 | 0.097 | 0.133 | 0.269 | 0.544 | 0.403 | 0.158 | 0.095 | 0.92 | 0.073 | 0.078 | 0.102 | 0.114 | 0.136 |
| | 1.5-3.0 | 0.175 | 0.083 | 0.078 | 0.085 | 0.143 | 0.294 | 1.23 | 0.818 | 0.432 | 0.422 | 0.371 | 0.485 | 0.446 | 0.4 | 0.325 | 0.158 |
| | 3.0-6.0 | 0.024 | 0.01 | 0.017 | 0.034 | 0.034 | 0.102 | 1.104 | 1.301 | 1.269 | 1.767 | 1.429 | 0.604 | 0.726 | 0.694 | 0.488 | 0.15 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0.015 | 0.002 | 0.121 | 0.502 | 0.548 | 0.33 | 0.167 | 0.015 | 0.017 | 0.024 | 0.015 | 0 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.184 | 0.068 | 0 | 0 | 0 | 0 | 0.002 | 0 | 9 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.034 | 0.002 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| F | 0.0-1.5 | 0.102 | 0.049 | 0.068 | 0.068 | 0.095 | 0.175 | 0.908 | 1.109 | 0.175 | 0.046 | 0.063 | 0.066 | 0.044 | 0.063 | 0.104 | 0.107 |
| | 1.5-3.0 | 0.019 | 0.01 | 0.07 | 0.007 | 0.17 | 0.085 | 0.946 | 0.694 | 0.243 | 0.211 | 0.112 | 0.136 | 0.121 | 0.133 | 0.126 | 0.083 |
| | 3.0-6.0 | 0 | 0 | 0 | 0 | 0 | 0.015 | 0.393 | 0.325 | 0.34 | 0.279 | 0.16 | 0.073 | 0.053 | 0.61 | 0.85 | 0.032 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.007 | 0.019 | 0.002 | 0 | 0 | 0.002 | 0 | 0 | 0 | 0 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| G | 0.0-1.5 | 0.036 | 0.046 | 0.068 | 0.041 | 0.066 | 0.153 | 0.769 | 1.344 | 0.24 | 0.067 | 0.061 | 0.078 | 0.049 | 0.051 | 0.075 | 0.058 |
| | 1.5-3.0 | 0.005 | 0.002 | 0 | 0.005 | 0.002 | 0.029 | 0.895 | 1.24 | 0.417 | 0.277 | 0.211 | 0.165 | 0.09 | 0.061 | 0.107 | 0.039 |
| | 3.0-6.0 | 0 | 0 | 0 | 0 | 0 | 0.005 | 0.216 | 0.267 | 0.296 | 0.403 | 0.119 | 0.017 | 0.019 | 0.015 | 0.015 | 0.002 |
| | 6.0-9.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.002 | 0.002 | 0 | 0 | 0 | 0 | 0 | 0 |
| | 9.0-12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | <12.0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |

**Air Quality**

The EPA regulates National Ambient Air Quality Standards for criteria pollutants as defined in the Clean Air Act Titles I through VI, which are designed to protect human health and welfare from adverse effects. Cattaraugus County falls within the Southern Tier West Intrastate district (Air Quality Control Region 164), with the following status of attainment: “Better than National Standards/Unclassifiable (cannot be classified).”

Radiological emissions are regulated under the National Emission Standards for Hazardous Air Pollutants regulations. Non-radiological air emissions are regulated by the NYSDEC whose regulations dictate monitoring and compliance of stationary and mobile sources of air pollution. The WVDP was approved for a capping plan for non-radiological emissions. There were no cases where air permit or regulatory criteria were exceeded during calendar year 2007. (WVES and URS 2008)

## 3.5 Geology and Seismology

The geology and seismology of the site and surrounding areas are described in this section.

### 3.5.1 Regional Physiography

The Center is located within the glaciated northern portion of the Appalachian Plateau Province, a maturely dissected upland region underlain in western New York by shales and siltstones of Devonian age. This region is bounded on the north by the Erie Ontario Lowlands, on the east by the Tughill Upland, on the south by the unglaciated Appalachian Plateau, and on the west by the Interior Lowlands (Figure 3-51).

The Appalachian Plateau of western New York has been subjected to multiple glaciations during the Wisconsinan glacial period 38,000 to 14,500 years ago, that resulted in the deepening and oversteepening of many pre-glacial valleys and in the accumulation in those valleys of as much as 500 feet of glacial tills, lacustrine, and glaciofluvial sediments. The Center is situated within one of these north-trending valleys (Figure 3-3).

### 3.5.2 Site Stratigraphy

The Center is located in a glacial valley filled with upwards of 500 feet of Pleistocene age glacial tills, lacustrine, and glaciofluvial sediments that were deposited during the Wisconsinan glacial period. The thickness of glacial deposits at the site ranges from five feet or less on the uplands to 500 feet along the axis of the valley. These glacial sediments were deposited on shales and siltstones of the Middle Devonian Conneaut and Canadaway Groups which comprise the uppermost portion of the Paleozoic bedrock that underlies the Center.

The Paleozoic section in the vicinity of the Center is approximately 7,500 feet thick and is comprised predominantly of shales, siltstones, sandstones, carbonates, and evaporites of Cambrian through Devonian age (Table 3-12). Bedrock stratification in the area is nearly flat and essentially undeformed. However, bedrock is tilted to the south at an average dip of six to eight meters per kilometer (approximately 32 to 42 feet per mile). The Paleozoic bedrock underlying the Center was deposited on a basement of older Precambrian-age rocks that are part of the Grenville Orogenic Belt which extends from eastern Canada, through the United States, and into Mexico.

**Table 3-12. Generalized Paleozoic Stratigraphic Section for Southwestern New York[(1)]**

| System | Series | Group | Unit | Lithology | Thickness (ft) |
|---|---|---|---|---|---|
| Pennsylvanian | | Pottsville | Olean | Ss, Cgl | 75 – 100 |
| Mississippian | | Pocono | Knapp | Ss, Cgl | 50 – 100 |
| Devonian | Upper | Conewango | | Sh, Ss, Cgl | 700 |
| | | Conneaut | Chadakoin | Sh, Ss | 700 |
| | | Canadaway | Undiff | Sh, Ss | 1100 – 1400 |
| | | | Perrysburg | Sh, Ss | |
| | | West Falls | Java | Sh, Ss | 375 – 1250 |
| | | | Nunda | Sh, Ss | |
| | | | Rhinestreet | Sh, Ss | |
| | | Sonyea | Middlesex | Sh | 0 – 400 |
| | | Genesee | | Sh | 0 – 450 |
| | Middle | | Tully | Ls | 0 – 50 |
| | | Hamilton | Moscow | Sh | 200 – 600 |
| | | | Ludlowville | Sh | |
| | | | Skaneateles | Sh | |
| | | | Marcellus | Sh | |
| | | | Onondaga | Ls | 30 – 235 |
| | Lower | Tristates | Oriskany | Ss | 0 – 40 |
| | | Helderberg | Manlius | Ls | 0 – 10 |
| | | | Rondout | Dol | |
| Silurian | Upper | | Akron | Dol | 0 – 15 |
| | | Salina | Camillus | Sh, Gyp | 450 – 1850 |
| | | | Syracuse | Dol, Sh, Salt | |
| | | | Vernon | Sh, Salt | |
| | | Lockport | Lockport | Dol | 150 – 250 |
| | | Clinton | Rochester | Sh | 125 |
| | | | Irondequoit | Ls | |
| | Lower | | Sodus | Sh | 75 |
| | | | Reynales | Ls | |
| | | | Thorold | Ss | 2 – 8 |

**Table 3-12. Generalized Paleozoic Stratigraphic Section for Southwestern New York[1]**

| System | Series | Group | Unit | Lithology | Thickness (ft) |
|---|---|---|---|---|---|
| | | Medina | Grimsby | Sh, Ss | 75 – 160 |
| | | | Whirlpool | Ss | 0 – 25 |
| Ordovician | Upper | | Queenston | Sh | 1100 – 1500 |
| | | | Oswego | Ss | |
| | | | Lorraine | Sh | 900 – 1000 |
| | | | Utica | Sh | |
| | Middle | Trenton-Black River | Trenton | Ls | 425 – 625 |
| | | | Black River | Ls | 225 – 550 |
| | Lower | Beekmantown | Tribes Hill /Chuctanunda | Ls | 0 – 550 |
| Cambrian | Upper | | Little Falls | Dol | 0 – 350 |
| | | | Galway (Theresa) | Dol, ss | 575 – 1350 |
| | | | Potsdam | Ss, Dol | 75 – 500 |
| Precambrian | | | | Meta Rx | |

NOTE: (1) From Jacobi and Fountain 1993.

LEGEND: Cgl = conglomerate, Dol = dolomite, Gyp – gypsum, Ls = limestone, Sh = shale, Ss = sandstone, Meta Rx = metamorphic rocks

**Site Glacial Stratigraphy**

The WVDP is underlain by upwards of 500 feet of Pleistocene-age glacial sediments that were deposited in a northwest-trending bedrock valley (Figure 3-52). The principal glacial units are identified below.

**Surficial Sand and Gravel Unit**

The surficial sand and gravel unit is a silty, sandy gravel deposit that incorporates two overlapping units of different ages and origins. The older unit, the slack-water sequence, is a Wisconsinan glaciofluvial deposit deposited in Buttermilk Creek Valley by draining glacial meltwaters of Lavery-age ice. The younger unit, the thick-bedded unit, is a post-glacial Holocene-age alluvial fan deposited by streams entering Buttermilk Creek Valley.

This unit is found at grade in the north plateau area of the Center where it has a maximum thickness of 41 feet in the center of the plateau. The sand and gravel unit thins to a few feet towards the northern, eastern, and southern margins of the north plateau where it has been truncated by the downward erosion of stream channels bounding the north plateau. The Process Building, Vitrification Facility, and adjacent facilities were built on these alluvial and glaciofluvial deposits (Figure 3-5).

The composition of the sand and gravel unit varies, but on the average it is a mixture of gravel (41 percent), sand (40 percent), silt (11 percent), and clay (8 percent). X-ray diffraction analysis indicates the mineralogy of this unit is dominated by quartz, illite, chlorite, and plagioclase with subordinate amounts of calcite and dolomite.

Surficial sands and gravels that are equivalent to the surficial sand and gravel unit in the north plateau are located in a number of areas within the Center (Figure 3-53). These sands and gravels have been quarried for gravel in three locations within the Center. Two of the gravel pits are located west of the Process Building on the west side of Rock Springs Road (Figure 3-8). These gravel pits are no longer in operation and were closed in accordance with NYSDEC regulations. The third gravel pit was located on the southeastern margin of the Center (Figure 3-9). This gravel pit was quarried by the Town of Ashford. The three gravel pit quarries do not contain any residual radioactive contamination from NFS or WVDP operations.

**Lavery Till**

The Lavery till is predominantly an olive-gray, silty-clay, glacial till with lenses of sand, gravel, silt, and rhythmic clay-silt laminations (Albanese, et al. 1983). This unit underlies the surficial sand and gravel unit in the north plateau and is exposed at the surface in the south plateau (Figure 3-53). As noted previously, the Lavery till is the host unit for both the SDA and the NDA.

The thickness of the Lavery till ranges from a few feet at its western margin to upwards of 130 feet to the east towards Buttermilk Creek. The Lavery till is a mixture of clay (50 percent), silt (30 percent), sand (18 percent), and gravel (two percent) (WVNSCO 1993e). The mineral composition of the till largely resembles that of local bedrock.

On the south plateau, the upper three to 16 feet of the Lavery till is weathered to a brown color and it contains root tubes and numerous fractures whose number decrease with depth. This upper layer is referred to as the weathered Lavery till and it is principally found in the south plateau of the Center. The weathered Lavery till is either absent or only a few inches thick on the north plateau.

X-ray diffraction analysis indicates the mineralogy of the weathered Lavery till is composed mainly of illite, quartz, calcite, kaolinite, plagioclase feldspar, and dolomite in decreasing quantities. The mineralogy of the unweathered Lavery till is composed mainly of quartz, illite, calcite, and kaolinite in decreasing abundance.

A borrow pit excavated into the Lavery till is located on the south plateau east of the SDA between Franks Creek and Buttermilk Creek (Figure 3-9). Clay was excavated from this pit beginning in the 1970's to provide clay fill for use at the SDA. The borrow pit did not contain any residual radioactive contamination from NFS or WVDP operations. The pit covered an area of less than one acre and it was closed by backfilling and grading in accordance with the NYSDEC Mined Land Reclamation Program in the early 2000's.

**Lavery Till-Sand Unit**

The Lavery till-sand unit is a lenticular shaped, silty, sand layer that is locally present within the Lavery till in the north plateau of the Center, immediately southeast of the

Process Building. It is thought to be either a pro-glacial sand deposit or a reworked kame deposit.

The till-sand is limited in areal extent, occurring on the north plateau in an east-west band approximately 750 feet wide. It lies within the upper 20 feet of the Lavery till (Figure 3-6) and is up to seven feet in thickness.

Re-examination of borehole logs from the north plateau in 2007 resulted in a re-evaluation of the areal extent of the Lavery till sand. From 1991 to 2007, the Lavery till sand was inferred to be present to the west, south, and southeast of the Process Building in a location that was hydraulically upgradient and cross-gradient to the north plateau groundwater plume. Earlier interpretations of the borehole logs considered a prominent clay-rich geologic horizon up to several feet in thickness as part of the unweathered Lavery till and the underlying sandy unit as the Lavery till sand.

Following the completion of the 1993 soil boring program to support the RCRA Facility Investigation, the 1993 borehole data indicated that the sand and gravel unit was composed of two distinct subunits, the thick-bedded unit and the underlying slack water sequence which are separated by the prominent clay-rich geologic horizon mentioned earlier. In 2007 it was noted that the elevation of the original Lavery till sand west and southwest of the Process Building was much shallower in elevation than the Lavery till sand to the southeast of the Process Building. It was determined that this western and southwestern portion was more consistent with the elevation of the slack water sequence of the sand and gravel unit and it was reclassified as part of the slack water sequence. As a result, the areal extent of the Lavery till sand was substantially reduced and it is now located southeast of the Process Building away from the north plateau groundwater plume as shown in Figure 3-64.

**Kent Recessional Sequence**

The Kent Recessional Sequence underlies the Lavery till on both the north and south plateaus and it includes both lacustrine and kame delta deposits; it is 30 to 60 feet thick at the WVDP. Lacustrine strata composed of laminated silt and clay forms the lower 30 feet of the Kent Recessional Sequence, which is present in the subsurface across the entire WVDP.

The lacustrine section is interpreted as forming in a pro-glacial lake that formed after the recession of the Kent ice margin (LaFleur 1979). The lacustrine section is composed mainly of quartz, illite, calcite, dolomite, and plagioclase feldspar in decreasing abundance. Calcite and dolomite together make up 12 to 20 percent of the lacustrine section by weight.

The lacustrine section in the eastern portion of the WVDP is overlain by upwards of 30 feet of sand and gravel believed to represent several kame deltas. (Figure 3-6) Several of these kame deltas are exposed along Buttermilk Creek and extend into the WVDP west of the NDA (Bergeron, et al. 1987).

The kame deltas were deposited during pauses in the recession of the Kent glacier through a pro-glacial lake that allowed the accumulation of kame deltas over lakebed silts and clays. This unit is underlain by at least two older silty-clay tills, the Kent till and the Olean till, which also are separated by similar lacustrine and glaciofluvial deposits (LaFleur 1979).

### 3.5.3 Site Geomorphology

Karst terrains are not developed at the Center as there are no occurrences of carbonate bedrock in the vicinity of the site. Natural subsidence of surficial soils has not been observed at the Center. However, small scale subsidence has been observed over some of the burial holes in the NDA and SDA during their operating history which are believed related to collapse and compaction of buried waste.

Geomorphological studies at the WVDP have focused on the major erosional processes acting on Buttermilk Creek and Franks Creek drainage basins near the WVDP. This section describes these processes – channel incision, slope movement, and gullying – and details where they occur. The erosion rates from these processes have been measured at numerous locations throughout the drainage basins, as summarized in Table 3-13. Results vary based on location and methodology used in the measurements.

#### Channel Incision

The streams in the vicinity of the WVDP are at a relatively young stage of development and are characterized by steep profiles, V-shaped cross-sections, and little or no floodplains. At this stage, streams are able to move large quantities of sediment and erode their channels, a process referred to as channel incision or stream downcutting. The channel incision process is greatest during high-flow, high-energy rainfalls from prolonged soaking storms and brief, high-intensity thunderstorms.

These streams are also actively elongating their stream course or profiles through erosion upstream, a process referred to as headward advance. Headward advance starts when the movement of channel sediment is blocked by debris in the stream channel, which results in an abrupt change in the longitudinal profile of the stream bed, referred to as a knickpoint.

The stream erodes the knickpoint area by simple basal scour due to an attached impinging jet which undercuts the knickpoint face. Large blocks of material are then removed by cantilever mass failure and are then dispersed and washed downstream.

The shape of the channel cross-section changes from a U-shape, or flatbottom, with a low erosion rate to a V-shaped channel with a higher erosion rate. The knickpoint migration rate has been measured at 10.7 feet per year along Erdman Brook and 7.5 feet per year along Franks Creek (WVNSCO 1993d).

#### Slope Movement

Slope erosion within the Buttermilk Creek and Franks Creek drainage basin has been dominated by the formation of slump blocks along the stream valley wall. Slumps develop when water infiltrates into fractures within stream banks, causing an increase in soil pore pressures, which reduces the soil strength until the slope slumps down into the stream valley. Slumps also occur on the outside of a stream meander loop, where the increased stream flow velocity undercuts the base of the slope, decreasing the slope stability and accelerating the slumping process.

Three slump blocks have been identified along Franks Creek, one on Erdman Brook, and one on Quarry Creek. The blocks vary in length from about five feet to greater than 100 feet and tend to be about three to four feet in height and width when they initially form (WVNSCO 1993d).

On the basis of data collected from 1982 to 1991, the rate of downslope movement within the slump blocks on Erdman Brook is reported to range from 0.09 and 0.16 feet per year, which equates to a stream valley rim widening rate of approximately 0.07 to 0.12 feet per year.

**Gullying**

The steep walls of the stream channels within the Buttermilk Creek and Franks Creek drainage basin are susceptible to gully formation. Gullies are most likely to form along stream banks, where slumps and deep fractures are present, groundwater seeps are flowing, and the toe of the slope intersects the outside of a stream meander loop.

Gully formation occurs during thaws and after thunderstorms, where a concentrated stream of water flows over the side of a plateau, which is great enough to promote entrainment and removal of soil particles from the base of the gully. Surface water runoff into the gully contributes to gully growth by removing fallen debris at the base of the scarp.

More than 20 major and moderate-sized gullies have been identified near the WVDP. The initiation and growth of gullies may be the most rapid means for eroding the north and south plateaus. Gully advance was calculated at 1.2 feet per year near the SDA on the south plateau, and at 2.2 feet per year for two areas on the north plateau (WVNSCO 1993d).

**Table 3-13. Summary of Erosion Rates Near the WVDP**

| Location | Erosion Rate (m/y) | Reference | Method |
|---|---|---|---|
| Sheet and Rill Erosion | 0 to 0.0045 | URS 2001 | Erosion frame measurements (11-year average rate) |
| Deepening of Buttermilk Creek | 0.0015 to 0.0021 | LaFleur 1979 | Carbon-14 date of terrace - depth of stream below terrace |
| Deepening of Buttermilk Creek | 0.005 | Boothroyd, et al. 1982 | Carbon-14 date of terrace - depth of stream below terrace |
| Deepening of Quarry Creek, Franks Creek, and Erdman Brook | 0.051 to 0.089 | Dames & Moore 1992 | Difference from 1980 to 1990 in stream surveys |

**Table 3-13. Summary of Erosion Rates Near the WVDP**

| Location | Erosion Rate (m/y) | Reference | Method |
|---|---|---|---|
| Downcutting of Buttermilk Creek | 0.0032 | USGS 2007 | Optically stimulated luminescence age dating of 9 terraces along Buttermilk Creek |
| Buttermilk Creek Valley Rim Widening | 4.9 to 5.8 | Boothroyd, et al. 1979 | Downslope movement of slump block over 2 years |
| Valley Rim Widening of Buttermilk and Franks Creeks and Erdman Brook | 0.05 to 0.13 | McKinney 1986 | Extrapolate Boothroyd data for 500 years |
| Erdman Brook Valley Rim Widening | 0.02 to 0.04 | Dames & Moore 1992 | Downslope movement of stakes over 9 years |
| Downcutting of Franks Creek | 0.06 | Dames & Moore 1992 | Stream profile, knickpoint migration 1955 to 1989 |
| SDA Gully Headward Advancement | 0.4 | Dames & Moore 1992 | Gully advancement Soil Conservation Service TR-32 method |
| NP3 Gully Headward Advancement | 0.7 | Dames & Moore 1992 | Gully advancement Soil Conservation Service TR-32 method |
| 006 Gully Headward Advancement | 0.7 | Dames & Moore 1992 | Gully advancement Soil Conservation Service TR-32 method |

**Slope Stability**

Landslides provide an active mechanism to headward erosion for altering the landform in Buttermilk Creek Valley. Since landslides typically occur on slopes that have a relief of more than 10 feet, all currently eroding surfaces except the upland flats have potential for landslide development. Landslides range from three feet to 65 feet in height. Landsliding has been recognized since the mid-1970s along the small streams bordering the burial areas.

Stratigraphy affects both landslide location and development. Landsliding takes place along Buttermilk Creek where the Lavery till unit is dissected and the underlying lower sand and gravel of the Kent Recessional Sequence is exposed. These unconsolidated sands and gravels are removed by stream erosion, leaving the overlying till unsupported, followed by bank collapse, bringing down large blocks of the valley wall.

Landslides on the smaller streams draining the WVDP tend to occur as the channel cuts downward through the Lavery till, increasing the steepness of the stream banks, which eventually results in a series of short slide blocks. The blocks tend to be less than four feet high and occur along the slope from the edge of the plateau to the edge of the stream channel.

Creep occurs on the slopes of Buttermilk Creek and its tributaries at relatively slow rates of a few centimeters per year. A slope may have surface layers a few centimeters thick that move a few centimeters per year. If highly charged with water, the surface soils may liquefy and then move down-slope as mudflows. These mudflows occur most frequently in conjunction with landsliding.

Down-slope movement of till in the Buttermilk Creek Valley by landslides, slumping, and earthflow appears to be a continuous process measured at an average rate of five feet per year (Boothroyd, et al. 1982). The average volume of material delivered to Buttermilk Creek has been estimated to be 5,250 cubic feet per year (Boothroyd, et al. 1982).

Landslide mapping and monitoring suggests areas most susceptible to failure have the following characteristics: surface slopes exceeding eight degrees, slopes composed of silty and clayey tills or alluvial fan material, an active stream channel at the foot of slope, and little or no vegetative cover or heavy overburden (WVNSCO 1993c).

### 3.5.4 Regional Structure and Tectonics

The bedrock in the immediate vicinity of the Center is composed of interbedded shales, siltstones, and sandstones of the Upper Devonian Canadaway and Conneaut Groups (Rickard 1975). These and underlying Paleozoic sediments were deformed by compressive stresses originating from the Pennsylvanian-Permian Alleghanian orogeny which was the last major orogenic episode affecting the Appalachian mountain belt.

The major manifestations of this Alleghanian deformation are the prominent regional folds, thrust faults, and metamorphism that are found to the southeast in the Appalachian Valley and Ridge, Blue Ridge, and Piedmont Provinces (Figure 3-51). However, Alleghanian deformation did extend into the Appalachian Plateau Province of western New York where geologic structure such as joints, low amplitude folds, and thrust faults with small stratigraphic separation were developed in Paleozoic bedrock.

#### Alleghanian Folds and Thrust Faults

The Alleghanian deformation within the Appalachian Plateau of western New York principally affected the Upper Silurian Salina Group and overlying Devonian-age rocks (Table 3-14). During the Alleghanian orogeny, Paleozoic strata overlying the Salina Group was detached from underlying older strata by a decollement in the Salina Group. The stratigraphic section overlying this decollement was deformed, shortened, and translated to the northwest during the Alleghanian orogeny. The deformation of the strata overlying the decollement was manifested in the development of thrust faults, folds, and systematically oriented bedrock fractures.

The thrust faults that splayed off of the Salina decollement into the Lower to Middle Devonian section displaced and folded overlying bedding, producing an arcuate fold belt in western and central New York (Figure 3-54). The trend of this fold belt changes across New York State. Anticline fold axes, which trend roughly northeast-southwest in Chautauqua, Cattaraugus, and Allegany Counties, are observed to rotate to the east and become more east-west trending in Steuben and Chemung Counties.

These folds have low amplitudes with limb dips that are generally 1 to 2 degrees (Wedel 1932, Engelder and Geiser 1980). The low amplitudes of these folds are related to the small amount of stratigraphic separation that occurs across the thrust faults forming these folds. Higher amplitude folds, with corresponding higher limb dips and larger amount of separation across thrust faults, are found in the Valley and Ridge Province of Pennsylvania (Figure 3-51).

The Bass Islands Trend, a northeast trending, oil and gas producing structure extending from northeastern Ohio into western New York, is an example of an Alleghanian foreland fold and thrust structure. The Bass Islands Trend extends from the southwest corner of New York State, through Chautauqua Lake, northwestern Cattaraugus County, and into southern Erie County (Figure 3-55). The Bass Islands Trend is a regional fold that formed as the result of a thrust fault ramping up-section from the Salina Group into the overlying Lower Devonian section.

Bedrock mapping in the south branch of Cattaraugus Creek, approximately 12 miles west of the WVDP, indicates the presence of northeast-striking inclined bedding, folds, and faults which are attributed to faults associated with the Bass Islands Trend (Baudo and Jacobi 1999, Jacobi and Zhao 1999). Recent field mapping in the Ashford Hollow quadrangle, in which the Center is located, indicates the presence of northwest and northeast striking fractures that represent typical Alleghanian age cross-fold and fold-parallel fracture sets (Tober and Jacobi 2000).

**Table 3-14. Summary of Observed Faults on Seismic Lines WVN-1 and BER83-2A(1)**

| Seismic Line | Shot Point Location Top of Fault | Displacement (feet) | Shot Point Location Base of Fault | Fault Apparent Dip Angle | Fault Type | Displace Trenton |
|---|---|---|---|---|---|---|
| WVN-1 | 155.5 | | 156.5 | 82.1E | Reverse | No |
| | 204.5 | 75 | 206.0 | 85.4E | Normal | No |
| | 241.5 | 35 | 239.0 | 84.6W | Reverse | No |
| | 265.0 | 23 | 264.5 | 88.9W | Reverse | ? |
| | 467.0 | 47 | 465.0 | 81.4W | Normal | No |
| | 478.5 | 23 | 484.0 | 81.7E | Reverse | No |
| | 486.0 | 35 | 502.0 | 50.9E | Reverse | No |
| | 522.5 | 47 | 506.5 | 62.9W | Reverse | ? |
| | 557.0 | | | | | |
| | 601.0 | 70 | 585.0 | 61.3W | Reverse | Yes |
| | 621.5 | 35 | 622.0 | 88.0E | Normal | No |
| | 633.0 | 58 | 631.0 | 86.2W | Reverse | Yes |
| | 668.5 | 58 | 667.5 | 87.7W | Reverse | Yes |

**Table 3-14. Summary of Observed Faults on Seismic Lines WVN-1 and BER83-2A[1]**

| Seismic Line | Shot Point Location Top of Fault | Displacement (feet) | Shot Point Location Base of Fault | Fault Apparent Dip Angle | Fault Type | Displace Trenton |
|---|---|---|---|---|---|---|
| | 699.0 | 10 | 699.5 | 88.7E | Reverse | ? |
| | 740.0 | 28 | 737.5 | 87.6W | Normal | Yes |
| | 766.0 | 287 | 764.5 | 88.6W | Normal | Yes |
| | 797.5 | 57 | 792.0 | 65.7W | Reverse | No |
| | 871.0 | 48 | 859.5 | 65.0W | Normal | Yes |
| BER83-2A | 412.0 | 51 | 421.5 | 75.9S | Normal | Yes |
| | 451.5 | 38 | 457.0 | 84.3S | Normal | Yes |
| | 452.5 | 102 | 457.0 | 85.3S | Normal | Yes |
| | 519.0 | | 521.0 | 81.0S | Normal | No |
| | 681.0 | | 684.0 | 84.3S | Normal | No |
| | 709.5 | 13 | 714.0 | 85.0S | Normal | Yes |
| | 748.0 | | 752.0 | 83.4S | Normal | No |
| | 779.5 | 26 | 791.5 | 70.1S | Reverse | No |
| | 800.0 | 39 | 822.0 | 60.7S | Reverse | No |
| | 828.0 | 12 | 842.0 | 87.2S | Normal | No |

NOTE: (1) From Bay Geophysical 2001.

The presence of northeast trending fracture intensification domains suggest thrust faults associated with the Bass Island Trend or other Alleghanian thrust faults may extend eastward into the Ashford Hollow quadrangle (Tober and Jacobi 2000). Alleghanian folds and thrust faults are no longer tectonically active or seismically active. As a result there is no rate of deformation associated with these structures.

### Bedrock Fractures

Fractures are ubiquitous in the Paleozoic bedrock of western New York. Systematically oriented fracture or joint sets have been identified in the Paleozoic bedrock of the Appalachian Plateau of western New York (Engelder and Geiser 1980, Fakundiny, et al. 1978, Geiser and Engelder 1983, McKinney, Gross and Engelder 1991, Jacobi, et al. 1996, Zhao and Jacobi 1997). These joint sets are part of a regional fracture system that formed primarily in response to compressive stresses originating during the Pennsylvanian-Permian Alleghanian Orogeny. However, other joint sets identified in bedrock in western New York may have originated in response to the contemporary east-northeast regional stress field currently affecting eastern North America (Engelder and Geiser 1980, Geiser and Engelder 1983, Gross and Engelder 1991), or post-Precambrian movements along the Clarendon-Linden Fault System (Jacobi, et al. 1996, Zhao and Jacobi 1997).

Three vertical joint sets in Paleozoic bedrock from western New York, including rocks from the Upper Devonian Canadaway and Conneaut Groups have been identified (Engelder and Geiser 1980). Two of these joint sets, trending approximately north 45° west (N45W) and N45E, were produced from the compressive stresses generated during the Alleghanian orogeny (Figure 3-54).

The N45E joint set parallels fold axes in the Appalachian plateau and formed during the Alleghanian-age compression that produced these folds. The N45W joint set is generally perpendicular to fold trends in this area and was produced before the folding of bedrock in the Appalachian Plateau (Figure 3-54). A third set trending N60E is found throughout New York and probably formed under the current east-northeast regional compressive stress field. These joints sets are cells found in the Devonian bedrock in and around the Center.

Eight systematic joint sets were identified in rocks from the Canadaway and Conneaut Groups in Allegany County (Engelder and Geiser 1980, Zhao and Jacobi 1997). The strike of these joint sets ranged from west-northwest to east-northeast and they were produced at various stages of the Alleghanian deformation that affected western New York. The orientation of these joint sets reflects changes in the orientation of the principal stresses that were associated with the deformation of the Appalachian plateau of western New York, beginning with north-northwest trending cross fold joints followed by the progressive development of joint sets to the east and west.

**Regional Northwest Trending Lineaments and Structures**

Regional northwest trending lineaments have been identified across the eastern United States based on analyses of regional gravity and magnetic anomaly trends. These lineaments are typically hundreds of kilometers in length and are believed to be the surface expression of regional crustal fracture zones that extend into the crust and which juxtapose rocks of differing densities and magnetic susceptibility. Examples of these lineaments include the Tyrone-Mt. Union lineament in Pennsylvania and the Lawrenceville-Attica lineament in New York (Figure 3-56).

The Tyrone-Mt. Union lineament is believed to extend southeast from Lake Erie to beyond the Atlantic coastline of the United States where it is thought to coincide with transform faults associated with the mid-Atlantic ridge system. Subsurface geologic mapping and analysis of regional magnetic and gravity patterns suggest significant lateral displacement of at least 31 to 37 miles across this lineament.

The Lawrenceville-Attica lineament in western New York extends northwest from Lawrenceville, New York through Attica, New York and into western Lake Ontario. The Lawrenceville-Attica lineament may be contiguous with the Georgian Bay Linear Zone, a northwest-trending zone extending from Georgian Bay in southern Ontario southeastward in western New York State.

The Georgian Bay Linear Zone is an 18.6-mile wide structural zone that extends from Georgian Bay to the southeast across southern Ontario, western Lake Ontario, and into western New York (Figure 3-56). The Georgian Bay Linear Zone has been delineated by a set of northwest-trending aeromagnetic lineaments, one of which parallels the straight eastern shoreline of Georgian Bay.

A variety of neotectonic structures and features have been identified in surficial bedrock and in lake bed sediments within the Georgian Bay Linear Zone. These include faults and bedrock pop-ups and linear pockmarks and linear acoustic backscatter anomalies imaged on seismic sidescan profiles in lake bed sediments that may represent bedrock fractures and faults.

**Clarendon-Linden Fault System**

The Clarendon-Linden Fault System is located approximately 19 miles east of the Center (Figure 3-56) and is comprised of at least five north-south striking, high-angle faults which extend southward from Lake Ontario through Orleans, Genesee, and Wyoming Counties, and into Allegany County.

Stratigraphic relationships indicate that the overall sense of movement across the Clarendon-Linden Fault System is consistent with reverse faulting from east to west with up to 330 feet of stratigraphic separation across the Clarendon-Linden Fault System. Recent bedrock mapping and soil gas surveying in Allegany County suggests the Clarendon-Linden Fault System extends further south into Allegany County based on the presence of at least seven north-south striking fracture intensification domains and associated soil gas anomalies.

The southwest trending Attica Splay has been interpreted to splay off of the western north-south trending fault approximately 0.75 mile south of Batavia (Figure 3-56) and to continue to the southwest through Alexander and Attica, New York to a point approximately 1.25 miles northwest of Varysburg, New York. Seismic reflection data suggest the presence of at least two east-dipping faults extending from the Precambrian basement into the Paleozoic section forming a graben structure with a stratigraphic separation of 74 - 148 feet (Fakundiny, et al. 1978). The eastern fault is a reverse fault showing east to west movement and the western fault is a normal fault showing west to east movement.

Seismic reflection profiling suggests that the faults comprising the Clarendon-Linden Fault System are contiguous with faults located within the Grenville Province Central Metasedimentary Belt which underlies the Paleozoic bedrock of western New York. The Central Metasedimentary Belt has been subdivided into two distinct terrains, the Elzevir terrain and the Frontenac terrain, which are separated by the Elzevir-Frontenac Boundary Zone, a northeast trending six- to 22-miles wide crustal shear zone. The eastern boundary of the Elzevir-Frontenac Boundary Zone, which is known as the Maberly shear zone in southern Ontario, appears contiguous with the Clarendon-Linden Fault System in Western New York.

The Clarendon-Linden Fault System has been active at least since the Middle Ordovician and has displayed a complicated movement history alternating from normal or extensional faulting, to reverse or compressional faulting during the Paleozoic. The episodic movement along the Clarendon-Linden Fault System during the Paleozoic occurred in response to orogenic induced subsidence of the Appalachian basin. Normal faulting with down-to the-east motion occurred when the basin axis was located east of the Clarendon-Linden Fault System. Reverse faulting with east to west movement sense occurred when the basin axis was located west of the Clarendon-Linden Fault System.

**WVDP Seismic Reflection Survey**

In June 2001, the WVDP collected nearly 18 miles of seismic reflection data along an east-west line in southern Erie County, approximately 5 miles north of the Center (Bay Geophysical 2001). (See Figure 3-57.) This seismic survey was designed to image any north or northeast-trending structures in the Precambrian basement and overlying Paleozoic bedrock.

The WVDP also reviewed approximately 16 miles of reprocessed seismic reflection data collected in 1983 along a north-south line along Route 219 in Erie and Cattaraugus Counties. This line was reviewed to evaluate whether any east-west trending structures were present in the Precambrian basement and Paleozoic bedrock near the Center.

Both seismic lines indicate the presence of numerous high-angle faults originating in Grenville-age basement which extend up-section into Middle Ordovician or Middle Devonian strata. (See Figure 3-57) The majority of these faults terminate near the Middle Ordovician Trenton Group. These faults have apparent dips of 50 to 89.45° to the west, east, or south, show reverse and normal offset of bedding, and have up to 300 feet of stratigraphic separation.

Strata overlying some of the fault terminations are folded above the Middle Devonian Onondaga Formation, suggesting that these faults were emplaced or reactivated after the deposition of the uppermost folded unit. The most recent period of movement along these faults cannot be determined based on a lack of definitive age-dating relationships. Two faults near Sardinia, New York were interpreted to continue up-section through the Middle Devonian Onondaga Formation. These west-dipping normal faults show up to 300 feet of estimated stratigraphic separation (Figure 3-57).

A series of east- and south dipping high-angle faults spaced at intervals of 500 to 4,500 feet were interpreted in the Silurian to Devonian section northwest of Springville, New York. These faults originate in the Silurian Salina Group and cut up-section to the northwest through the Middle Devonian Onondaga Formation. These are believed to be thrust faults associated with the Bass Islands Trend.

### 3.5.5 Historical Seismicity

Earthquake catalogs maintained by the U.S. Geological Survey's National Earthquake Information Center were used to identify historical earthquakes with a magnitude of three or greater and a Modified Mercalli Intensity of IV or more within a 200-mile radius of the site. Three of the National Earthquake Information Center earthquake catalogs were queried to obtain information on earthquake activity in western New York. These included the Preliminary Determination of Epicenters, the Significant U.S. Earthquakes, and the Eastern, Central, and Mountain States of the United States catalogs. The historical seismicity search also utilized historical events identified in the Safety Analysis Report for Waste Processing and Support Activities (WVNSCO 2007). Historical seismicity within 200 miles of the site is summarized in Table 3-15. Table 3-15 also lists the date, location, time, depth, intensity, magnitude, distance, and information source.

From 1840 to 2003, there have been 45 recorded earthquakes with epicentral magnitudes of 3 or greater and Modified Mercalli Intensity of IV or greater within 200 miles of the WVDP. None of these earthquakes were reported to have caused landsliding or liquefaction events in the vicinity of the site. The geographic distribution of this seismicity is shown on Figure 3-55.

**Table 3-15. Historical Seismicity Within 320 Kilometers (200 Miles) of Site[(1)]**
(Only events with a magnitude ≥ 3 or a MMI intensity ≥ IV are listed)

| Date | Latitude (°N) | Longitude (°W) | Origin Time | Depth (km) | Intensity (MMI) | Magnitude ($m_b$) | Distance (km) | NEIC Catalog |
|---|---|---|---|---|---|---|---|---|
| 1840 9/10 | 43.20 | 79.90 | - | - | 5← | - | 113.7 | Unk |
| 1853 3/12 | 43.70 | 75.50 | - | - | 6← | - | 302.3 | Unk |
| 1853 3/13 | 43.10 | 79.40 | - | - | 5← | - | 74.9 | Unk |
| 1857 10/23 | 43.20 | 78.60 | 2015 | - | 6← | 4.3 FA | 83 | USHIS |
| 1873 7/6 | 43.00 | 79.50 | - | - | 6← | - | 73.6 | Unk |
| 1900 4/9 | 41.40 | 81.90 | 14 | - | 6← | 3.4 FA | 293 | USHIS |
| 1906 6/27 | 41.40 | 81.60 | - | - | 5 | 4.2 | 269.8 | Unk |
| 1912 5/27 | 43.20 | 79.70 | - | - | 5← | - | 100.6 | Unk |
| 1914 02/10 | 44.98 | 76.92 | 1831 | - | 7 | 5.20 FA | 313 | Unk |
| 1927 1/29 | 40.90 | 81.20 | - | - | 5 | - | 275.8 | Unk |
| 1928 9/9 | 41.50 | 82.00 | 21 | - | 5 | 3.70 FA | 297 | SRA |
| 1929 8/12 | 42.91 | 78.40 | 112448.70 | 9 | 8← | 5.20 Mn | 54* | SRA/ USHIS |
| 1929 12/2 | 42.80 | 78.30 | - | - | 5← | - | 47.4* | Unk |
| 1932 1/21 | 41.10 | 81.50 | - | - | 5 | - | 280.9 | Unk |
| 1934 10/29 | 42.00 | 80.20 | - | - | 5← | - | 134.9 | Unk |

**Table 3-15. Historical Seismicity Within 320 Kilometers (200 Miles) of Site**[1]
(Only events with a magnitude ≥ 3 or a MMI intensity ≥ IV are listed)

| Date | Latitude (°N) | Longitude (°W) | Origin Time | Depth (km) | Intensity (MMI) | Magnitude ($m_b$) | Distance (km) | NEIC Catalog |
|---|---|---|---|---|---|---|---|---|
| 1938 7/15 | 40.68 | 78.43 | 224612 | - | 6← | 3.30 FA | 233 | SRA/ USHIS |
| 1943 3/09 | 41.63 | 81.31 | 032524.90 | 7 | 5 | 4.50 Mn | 238 | SRA/ USHIS |
| 1951 12/03 | 41.60 | 81.40 | 0702 | - | 4 | 3.20 FA | 246 | SRA |
| 1954 01/31 | 42.90 | 77.3 | 12:30:00 | - | 4 | 3.1 | 121 | NCEER |
| 1954 02/1 | 43.03 | 76.65 | 00:37:50 | - | | 3.3 | 178 | NCEER |
| 1954 02/21 | 41.20 | 75.90 | - | - | +7← | - | 288.5 | Unk |
| 1954 04/27 | 43.10 | 79.20 | 02:14:08 | - | | 4.1 | 85 | NCEER |
| 1955 5/26 | 41.50 | 81.70 | - | - | 5 | 3.8 | 272.0 | Unk |
| 1955 6/29 | 41.50 | 81.70 | - | - | 5 | 3.8 | 272.0 | Unk |
| 1955 8/16 | 42.90 | 78.30 | - | - | 5 | - | 53.5* | Unk |
| 1958 5/1 | 41.50 | 81.70 | - | - | 5 | 4.0 | 272.0 | Unk |
| 1958 07/22 | 43.00 | 79.50 | 01:46:40 | - | | 4.4 | 92 | NCEER |
| 1958 08/4 | 43.13 | 80.00 | 20:25:58 | - | 4 | 3.8 | 134 | NCEER |
| 1958 08/22 | 43.00 | 79.00 | 14:25:05 | - | | 3.6 | 67 | NCEER |
| 1962 3/27 | 43.00 | 79.30 | - | - | 5← | 3.0 | 61.0 | Unk |
| 1963 01/30 | 44.00 | 75.90 | 1450 | - | 4 | 3.00 ML | 281 | SRA |
| 1964 02/13 | 40.38 | 77.96 | 194640.80 | 1 | 5 | 3.30 Mn | 237 | SRA |
| 1964 05/12 | 40.30 | 76.41 | 064510.70 | 1 | 6 | 4.50 mb | 303 | SRA/ USHIS |
| 1965 07/16 | 43.20 | 78.50 | 110655 | - | 4 | 3.50 ML | 84 | SRA |
| 1965 08/28 | 43.00 | 78.10 | 0155 | - | 4 | 3.10 ML | 75 | SRA |
| 1966 1/1 | 42.84 | 78.25 | 132339 | 0 | 6← | 4.70 mb | 54* | SRA/ USHIS |
| 1967 6/13 | 42.84 | 78.23 | 190855.50 | 1 | 6← | 4.40 Mn | 54* | SRA/ USHIS |
| 1980 6/6 | 43.56 | 75.23 | 131552 | 1 | 5 | 3.80 UK | 304 | PDE |
| 1980 6/6 | 43.57 | 75.14 | 131552.90 | 1 | 5 | 3.80 Mn | 311 | SRA |
| 1983 10/4 | 43.44 | 79.79 | 171840 | 2 | 4 | 3.10 Mn | 144 | PDE |

**Table 3-15. Historical Seismicity Within 320 Kilometers (200 Miles) of Site**[(1)]
(Only events with a magnitude ≥ 3 or a MMI intensity ≥ IV are listed)

| Date | Latitude (°N) | Longitude (°W) | Origin Time | Depth (km) | Intensity (MMI) | Magnitude ($m_b$) | Distance (km) | NEIC Catalog |
|---|---|---|---|---|---|---|---|---|
| 1986 1/31 | 41.65 | 81.16 | 164642.30 | 2 | 6 | 5.00 mb | 226 | SRA/ USHIS |
| 1986 1/31 | 41.65 | 81.16 | 164643.33 | 10 | 6 | 5.00 mb | 226 | PDE |
| 1987 7/13 | 41.90 | 80.77 | 054917.43 | 5 | 4 | 3.80 Mn | 185 | PDE |
| 1991 1/26 | 41.54 | 81.45 | 032122.61 | 5 | 5 | 3.40 Mn | 253 | PDE |
| 1991 8/15 | 40.79 | 77.66 | 071607.15 | 1 | 5 | 3.00 Mn | 202 | PDE |
| 1992 3/15 | 41.91 | 81.25 | 061355.22 | 5 | 4 | 3.50 Mn | 222 | PDE |
| 1993 10/16 | 41.70 | 81.01 | 063005.32 | 5 | 4 | 3.60 Mn | 212 | PDE |
| 1995 5/25 | 42.99 | 78.83 | 142232.69 | 5 | 4 | 3.00 Mn | 62 | PDE |
| 1998 9/25 | 41.49 | 80.39 | 195252.07 | 5 | 6 | 5.20 Mn | 179 | PDE |
| 2001 1/26 | 41.94 | 80.80 | 030320.06 | 5 | 5 | 4.40 Mn | 186 | PDE |
| 2003 6/30 | 41.80 | 81.20 | 192117.20 | 4 | 4 | 3.60 Mn | 223 | PDE |
| 2005 10/20 | 44.68 | 80.48 | 211628.75 | 11 | | 4.20 Mn | 316 | PDE |
| 2006 6/20 | 41.84 | 81.23 | 201118.54 | 5 | | 3.80 Mn | 239 | PDE |
| 2007 3/12 | 41.28 | 81.38 | 231816.41 | 5 | | 3.70 Mn | 271 | PDE |

NOTE: (1) From earthquake catalogs of the U.S. Geological Survey's National Earthquake Information Center. The coordinates used in the search criteria were latitude 42.450N and longitude 78.654W, which correspond to a point near the process Building.

LEGEND: ←Could have been felt at site * Associated with Clarendon-Linden Structure

Origin time is the time the earthquake occurred.

PDE = NEIC Preliminary Determination of Epicenters
NCEER = National Center for Earthquake Engineering Research
USHIS = NEIC Significant U.S. Earthquakes
SRA = NEIC Eastern, Central, and Mountain States of the United States
MMI = Modified Mercalli Intensity
Mn = Nuttli magnitude
ML = Local magnitude
Mb = Compressional Body Wave (P-wave) Magnitude
FA = Felt Area Magnitude
UK = Unknown Magnitude

The Buffalo-Lockport earthquake of October 23, 1857 affected an area of approximately 18,000 square miles. The epicentral intensity of VI was felt in an area 75 miles long, from north-northeast to south-southwest, and 62 miles wide. This earthquake was felt at Hamilton, Petersborough, and Port Hope in Ontario and at Rochester, New York, Warren, Pennsylvania, and Dayton, Ohio.

The August 12, 1929 earthquake occurred near Attica, New York, about 30 miles northeast of the WVDP. The affected area of approximately 50,000 square miles included parts of Canada. The earthquake was felt most strongly in the eastern part of the city of Attica and immediately to the east. There was less effect on structures immediately to the south of the epicenter, but changes in groundwater conditions were noted. Based on the reported damage, an epicentral intensity of VII and a Compressional Body Wave magnitude $m_b$ = 5.2 was assigned to the 1929 Attica event (WVNSCO 2007).

The Attica earthquakes of January 1, 1966 (Modified Mercalli Intensity VI) were felt over approximately 3,500 square miles of western New York, northwestern Pennsylvania, and southern Ontario, and the main shock was most strongly felt at Varysburg, about eight miles southwest of Attica. The Attica earthquake of June 13, 1967 (Modified Mercalli Intensity VI) was felt over an area of about 3,000 square miles in western New York. Slight damage was sustained at Attica and at Alabama, New York, where the shock was felt by many people. Focal mechanism solutions of these earthquakes indicate focal depths of approximately 1.2 to 1.9 miles and a combination of right-lateral strike-slip and reverse faulting on planes parallel to the northerly trend of the Clarendon-Linden Structure (Herrmann 1978).

### 3.5.6 Evaluation of Seismic Hazard

A site-specific probabilistic seismic hazard analysis of the Center was performed to estimate the levels of horizontal ground motions that could be exceeded at specified annual return periods at the site (Wong, et al. 2004). The hazard for the site was computed for a hard rock condition. Site response analyses were also performed for the north and south plateau areas of the site to evaluate the potential ground motion amplification resulting from soils and unconsolidated sediments that underlie the site, such as the Surficial Sand and Gravel Unit, Lavery till, and Kent Recessional Sequence.

A total of 19 seismic sources were included in the probabilistic hazard analysis, including four fault systems or fault zones and 15 regional seismic source zones. The fault systems considered in the analysis included the Clarendon-Linden fault zone, the Charleston fault zone, the New Madrid fault system, and the Wabash Valley fault system. The analysis considered the Southern Great Lakes seismic source zone in which the Clarendon-Linden fault zone is located. Regional seismic source zones were included in the analysis to incorporate the hazard associated with earthquakes affiliated with buried or unknown faults.

Peak horizontal ground acceleration and 0.1 and 1.0 second horizontal spectral accelerations) were calculated for bedrock at the Center for three DOE-specified return periods (Table 3-16). Figure 3-58 shows the various hazard curves for peak ground acceleration at the site including the mean and median curves. The hazard curves for the 1.0 second SA are shown in Figure 3-59.

The analysis indicates the largest contributor to the hazard at the Center is the Clarendon-Linden fault zone at almost all return periods, whereas seismicity within the Southern Great Lakes seismic source zone is the second most important contributor to seismic hazard at the site (Figure 3-60).

**Table 3-16 Site-Specific Mean Spectral Accelerations on Hard Rock (g's)(1)**

| Return Period (yrs) | PGA | 0.1 sec SA | 1.0 sec SA |
|---|---|---|---|
| 500 | 0.04 | 0.07 | 0.02 |
| 1,000 | 0.05 | 0.11 | 0.03 |
| 2,500 | 0.10 | 0.20 | 0.06 |

NOTE: (1) From Wong, et al. 2004.
LEGEND: PGA = peak ground acceleration, SA = spectral acceleration.

Site response analyses were performed for the north and south plateau areas for return periods of 500 and 2,500 years to evaluate potential ground motion amplification resulting from the unconsolidated glacial sediments underlying these areas (Tables 3-17 and 3-18). The increased peak ground acceleration in the north plateau evaluation suggests slight amplification of ground motions in the north plateau area of the site (Tables 3-16 and 3-17). The south plateau evaluation suggests ground motions for the 500 year return period are deamplified whereas ground motions are slightly amplified for the 2,500 year return period (Tables 3-16 and 3-18).

**Table 3-17 Site-Specific Mean Spectral Accelerations on Soil (g's) for the North Plateau(1)**

| Return Period (yrs) | PGA | 0.1 sec SA | 1.0 sec SA |
|---|---|---|---|
| 500 | 0.05 | 0.09 | 0.04 |
| 2500 | 0.14 | 0.24 | 0.11 |

NOTE: (1) From Wong, et al. 2004.
LEGEND: PGA = peak ground acceleration, SA = spectral acceleration.

**Table 3-18 Site-Specific Mean Spectral Accelerations on Soil (g's) for the South Plateau**

| Return Period (yrs) | PGA | 0.1 sec SA | 1.0 sec SA |
|---|---|---|---|
| 500 | 0.03 | 0.08 | 0.05 |
| 2500 | 0.11 | 0.22 | 0.14 |

NOTE: (1) From Wong, et al. 2004.
LEGEND: PGA = peak ground acceleration, SA = spectral acceleration.

## 3.6 Surface Hydrology

### 3.6.1 Hydrologic Description

The WVDP watershed is drained by three named streams: Quarry Creek, Franks Creek, and Erdman Brook (see Figure 3-3). Erdman Brook and Quarry Creek are tributaries to Franks Creek, which in turn flows into Buttermilk Creek. The WVDP drainage basin is approximately 1,200 acres.

The point where all surface runoff from the site reaches a single stream channel (the watershed outfall) is located at the confluence of Franks Creek and Quarry Creek, north of the main project facilities. On the WVDP site, numerous drainage ditches and culverts direct flow away from roadways and facilities to the channels of the stream headwaters that are located on or around the site. The most significant of these ditches and culverts would be those associated with the site railroad spur and Rock Springs Road.

Erdman Brook has a 140-acre drainage area and drains the central portion of the developed project premises, including a large portion of the disposal areas, the areas surrounding the lagoon system, the Process Building, warehouse areas, and a major part of the parking lots. Following treatment, the project's waste waters are also discharged to this brook.

Erdman Brook flows from a height of over 1,400 feet above mean sea level west of Rock Springs Road to 1,305 feet above mean sea level at the confluence with Franks Creek northeast of the lagoons. It flows through the project facilities for about 3,000 feet.

Quarry Creek drains the largest area of the three named streams (740 acres) and receives runoff from the HLW Tank Farm, the north half of the northern parking lot, and the Lag Storage Buildings. It flows from an elevation of 1,930 feet west of Dutch Hill Road to 1,245 feet at its confluence with Franks Creek. The segment that flows along the north side of the project is about 3,500 feet in length.

Franks Creek has a drainage area of 295 acres and receives runoff from the east side of the project, including the Drum Cell, part of the SDA, and the Construction and Demolition Debris Landfill. Franks Creek flows into Buttermilk Creek about 2,000 feet downstream of its confluence with Quarry Creek. It flows from an elevation of 1,790 feet above mean sea level west of Rock Springs Road, to 1,245 feet at the Quarry Creek confluence, to 1,180 feet at the Buttermilk Creek confluence. About 6,000 feet of its length lies adjacent to WVDP facilities. (WVNSCO 1993c)

Buttermilk Creek, shown in Figure 3-2, roughly bisects the Center property and flows in a northwestwards direction to its confluence with Cattaraugus Creek at the northwest end of the Center. Several tributary (perennial) streams flow into Buttermilk Creek in the Center (Figure 3-61).

The flow length of Buttermilk Creek through the Center is about 4.7 miles. Within the Buttermilk Creek watershed, a small 18-acre sub-basin on the east side of Buttermilk Creek drains the area around the Bulk Storage Warehouse.

Buttermilk Creek lies in a deep, narrow valley cut into glacial deposits, with a downstream portion down-cut to shale bedrock. The reach of stream to the east of the WVDP facilities has down-cut through the Lavery till and the underlying Kent Recessional Sequence, and is presently incising the Kent till. The Kent Recessional Sequence is discussed below.

The stream invert drops from an elevation of 1,310 feet above mean sea level at the southern Center boundary, to 1,215 feet at the northern edge of the Project facilities, to 1,110 feet at the confluence with Cattaraugus Creek. The drainage area of the Buttermilk Creek basin has been estimated to be 19,600 acres (Boothroyd, et al. 1982).

Buttermilk Creek flows at an average rate of 46 cubic feet per second to its confluence with Cattaraugus Creek. Peak flows were 340.3 cubic feet per second at the confluence of Quarry Creek and Franks Creek, 161 cubic feet per second where Franks Creek leaves the project premises, and 60 cubic feet per second in Erdman Brook downstream of the SDA. Peak flow measured at the U.S. Geological Survey USGS gauge station at the Bond Road Bridge over Buttermilk Creek (which operated from 1962 to 1968) was 3,910 cubic feet per second on September 28, 1967. The historic high-water level of 1,358.6 feet above mean sea level in the reservoirs was recorded on the same day.

Cattaraugus Creek flows westward generally at a rate of 353 cubic feet per second from the Buttermilk Creek confluence to Lake Erie, 39 miles downstream. The total drainage area is estimated to be 524 square miles. A gauging station has been maintained at Gowanda, New York since 1939. The drainage basin to this point is estimated to be about 432 square miles. The drainage area of Cattaraugus Creek upstream of the Buttermilk Creek confluence is an estimated 220 square miles.

A small hydroelectric dam and water impoundment is located on Cattaraugus Creek about 1,000 feet upstream of where the Scoby Road bridge was located, southwest of Springville, New York. Neither Buttermilk Creek nor Cattaraugus Creek downstream of the WVDP are used as a regular source of potable water. Cattaraugus Creek downstream of Buttermilk Creek is a popular fishing and canoeing/rafting waterway. As such, Cattaraugus Creek water, fish, and sediments are monitored as part of the WVDP environmental monitoring program.

The WVDP obtains potable and process water from two water supply reservoirs located south of the main plant facilities (see Figure 3-12). The reservoirs were formed by damming headwater tributaries to Buttermilk Creek and collect drainage from numerous small streams over a 3,100-acre drainage basin, of which 2,000 acres drain directly to Reservoir 1 and 1,100 acres drain directly to Reservoir 2. The storage capacity of the reservoirs is 19,815,435 cubic feet at 1,353 above sea level, and 17,857,265 cubic feet at 1,350.5 above sea level. An emergency spillway is located at the south end of Reservoir 1.

As explained in Section 3.1.3, the Low Level Waste Treatment Facility includes four in-series lagoons (lagoons 2, 3, 4, and 5). The largest is Lagoon 3, which has a capacity of 467,900 cubic feet. Lagoon 3 is the final lagoon in the system before the wastewater is discharged into Erdman Brook.

The site Sewage Treatment Plant discharges to a gully that flows into Erdman Brook. A former equalization basin for the Sewage Treatment Plant in 2004 served as a sludge pond for utility room discharges.

### 3.6.2 WVDP Effluents

WVDP effluents discharged to surface waters must meet limits prescribed by the NYSDEC for non-radiological parameters in a State Pollutant Discharge Elimination System permit and by DOE for radiological parameters. Discharges are monitored to ensure that all standards are met. Monitoring is performed at the point of effluent discharge and several surface water drainage locations. There are two permitted discharge locations at the WVDP:

- Outfall 007 (WNSP007) with an average daily flow of approximately 10,000 gallons (WVES and URS 2008). This outfall includes waters from the site sanitary and industrial wastewater treatment facility, and
- Outfall 001 (WNSP001) is batch discharged from lagoon 3. Approximately seven batches are discharged annually, totaling approximately 13.5 million gallons per year, including water from the Low Level Waste Treatment Facility.

### 3.6.3 Influence of Flooding on Site

Franks Creek, Quarry Creek, and Erdman Brook are located in deep steep-sided valleys bounding the north and south plateaus. Historical evidence and computer modeling indicate that flood conditions, including the probable maximum flood, will not result in stream flows overtopping their banks and flooding the north or south plateau. Therefore, the effects on the WVDP of flooding by these creeks are negligible, as supported by historical data. Figure 3-4 shows the 100-year floodplains of these streams.

An analysis of the probable maximum flood has been evaluated (URS 2008). The probable maximum flood is generally more conservative than the 500-year flood because it is defined as the flood resulting from the most severe combination of meteorological and hydrological conditions (DOE 2002).

Peak discharges of the probable maximum flood were generated for the sub-areas constituting the watershed using the SCS TR-20 computer modeling program (USSCS 1983). These discharges were then used to determine the depth of flow at four stream locations adjacent to site facilities. The results of these analyses demonstrate that the depths of flow associated with the probable maximum flood on area streams are well below the elevations of site facilities

The results of this analysis indicate that the probable maximum flood floodplain is very similar to the 100-year floodplain, particularly in areas adjacent to the developed portions of the site including areas where waste is stored or buried (URS 2008). Most of the stream channels near the developed portions of the north plateau area have relatively steep sides and the probable maximum flood flow remains in these channels. The probable maximum flood floodplain is wider than the 100-year floodplain in areas where the topography is relatively flat, such as the extreme upper reaches of Erdman Brook and Franks Creek.

Indirect short-term impacts, including stream bank failure and gully head advancement in the event of high stream flows could impact Lagoons 2 and 3 in WMA 2, the NDA in WMA 7, and site access roads in several locations of the project premises.

### 3.6.4 Water Use

#### Current Water Use of Buttermilk Creek

The project premises lies entirely within the Buttermilk Creek watershed. The Center property is adjacent to Buttermilk Creek nearly the entire stream length from its intersection with the Buffalo and Pittsburgh Railroad to its outlet into Cattaraugus Creek, approximately 3,000 feet upstream of the Felton Bridge. There is no public or private use of stream water within the Center property.

**Current Water Use of Cattaraugus Creek**

From the Buttermilk Creek outlet, Cattaraugus Creek flows approximately 39 miles to Lake Erie. The use of water within Cattaraugus Creek varies along the length of the stream.

Downstream of the Buttermilk outlet, Cattaraugus Creek flows through the Zoar Valley Multiple Use Area, Deer Lick Nature Sanctuary, the town of Gowanda, the Cattaraugus Indian Reservation, the town of Versailles, the town of Irving, and the town of Hanover, and outlets into Lake Erie at the hamlet of Sunset Bay. Cattaraugus Creek is not used as a source of public drinking water, as noted previously. Land use adjacent to Cattaraugus Creek is comprised of agricultural, forest, residential, recreational, and commercial. Some water is taken from Cattaraugus Creek for irrigation purposes.

The segment of Cattaraugus Creek which flows through the Zoar Valley Multiple Use Area is used for unsupervised swimming, rafting, and canoeing where water depth permits. Motorized boating is generally limited to within two miles of Lake Erie. Sunset Bay at the mouth of Cattaraugus Creek is a dense residential area with mixed recreation such as swimming beaches, marinas, boating and fishing.

Cattaraugus Creek downstream of the Springville dam provides habitat for lake-based fisheries, is a popular recreational fishing area, and is a top salmonid spawning stream within the Lake Erie drainage basin. Since 1994, New York has stocked Cattaraugus Creek with walleye, steel head trout, and brown trout.

**Current Water Use of Lake Erie**

Lake Erie is used for transportation, industrial, commercial, and recreational purposes. Recreational activities include sailing, boating, jet skiing, fishing, and swimming beaches.

Recent information on commercial fishing in the New York waters of Lake Erie is contained in the New York State Department of Environmental Conservation (NYSDEC) Annual Report to the Great Lakes Fishery Commission's Lake Erie Committee (NYSDEC 2004).

This report indicates that rainbow smelt currently are the target of a major commercial fishing industry on the Ontario, Canada side of Lake Erie, but are fished less in the United States waters. Since 1960, New York commercial fishing efforts have focused on walleye and yellow perch. However, yellow perch and walleye production from New York is a small fraction (less than five percent) of total Lake Erie landings for those species.

Open lake sport fishing in 2003 measured 352,128 angler-hours, the second lowest total in 16 years. Peak fishing activity occurred in July and Dunkirk Harbor was the most frequently used access site. Harvested fish include walleye, smallmouth bass, yellow perch, and lake trout. Electro-fishing surveys within Cattaraugus Creek document high densities of spawning-phase walleye, and continued stocking efforts are planned.

## 3.7 Groundwater Hydrology

Groundwater hydrology in the WVDP area is summarized below.

### 3.7.1 Description of the Saturated Zone

The subsurface of the WVDP has been investigated since the early 1960's, resulting in hundreds of borings and installation of groundwater wells and other subsurface monitoring equipment. As explained previously, the hydrogeology of the WVDP site includes a sequence of glacial sediments underlain by shale bedrock. In chronologically descending order, this sequence is composed of an alluvial-glaciofluvial sand and gravel unit on the north plateau underlain by a sequence of up to three relatively impermeable glacial tills of Lavery, Kent, and possibly Olean age, separated by stratified fluvio-lacustrine deposits, which are in turn underlain by shale bedrock.

The sediments above the Kent till – the Kent Recessional Sequence, the weathered and unweathered Lavery till, the Lavery till-sand, and the surficial sand and gravel – are generally regarded as containing all of the potential routes for the migration of contaminants (via groundwater) from the WVDP site. Figures 3-6 and 3-7 are generalized cross-sections across the north and south plateaus showing the relative locations of these sediments. The Lavery till, the Kent Recessional Sequence, and the Kent till are common to both the north and south plateaus. Detailed geologic cross sections have been constructed using lithologic data collected from boreholes installed from 1961 to the present.

The WVDP does not use groundwater for drinking or operational purposes, nor does it discharge effluent directly to groundwater. No public water supplies are drawn from groundwater downgradient of the WVDP or from Cattaraugus Creek downstream of the WVDP. However, groundwater upgradient of the WVDP is used for drinking water by local residents.

**Sand and Gravel Unit**

As explained previously, the sand and gravel unit is unique to the north plateau and is a silty sand and gravel layer composed of younger Holocene alluvial deposits, the thick-bedded unit, that overlie older Pleistocene-age glaciofluvial deposits, the slack-water sequence. Together these two layers range up to 41 feet in thickness near the center of the plateau and pinch out along the edges of the plateau, where they have been truncated by the sidewall of the bedrock valley or the downward erosion of stream channels.

Disturbed materials and fill from construction activities also exist to varying depths on the developed portions of the north plateau. These are typically composed of re-compacted original sediment.

Depth to groundwater within the sand and gravel unit varies from 0 to 16 feet, being deepest generally beneath the central area of the north plateau, decreasing to the west, east, and north, and intersecting the ground surface farther northeastward toward the security fence.

Groundwater in this unit generally flows northeastward toward Franks Creek (Figure 3-62). Groundwater near the northwestern and southeastern margins of the sand and gravel layer also flows radially outward toward Quarry Creek and Erdman Brook, respectively.

In areas upgradient of the north plateau groundwater plume, recharge is limited by run-off diversions and culverts that channel surface flow to distant parts of the plateau. There is minimal groundwater flow downward into the underlying Lavery till. The overall hydraulic gradient across the north plateau has been calculated at 0.031; gradients up to 0.049 and as little as 0.026 exists in localized areas. An average groundwater velocity of 61.0 feet per year has been calculated for this unit (WVNSCO 1993e).

Recharge to the north plateau has been estimated as ranging from 3.0 inches to 13.5 inches and averaging 6.8 inches per year. Precipitation and bedrock underflow are the largest contributors to this recharge. Discharge occurs through evapotranspiration and drainage to streams, seeps, and springs along the edge of the north plateau, with a negligible amount as downward flow into the underlying Lavery till.

**Weathered and Unweathered Lavery Till**

Groundwater flow in the weathered till has both horizontal and vertical components. Groundwater typically flows laterally across the south plateau before moving downward or discharging to nearby incised stream channels. A lateral groundwater velocity has been calculated at 4.4 feet per year in this unit.

Groundwater elevation contours of the weathered Lavery till illustrate a potentiometric surface that dips generally to the northeast (Figure 3-63), with the exception of the northern section of the NDA, which is controlled by the operation of the interceptor trench. Groundwater in areas next to the trench flows directly toward and into the trench. Once inside the trench, laterals along the bottom of the trench drain the water toward the manhole sump (monitoring location NDATR on Figure 3-63), where it is pumped regularly to Lagoon 2.

On the north plateau, the weathered Lavery till is much thinner or nonexistent, and the sand and gravel unit typically immediately overlies the unweathered Lavery till, as noted previously. Hydraulic head distributions in the unweathered Lavery till indicate that groundwater flow is predominantly vertically downward at a relatively slow rate, toward the underlying Kent Recessional Sequence. A vertical groundwater velocity of 0.2 feet per year has been calculated for this unit.

**Lavery Till-Sand Unit**

The Lavery till-sand is a sandy unit of limited areal extent that is up to 16 feet thick within the Lavery till, primarily beneath the southeastern portion of the north plateau. The potentiometric surface of the Lavery till-sand is characterized by a variably sloping surface that generally dips to the east and southeast across the entire unit towards Erdman Brook (See Figure 3-64). Surface discharge locations have not been identified.

**Kent Recessional Sequence**

The Kent Recessional Sequence is a fine-grained lacustrine unit of interbedded clay and silty clay layers locally overlain by coarse-grained glacial sands and gravels. These deposits are found below the Lavery till beneath most of the site and range up to 75 feet in thickness beneath the eastern portions of the site (WVNSCO 1993e).

Groundwater flow in the Kent Recessional Sequence is predominantly to the northeast, toward Buttermilk Creek (Figure 3-65). Recharge comes primarily from bedrock in-flow in the southwest, with limited recharge from the overlying Lavery till. The Kent Recessional Sequence discharges to Buttermilk Creek. Because of the limited recharge received from the overlying Lavery till, the upper portions of the Kent Recessional Sequence are unsaturated. The deeper portions are saturated, and the groundwater velocity has been calculated at 0.4 feet per year (WVNSCO 1993e).

Groundwater elevation contours of the Kent Recessional Sequence illustrate a potentiometric surface that dips to the northeast. The steepest gradient is found in the southwestern portion of the south plateau, where the shoulder of the underlying bedrock valley slopes steeply to the northeast. Toward the middle of the south plateau, the glacial sediments filling the valley thicken, and the groundwater contours flatten somewhat and begin to slope to the north-northeast.

**Shale Bedrock**

The bedrock underlying the site occurs as a U-shaped valley of upper Devonian shales and siltstones. The upper 10 feet of rock is weathered and fractured. Bedding in these units generally dips 0.5 degree southward.

### 3.7.2 Monitoring Wells

**Monitoring Equipment Inventory**

There are currently 286 wells, well points, piezometers, seepage points, manholes, and surface water elevation hubs in the WVDP groundwater monitoring equipment inventory. Of this total, 222 devices are actively used for various monitoring purposes, and 64 are considered inactive (i.e., not used for any purpose). A total of 235 monitoring devices have previously been removed from service via approved decommissioning protocols. The monitoring equipment inventory includes equipment installed since 1960.

Aquifer tests were performed at the WVDP to support development of the North Plateau Groundwater Recovery System and the pilot Permeable Treatment Wall in 1996 and in 2003, respectively. Slug tests are also routinely performed on selected groundwater monitoring wells as part of a site-wide well maintenance program. This information is used to determine if degradation of a well has occurred, indicating that redevelopment is needed.

### 3.7.3 Physical Hydrogeologic Parameters in the Saturated Zone

**Saturated Hydraulic Conductivity**

The WVDP performs hydraulic conductivity testing of selected wells on an annual basis in accordance with approved site procedures and good engineering practices. A rotational system of testing a different group of selected wells every year ensures that most wells are tested periodically.

A summary of averaged hydraulic conductivity results for the five hydrogeologic units, based on testing performed from 1987 through 2004, is provided in Table 3-19.

**Table 3-19. WVDP Hydraulic Conductivity (K) Testing Summary Table**[(1)]

| Geologic Unit | Sub-Unit | Maximum K (cm/s) | Average K (cm/s) | Minimum K (cm/s) |
|---|---|---|---|---|
| Sand and Gravel Unit | Thick-Bedded Unit | 3.78 E-02 | 4.43 E-03 | 1.25 E-04 |
| | Slack Water Sequence | 1.13 E-01 | 2.44 E-02 | 8.19 E-04 |
| Weathered Lavery Till | NA | 1.50 E-03 | 3.36 E-04 | 4.87 E-07 |
| Unweathered Lavery Till | Upper 3 meters | na | 1.00 E-06 | na |
| | Below 3 meters | na | 6.00 E-08 | na |
| Lavery Till-Sand | NA | 4.54 E-03 | 2.04 E-03 | 1.06 E-04 |
| Kent Recessional Sequence | NA | 1.62 E-03 | 7.03 E-04 | 2.98 E-06 |

NOTE: (1) From DOE and NYSERDA 2008.
LEGEND: NA = Not Applicable
na = not available

The WVDP does not regularly perform hydraulic conductivity tests on bedrock wells because so few onsite wells penetrate bedrock. The hydraulic conductivity of bedrock at the WVDP, based on values collected for similar rock types, is estimated to range from 1.0E-07 cm/s for unweathered rock to 1.0E-05 cm/s for the weathered zone (WVNSCO 1993e).

**Transmissivity**

The transmissivity of the sand and gravel unit varies across the north plateau due to the variability of its saturated thickness and hydraulic conductivity. The transmissivity ranges from 4.8 E-03 $cm^2/s$ to 6.8 E-03 $cm^2/s$ (WVNSCO 1993e).

### 3.7.4 Unsaturated Zone

**Description of the Unsaturated Zone**

The unsaturated zones (vadose zones) within the surficial sand and gravel layer and the weathered Lavery till have been characterized separately, due to their different lithologies.

Hydrologic data obtained from unsaturated zone monitoring arrays were used to determine response to wetting and drying events. These data indicate that a downward migrating wetting front is generated after significant precipitation, and is dependent upon the soil moisture, soil hydrogeology, and structural features in the soil. When the soil is near saturation, this front raises the water table; when the soil is dry, the front will either redistribute into or evapotranspire from the vadose zone before contacting the water table.

The vadose zone in the weathered Lavery till fluctuates an average of 10 feet (i.e., one foot to 11 feet from grade) and varies with the season; horizontal and vertical fracture flow occurs within the entire fractured zone during the wet season and in the lower weathered zone during the dry season.

Dry season matric potentials in the Lavery till create an upward flow gradient from grade to five feet, with widening fractures increasing this depth during the late discharge season. The capillary fringe of the Lavery till is approximately seven feet thick.

Due to a varying topography, the vadose zone of the sand and gravel layer fluctuates in thickness over a generally uniformly sloping water table that itself annually fluctuates an average of 30 inches. Water within this vadose zone flows vertically downward to the water table. Dry season and matric potentials in the surficial sand and gravel create an upward flow gradient from grade to 6.9 feet (WVNSCO 1993f). The capillary fringe of the sand and gravel varies between 8.3 inches to 16.7 inches, depending on local lithology (WVNSCO 1993f).

The unsaturated zone at the WVDP has been modeled with several different computer codes. Results of these efforts are available in WVNSCO 1992.

**Water Budget within the Unsaturated Zone**

Precipitation occurring from December through April is lost mainly to rapid runoff and infiltration. From May through November, precipitation is lost mainly to infiltration and subsequent evapotranspiration, with a minor portion going to runoff.

Maximum recharge to most soils occurs when the ratio of the infiltration rate to precipitation rate is equal to or less than 1.0. For dry Lavery till soils (<75 percent saturated), precipitation is almost immediately absorbed and stored in the soil as recharge. In wet or nearly recharged soils (>75 percent saturated), the capillary potential of the primary pores is low, and any fractures may show less conductivity due to soil swelling. Thus, for the same precipitation rate, the wet season infiltration rate is lower and recharge is governed by the saturated hydraulic conductivity of the soil matrix and, to a lesser extent, by any fracture flow. However, if the fractures are not yet fully closed (as occurs in the late fall), the absorptive capacity of the bulk soil volume can still be high, allowing horizontal flow of the meteoric water.

The local runoff to precipitation ratio is highest in spring since the ground is saturated from late fall rains, early winter snow melt, and spring rains that contribute new water to soil profile of high antecedent soil moisture. This ratio lowers throughout the late spring, summer, and early fall (April–October) due to a soil moisture deficit that is produced from increasing summer evapotranspiration rates, as indicated by tensiometric data.

### 3.7.5 Description of Unsaturated Zone Monitoring Stations

In addition to groundwater monitoring wells, the WVDP maintains 11 surface water monitoring hubs (SE001 through SE011) to collect surface water elevations in areas of the north plateau where the water table in the sand and gravel unit intersects the ground surface. This information is correlated with groundwater well data and is used to define the water table surface in areas where monitoring well coverage is sparse or nonexistent.

### 3.7.6 Physical Parameters

#### Total and Effective Porosity

Total porosity of the sand and gravel unit has been calculated and ranges from 21.0 percent to 22.8 percent with an average value of 21.9 percent (WVNSCO 1993e).

#### Specific Yield

The specific yield ($S_y$) of the sand and gravel unit has been calculated to range from 0.10 to 0.25 (WVNSCO 1993e). Lower values reflect areas of poor sorting, and higher values reflect areas characterized by well sorted sands and gravels.

#### Specific Storage

The specific storage of the unweathered Lavery till has been calculated through consolidation tests, and was observed to decrease with depth from a maximum of 1.6E-05 per cm (6.3E-06 per inch) to a minimum of 2.0 E-06 per cm (7.9E-07 per inch), with an average of 8.0E-06 per cm (3.15 E-06 per inch) (WVNSCO 1993e).

### 3.7.7 Numerical Analysis Techniques

Three-dimensional far-field and near-field groundwater flow and transport models were developed to support the preparation of the Decommissioning EIS. These models were developed to evaluate site-wide groundwater flow patterns across the project premises and underlying geologic units, evaluate local changes in groundwater hydrology resulting from the proposed EIS closure alternatives, and identify transport parameters required to complete the performance assessments for the closure alternatives.

The three-dimensional site-wide groundwater flow model was the Finite Element Heat and Mass Transfer Code (FEHM), a finite element code developed by the DOE's Los Alamos National Laboratory (LANL 2003). The FEHM model used in the preparation of the Draft EIS was an improvement over earlier models developed for the site which were limited to evaluating groundwater flow in the surficial sand and gravel unit in the north plateau of the Center. The FEHM model evaluated groundwater flow over a larger lateral and vertical extent of the Center, including the glacial geologic units underlying the surficial sand and gravel unit. The lateral and vertical boundaries of the site-wide FEHM model are as follows:

- Northern Boundary – from Quarry Creek eastward to Franks Creek downstream to its confluence with Buttermilk Creek,
- Western Boundary – follows the 1,450 foot surface elevation contour along Rock Springs Road between Quarry Creek and Franks Creek to the south,
- Southern Boundary – follows Franks Creek along the southern boundary of the South Plateau and continues as an imaginary line to Buttermilk Creek,
- Eastern Boundary – follows Buttermilk Creek from the confluence with Franks Creek to the north, to the intersection of the Southern Boundary with Buttermilk Creek in the south,
- Upper Boundary – the upper surface of the model domain follows the ground surface, and
- Bottom Boundary – the bottom surface of the model domain is at an elevation of 525 feet above sea level.

The finite-element grid used in the site-wide model used a total of 955 grid blocks with a uniform dimension of 140 feet in the x-y plane with a node located in the center of each grid block. The model was subdivided vertically into 23 discrete layers to represent the varying thicknesses of the 10 geologic units being modeled (thick-bedded unit, slack-water sequence, weathered Lavery till, unweathered Lavery till, Kent Recessional Sequence, Kent till, Olean Recessional Sequence, Olean till, weathered bedrock, and bedrock). The site-wide model has a total of 21,965 nodes with 955 in each model layer.

The site-wide model was calibrated both manually and with the automated calibration code, Parameter Estimation (PEST) (Doherty 2004). The manual calibration involved the comparison of model predicted heads with the median of observed groundwater level elevations from 56 well locations, and comparison of model predicted seepage flows with actual estimated seepage flows. The model simulated water table contours generated for the thick-bedded unit in the north plateau and the weathered Lavery till in the south plateau are in close agreement in most areas with the observed fourth quarter water table for the north plateau and south plateau. Differences were noted in several areas of the north and south plateaus that are partly attributed to the model grid size.

The site-wide FEHM groundwater flow model was not well suited for evaluating flows associated with the proposed small-scale close-in-place alternative and phased decision-making alternative engineered structures. A three-dimensional near-field groundwater flow model, the Subsurface Transport Over Multiple Phases Code (STOMP), was used to evaluate rates and directions of groundwater flow in the surficial sand and gravel unit that would be affected by the proposed engineered barriers associated with the close-in-place and phased decision-making alternatives. STOMP is a finite difference code developed by the Pacific Northwest National Laboratory (PNNL 2000). The stratigraphy and boundary conditions used in the FEHM far-field model were incorporated into the STOMP model to the maximum extent. The results of the STOMP near-field groundwater flow modeling associated with the WMA 1 and WMA 2 hydraulic barriers are described in Appendix D.

### 3.7.8 Distribution Coefficients.

An important aspect of site hydrogeology is the mobility of a contaminant in the various soil layers under the influence of groundwater. The distribution coefficient, also called partition coefficient or $K_d$, is used to describe the decrease in concentration of a contaminant in solution through interactions with geologic media in a soil-groundwater system. The $K_d$ is defined as the ratio of the concentration (or activity in the case of radionuclides) of a species sorbed on the soil, divided by its concentration (or activity) in solution under steady-state conditions. It is an empirical parameter and its use in a given situation implies that the soil-groundwater system under study is in equilibrium.

The set of elements whose sorption onto West Valley geologic media have been studied over the years is representative in several respects. First, most of the elements considered have radioisotopes typically identified as key in post-closure performance assessments. The elements considered are also representative in that, based on location in the periodic table, several potentially different chemical behaviors are considered, such as monovalent and multivalent cations, chelation, formation of anionic species, and actinides.

$K_d$ values for several important radionuclides have been determined for materials from those hydrogeological units of primary interest – the surficial sand and gravel unit on North Plateau, the weathered Lavery till, and the unweathered Lavery till. There are fewer results for the lacustrine unit and no data for the Kent Recessional till or bedrock.

Finally, $K_d$ values at West Valley have been estimated by a variety of different techniques – batch studies, experimental sorption isotherms, column studies, and the analysis of contaminant migration in soil cores taken from the site.

**$K_d$ Studies at the Center**

Five studies have been performed, as described below.

***Brookhaven studies – Chemical Environment***. $K_d$ values for Cs, Co, Sr, Am, and Eu were determined in a series of experiments at the Brookhaven National Laboratory for four West Valley geochemical environments: the Lavery till, the lacustrine unit, overland flow, and the waste mass in the disposal trenches (Pietrzak et al. 1981). Samples of unweathered Lavery till collected at a depth of 35 feet in the SDA were tested for their sorption characteristics in the presence of trench leachate collected from sumps and well points. Batch $K_d$ determinations were conducted in both oxic and anoxic environments. This study was sponsored by NRC.

A description of the equipment and procedures employed in the Brookhaven study, and preliminary results and conclusions, were reported in Columbo and Weiss 1979 and subsequently expanded by Pietrzak et al. 1981. The latter report includes $K_d$ values for europium and americium as well as cesium, strontium, and cobalt, and discusses the observed effects of each of several variables on the sorption characteristics of the till.

In addition to quantifying distribution coefficients, the Brookhaven studies clearly demonstrate both the effects of anoxic or reducing environments on sorption, and the effect of complexing agents, i.e., organics in the trench water, on sorption. The studies also indicated that the soil disaggregation technique used in an experiment has an impact on the $K_d$. Hence, there is an element of uncertainty in the observed $K_d$ values due to experimental method, as well as to natural variation, in the Brookhaven numbers.

***NFS Sorption Studies – Variation With Depth***. In 1974, Duckworth (Duckworth, et al. 1974) reported percentage sorption for Cs-137, Sr-85, Ru-106, and Co-60 on a total of 37 samples of weathered and unweathered Lavery till taken from the SDA at depths of four to 51 feet. Iodine sorption percentages were also determined for 10 samples of weathered and unweathered till. Later, the WVDP used these data to calculate the distribution coefficients for the radioisotopes studied (WVNSCO 1993a).

The number and distribution of the samples tested clearly indicate differences between sorption on weathered and sorption on unweathered till but for not all radionuclides. This pattern is illustrated in Figures 3-66 through 3-68.

The right half of each figure shows stripplots[6] of the $K_d$ values determined at four increasing depths: 10 feet, 25 feet, 30 feet, and 50 feet. The 10-foot $K_d$ values are for weathered till and the remaining $K_d$ values are for unweathered till. The left half of each figure shows the normal probability plot[7] of all of the $K_d$ values where the weathered (10-foot) $K_d$ values are solid black circles and the unweathered till $K_d$ values are solid gray circles.

In the figures, cesium and strontium – and possibly iodine – show variation of the $K_d$ with soil type (i.e., by depth). (The iodine data show a similar variation by soil type, but this trend is less statistically significant in light of the smaller number of samples involved.) Neither the ruthenium nor the cobalt $K_d$ values vary with depth.

Finally, there is one drawback to this set of distribution coefficients: the longest contact time in the batch experiments was 16 hours, and it is unlikely that equilibrium was attained. However, shorter contact times lead, in principle to lower (more conservative) $K_d$ values.

***Oak Ridge National Laboratory Study - Competitive Sorption on the Lavery Till***. Lavery Till samples from 1961 were submitted to Oak Ridge National Laboratory for batch-test radionuclide sorption studies. The locations and sampling depths were selected to provide coverage at both shallow to intermediate depths within the till, providing a comparison of the weathered and unweathered materials (WVNSCO 1993a).

The study results for cesium and strontium were numerically similar[8] to the results from Duckworth's data, showing that the Lavery till has a high affinity for cesium and a lower affinity for strontium. Cobalt-60 was almost completely sorbed by both weathered and unweathered tills with cobalt exhibiting no selectivity for either material.

[6] Individual $K_d$ determinations are plotted and grouped by weathered or unweathered.

[7] A normal probability plot presents the ordered values of the $K_d$ versus the z-scores of the corresponding quantiles from the standard normal distribution. In these figures, the "Sample Quantiles" are just the $K_d$ values and the "Theoretical Quantiles" are the z-scores. (A z-score is a measure of the distance in standard deviations of a sample from the mean.)

[8] The Oak Ridge tests were 24 hour batch tests. The Kd's were higher but still comparable

Some tests were also run for ruthenium, but the results were not considered particularly meaningful because they were conducted using ruthenium which had percolated through the Oak Ridge soil and from which the sorbable and filterable portions had been removed. The Oak Ridge sorption percentages were much lower than those observed by Duckworth. Chelation or complexation of the ruthenium in the Oak Ridge solution is a plausible explanation for the lower sorption.

Competitive sorption effects – cesium/potassium and strontium/calcium – were also examined in the Oak Ridge study. In both cases, the presence of a competitor species slowed sorption. The introduction of potassium ions reduced the sorption of cesium by a factor of six. Similarly the sorption of strontium was found to be reduced fourfold by the presence of calcium in the leachate.

***United States Geological Survey Estimates****.* U.S. Geological Survey studies (Prudic 1986) on groundwater flow and contaminant transport in till immediately adjacent to the SDA have also included estimates of $K_d$ values for several elements – cesium, strontium, hydrogen, and carbon. In this study, the $K_d$ values were inferred from travel distances from the trench. The results for the carbon, cesium and strontium are consistent with the Brookhaven results for unweathered till under anoxic conditions. The tritium is assumed to be in the form of tritiated water and to experience no sorption[9] (i.e., a $K_d$ of 0).

***WVDP – North Plateau Sand and Gravel****.* In 1995 Dames and Moore reported the results for radionuclide sorption onto samples of the surficial North Plateau sand and gravel (Aloysius 1995 and Dames and Moore 1995). $K_d$ values were determined for strontium, technetium, iodine, cesium, europium, uranium, neptunium, plutonium and americium. Most of the determinations used either batch tests and/or plots of the sorption isotherms.

This study also examined several related phenomena of potential interest. The effect of having tributyl phosphate/n-dodecane present was investigated for both uranium sorption and americium sorption. No effects were observed for either radionuclide. Competitive effects between technetium and iodine were also studied, indicating that iodine is preferentially sorbed.

At the present, Sr-90 is the primary radionuclide of interest in the north plateau surficial aquifer. For this reason, strontium's sorption behavior was studied in great detail by the investigators. In addition to batch and isotherm testing, the $K_d$ of strontium was determined in column experiments and by the analyses of field data showing the distribution of Sr-90 in the surficial sand and gravel aquifer and the observed flow field of the aquifer. These dynamic estimates for the Sr-90 $K_d$ were consistent with the batch and isotherm determinations.

The effect of the chemical environment on strontium sorption was also investigated. The $K_d$ was found to be sensitive to small changes in pH and to increase with increasing pH. The strontium $K_d$ was observed to increase with increasing ionic strength, but decrease with increasing calcium concentrations, i.e., the calcium is preferentially sorbed. These

[9] This neglects absorption into pore-space deadwater.

experimental findings were corroborated with geochemical modeling using the MINTEQA2 code.

Table 3-20 summarizes the distribution coefficients quantifying the sorption of fourteen elements onto West Valley soils. The primary Brookhaven references are not available and values have been taken from citing documents. Where possible, the values have been entered as ranges.

**Table 3-20. Distribution Coefficients**

| Element | $K_d$(cm$^3$/g) | Geohydrological Unit | Notes | Reference |
|---|---|---|---|---|
| Hydrogen | 0 | Unweathered Lavery Till | Assumed zero (tritiated water) | Prudic 1986 |
| Carbon | 0.7 - 1.1 | Unweathered Lavery Till | Anoxic conditions, organic carbon | Prudic 1986 |
| | 3 – 12 | Unweathered Lavery Till | Anoxic conditions, inorganic carbon | Prudic 1986 |
| Cobalt | 1 – 5 | Unweathered Lavery Till | Anoxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 1.8 - 2.3 | Unweathered Lavery Till | Oxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 6400 | Unweathered Lavery Till | 16 hr batch | WVNSCO 1993a |
| | 5400 | Weathered Lavery Till | 16 hr batch | WVNSCO 1993a |
| Strontium | 4.5 | Surficial Sand and Gravel | North plateau | Aloysius 1995 |
| | 6.9 - 7.4 | Unweathered Lavery Till | Anoxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 25 – 32 | Unweathered Lavery Till | Oxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 1 – 7 | Unweathered Lavery Till | In-situ assessment, SDA, anoxic conditions | Prudic 1986 |
| | 30 | Unweathered Lavery Till | | WVNSCO 1993a |
| | 130 | Weathered Lavery Till | | WVNSCO 1993a |
| Technetium | 4.1 | Unweathered Lavery Till | Regression fit of linear isotherm | Aloysius 1995 |
| Ruthenium | 1300 | Unweathered Lavery Till | | WVNSCO 1993a |
| | 1200 | Weathered Lavery Till | | WVNSCO 1993a |
| Iodine | 0.4 - 3.4 | Lavery Till | | WVNSCO 1993a |
| Cesium | 48 – 260 | Unweathered Lavery Till | Anoxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 100 – 200 | Unweathered Lavery Till | Oxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 3350-4500 | Unweathered Lavery Till | | WVNSCO 1993a |

**Table 3-20. Distribution Coefficients**

| Element | $K_d (cm^3/g)$ | Geohydrological Unit | Notes | Reference |
|---|---|---|---|---|
| | 4900-8000 | Weathered Lavery Till | | WVNSCO 1993a |
| Europium | > 14,000 | Surficial Sand and Gravel | Based on detection limit | Aloysius 1995 |
| | 600 – 2100 | Unweathered Lavery Till | Anoxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 3700 – 4300 | Unweathered Lavery Till | Oxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| Radium | 195 | Unweathered Lavery Till | | Pietrzak, et al. 1981 cites Bergeron, et al. 1987 |
| Uranium | 9.1 - 9.6 | Unweathered Lavery Till | Regression fit of linear isotherm | Aloysius 1995 |
| | 11.9 | Unweathered Lavery Till | Regression fit of linear isotherm, TBP/n-dodecane present | Aloysius 1995 |
| Neptunium | 2.3 | Surficial Sand and Gravel | Recommendation | Aloysius 1995 |
| | 0.5 - 5.2 | Unweathered Lavery Till | Regression fit of linear isotherm | Aloysius 1995 |
| | 5.5 - 18.1 | Weathered Lavery Till | Regression fit of linear isotherm | Aloysius 1995 |
| Plutonium | 2600 | Surficial Sand and Gravel | Kinetic sorption experiment (120 hr batch) | Aloysius 1995 |
| | 27900 | Unweathered Lavery Till | Kinetic sorption experiment (120 hr batch) | Aloysius 1995 |
| | 5 – 56 | Unweathered Lavery Till | Anoxic trench water | Matuszek 1980 |
| Americium | 111000 | Unweathered Lavery Till | | Aloysius 1995 |
| | 77,000-272,000 | Unweathered Lavery Till | In presence of TBP/ n-dodecane | Aloysius 1995 |
| Americium | 420 – 1000 | Unweathered Lavery Till | Anoxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |
| | 4000 – 4700 | Unweathered Lavery Till | Oxic trench water | Pietrzak, et al. 1981 and Columbo and Weiss 1979 |

NOTE: (1) Range reflects differences due to experimental technique employed for soil disaggregation.

### 3.7.9 Hydraulic Properties

Prudic noted the abundant fractures in the weathered Lavery till zone, indicating that fractures with oxidized walls, spaced a few meters apart, extended down to about 14.7 feet (Prudic 1986). The oxidized zones bordering the fractures, as well as thin coatings of manganese and/or iron oxide, calcite, root hairs, and thin gray (reduced) zones on the inner surfaces of some fractures, clearly suggest water movement along the fractures.

The WVDP has total porosity data from several investigations. Table 3-21 shows results from samples obtained during monitoring well installation in the 1989-1990 period as reported in WVNSCO 1993e, which are representative of the available data. In the case of samples from the sand and gravel layer, the weathered Lavery till, and the unweathered Lavery till, total porosity was calculated using the equation:

$$P = [\, 1 - \rho / G \,] \times 100\ \%$$

where $P$ = total porosity

$\rho$ = bulk dry density

$G$ = specific gravity

An estimated bulk dry density of 2.1 g/cm$^3$ was used in the calculations for the sand and gravel layer and1.6 g/cm$^3$ for the Lavery till, both weathered and unweathered.

**Table 3-21. Total Porosity(1)**

| Geologic Unit | Range of Total Porosity (%) | Average Total Porosity (%) |
|---|---|---|
| Sand and Gravel(2) | 21 to 22.8 | 21.9 |
| Weathered Lavery Till(3) | 40.3 to 41 | 40.7 |
| Unweathered Lavery Till(4) | 41.4 to 42.5 | 41.7 |
| Lavery Till Sand(5) | NA | 25 |
| Kent Recessional Sequence(5) | NA | 25 |

NOTES: (1) From WVNSCO 1993a. The total porosity values were determined from boring samples collected during monitoring well installation in 1989 and 1990.

(2) From Table 2-1 of WVNSCO 1993e.

(3) From Table 3-1 of WVNSCO 1993e.

(4) From table 4-1 of WVNSCO 1993e.

(5) Estimated based on particle size and sorting.

## 3.8 Natural Resources

This section describes existing and potential natural resources at and in the vicinity of the WVDP. These resources include natural gas and oil, sand/gravel/clay deposits, surface water, groundwater, timber and two renewable energy sources–geothermal and wind energy.

### 3.8.1 Natural Gas and Oil

New York has proven natural gas and oil resources (NYSDEC 2001). The New York State Department of Environmental Conservation estimates that the state’s 2001 production was enough to heat approximate 353,000 homes. A significant portion of these resources are found in Chautauqua, Cattaraugus, and Erie Counties.

The annual production of natural gas and oil in New York State during 2001 is summarized in Table 3-22 along with production in nearby areas such as the Town of Ashford. New York produced 28 billion cubic feet of natural gas in 2001. Cattaraugus County and Erie County were the fourth and fifth largest producing counties in the state accounting for 9 percent of the production for that year. The largest Western New York producer of natural gas was Chautauqua County which was responsible for almost 23 percent of the State's production.

**Table 3-22. 2001 Natural Gas and Oil Production in Cattaraugus and Erie Counties, and the State of New York**[1]

| Location | County | Gas (1000s ft³) | Oil (barrels) | Active Gas Wells | Inactive Gas Wells | Active Oil Wells | Inactive Oil Wells |
|---|---|---|---|---|---|---|---|
| Ashford | Cattaraugus | 20,879 | 1,065 | 13 | 4 | 2 | 0 |
| East Otto | Cattaraugus | 6,133 | | 6 | 2 | 0 | 1 |
| Ellicottville | Cattaraugus | 6,344 | | 16 | 0 | 0 | 0 |
| Machias | Cattaraugus | 220 | | 1 | 1 | 0 | 0 |
| Yorkshire | Cattaraugus | 23,740 | | 18 | 3 | 0 | 0 |
| Colden | Erie | 6,374 | | 11 | 6 | 0 | 0 |
| Sardina | Erie | 19,228 | | 11 | 3 | 0 | 0 |
| Total | | 82,918 | 1,065 | 76 | 19 | 2 | 1 |
| Total Cattaraugus County | | 1,383,691 | 116,373 | 427 | 175 | 1,557 | 440 |
| Total Erie County | | 1,132,634 | 45 | 875 | 239 | 1 | 1 |
| New York State | | 28,020,207 | 175,666 | 5,949 | 843 | 3,373 | 1,416 |

NOTE: (1) From NYSDEC 2001.

Cattaraugus County was the top oil producing county in New York in 2001 contributing more than 66 percent to the state total. However, less than one percent of the county's contribution came from the Town of Ashford's two active oil wells. There are no active wells in any of the towns adjacent to Ashford.

Figure 3-69 shows the locations of all of the known wells associated with the production of natural gas and oil in Western New York. Figure 3-70 shows production in the Town of Ashford in Cattaraugus. The approximate location of the WVDP is indicated on Figure 3-72 by the black "**WV**." These two graphics clearly indicate that production occurs in the immediate vicinity of the site, but the site lies on the fringes of known resources. Most of the gas production occurs in a band paralleling Lake Erie west of the site, and most of the oil production occurs in the southern part of Cattaraugus County near the Pennsylvania state line.

### 3.8.2 Mineral Resources

#### Sand, Gravel, and Clay

As described above, the WVDP site and surrounding valley area are underlain by a sequence of glacial tills comprised mainly of clays and silts separated by sands and gravels. These materials are a potential mineral resource, although a determination of their classification (USGS 1980) as resource, reserve, marginal reserve, or sub-economic resource has not been evaluated. In any event these materials are currently restricted by virtue of the restricted access to the Center.

Sand and gravel mines are New York's most common type of mine. Construction sand and gravel is a high-volume, low-value commodity. The industry is highly competitive. Production costs vary widely depending on geographic location, the nature of the deposit, and the number and type of products produced. Transportation is a major factor in the delivered price of construction sand and gravel, and because of the high cost of transportation, construction sand and gravel continues to be marketed locally (NYSDEC 2005).

In 2001, there were 1931 active sand and gravel mines in the state producing more than 30 billion metric tons worth at least $162 million. Data for production by mine for that year are not available. However, based on permitted acreage two of New York's seven largest producers have mines in the vicinity of the WVDP (NYSDEC 2005). One is in the adjacent town of Machias, and the other in nearby Sardinia. There are approximately 20 mine sites within six miles of the WVDP. Approximately half of those were active in 2001.

The major clay minerals found in the site tills are illite and chlorite. Such clays are not particularly valuable for ceramic or industrial applications. There is one regulated clay mine in the Town of Concord which is within six miles of the site.

### 3.8.3 Water Resources

Both surface water and groundwater resources are found at the WVDP (see Sections 3.6 and 3.7). Buttermilk Creek Basin is a proven surface water resource. Its headwaters are located in and adjacent to the southern part of the site, and the creek flows northwest through the site. Two small water reservoirs were constructed on headwater tributaries to supply both potable and process water to Center and WVDP facilities.

Groundwater within the Center and the WVDP is not utilized for any purpose, as noted previously. However, groundwater is a proven if limited resource in the West Valley area as indicated by the use of several off-site residential wells. Approximately 259 homes within a 3.1-mile radius of the WVDP utilize groundwater as a potable water source. These wells utilize groundwater from surficial sand and gravel aquifers of limited areal extent, as well as weathered bedrock aquifers. Significant quantitative characterization of groundwater is limited to the WVDP, specifically the north plateau and south plateau. That effort has focused on contaminant hydrology as opposed to water resource characterization.

Using knowledge of the groundwater in the vicinity of the WVDP, one basin-wide aquifer is postulated, the weathered and fractured bedrock system. Lying above the competent, low permeability shale bedrock and below the low-permeability glacial tills, this system is recharged from the upland slopes bordering the valley. Discharge is largely to Buttermilk Creek which has cut through the till to bedrock in the valley floor. Little if any connection of the West Valley fractured bedrock aquifer with similar systems in the Connoissarauley and Broad Valleys is expected due to the intervening shale uplands.

Aquifers associated with the glacial drift are sand and/or gravel units of limited areal extent. The surficial sand and gravel unit of the north plateau receives significant recharge from infiltrating precipitation, is highly permeable, and lies on top of low-permeability clayey/silty till. However, it has limited lateral extent and discharges along much of its perimeter.

Subsurface sand and/or gravel units also appear to be limited in extent. Recharge to these units is poorly understood. Given the low permeability of the clayey/silty tills in which they are embedded, some connections with and recharge from the upland fractured-rock flow system at the valley periphery is plausible.

In sensitivity analyses with the three-dimensional site groundwater model, simulations have been run with and without the subsurface till sand unit which is situated on the north plateau east of the Project facilities. The simulations showed little sensitivity to the presence of this unit and the model fit was slightly better when it was left out. These results suggest that the flow associated with this system is not a significant participant in the overall scheme and this inference, by extension, implies that the unit (and others like it) are limited as water resources.

Finally, it is noted that the West Valley aquifer system is part of the Cattaraugus Creek Basin Aquifer System, designated as a sole source aquifer. Similar to West Valley, the sand and gravel aquifers in this system used as water sources tend to be local and limited in spatial extent. Generally, the gradient from the Cattaraugus sand and gravel aquifers is downward toward the fractured bedrock system or laterally to surface waters.

### 3.8.4 Timber Resources

The region's (Southern Tier) specific soil and climate help to produce several commercial species of hardwood timber including maple, ash, red oak and black cherry. The estimated annual net growth of timber amounts to over 1.6 million tons a year (STPRDB 2003). At present, about one third of this amount is being removed through harvesting, leaving a significant potential for future economic development, including the potential for increased domestic secondary use and export use.

Much of the Center is forested, as is characteristic of the region. A smaller portion of the WVDP is forest, however. The last sawtimber harvest occurred mid-century with cull, inferior, and smaller trees left. There has been no management in the interim. In 1978, the volume of sawtimber at the Center was estimated to be 3.2 million board feet having a total standing value of $313,000. Most of the value came from hardwoods. The annual growth rate was estimated to be low at 100 board-feet per acre per year. When corrected for inflation, the average stumpage rate of all eastern hardwoods increased by roughly 250 percent from 1978 to 1999 (Howard 2001). Neglecting new growth, degradation, the absence of management, changes in mix, etc., the current value of the Center forest would be $750,000.

### 3.8.5 Renewable Energy Resources

There are two renewable energy sources which are notable potential resources at or in the vicinity of the WVDP. These are geothermal energy and wind energy.

#### Geothermal

Geothermal energy is an inferred, i.e., unproven, resource at the Center. Recently development studies for the western Southern Tier (STPRDB 2003) have recognized the geothermal potential in that region. The reports indicate that low temperature geothermal wells are available in portions of Western New York. Analysis of bottom hole temperature data from Cambrian sandstones indicates the presence of extractable fluids in the low temperature geothermal target zone. The report notes that the potential of geothermal power has not yet been utilized in the region due to technological obstacles, high initial capital costs, and a reluctance to engage new resources. Low temperature geothermal resources may be used for direct heat, i.e., heat pumps, but not for the generation of electricity.

**Wind**

Recent work suggests that the hilltops to the west of the WVDP are suitable for the development of wind energy resources. In 2004, NYSERDA was engaged in wind energy research and recently has funded the development of wind resource maps for the entire state of New York (TrueWind 2005). Based on extensive meteorological data and numerical models, the maps rate every location in the state for wind energy potential. In these maps, locations along the ridge or hilltops separating West Valley from Connoissarauley Valley are rated as having a good potential for wind energy development.

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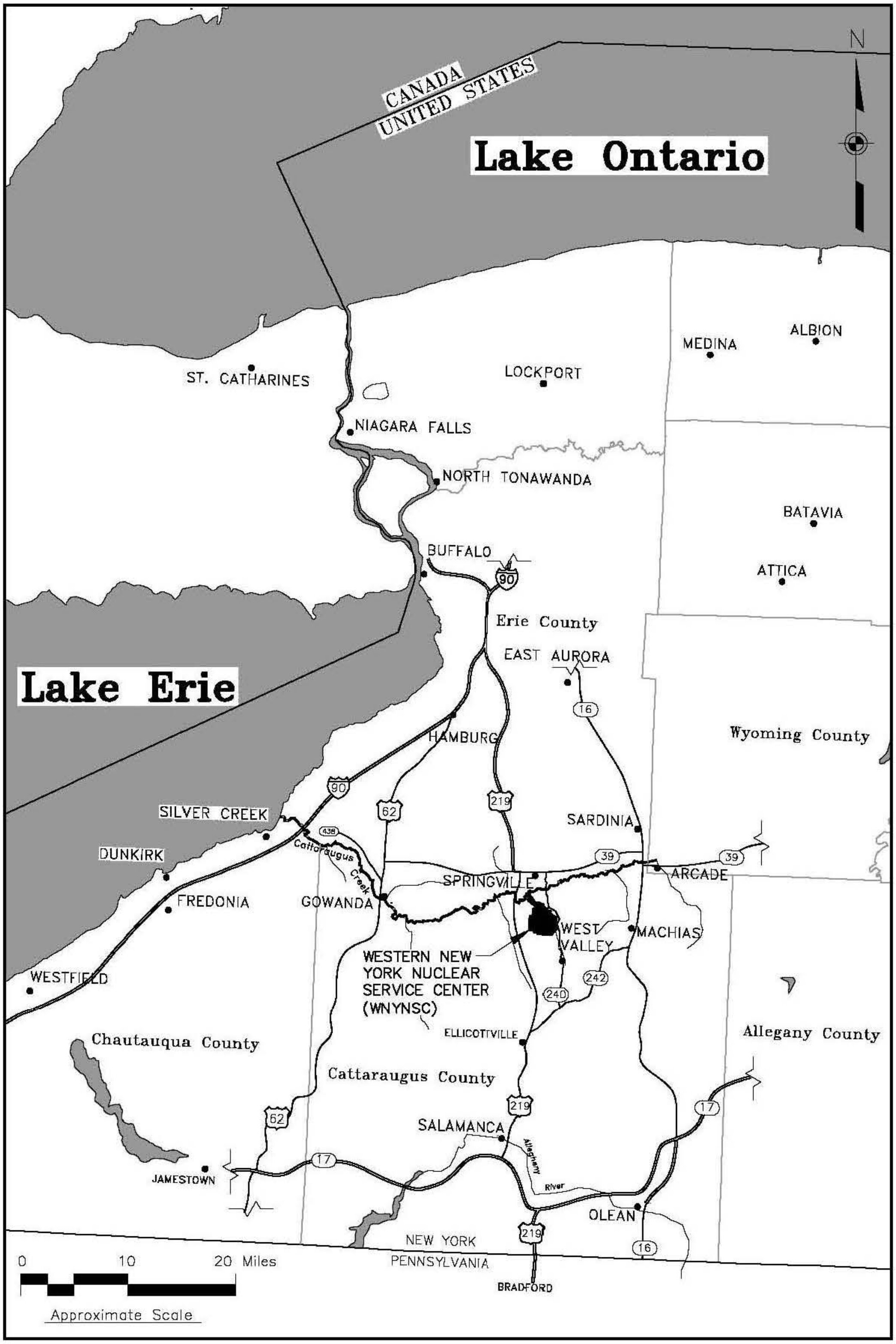



**Figure 3-1. Location of the Center in Western New York**

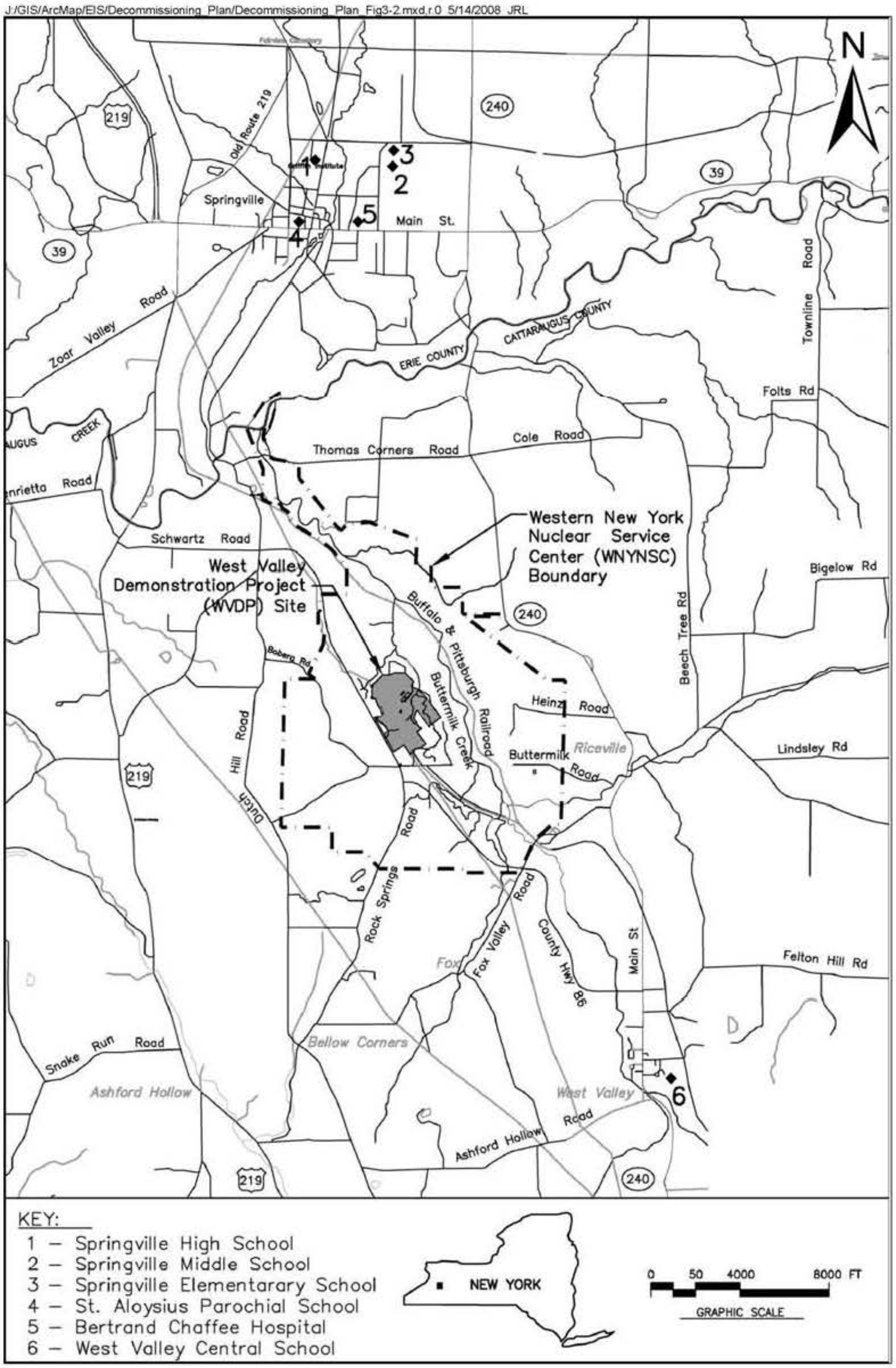



**Figure 3-2. The Center, the WVDP, and the Surrounding Area**

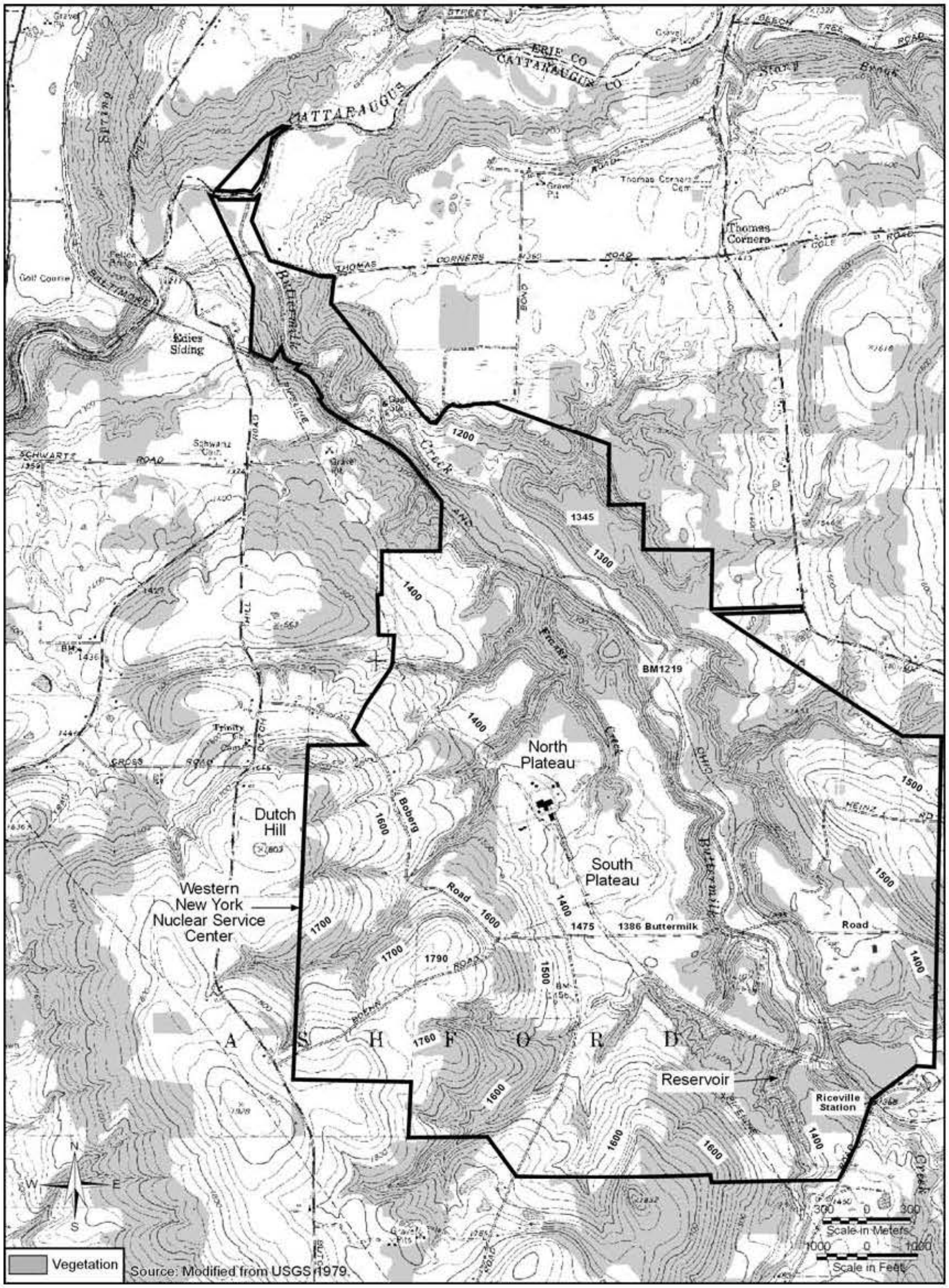



**Figure 3-3. Topography of the Western New York Nuclear Service Center**

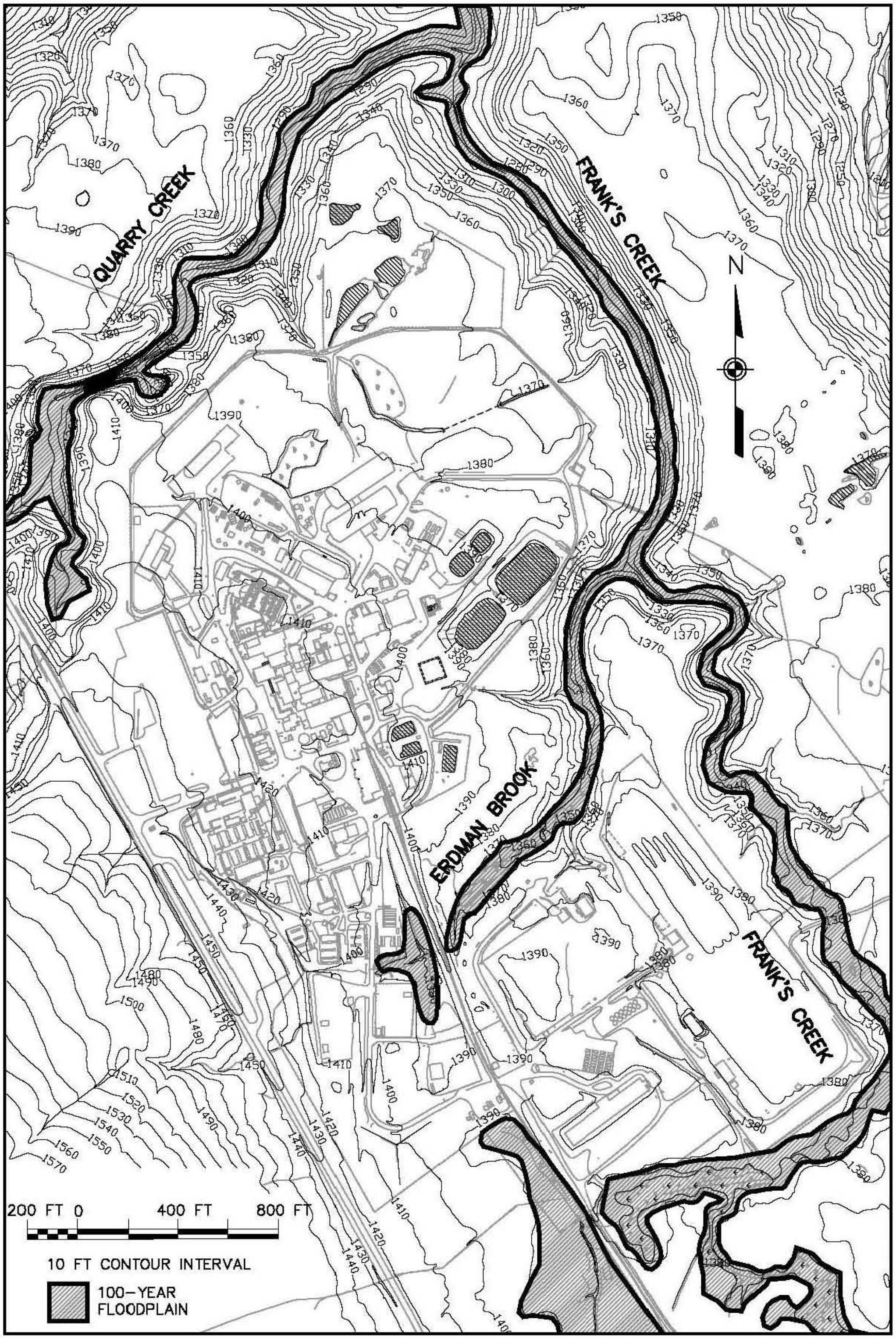



**Figure 3-4. Topography of the Project Premises, Showing 100-Year Floodplain**

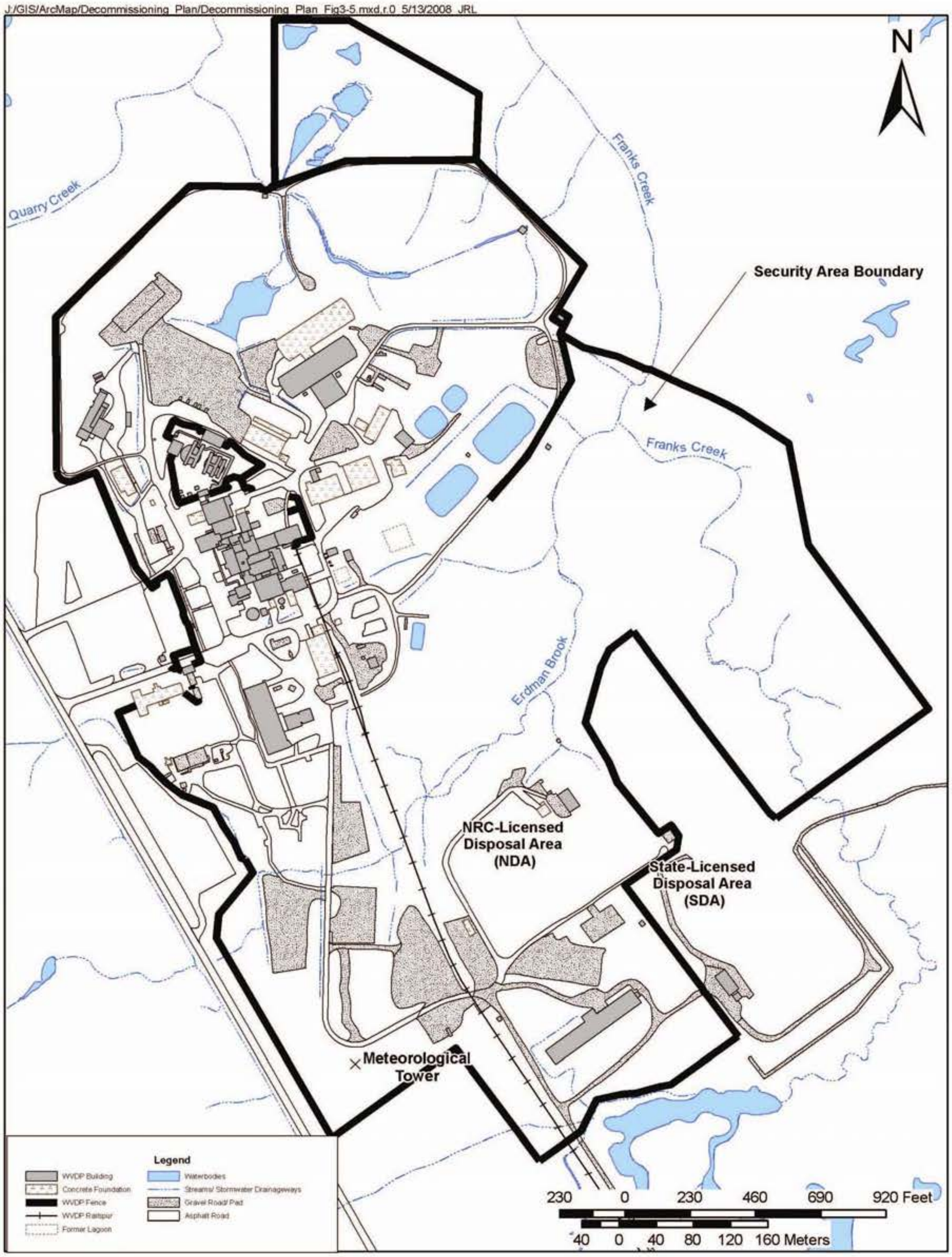



**Figure 3-5. Security Fence Around WVDP Premises Boundary**

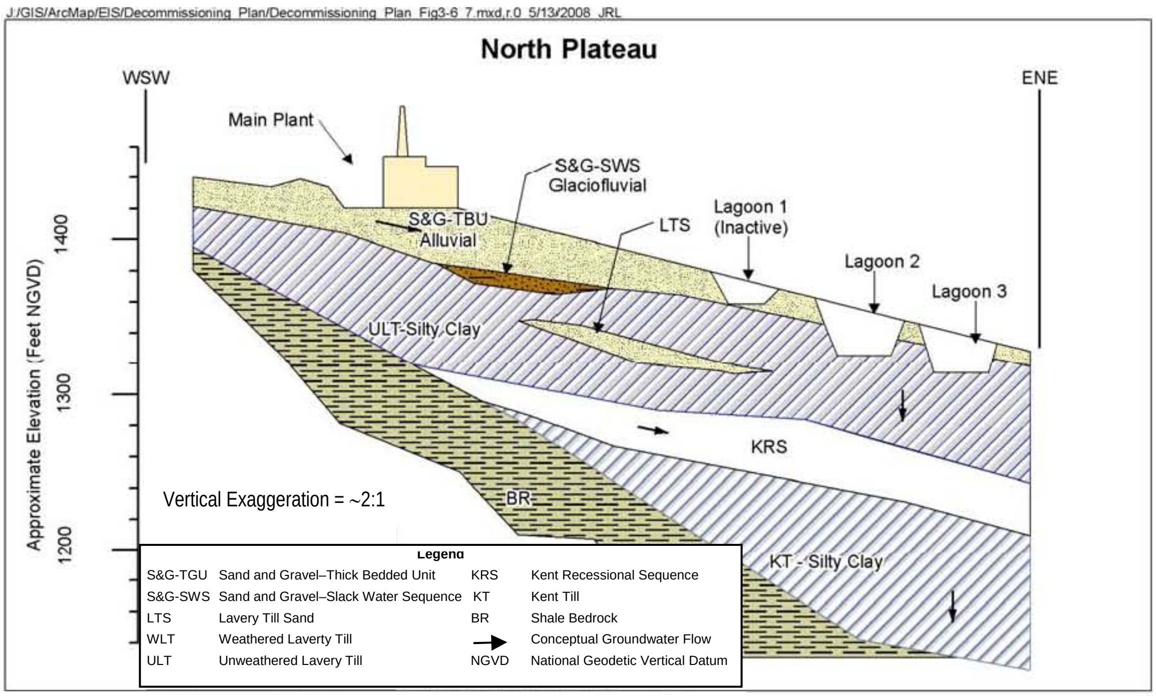



**Figure 3-6. North Plateau Geologic Cross Section**

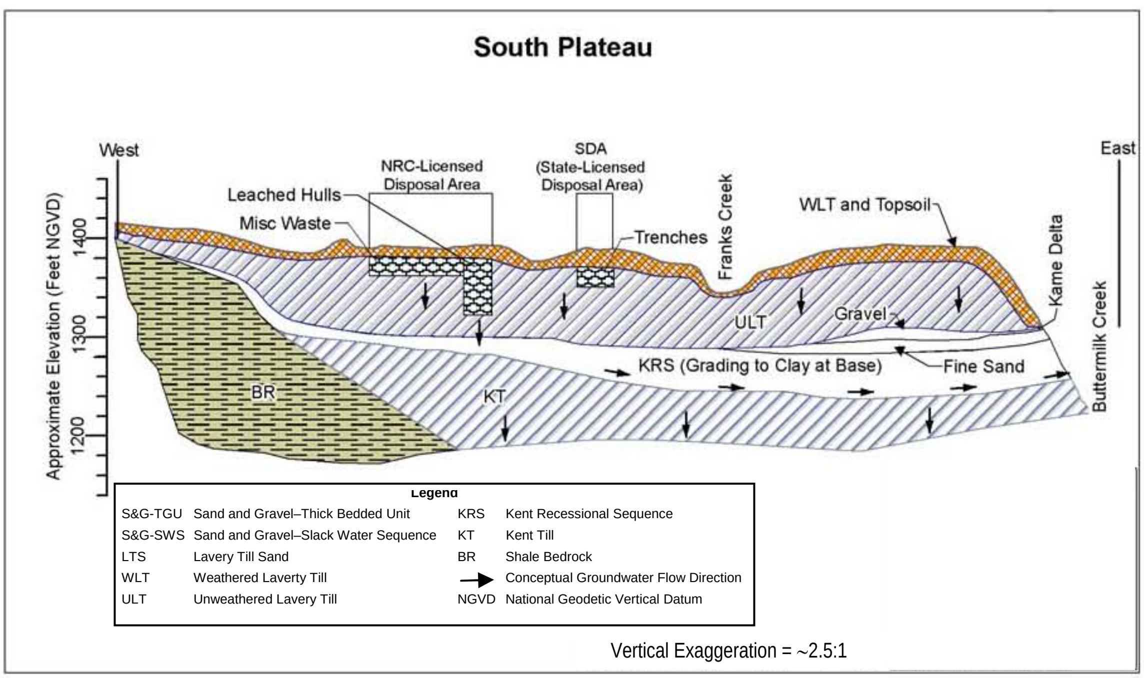



**Figure 3-7. South Plateau Geologic Cross Section**

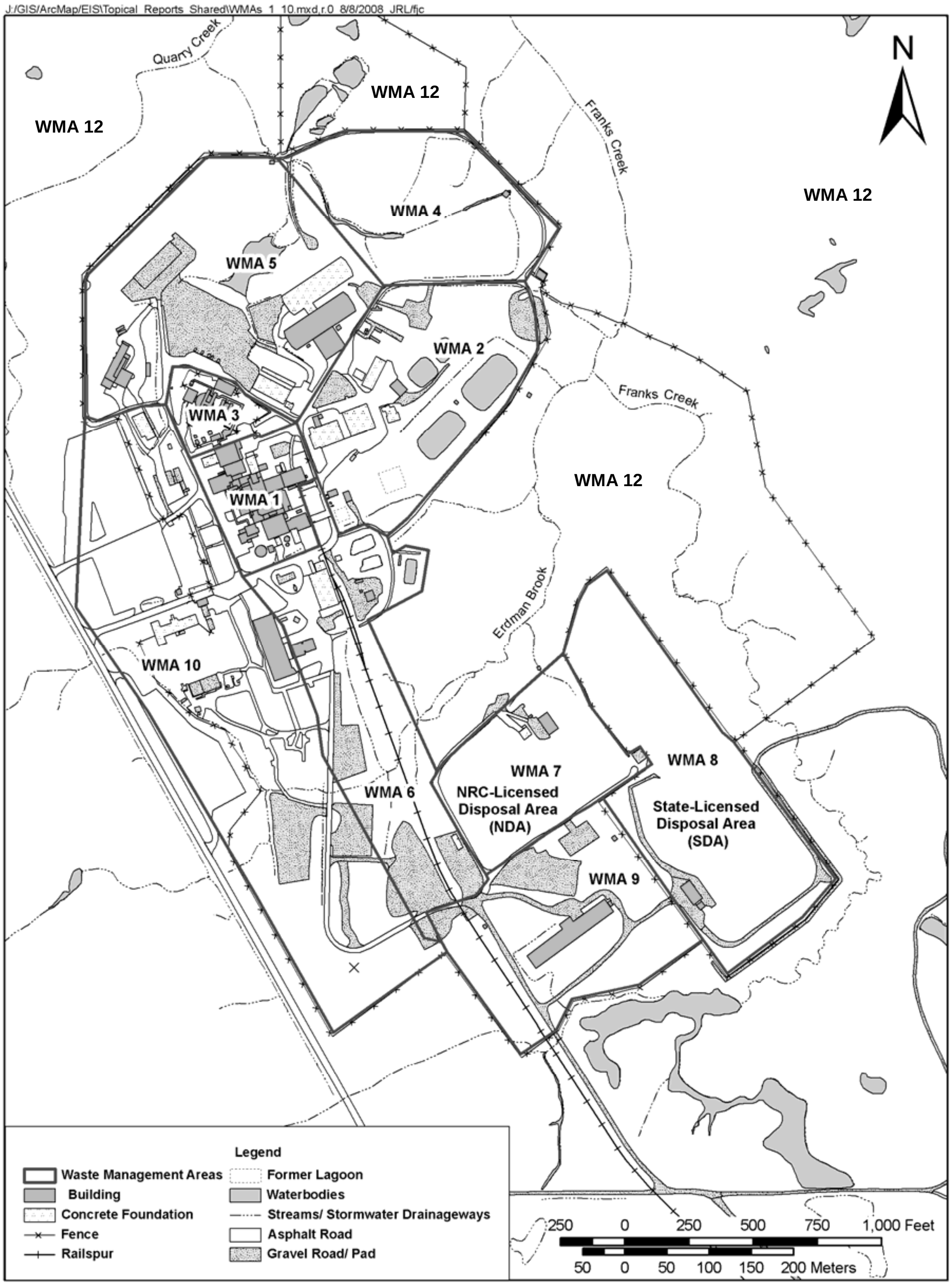



**Figure 3-8. WMAs 1 through 10**

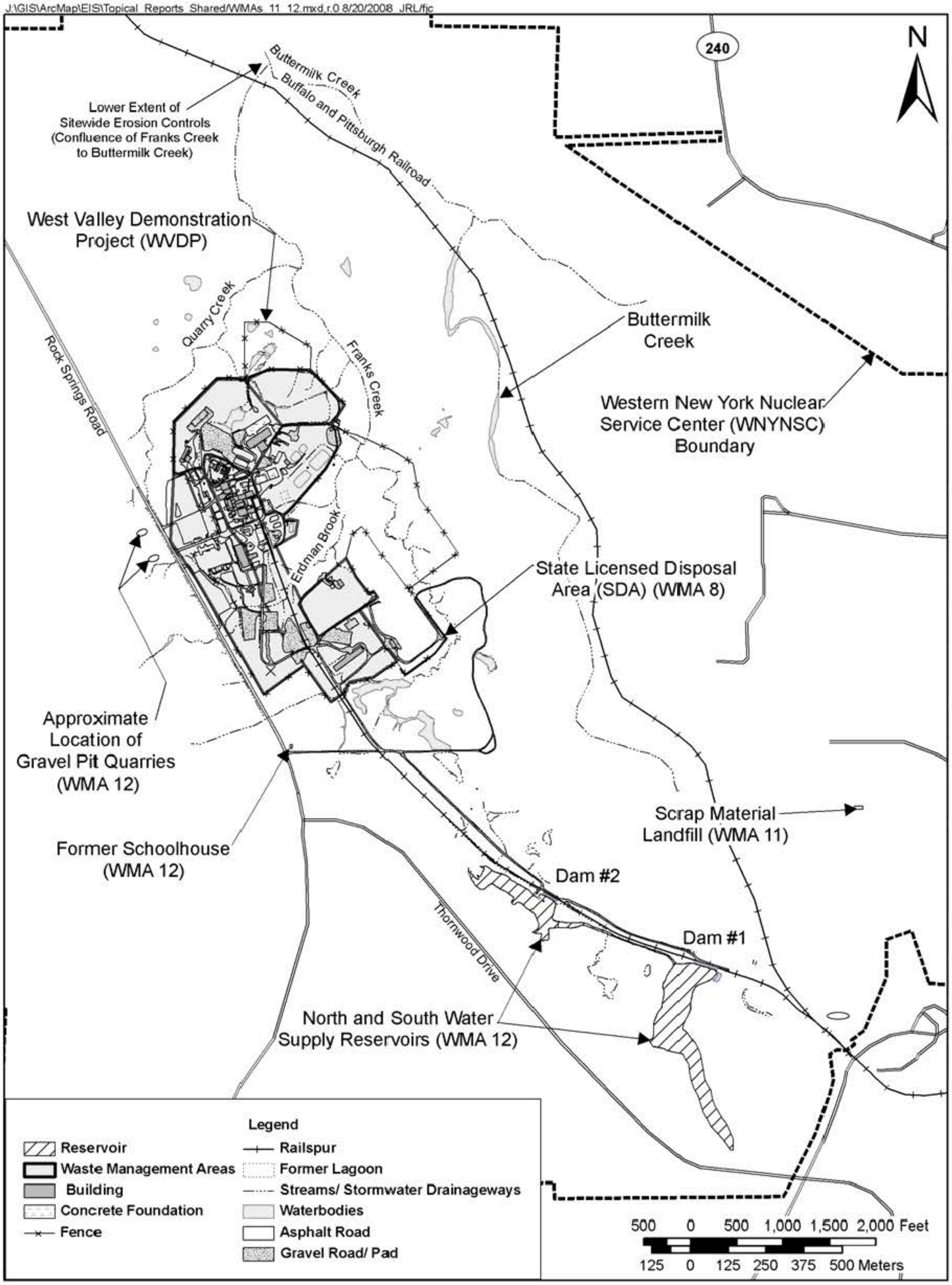



**Figure 3-9. WMAs 11 and 12**

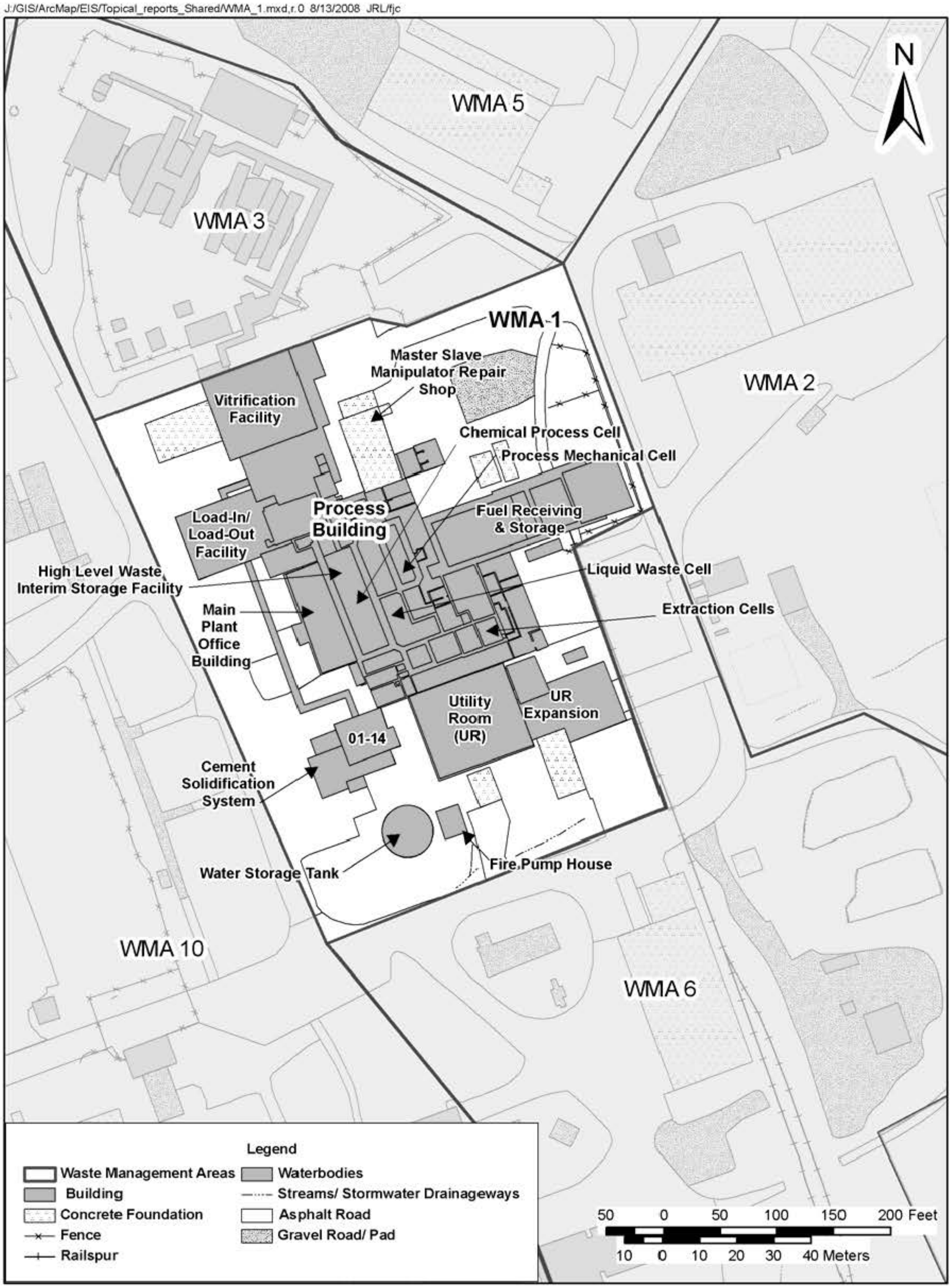



**Figure 3-10. WMA 1.** (The Phase 1 decommissioning activities will include removal of the facilities and the underlying north plateau groundwater plume source area.)

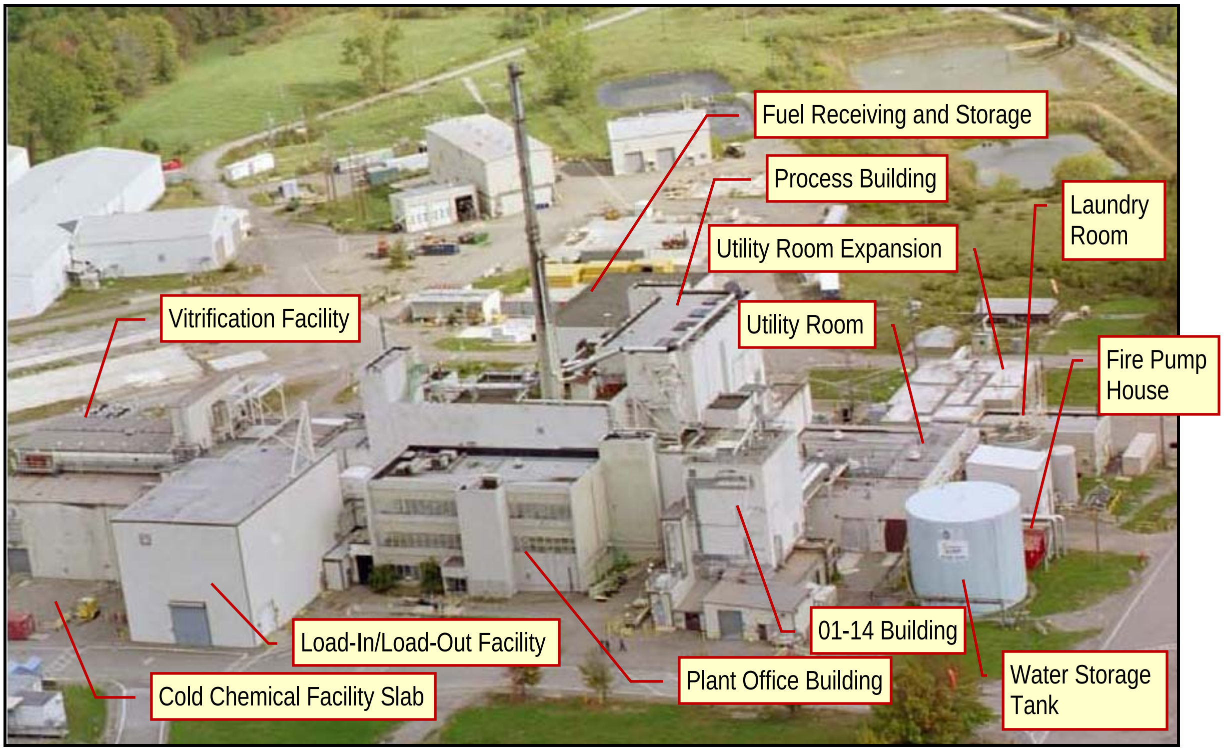



**Figure 3-11. Aerial View of the Process Building Area and Vitrification Facility Area in 2007.** (The Laundry Room will be removed before the Phase 1 of the decommissioning begins.)

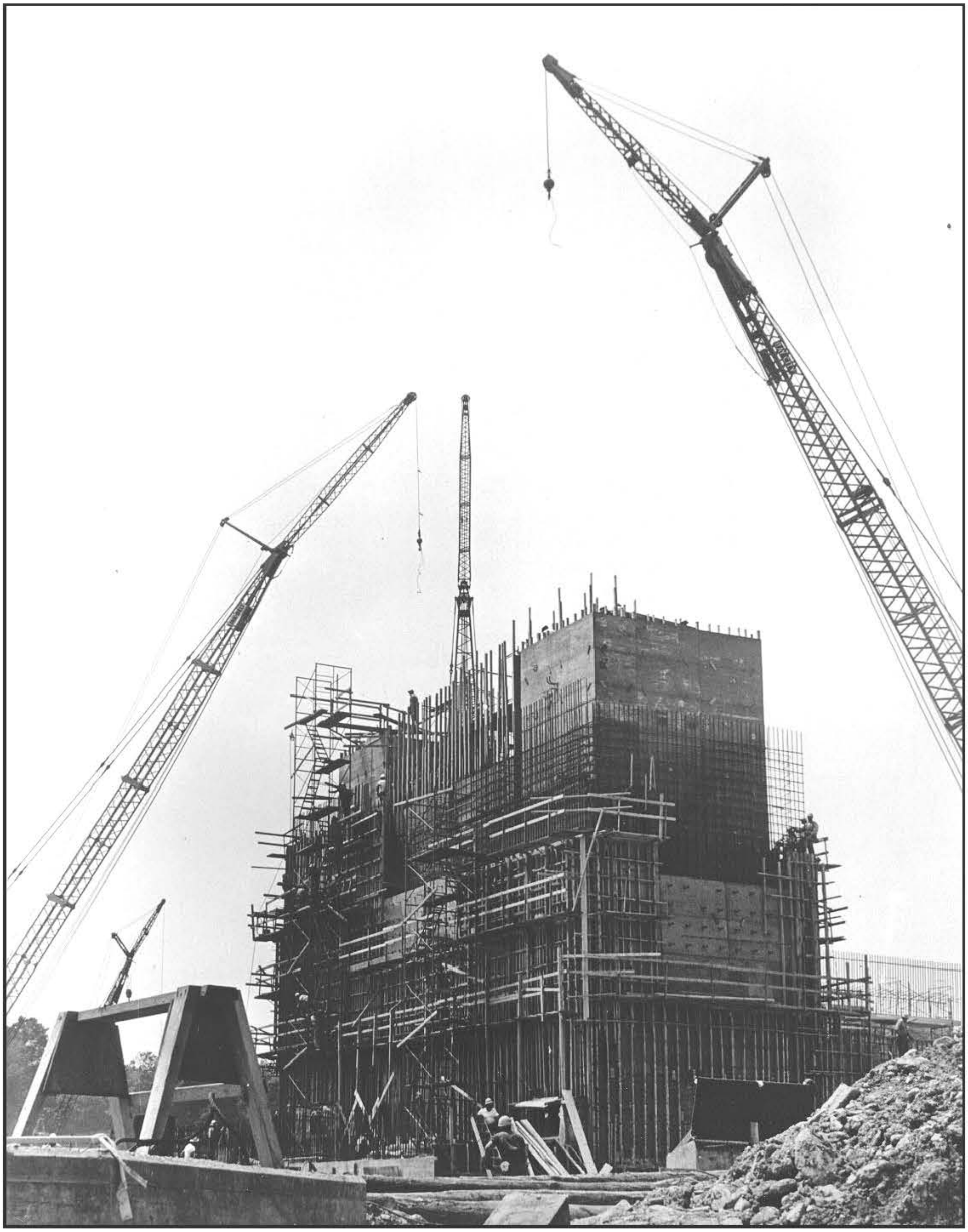
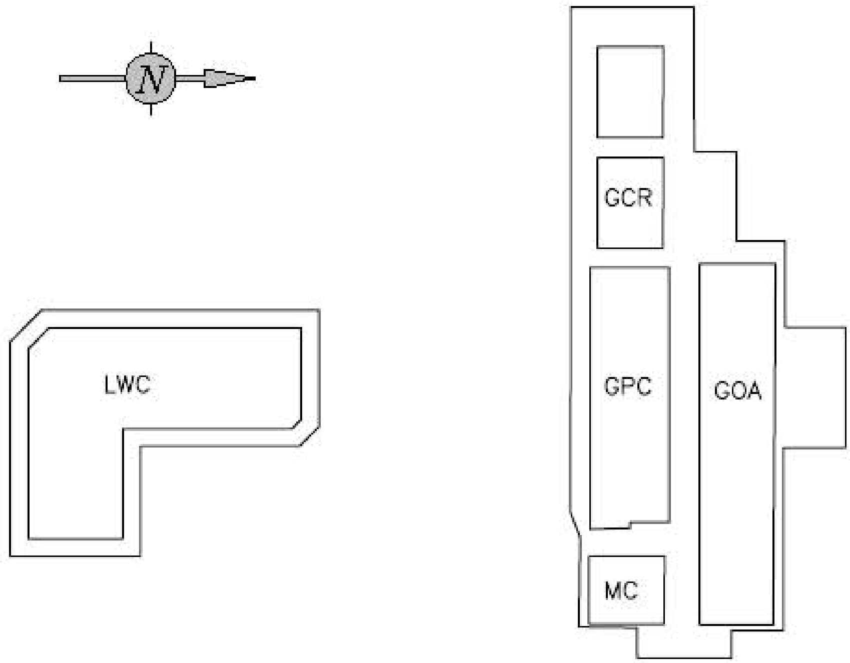


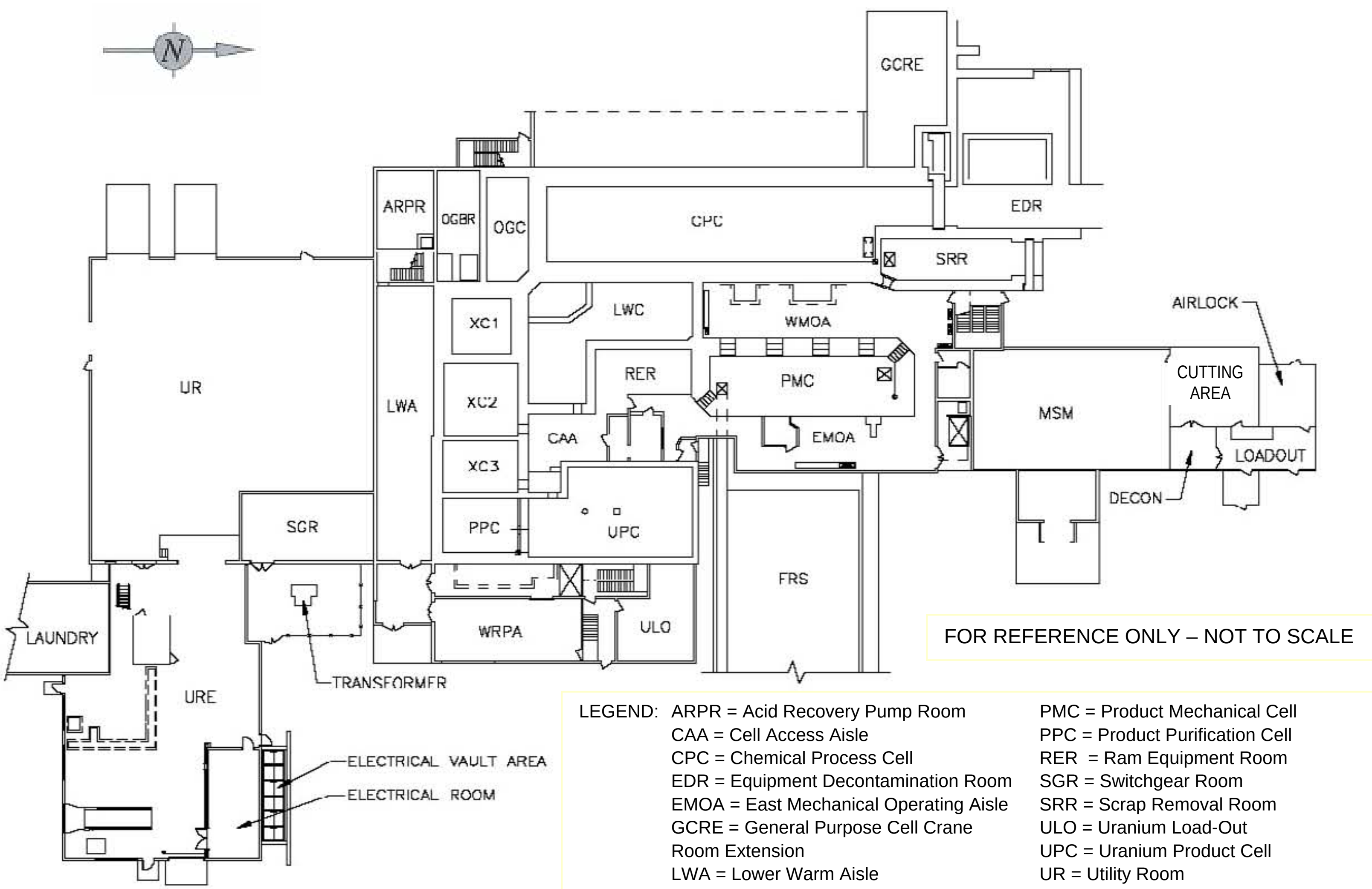
**Figure 3-12. Construction of the Process Building.**




FOR REFERENCE ONLY – NOT TO SCALE

LEGEND: GCR = General Purpose Cell Crane Room
GOA = General Purpose Cell Operating Aisle
GPC = General Purpose Cell
LWC = Liquid Waste Cell
MC = Miniature Cell

**Figure 3-13A. Process Building Layout – Below Grade**




FOR REFERENCE ONLY – NOT TO SCALE

LEGEND:

| | |
|---|---|
| ARPR = Acid Recovery Pump Room | PMC = Product Mechanical Cell |
| CAA = Cell Access Aisle | PPC = Product Purification Cell |
| CPC = Chemical Process Cell | RER = Ram Equipment Room |
| EDR = Equipment Decontamination Room | SGR = Switchgear Room |
| EMOA = East Mechanical Operating Aisle | SRR = Scrap Removal Room |
| GCRE = General Purpose Cell Crane Room Extension | ULO = Uranium Load-Out |
| LWA = Lower Warm Aisle | UPC = Uranium Product Cell |
| LWC = Liquid Waste Cell | UR = Utility Room |
| MSM = Manipulator Repair Shop | URE = Utility Room Expansion |
| OGBR = Off-Gas Blower Room | WMOA = West Mechanical Operating Aisle |
| OGC = Off-Gas Cell | WRPA = Waste Reduction & Packaging Area |
| | XC = Extraction Cell |

**Figure 3-13B. Process Building Layout at 100-Foot Elevation**

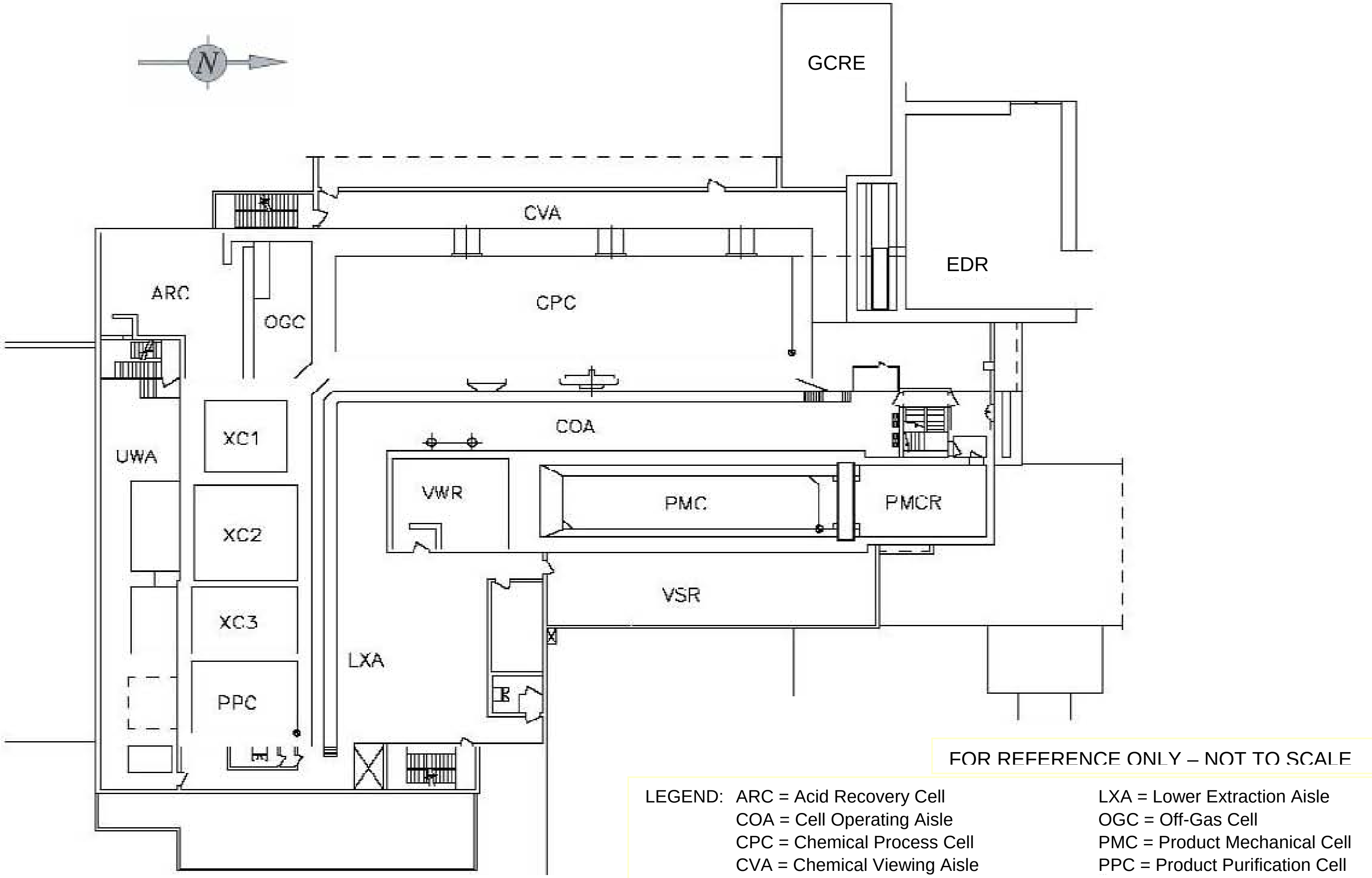
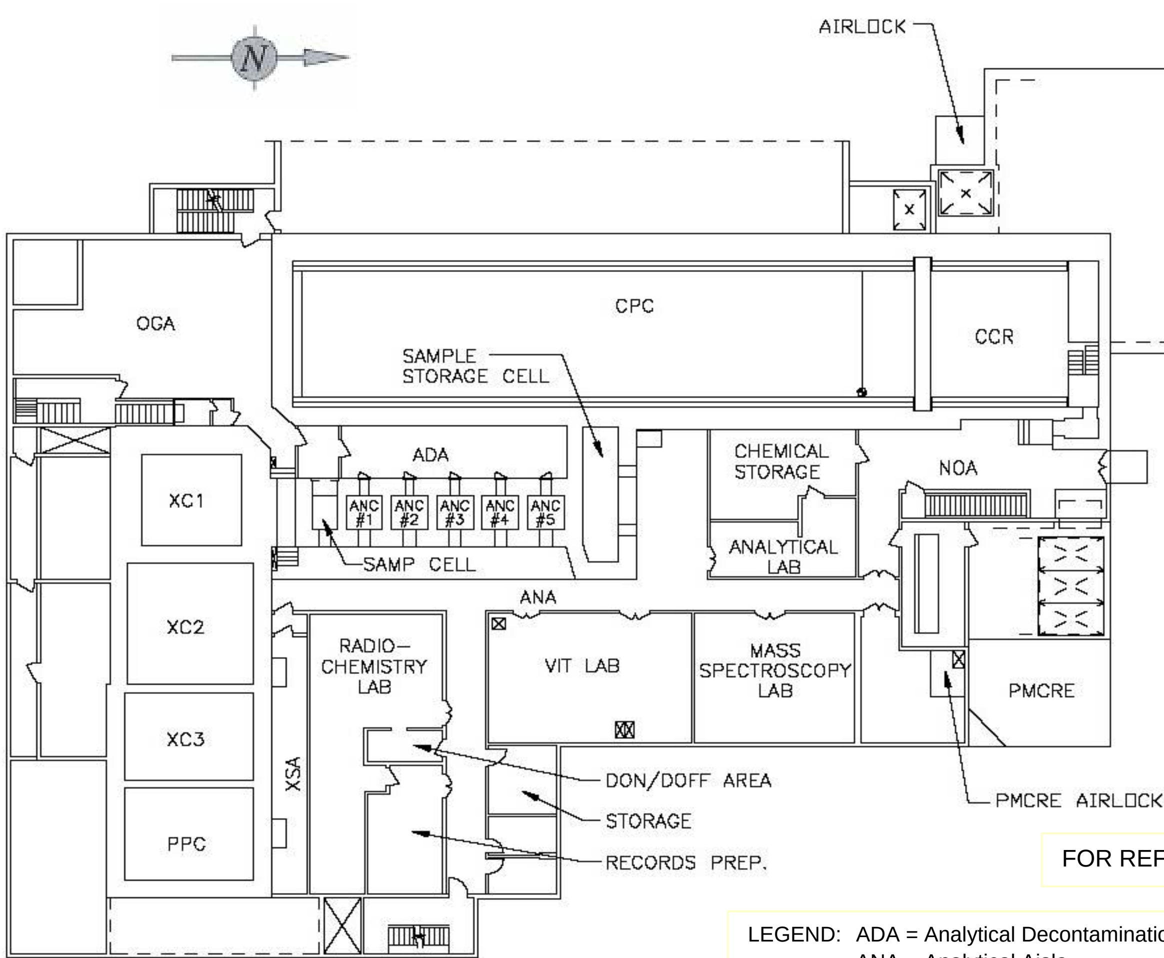



**Figure 3-13C. Process Building Layout at 114.5-Foot Elevation**

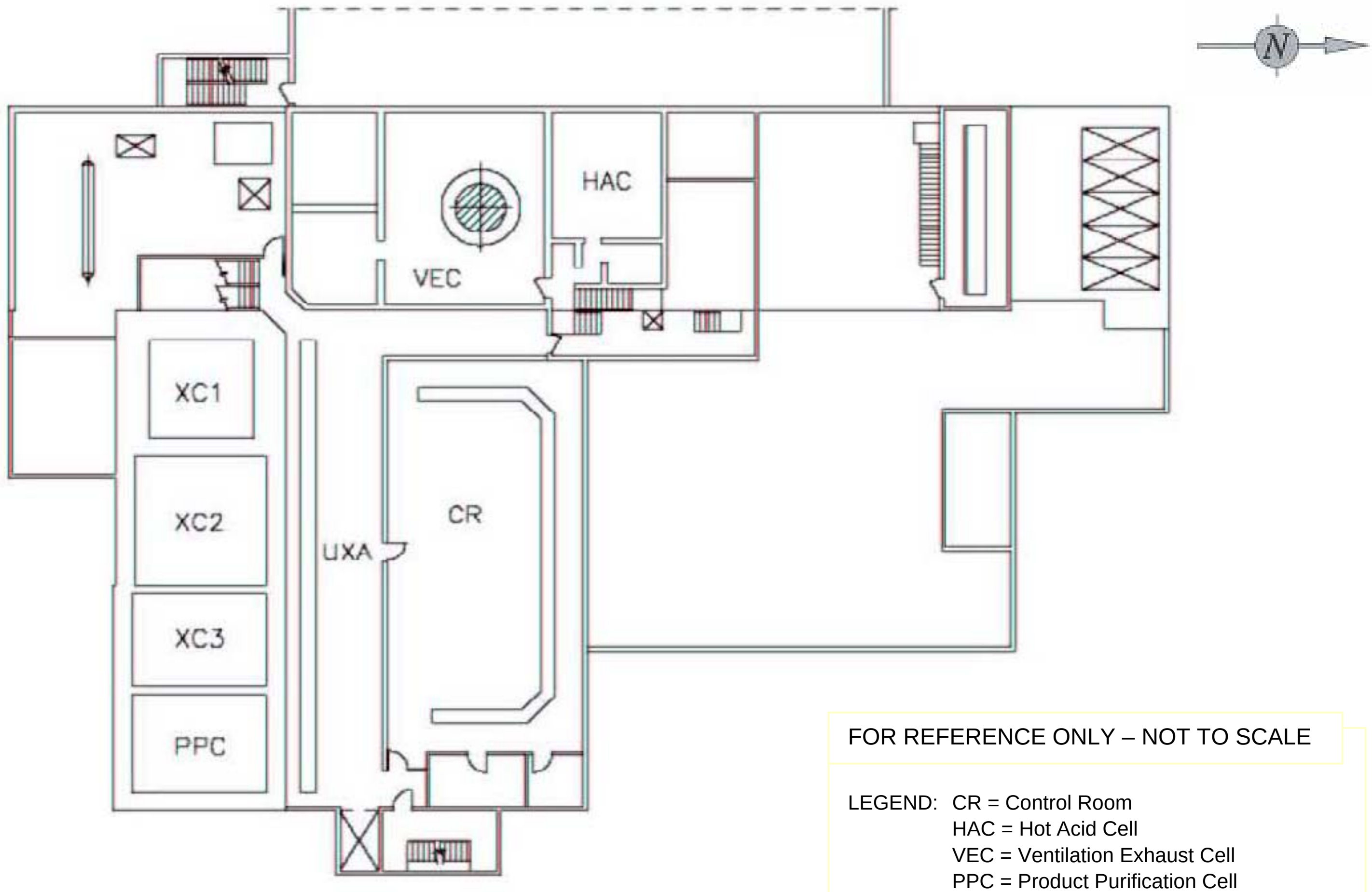
LEGEND:
- ARC = Acid Recovery Cell
- COA = Cell Operating Aisle
- CPC = Chemical Process Cell
- CVA = Chemical Viewing Aisle
- EDR = Equipment Decontamination Room
- GCRE = General Purpose Cell Crane Room Extension
- LXA = Lower Extraction Aisle
- OGC = Off-Gas Cell
- PMC = Product Mechanical Cell
- PPC = Product Purification Cell
- PMCR = PMC Crane Room
- VWR = Ventilation Wash Room
- XC = Extraction Cell




FOR REFERENCE ONLY – NOT TO SCALE

LEGEND:
ADA = Analytical Decontamination Aisle
ANA = Analytical Aisle
ANC = Analytical Sample Cell
CCR = CPC Crane Room
CPC = Chemical Process Cell
NOA = North Operating Aisle
OGA = Off-Gas-Acid Recovery Aisle
PMCRE = Process Mechanical Cell Crane Room Enclosure
PPC = Product Purification Cell
XC = Extraction Cell

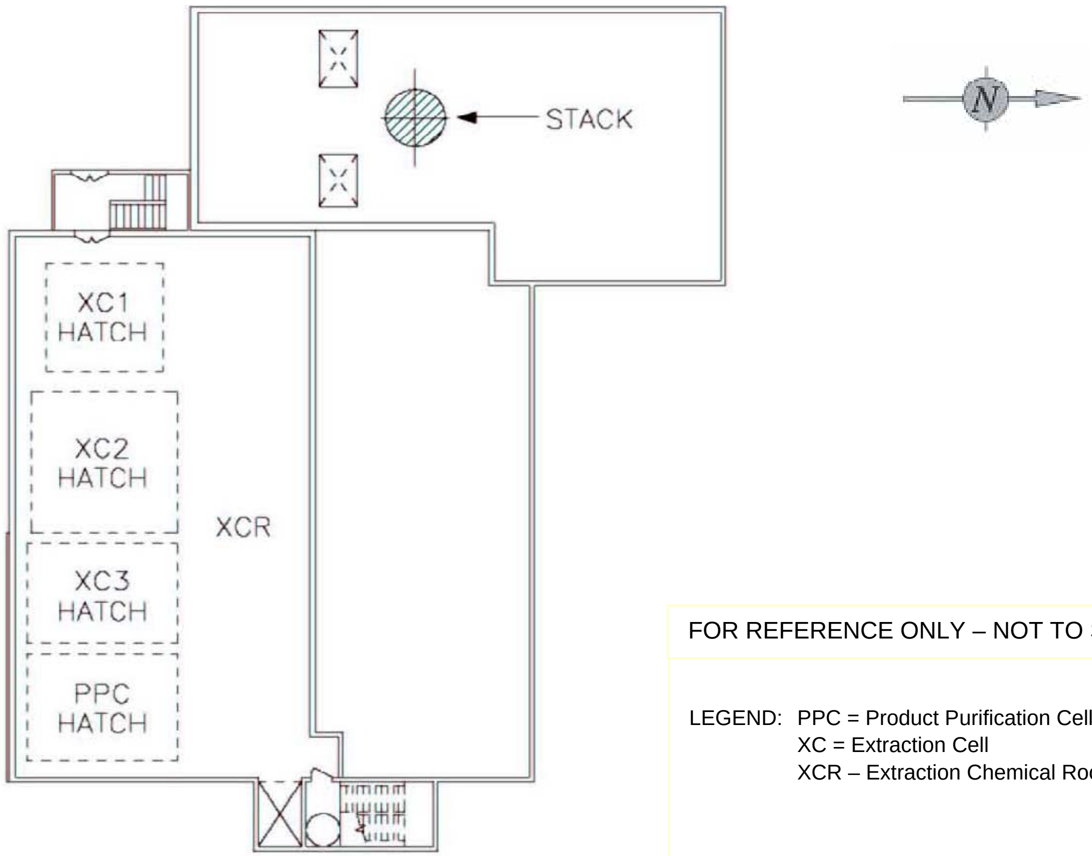
**Figure 3-13D. Process Building Layout at 131-Foot Elevation**




FOR REFERENCE ONLY – NOT TO SCALE

LEGEND: CR = Control Room
HAC = Hot Acid Cell
VEC = Ventilation Exhaust Cell
PPC = Product Purification Cell
UXA = Upper Extraction Aisle
VEC = Ventilation Exhaust Cell
XC = Extraction Cell

**Figure 3-13E. Process Building Layout at 144-Foot Elevation**




FOR REFERENCE ONLY – NOT TO SCALE

LEGEND: PPC = Product Purification Cell
XC = Extraction Cell
XCR – Extraction Chemical Room

**Figure 3-13F. Process Building Layout at 160-Foot Elevation**

**Figure 3-14. West Side of the Process Building.** (The building with windows is actually the Plant Office Building. The plant part of the Process Building is behind the Office Building

**Figure 3-15. Fuel Receiving and Storage Area.** (This facility is located on the east side of the Process Building.)

**Figure 3-16. HLW Canisters Stored in the HLW Interim Storage Area**

**Figure 3-17. Conditions in the General Purpose Cell in 1999.** (These were the conditions before the beginning of cleanup in connection with deactivation.)

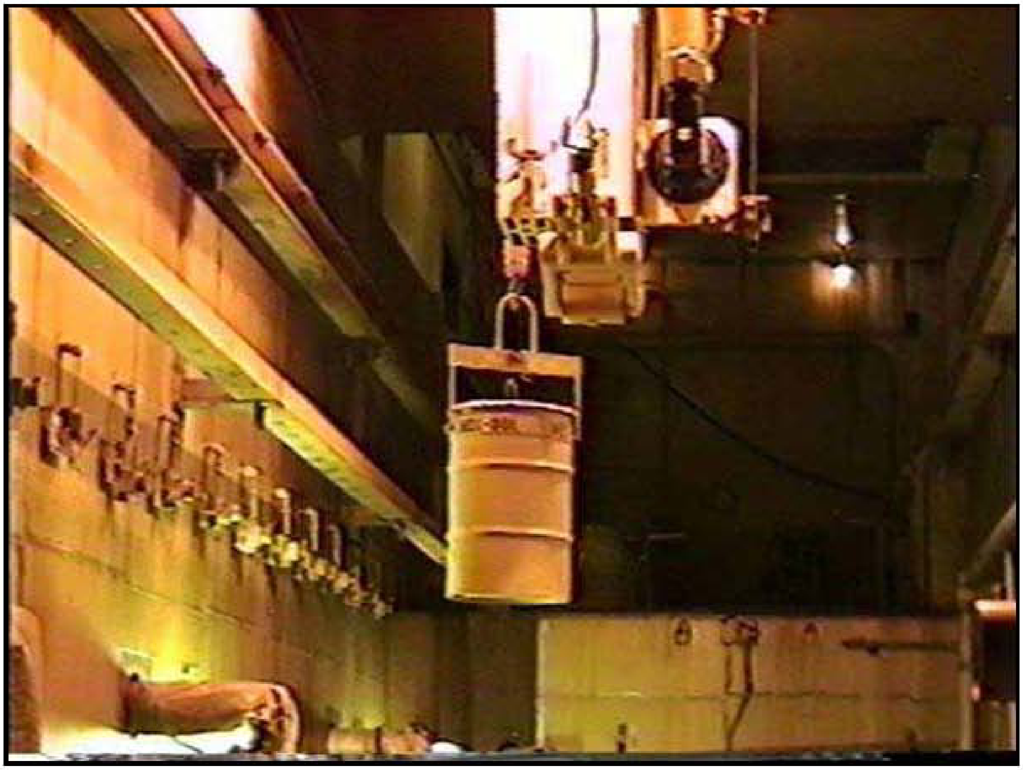


**Figure 3-18. Process Mechanical Cell During Deactivation**

**Figure 3-19. Extraction Cell 3** (After removal of processing equipment and before installation of the WVDP Liquid Waste Treatment System Equipment)

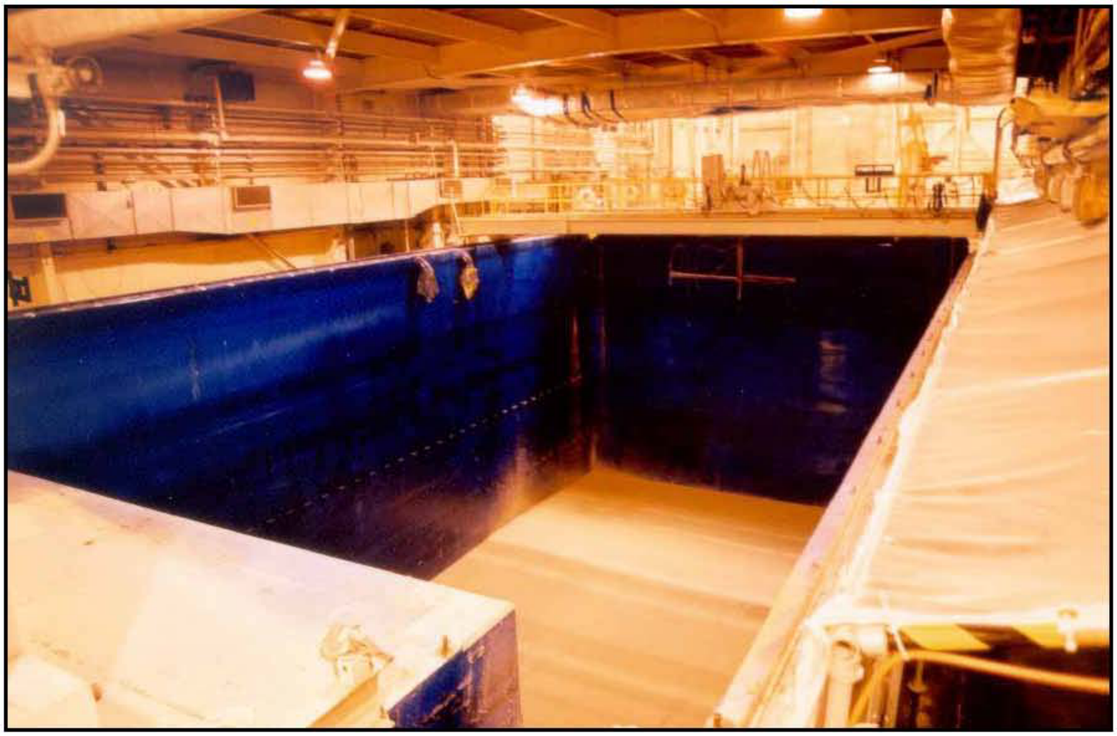


**Figure 3-20. The Spent Fuel Pool After Deactivation**

**Figure 3-21. Equipment Decontamination Room Before Cleanup**

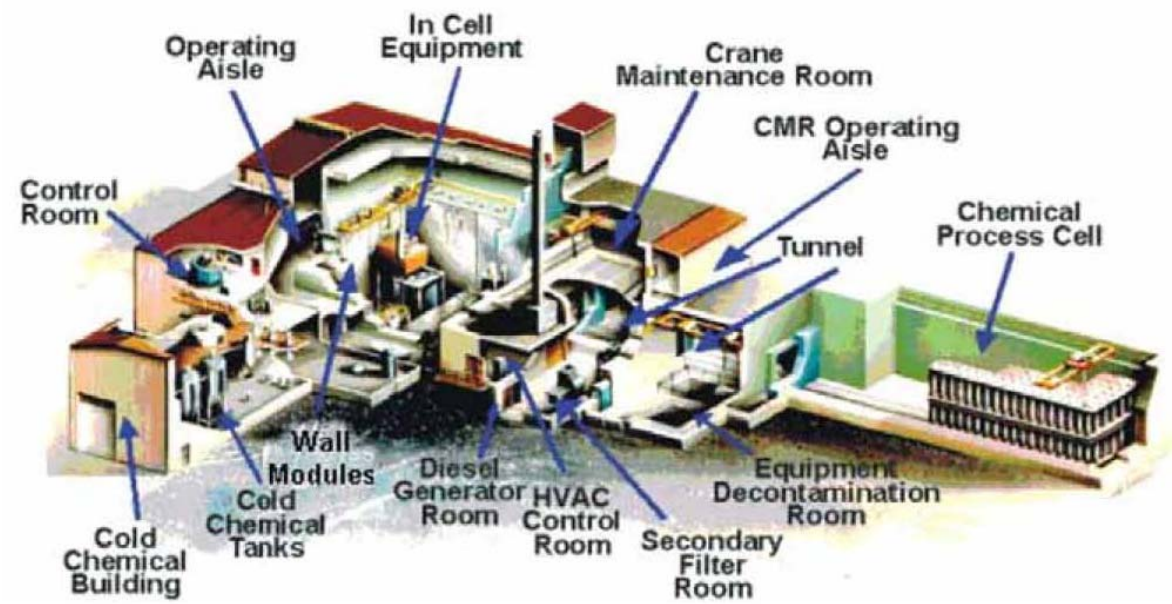



**Figure 3-22. Vitrification Facility General Arrangement**

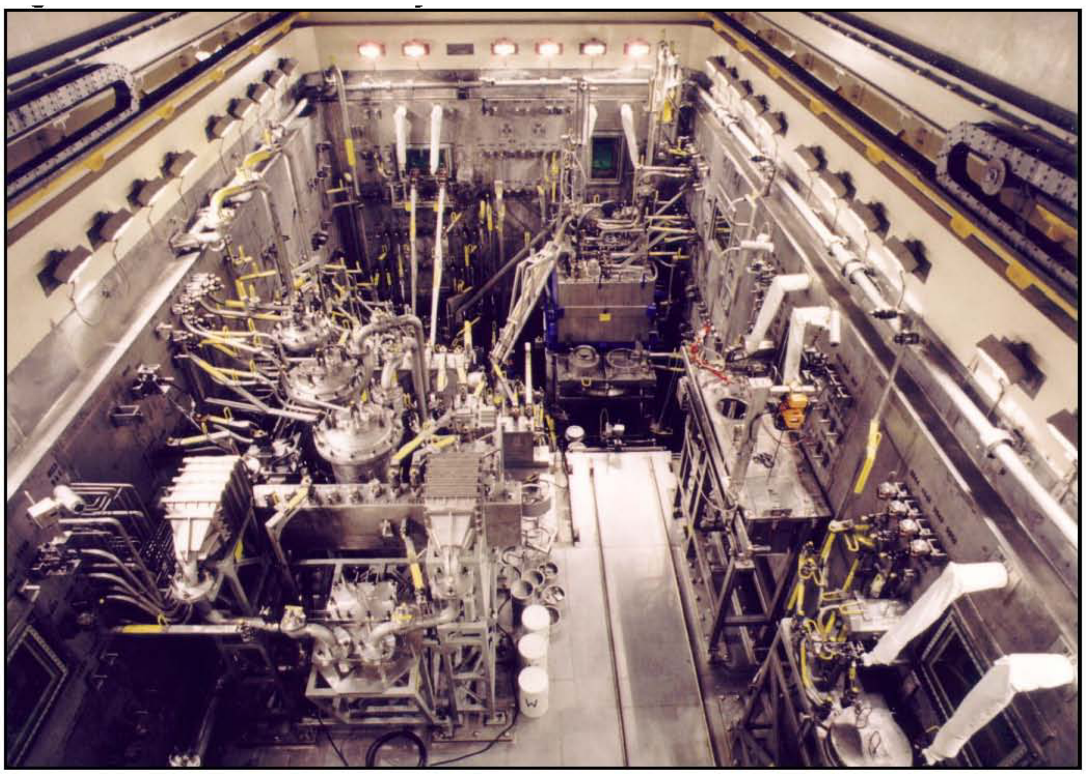


**Figure 3-23. Vitrification Cell at Time of Startup**

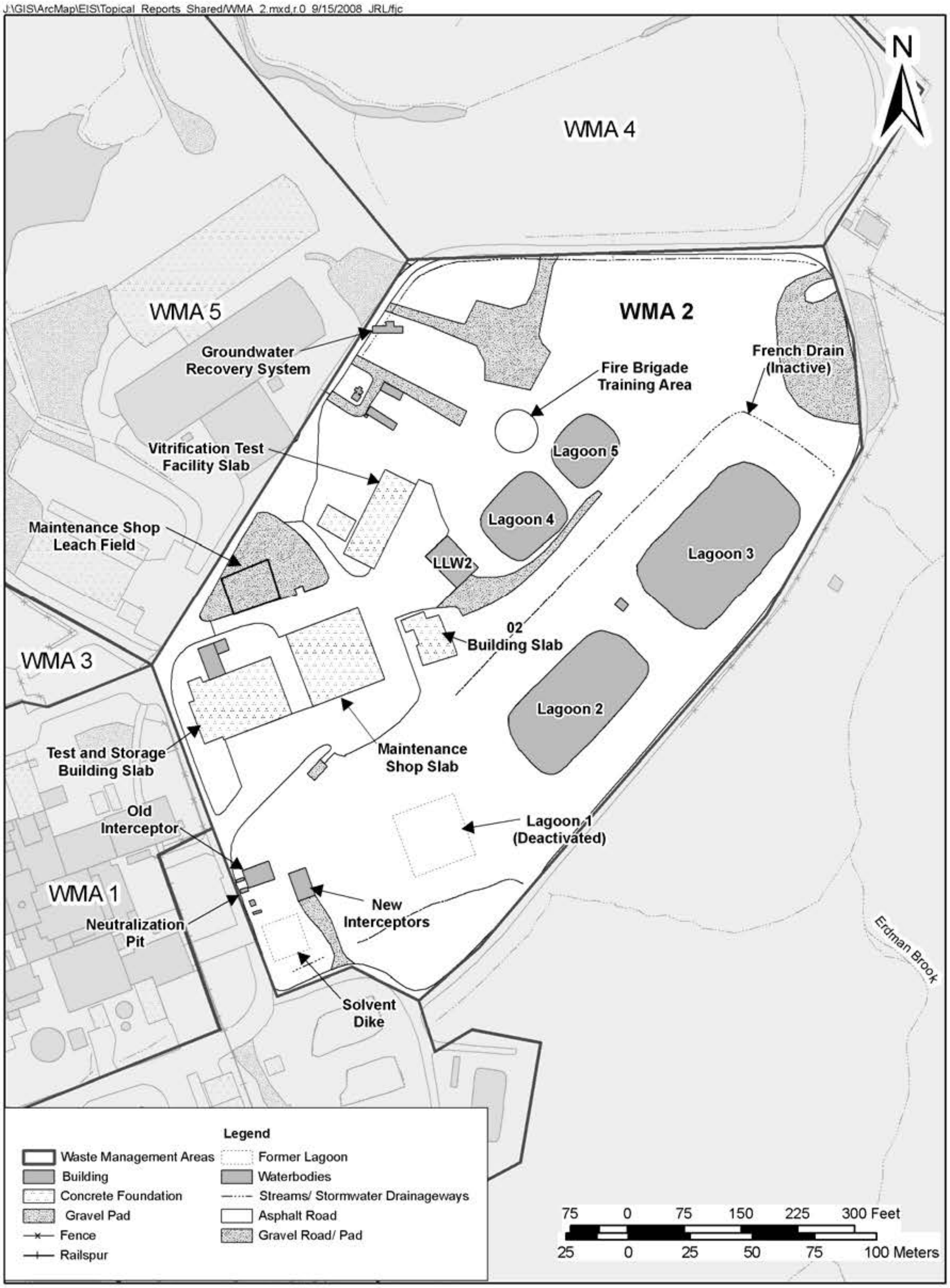



**Figure 3-24. WMA 2.** (The facilities to be removed during Phase 1 decommissioning activities include the Neutralization Pit, Interceptors, Lagoons, and remaining slabs.)


**Figure 3-25. The Low-Level Waste Treatment Facility. (**This photo shows the site in 1982, looking toward the southwest.)

**Figure 3-26. The LLW2 Building that Replaced the O2 Building**

**Figure 3-27. The Lagoon 1 Area.** (Radioactive debris was placed in Lagoon 1 when it was closed in 1985.)

**Figure 3-28. The New Interceptors.** (These are twin stainless-steel lined concrete holding tanks.)

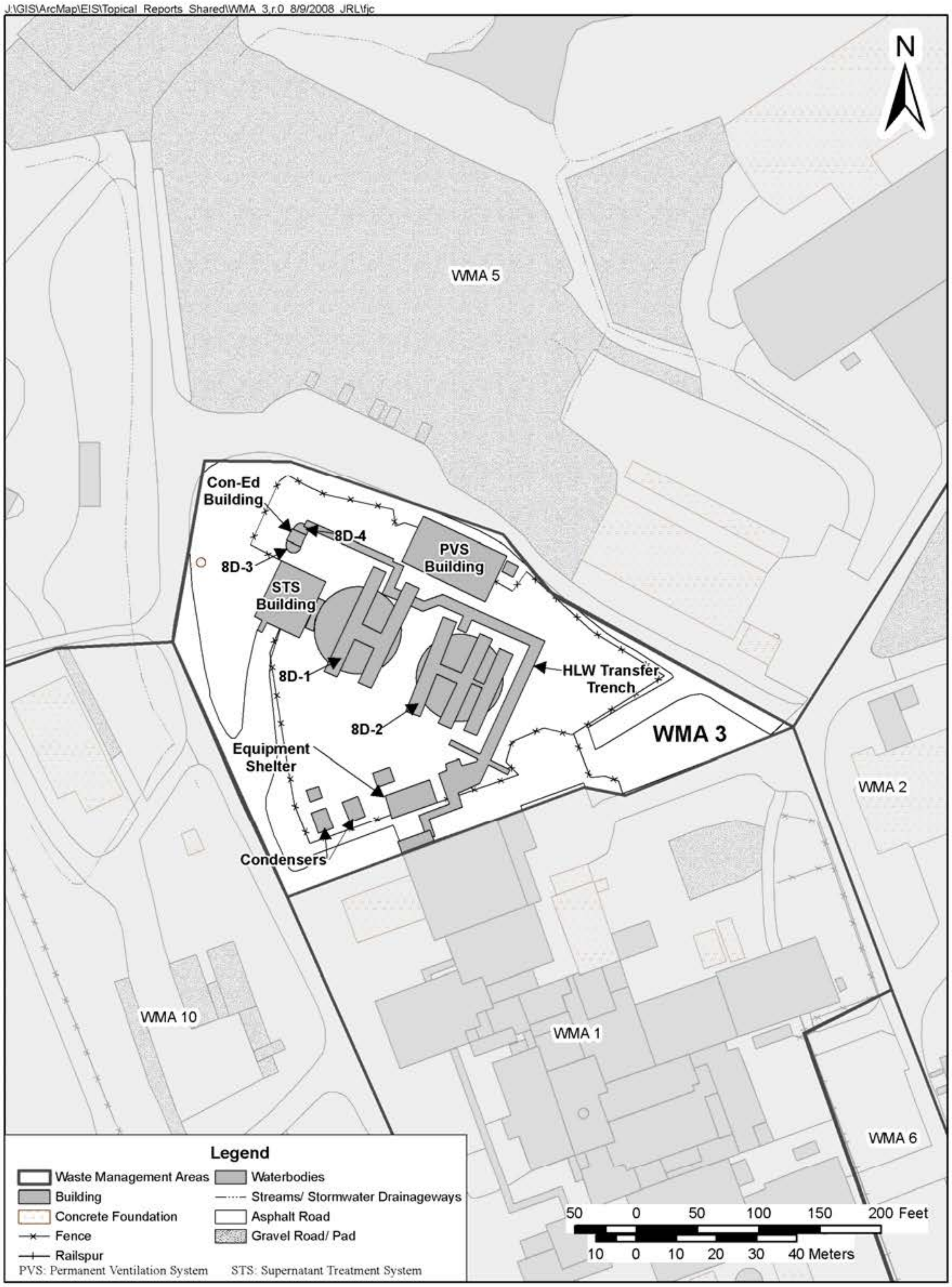



**Figure 3-29. WMA 3. (**Facilities to be removed during Phase 1 decommissioning activities include the Equipment Shelter, the condensers, the piping in the HLW transfer trench, and the Con-Ed Building.)

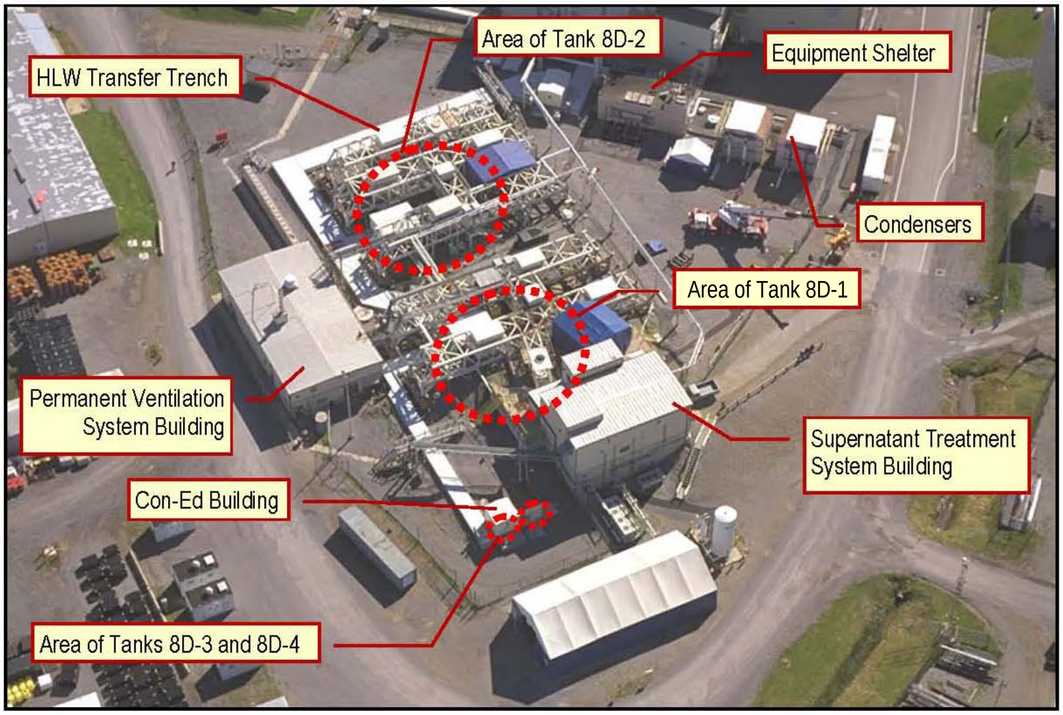



**Figure 3-30. Aerial View of WMA 3 Area**

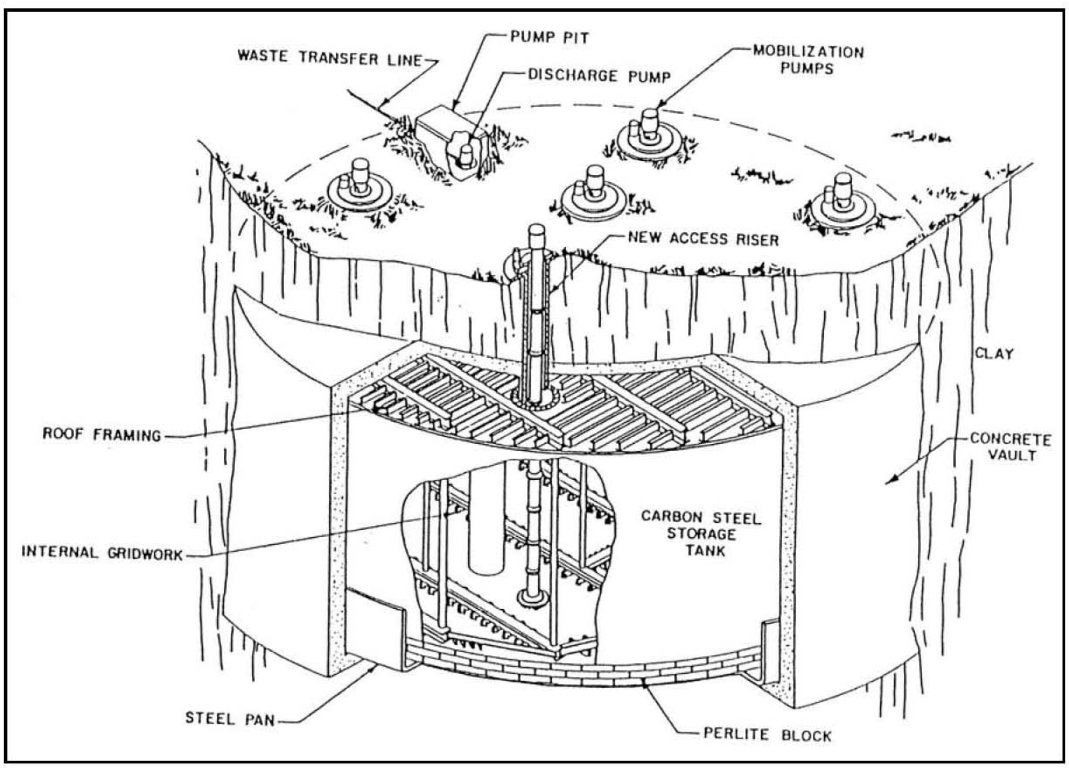
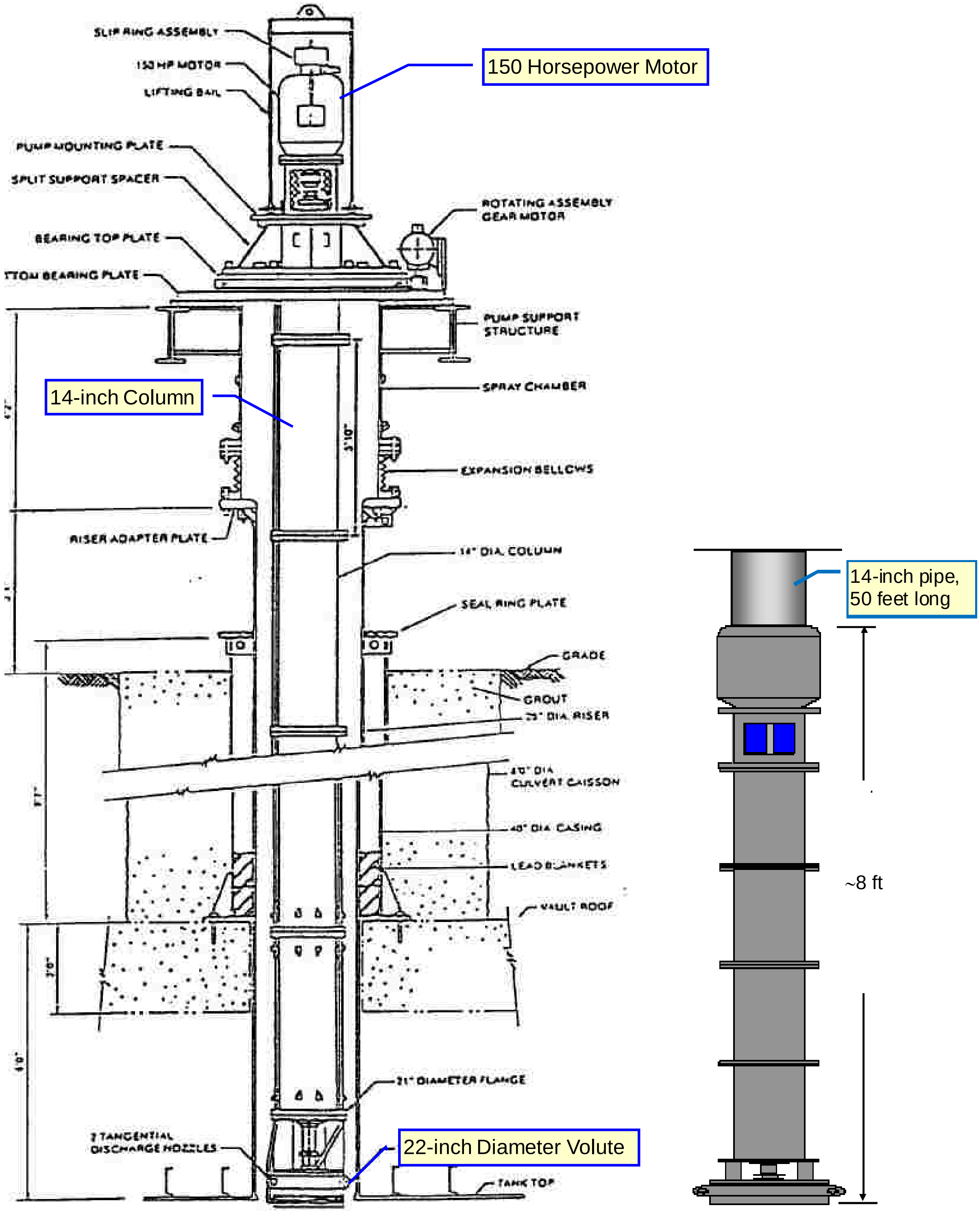



**Figure 3-31. Cutaway View of 750-Gallon Underground Waste Tank**




**Transfer Pump** **Mobilization Pump**

**Figure 3-32. HLW Transfer and Mobilization Pumps**

**Figure 3-33. HLW Transfer Trench Under Construction**

**Figure 3-34. Typical HLW Pump Pit**

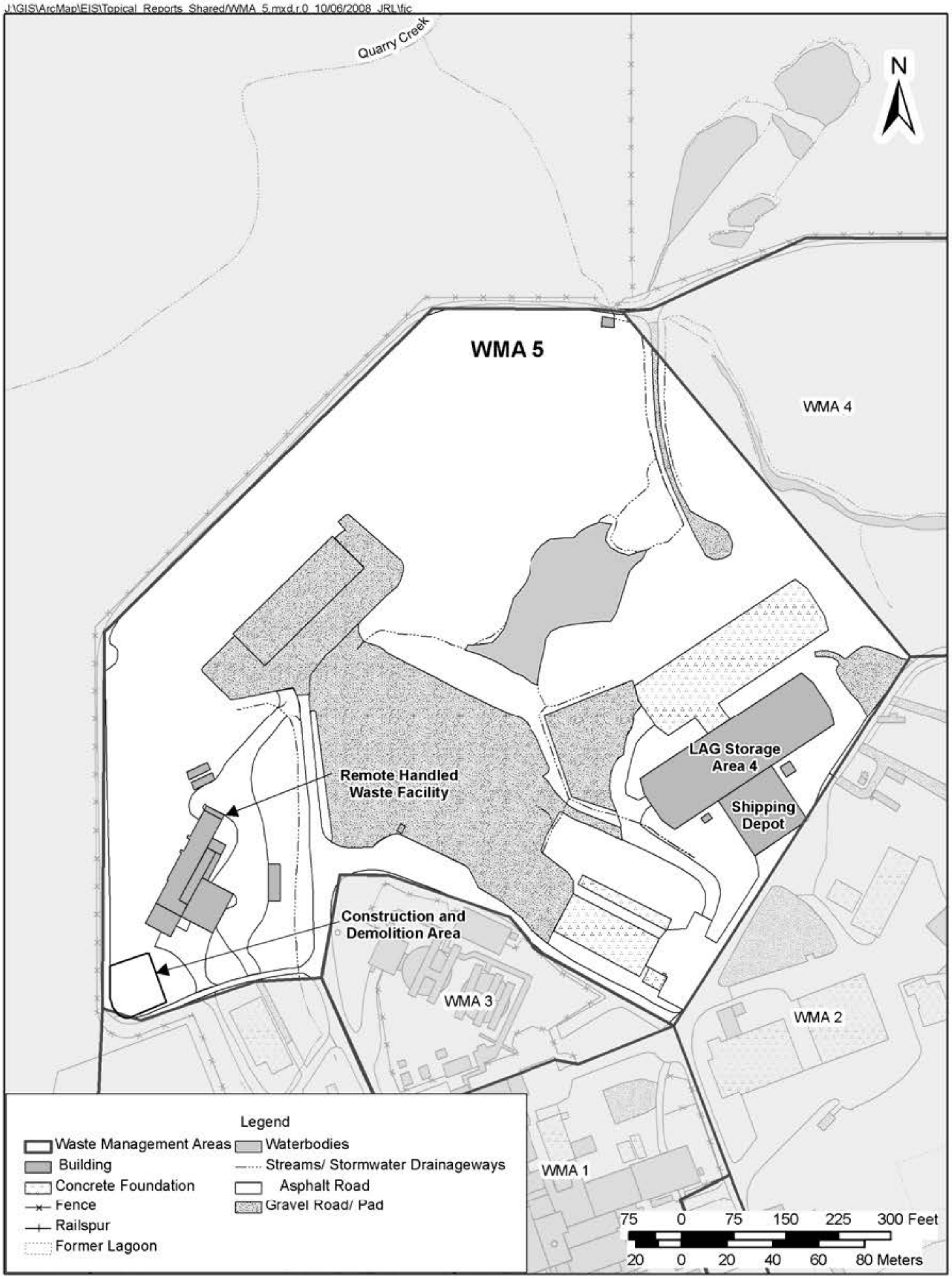



**Figure 3-35. WMA 5.** (Facilities to be removed during Phase 1 decommissioning include the Remote-Handled Waste Facility, Lag Storage Addition 4 and its Shipping Depot.)

**Figure 3-36. The Remote-Handled Waste Facility. (**Placed into service in 2004, this new building may contain significant contamination at the time it is removed.)

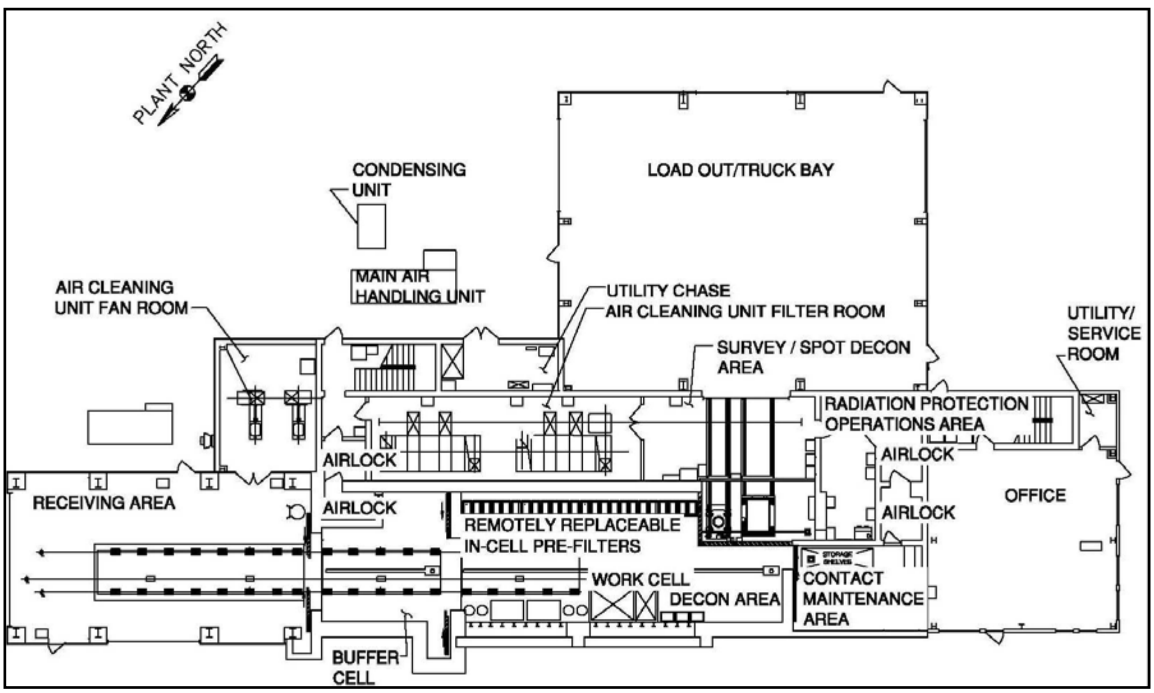



**Figure 3-37. The Remote-Handled Waste Facility First Floor Layout.**

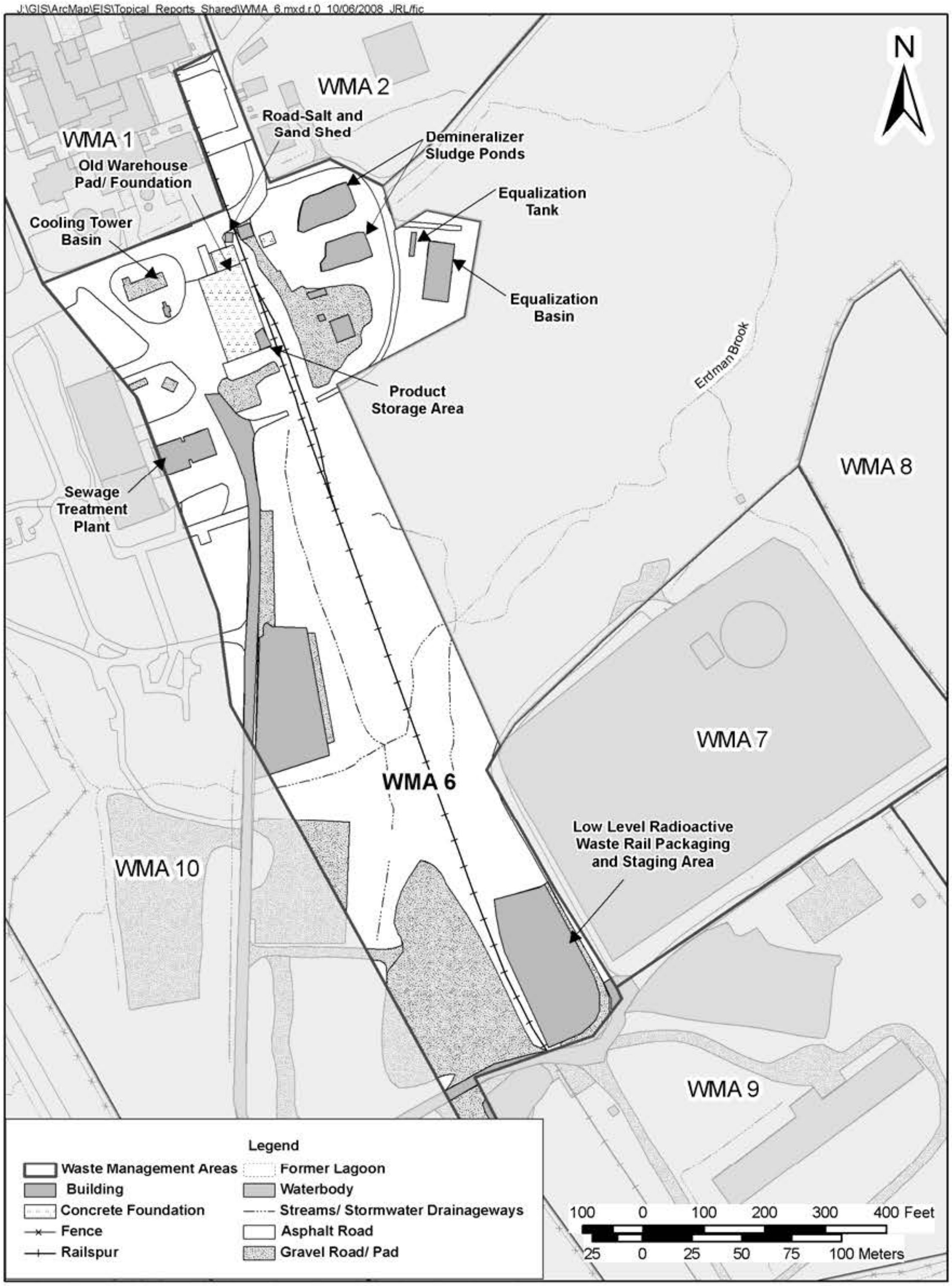



**Figure 3-38. WMA 6.** (Facilities to be removed during Phase 1 Decommissioning include the Demineralizer Sludge Ponds, the Sewage Treatment Plant, the Equalization Tank and Basin, the south Waste Tank Farm Training Platform, and the remaining slabs.)

**Figure 3-39. The Rail Spur.** (The rail spur leads to the Fuel Receiving and Storage Facility.)

**Figure 3-40. The New Cooling Tower.** (The cooling tower will be removed, except for its concrete basin, before Phase 1 decommissioning activities begin.)

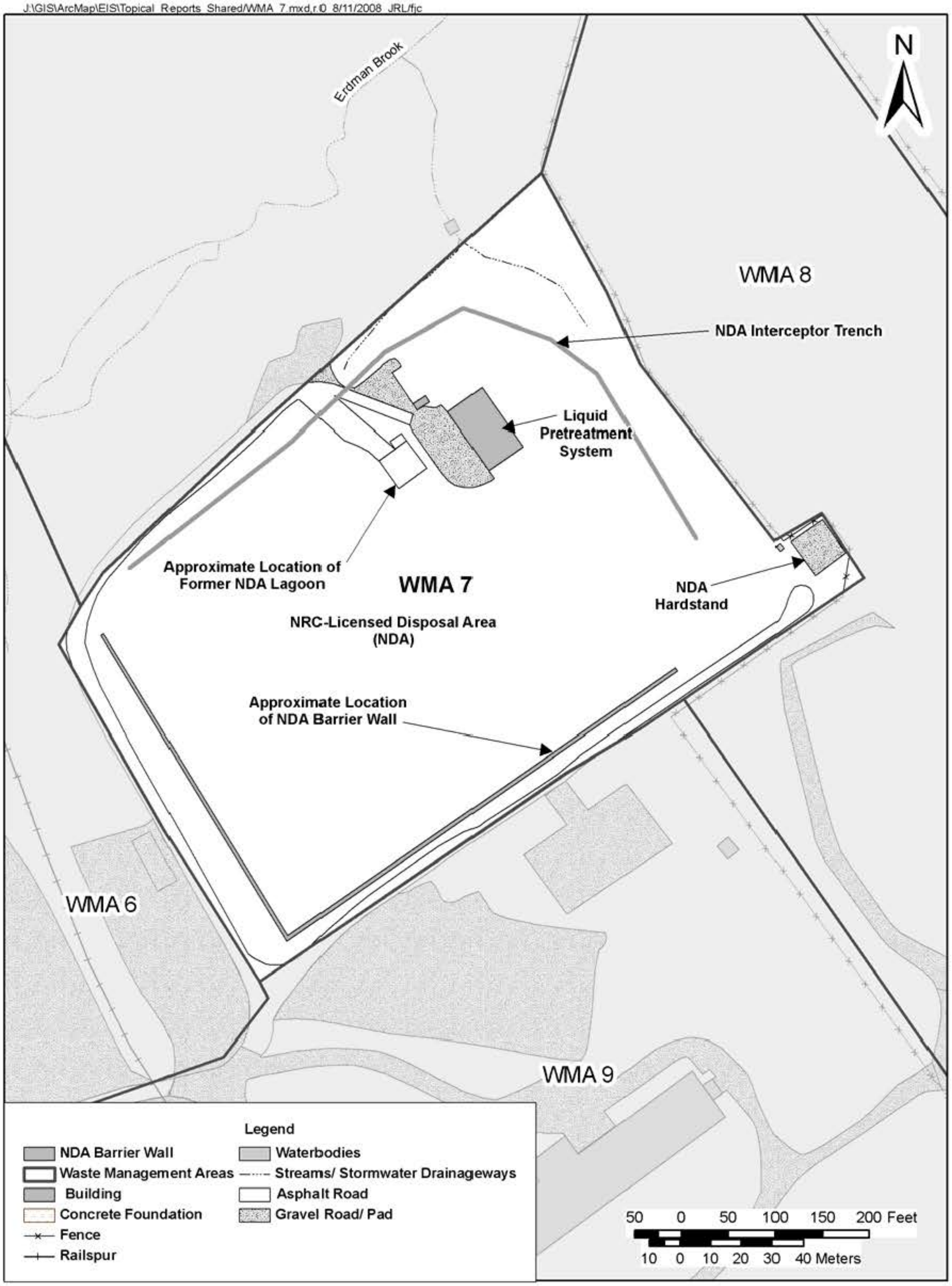



**Figure 3-41. WMA 7. (**The only facility to be removed during Phase 1 decommissioning is the NDA hardstand pad.)

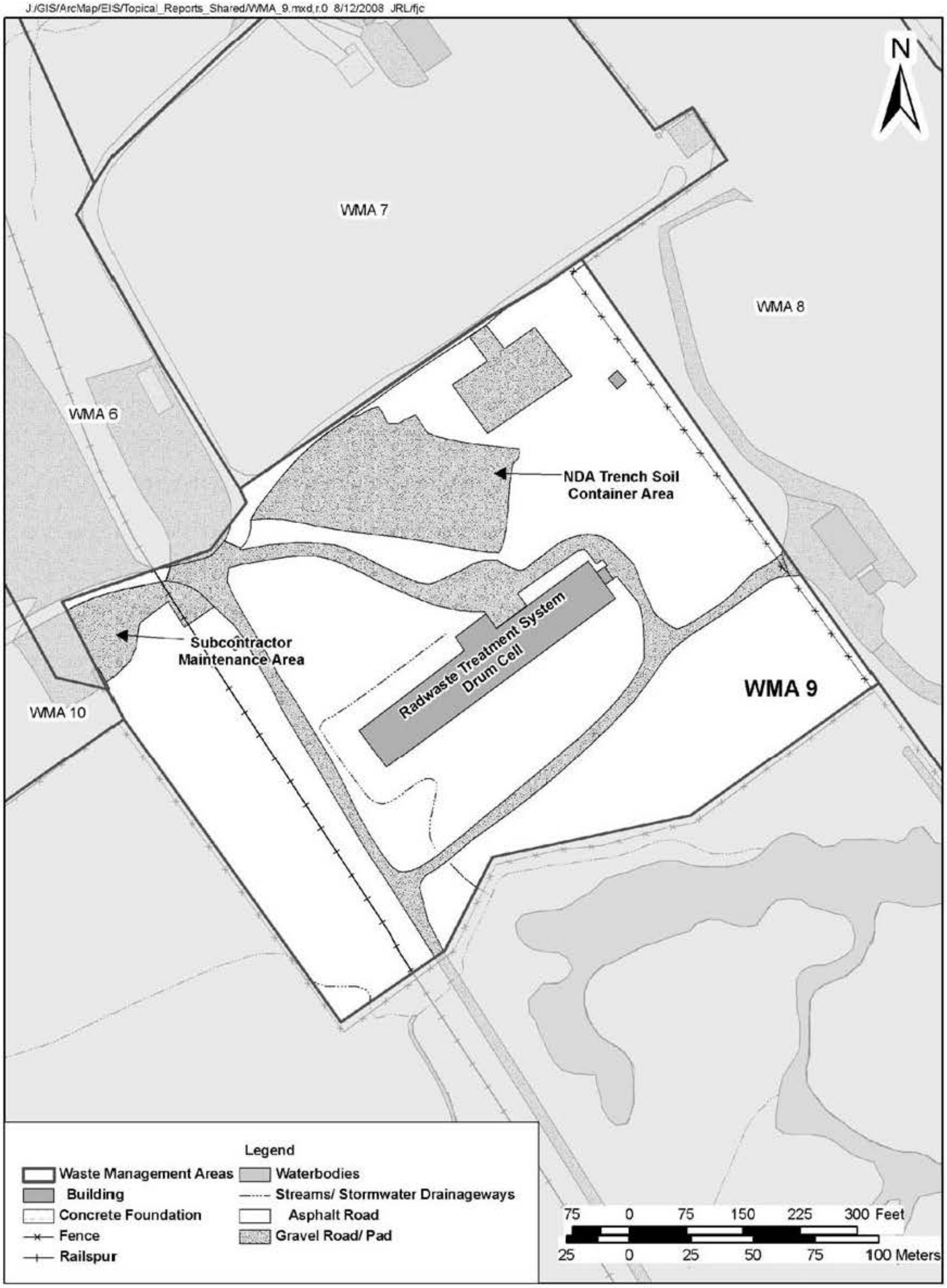



**Figure 3-42. WMA 9. (**The Drum Cell will be removed during Phase 1 decommissioning, along with NDA Trench Soil Container Area and the Subcontractor Maintenance Area.)

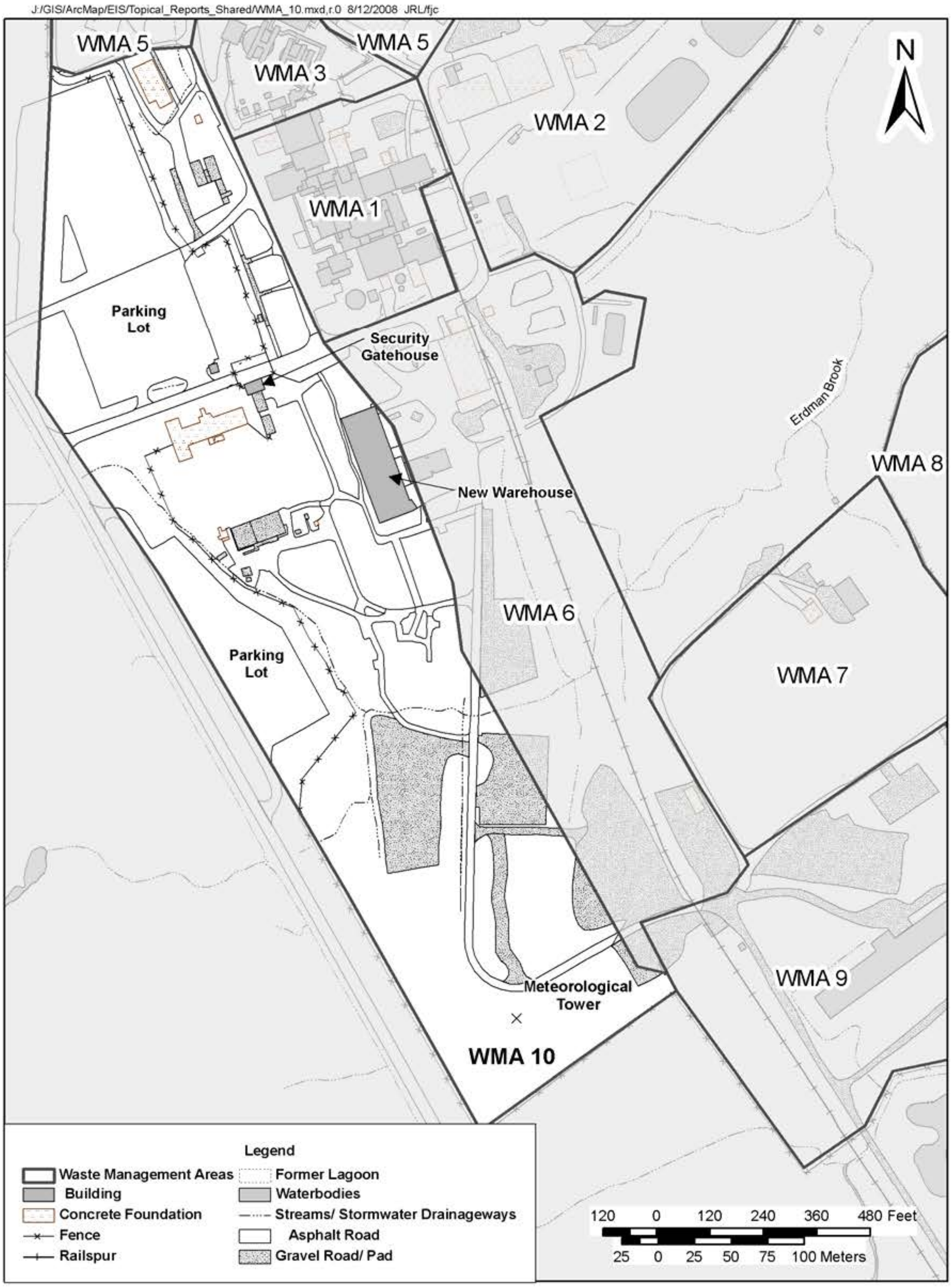



**Figure 3-43. WMA 10.** (Facilities to be removed during Phase 1 decommissioning include the New Warehouse and the remaining slabs and pads.)

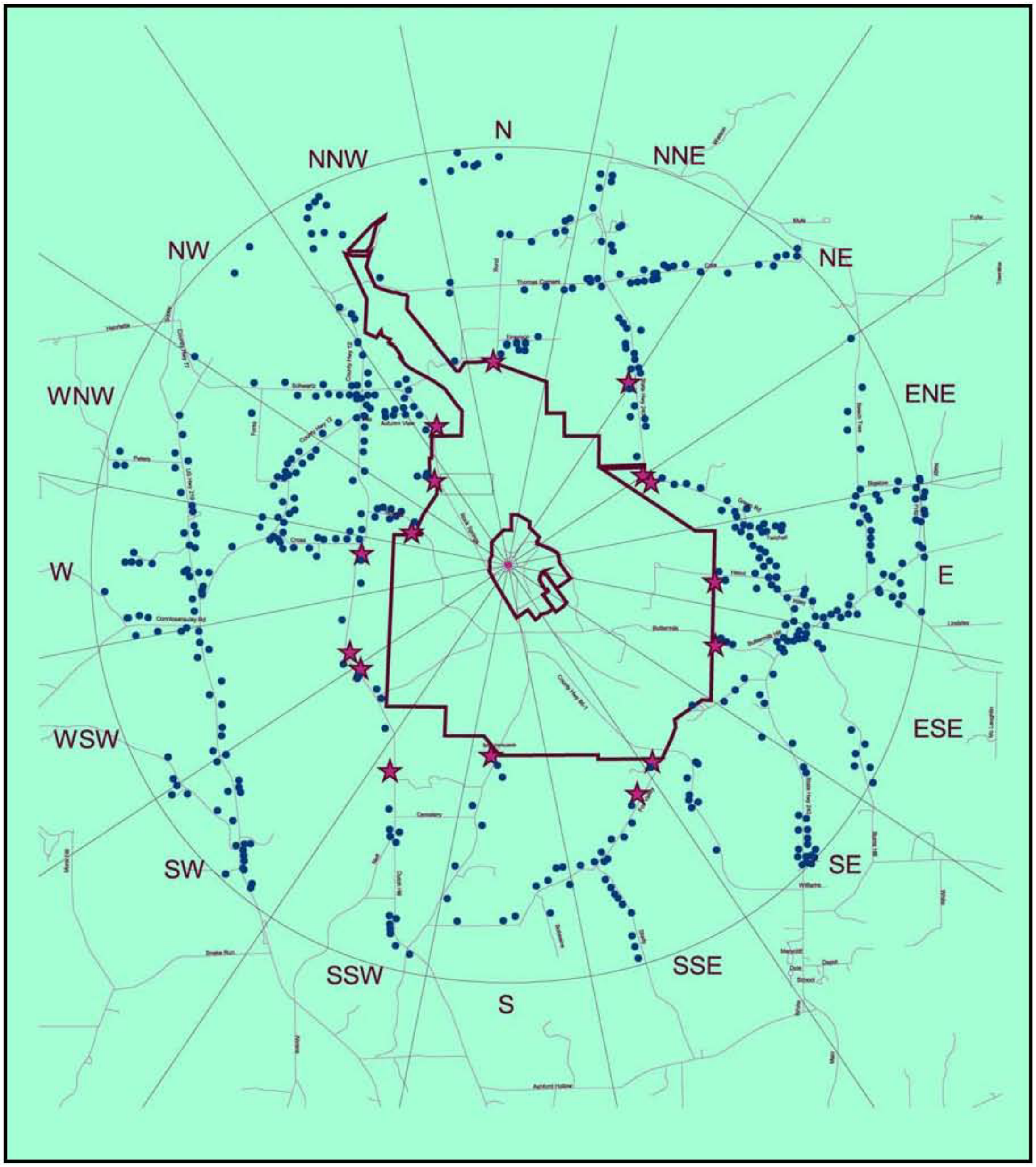



**Figure 3-44. Population Around the WVDP by Compass Vector**. (The dots represent residences. The stars show the nearest residences by compass vector.)

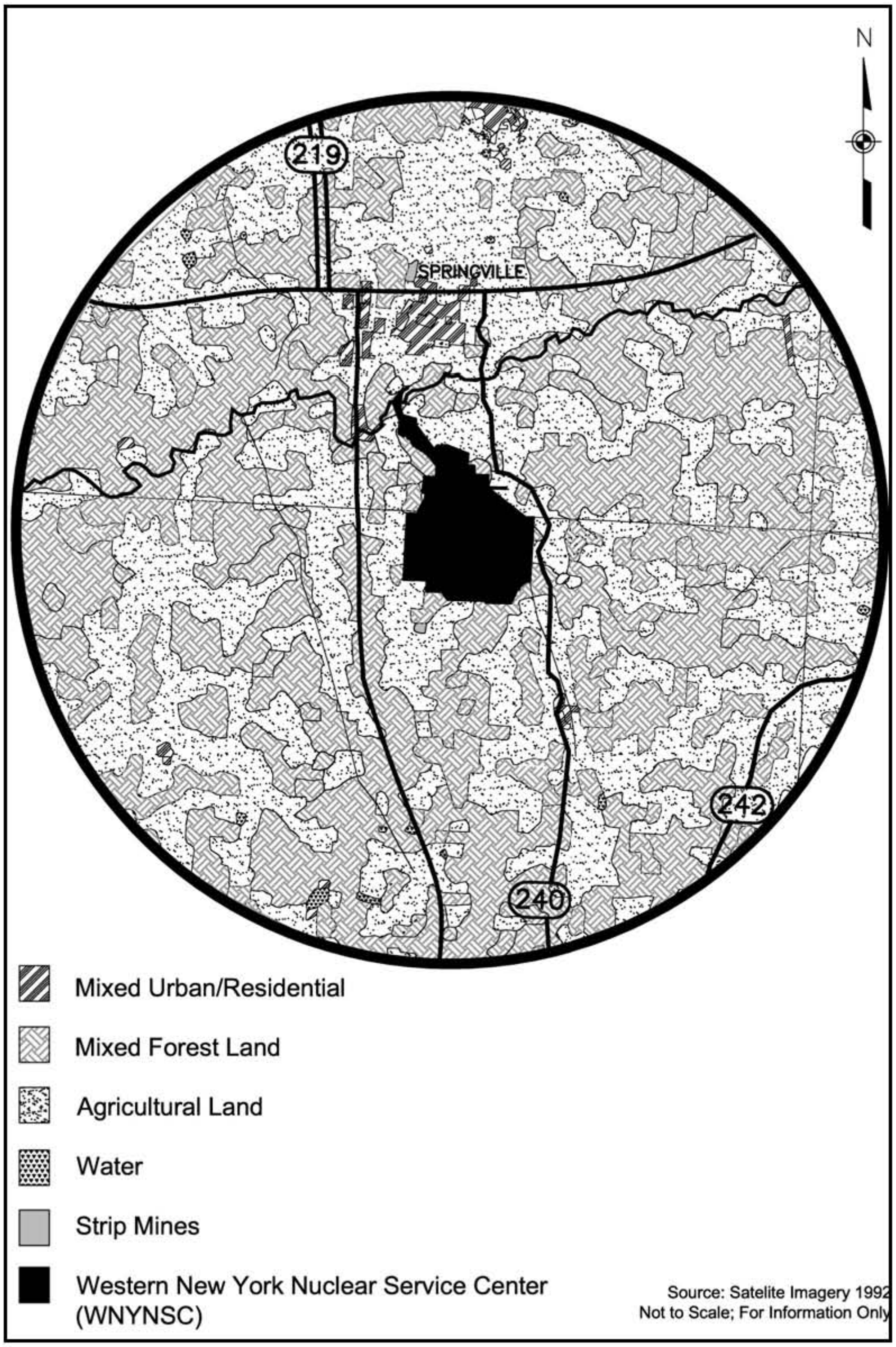



**Figure 3-45. Land Use in the Vicinity of the Center**

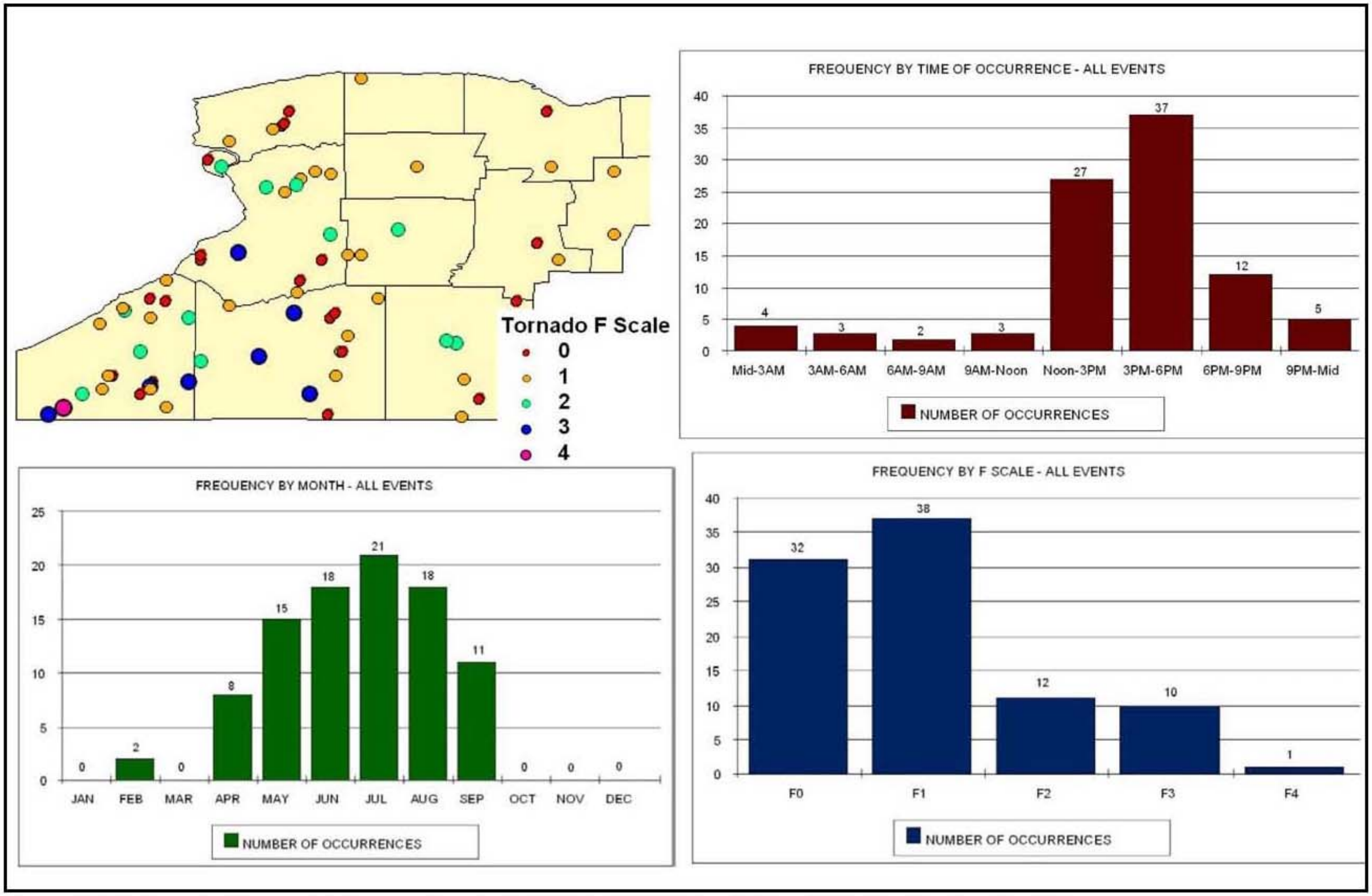



**Figure 3-46. Tornado Events in Western New York (1950 – 2002)** (From National Weather Service, Buffalo)

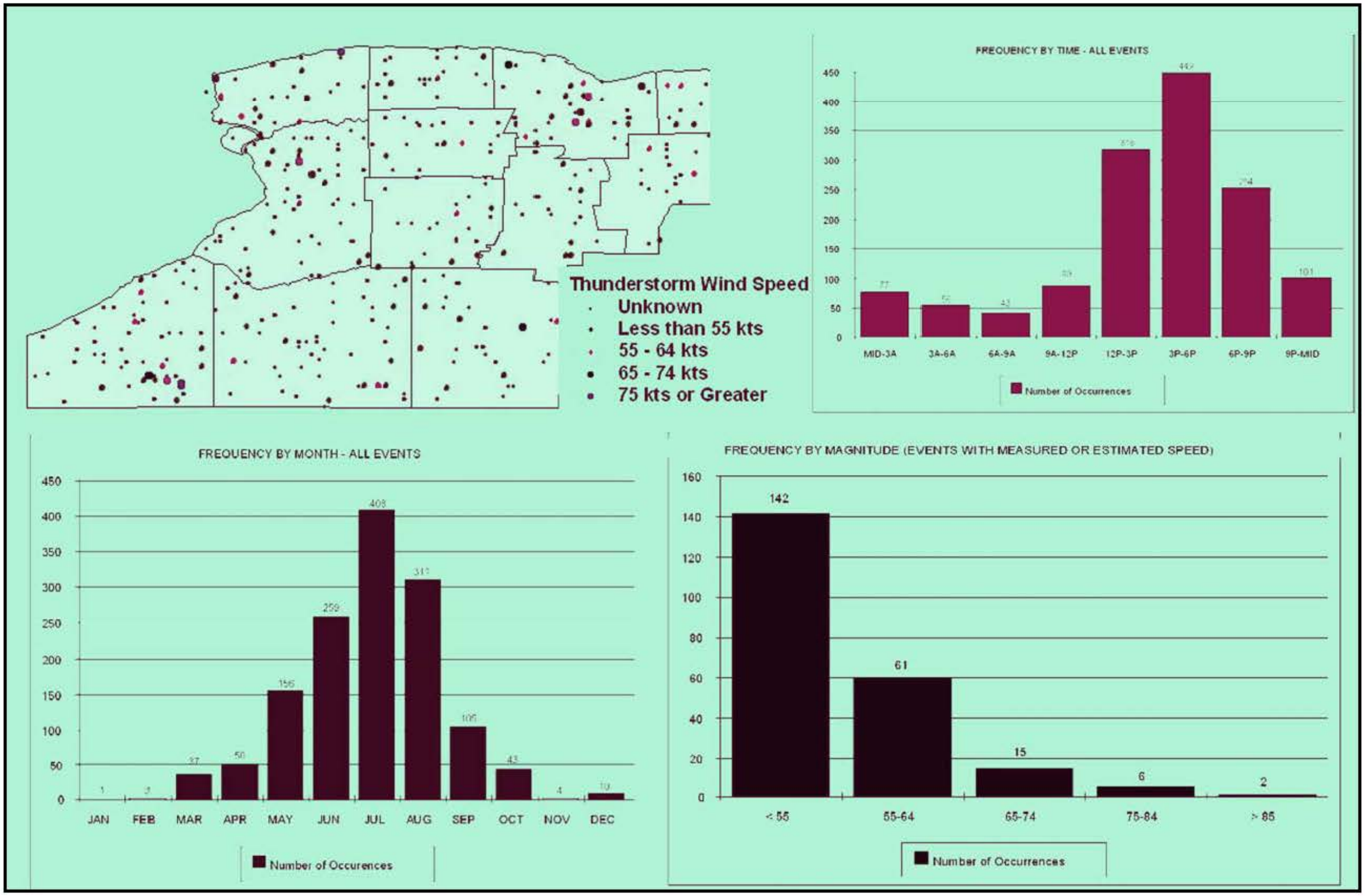



**Figure 3-47. Thunderstorm Wind Events in Western New York (1950 – 2002)** (From National Weather Service, Buffalo)

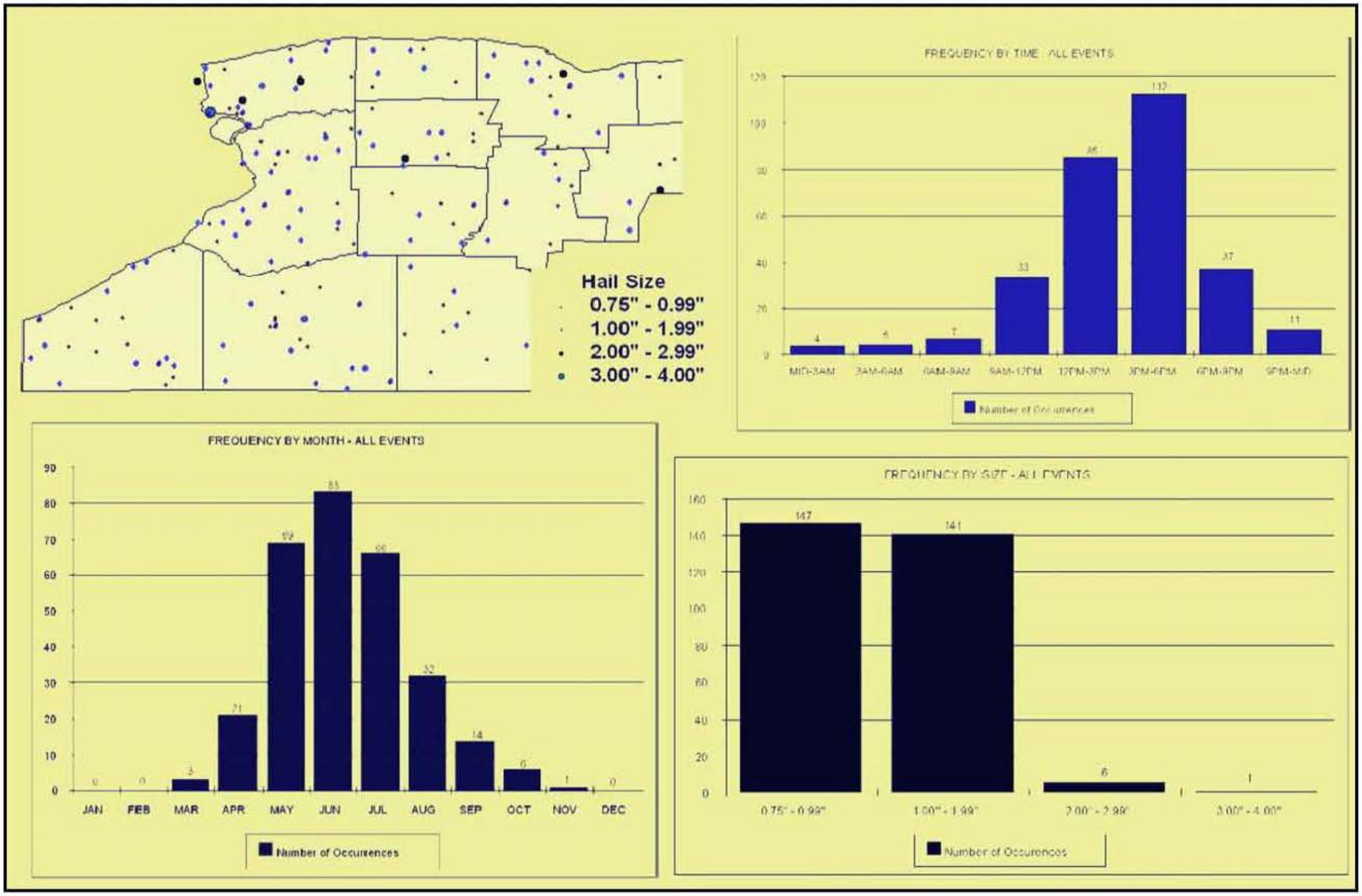



**Figure 3-48. Hail Events in Western New York (1950 – 2002)** (From National Weather Service, Buffalo)

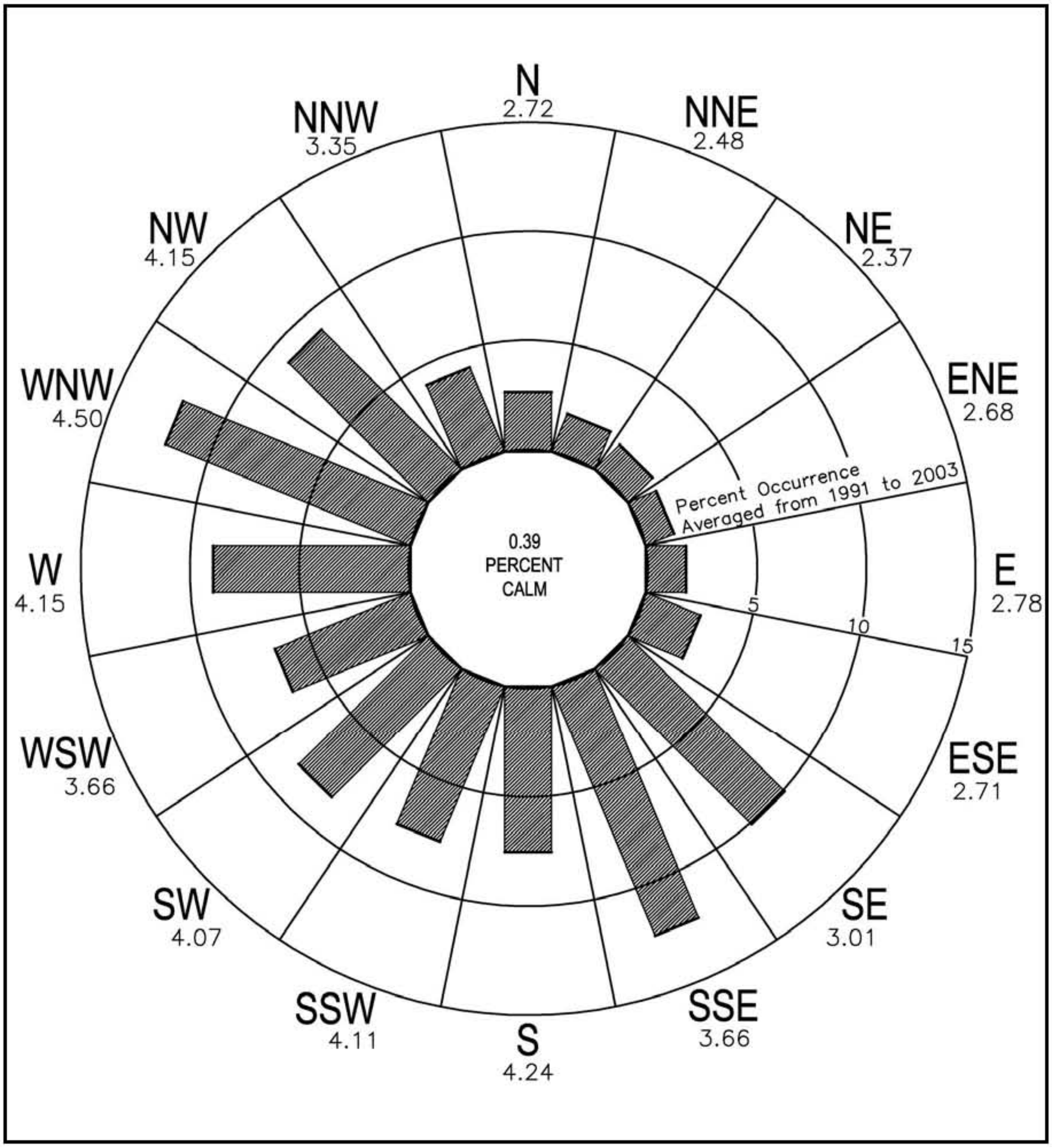



**Figure 3-49. Wind Rose Diagram.** (1991 – 2003 average head-wind direction and average wind speed in m/s)

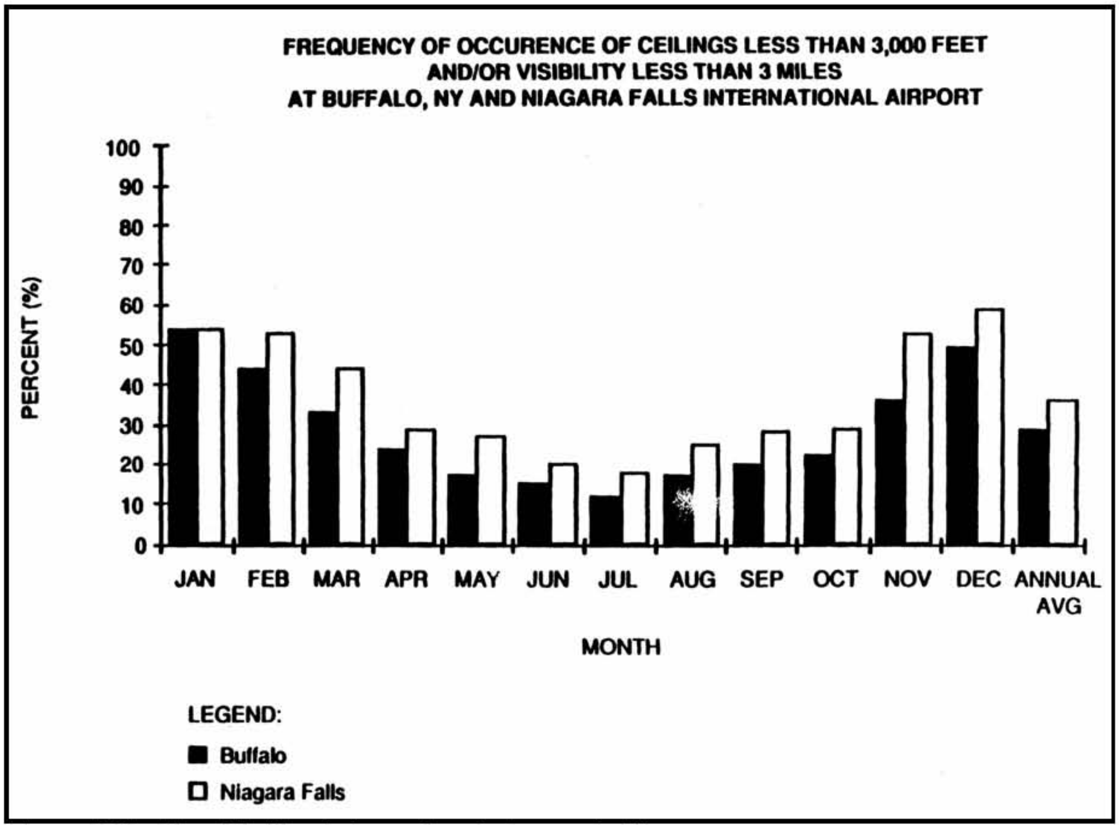



**Figure 3-50. Cloud Ceiling Information** (From reference 3-11)

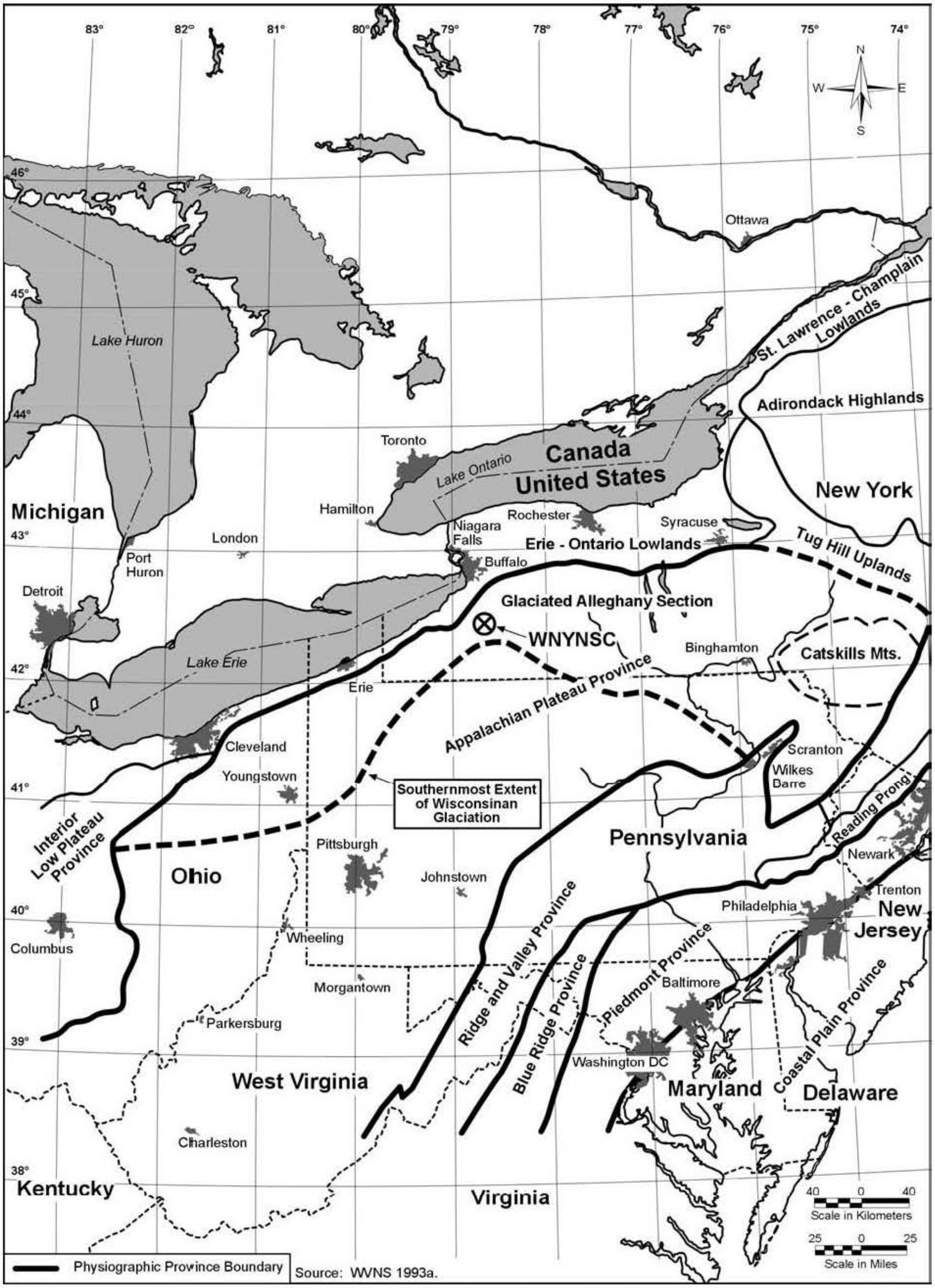



**Figure 3-51. Regional Physiographic Map**

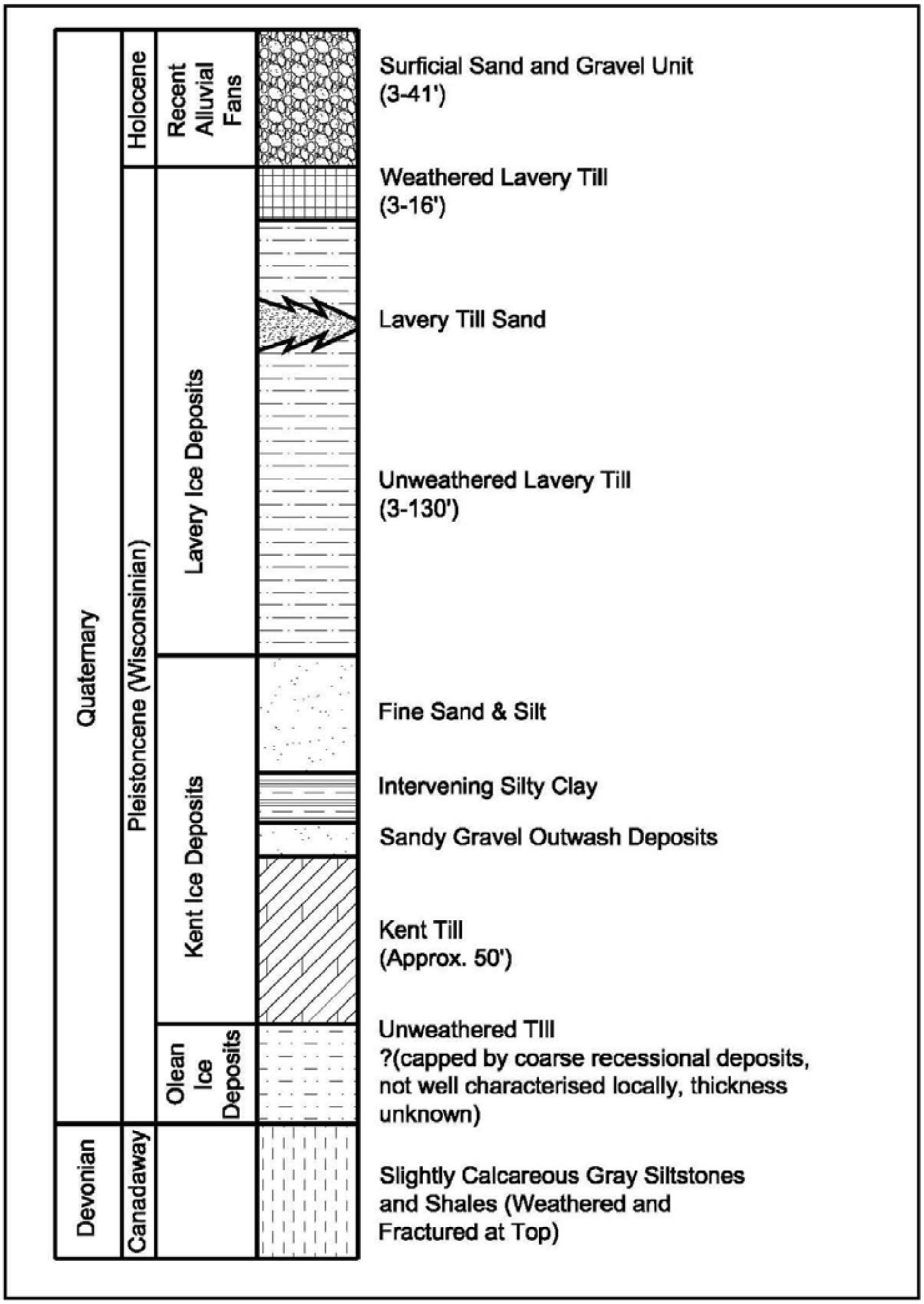



**Figure 3-52. Bedrock and Glacial Stratigraphy of the WVDP**

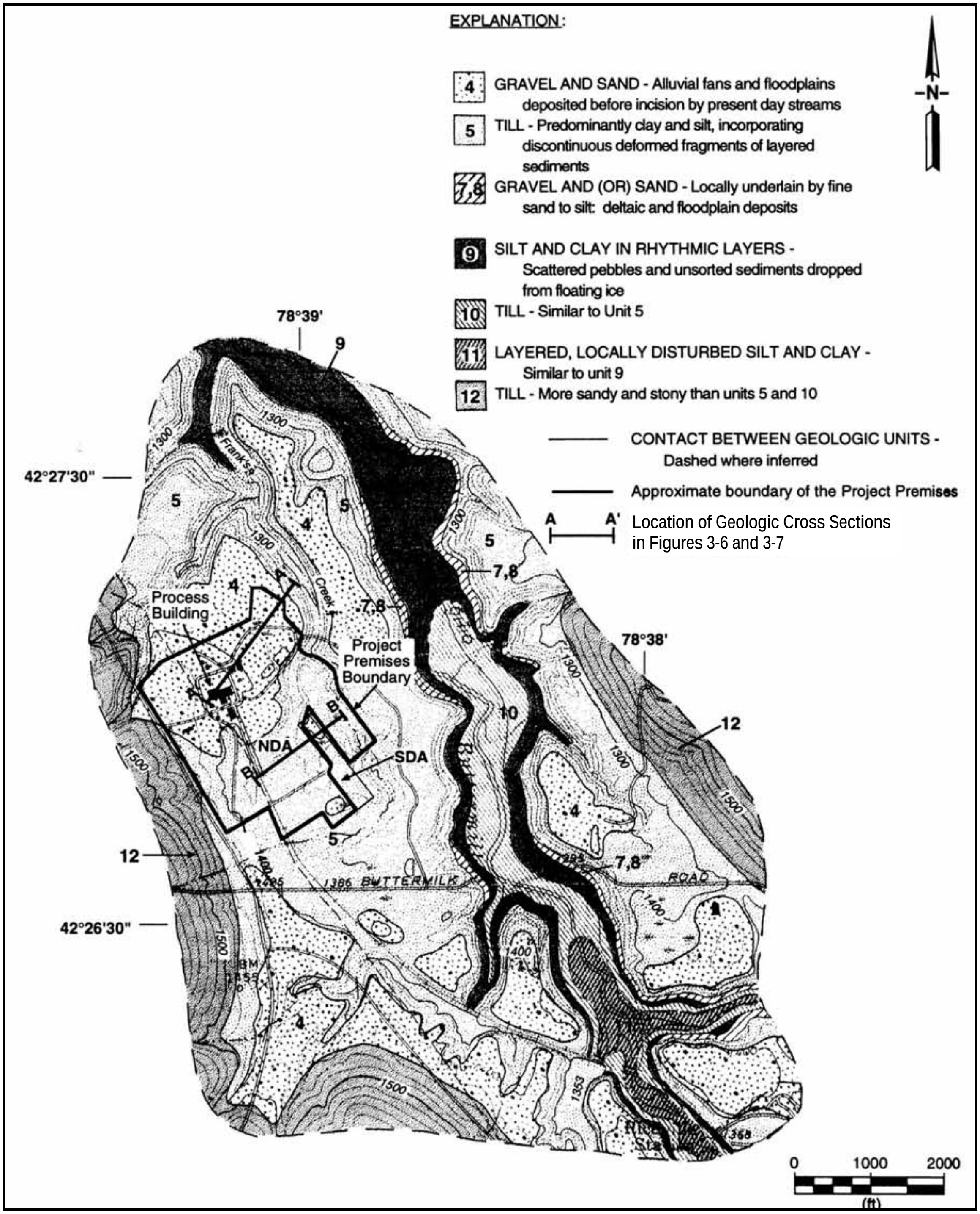



**Figure 3-53. Surface Geology of the Project Premises and the SDA**

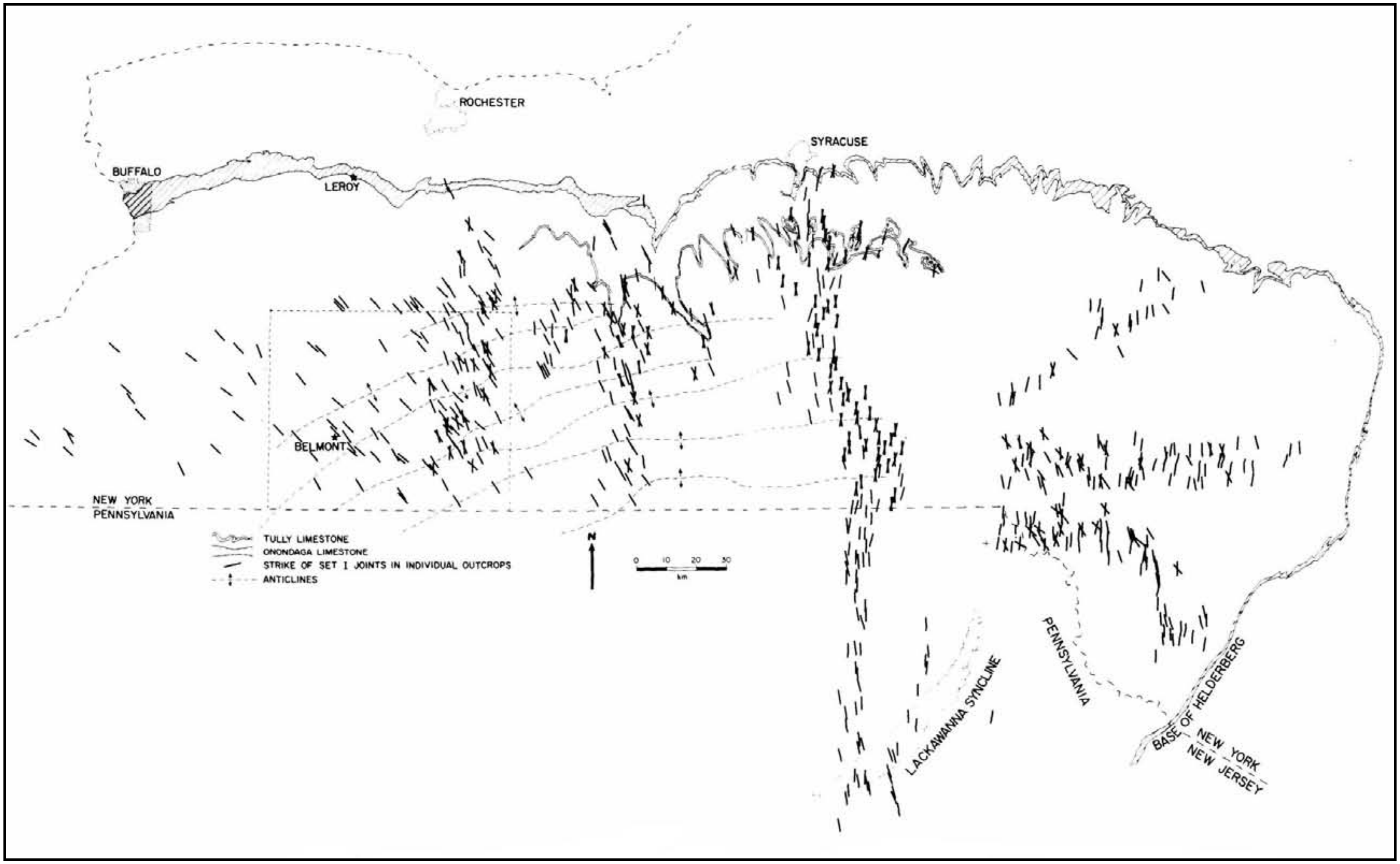



**Figure 3-54. Fold and Selected Joint Trends in the Appalachian Plateau of Western and Central New York**

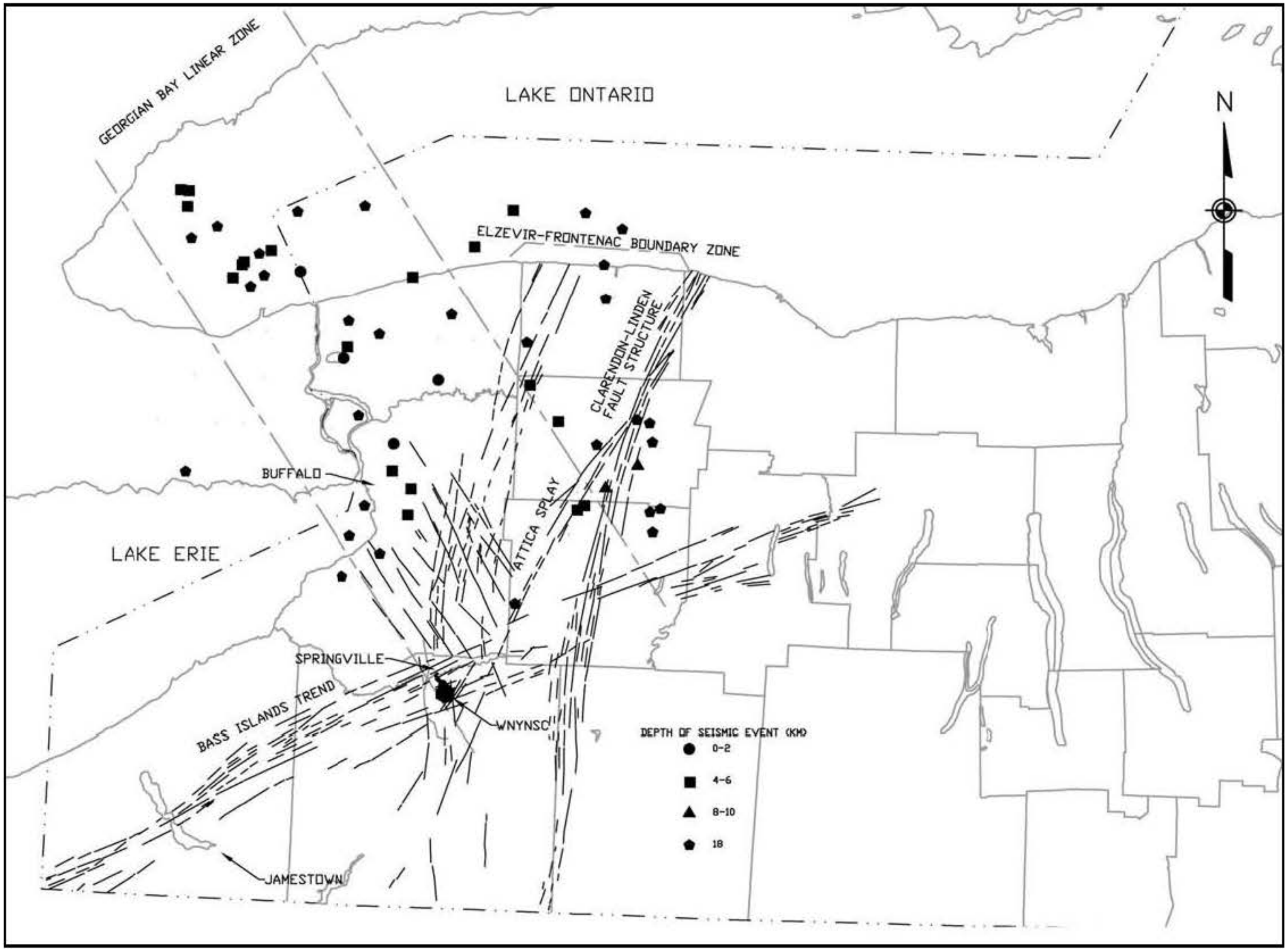



**Figure 3-55. Seismo-Tectonic Map of Western New York Showing Selected Regional Geologic Structures**

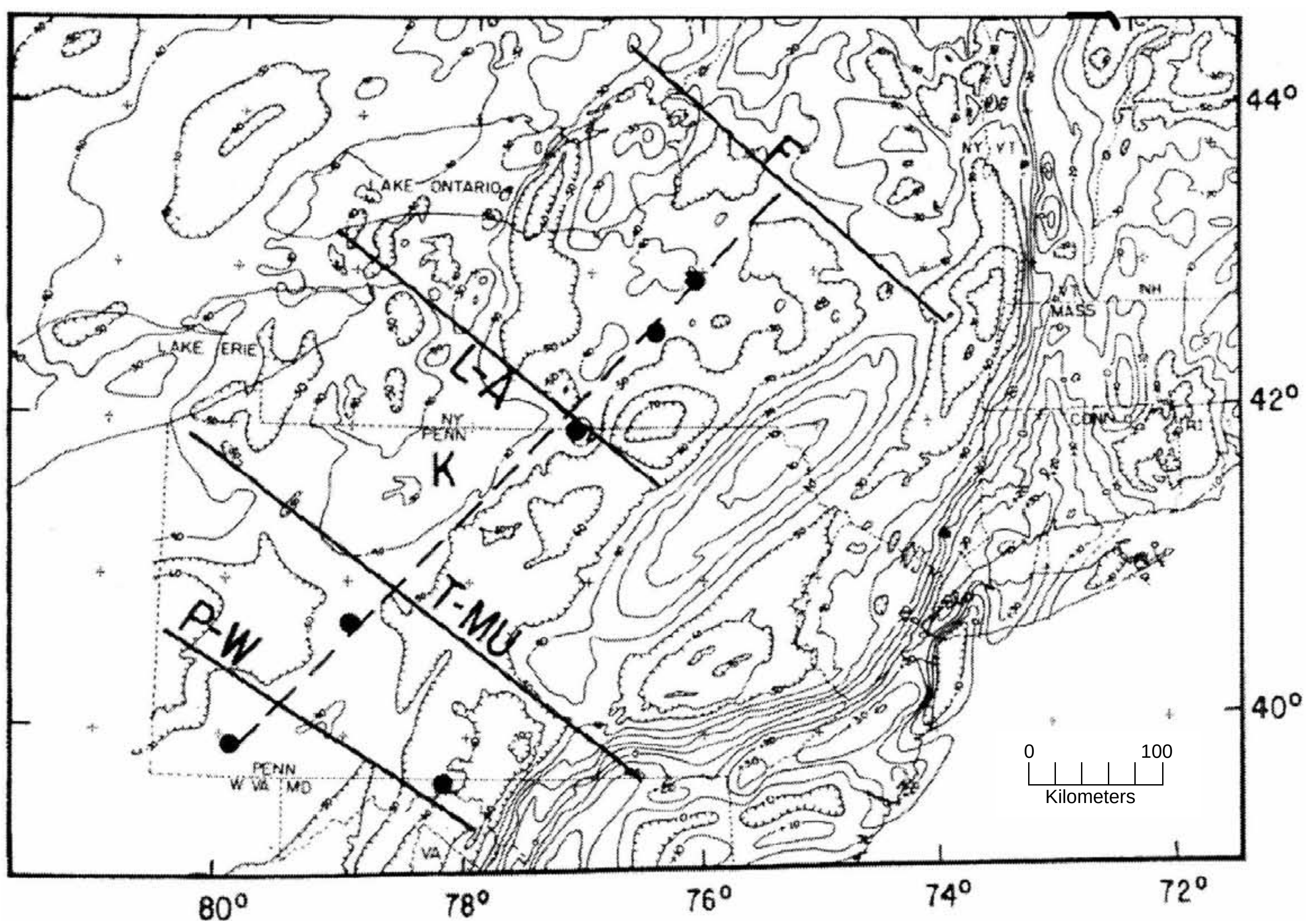



**Figure 3-56. Major Northwest Trending Lineaments in New York and Pennsylvania (**PW – Pittsburgh-Washington Lineament, T-MU – Tyrone-Mt. Union Lineament, L-A – Lawrenceville-Attica Lineament, F – F Lineament)

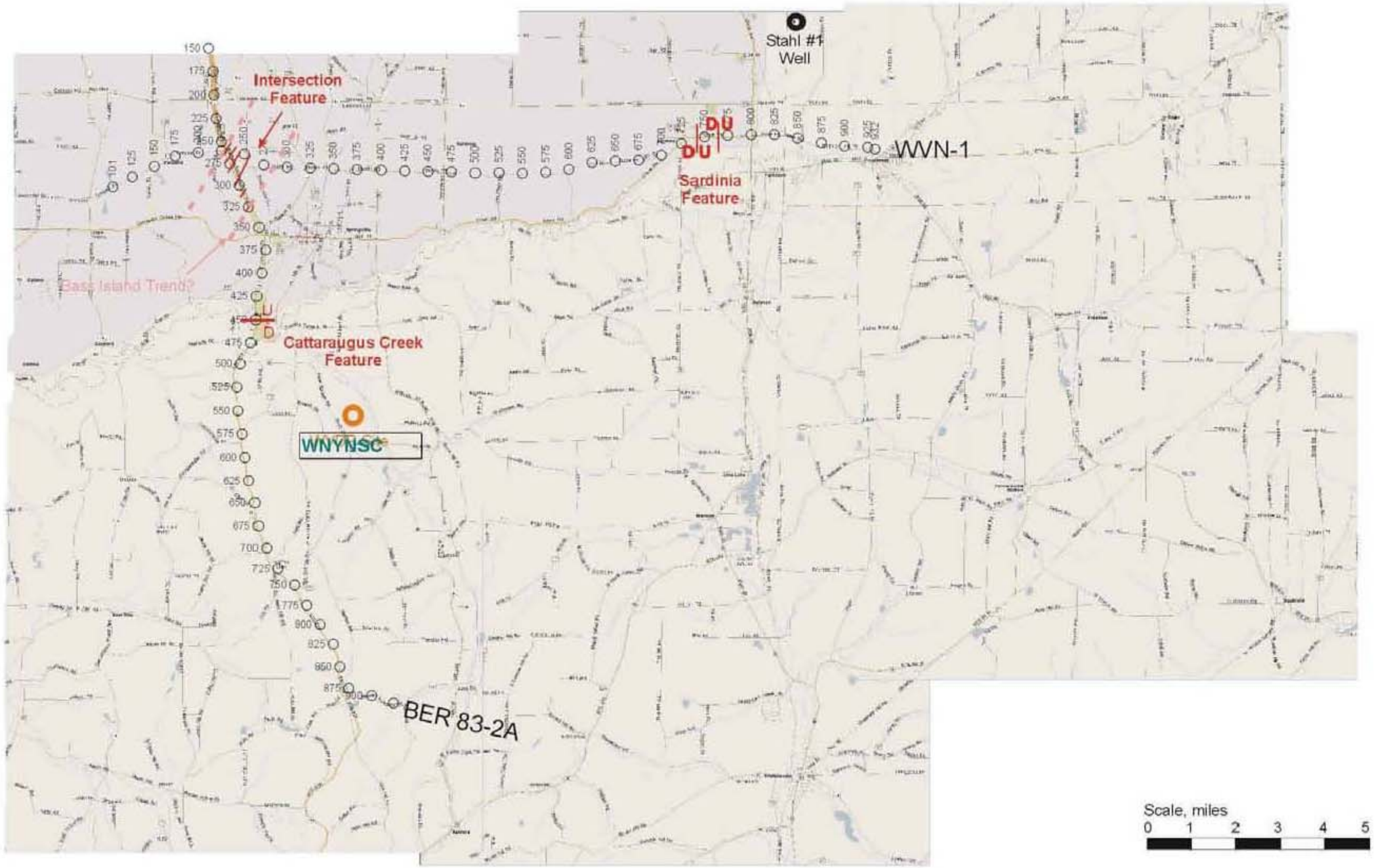



**Figure 3-57. Location of Seismic Lines WVN1 and BER 83-2A**

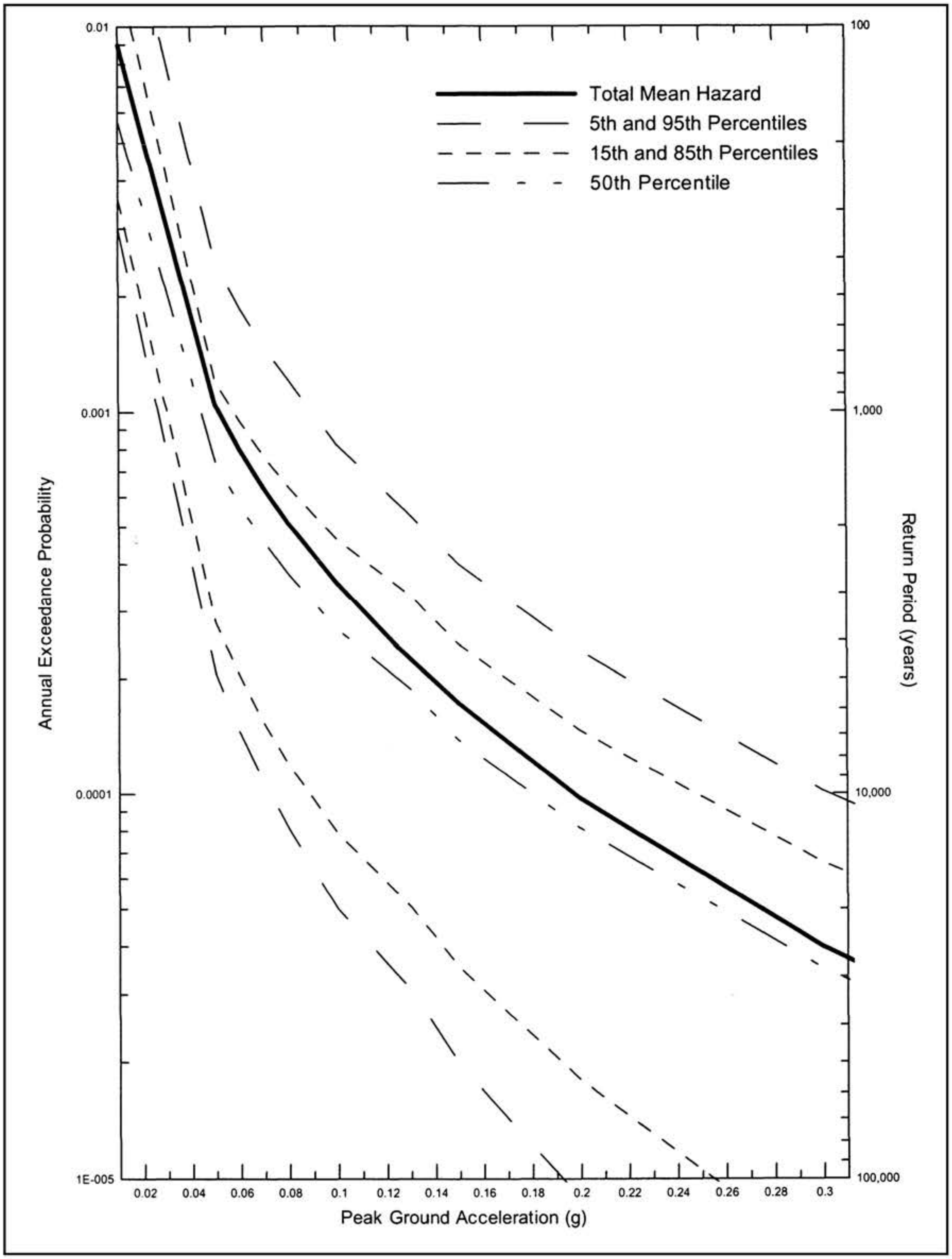



**Figure 3-58. Seismic Hazard Curves for Peak Horizontal Acceleration**

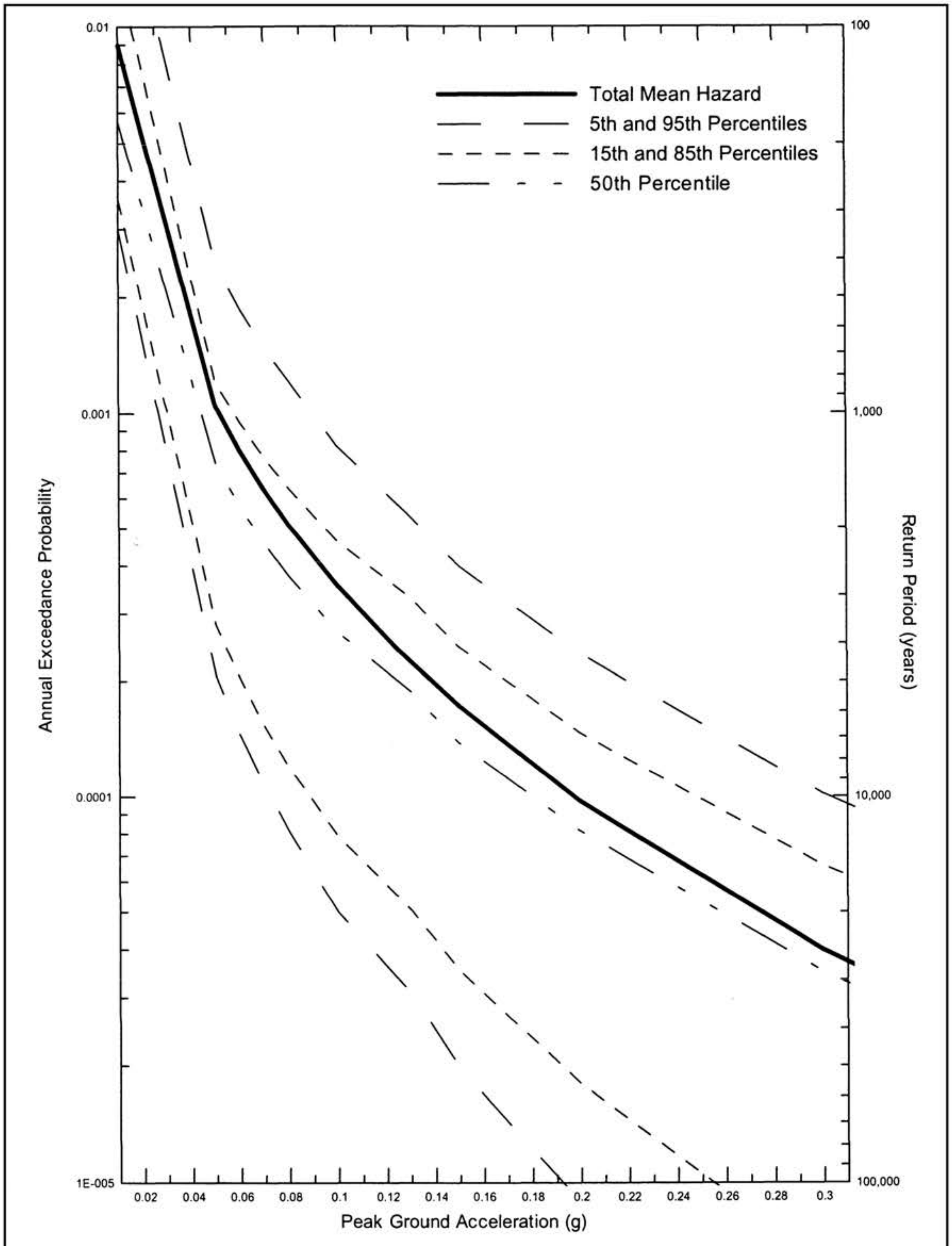



**Figure 3-59. Seismic Hazard Curves for 1.0 Second Horizontal Spectral Acceleration**

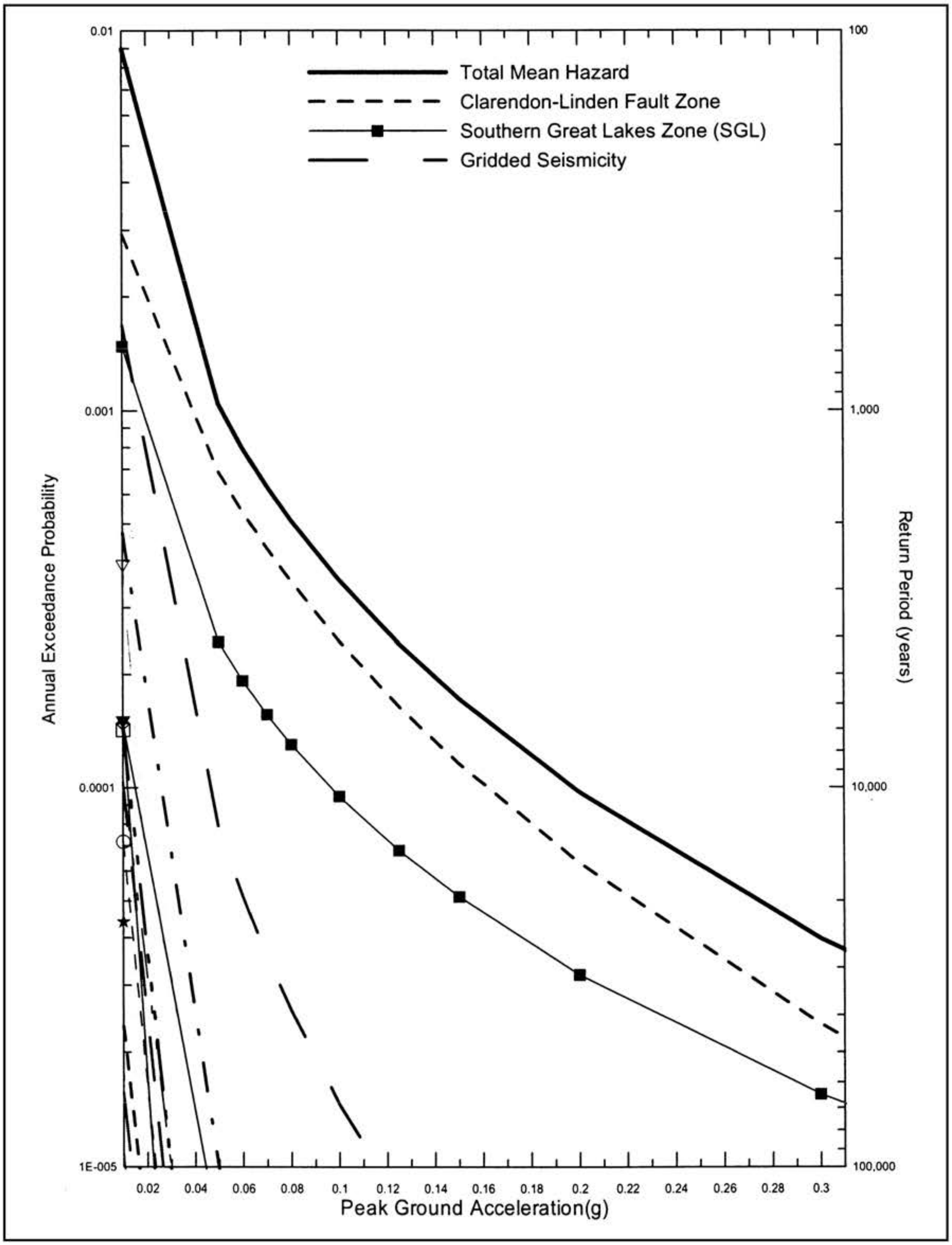



**Figure 3-60. Seismic Source Contributions to Mean Peak Horizontal Acceleration Hazard**

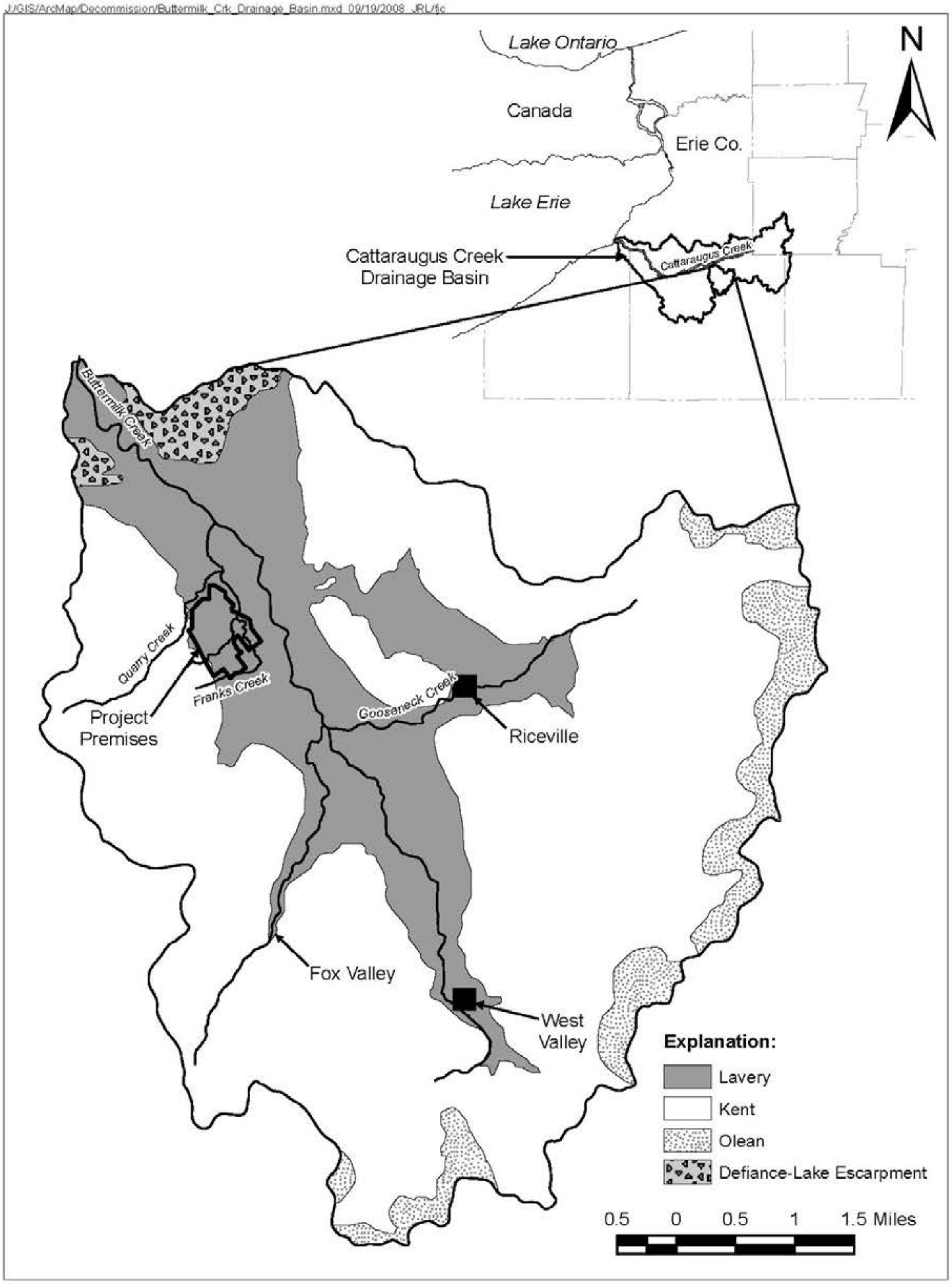



**Figure 3-61. Buttermilk Creek Drainage Basin**

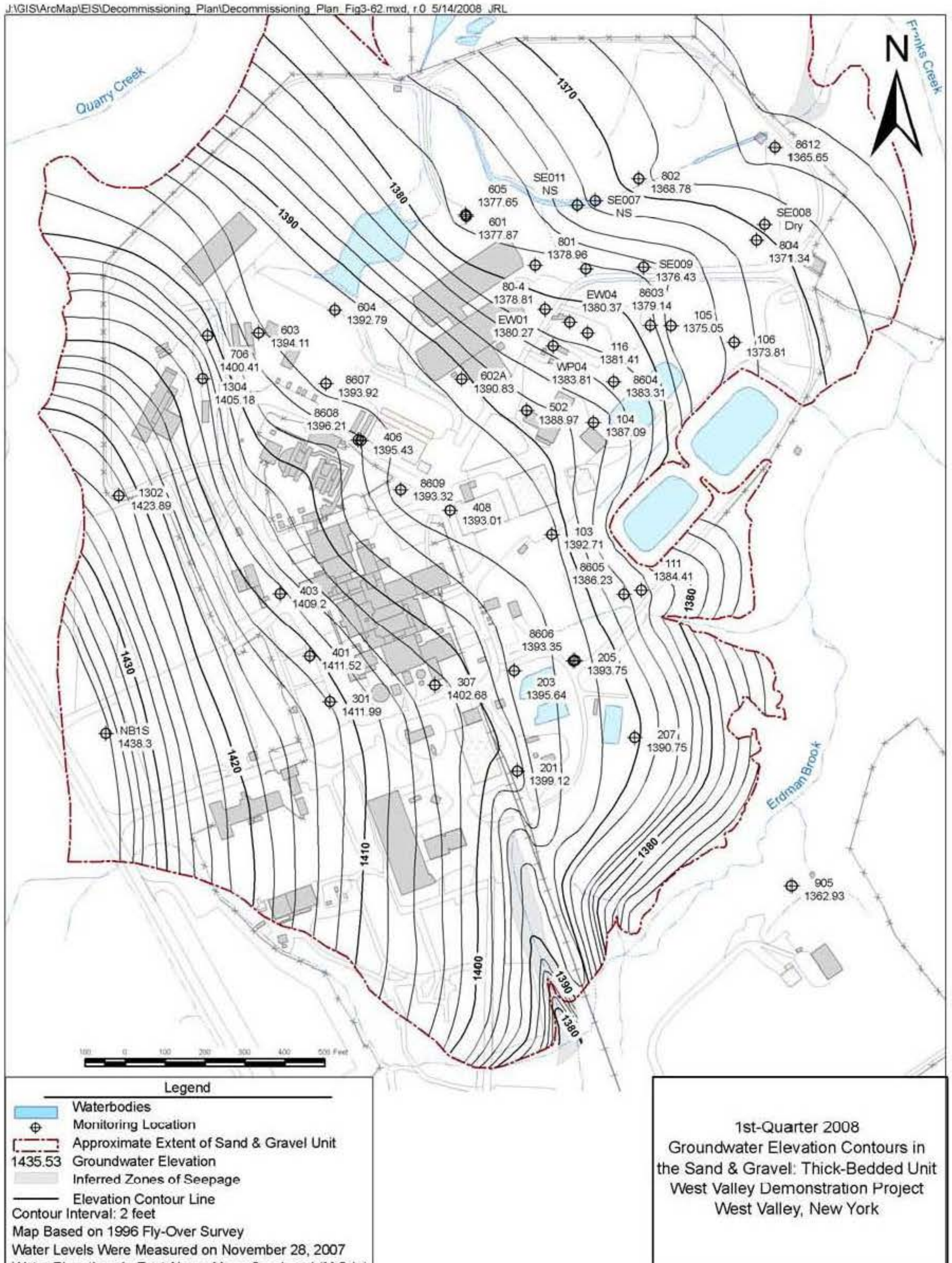



**Figure 3-62. Groundwater Elevation Contours of the Sand and Gravel Unit, First Quarter 2008**

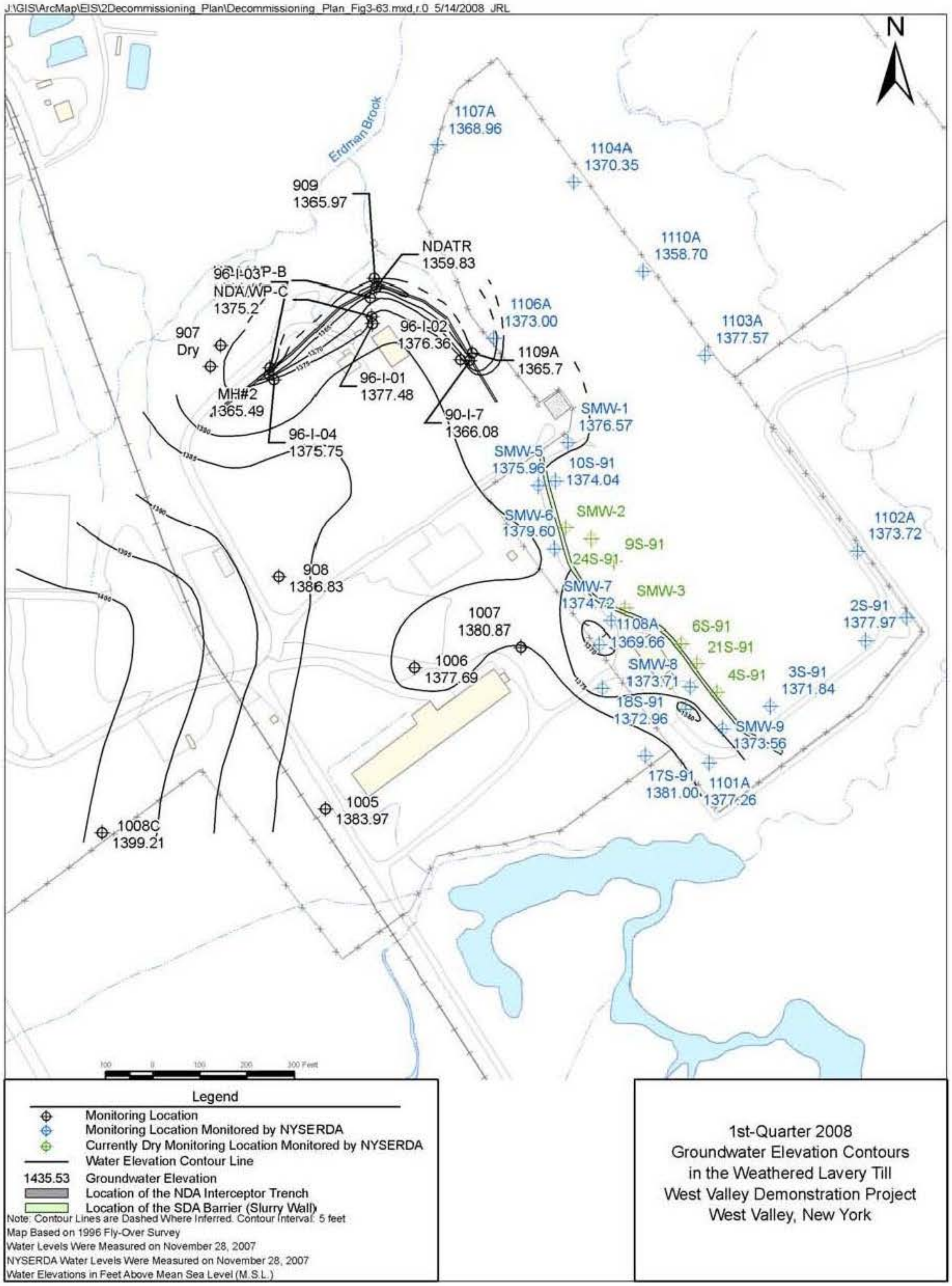



**Figure 3-63. Groundwater Elevation Contours of the Weathered Lavery Till, First Quarter 2008**

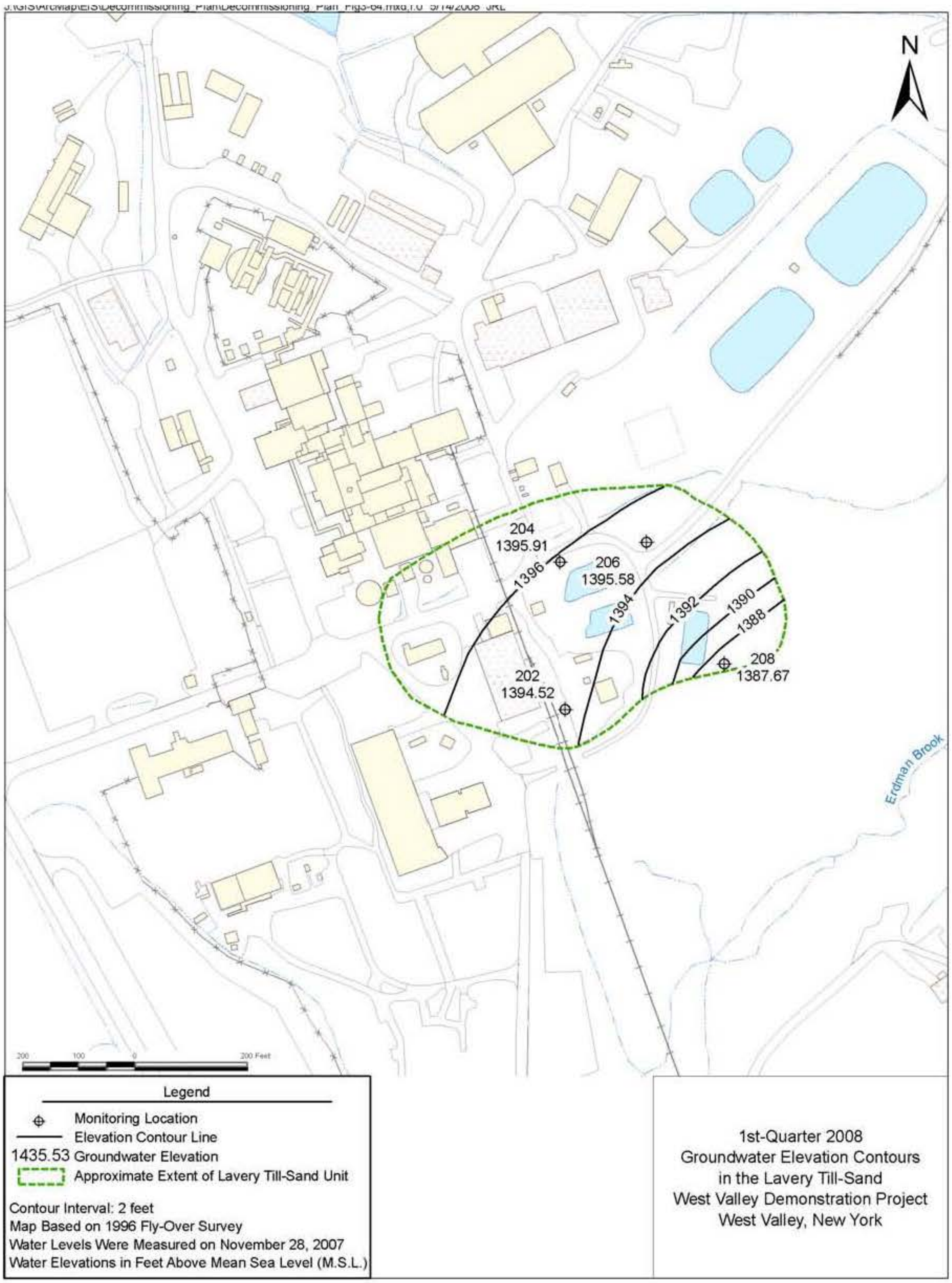



**Figure 3-64. Groundwater Elevation Contours of the Lavery Till Sand, First Quarter 2008**

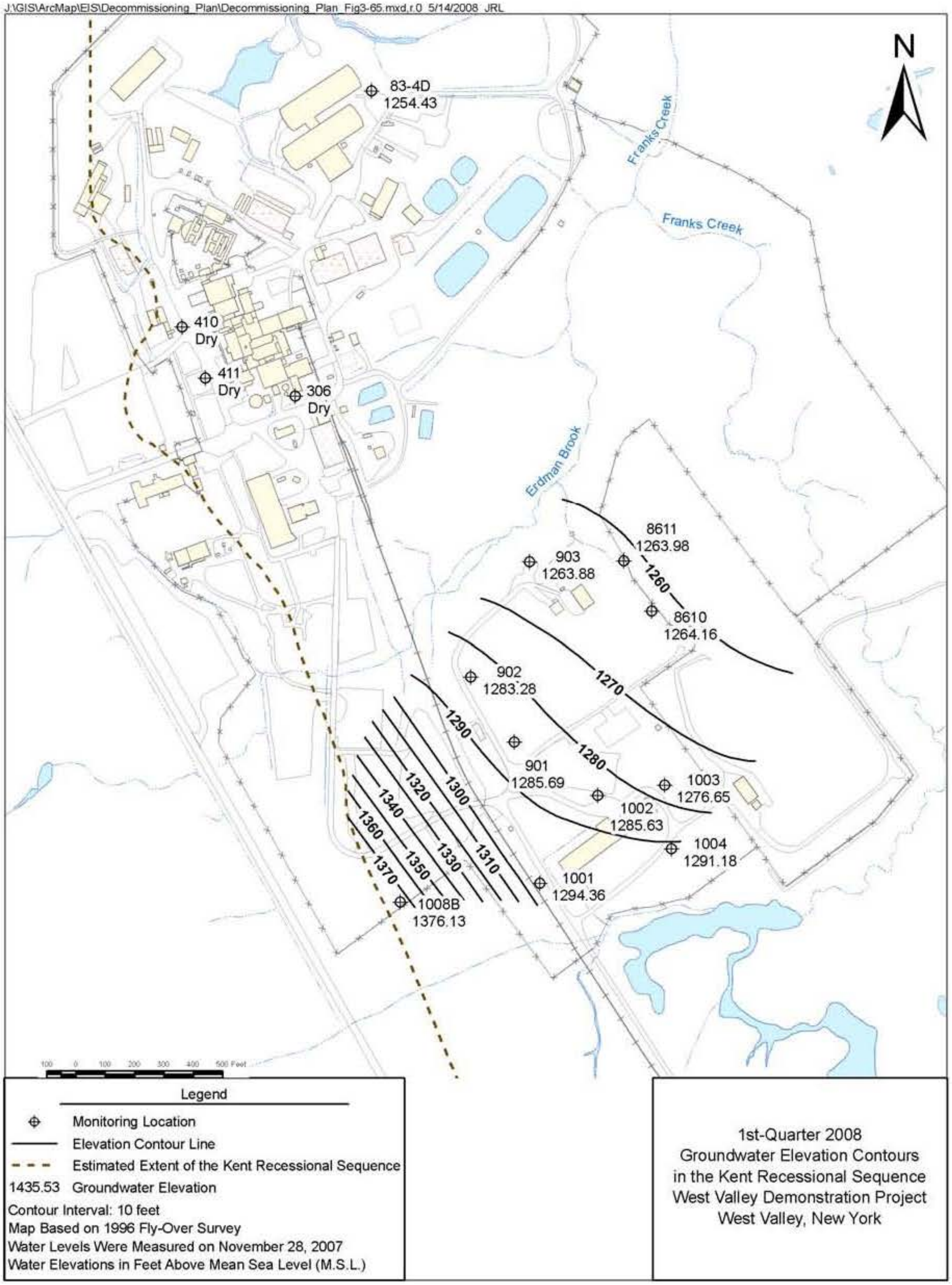



**Figure 3-65. Groundwater Elevation Contours of the Kent Recessional Sequence, First Quarter 2008**

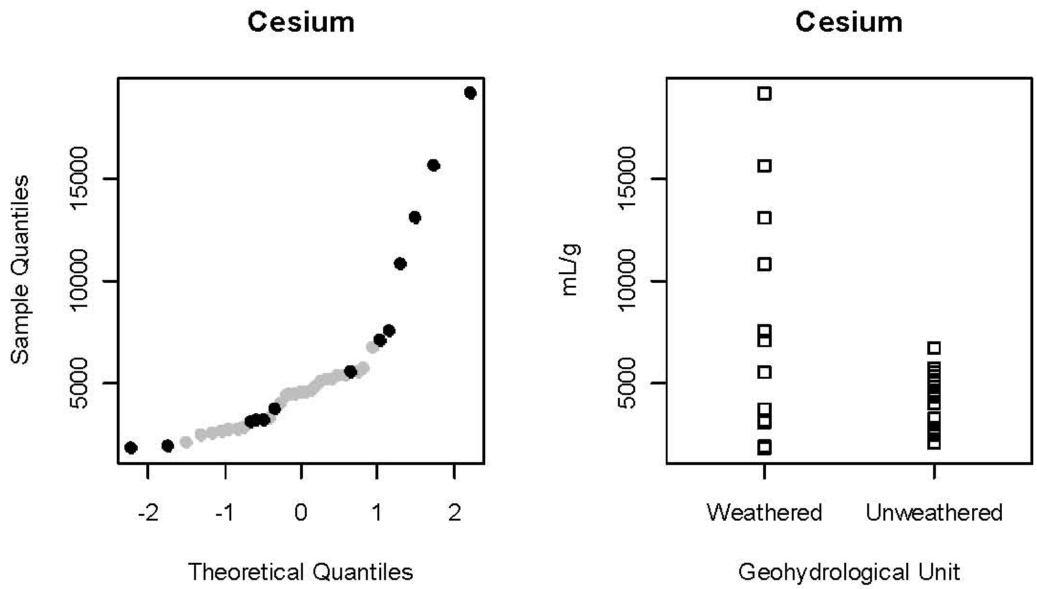



**Figure 3-66. Vertical Distribution of Cesium $K_d$ in the Weathered and Unweathered Tills** (WVNSCO 1993a)

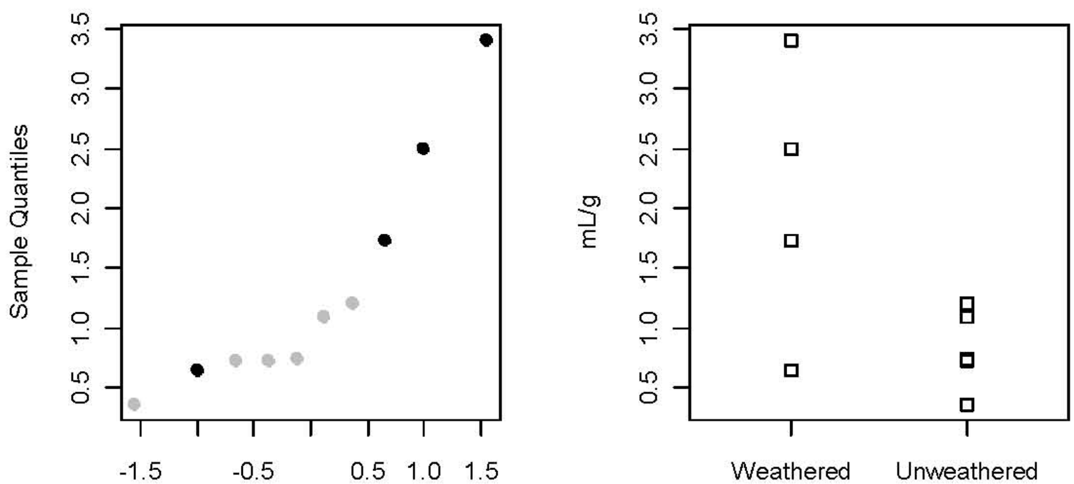



**Figure 3-67. Vertical Distribution of Iodine $K_d$ in the Weathered and Unweathered Tills** (WVNSCO 1993a)

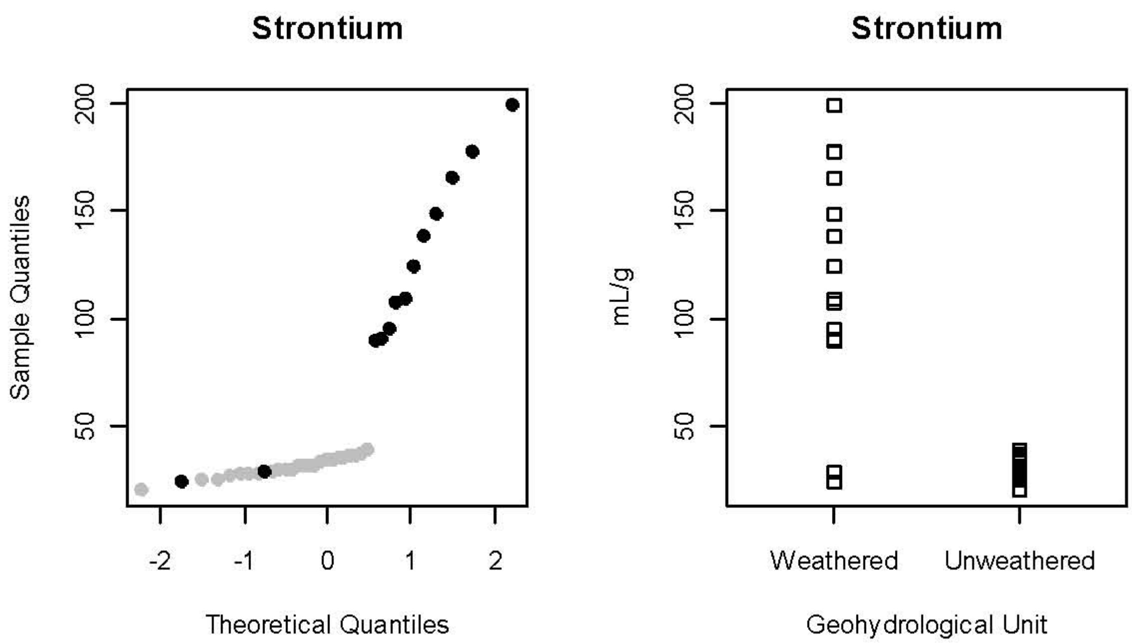



**Figure 3-68. Vertical Distribution of Strontium $K_d$ in the Weathered and Unweathered Tills** (WVNSCO 1993a)

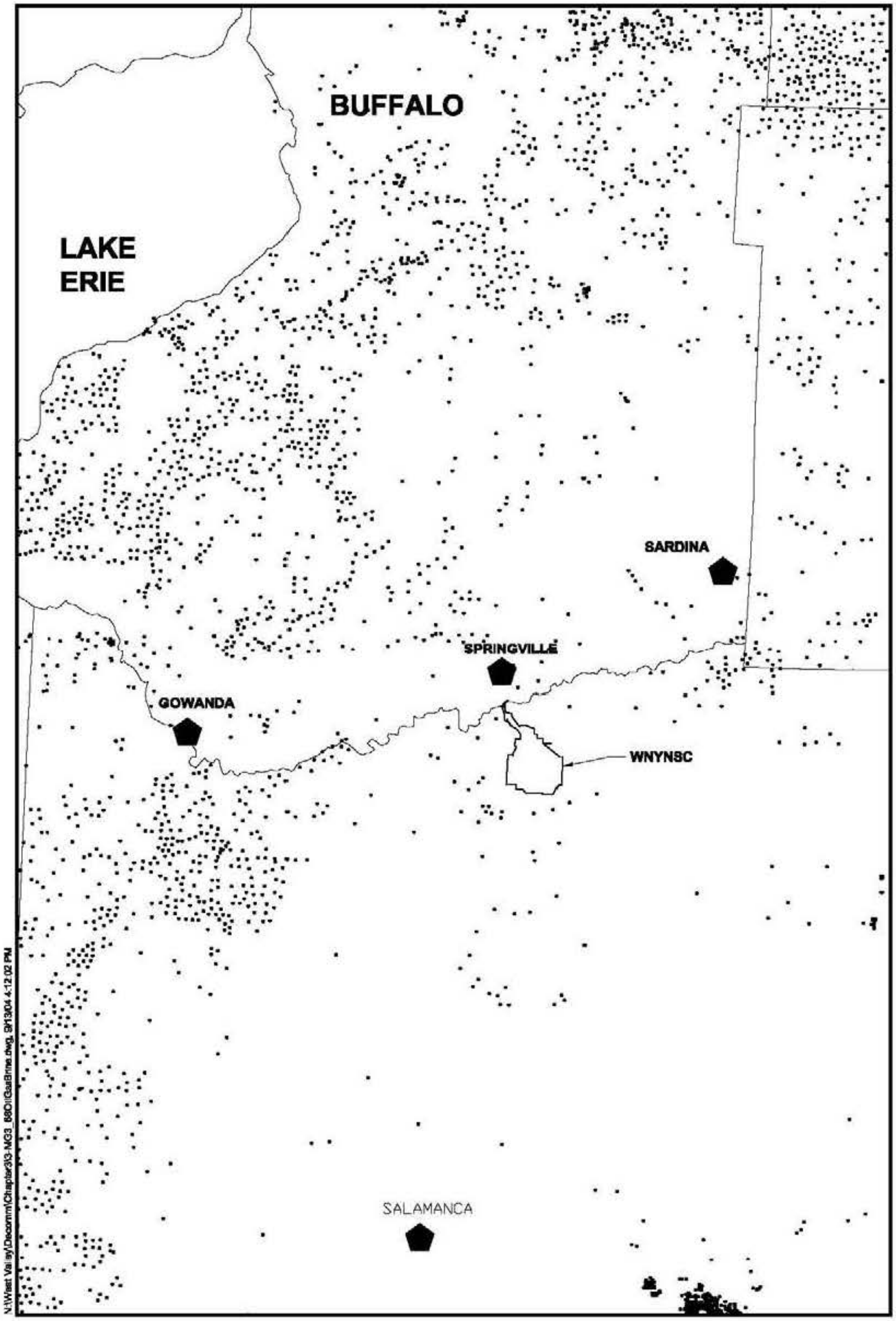



**Figure 3-69. Locations of Natural Gas and Oil Wells in Western New York**

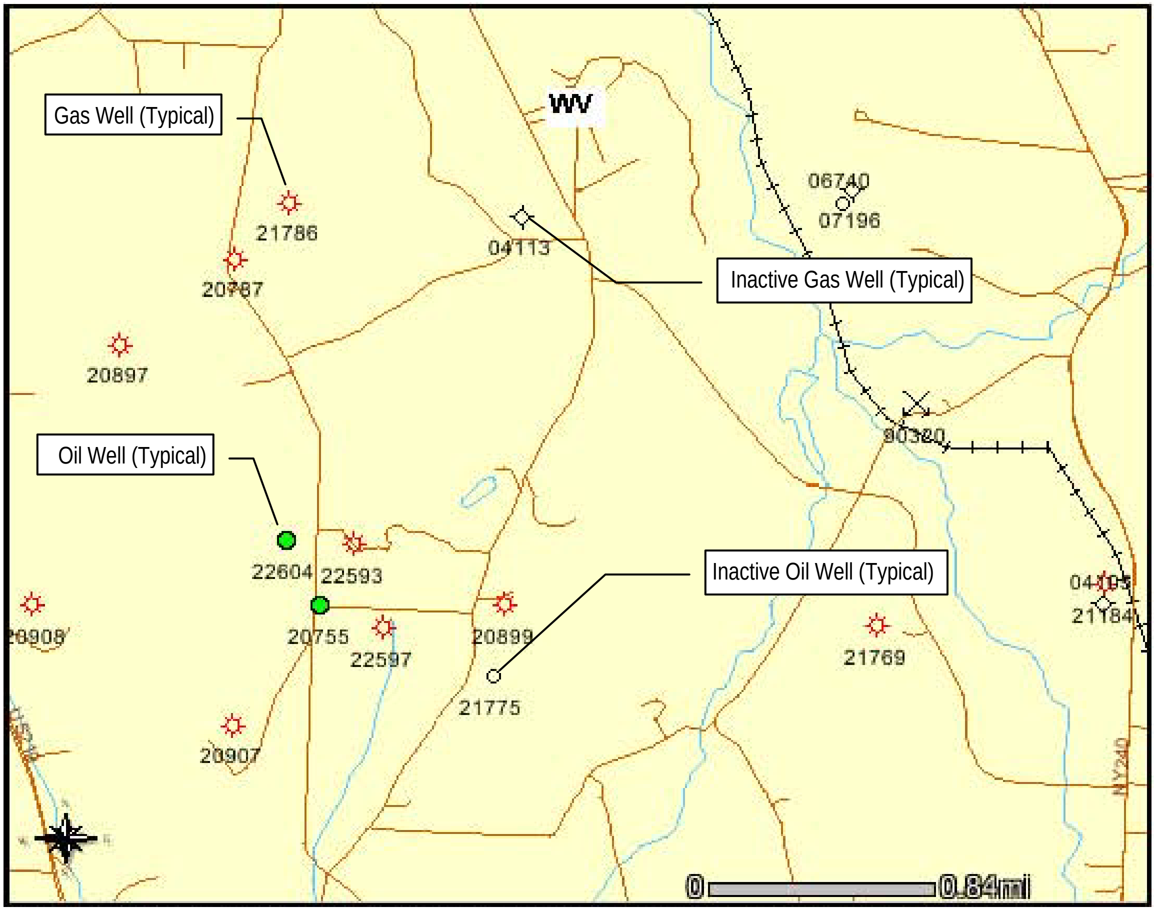



Figure 3-70. Locations of Natural Gas and Oil Wells in the Vicinity of the WVDP

# 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 2009.
- 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 will 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 will be performed in connection with Phase 1 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), which explain the characterization methodology. 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 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 will be performed in connection with Phase 1 decommissioning activities as described in Sections 7 and 9.)

**Background Radioactivity**

Limited data are available on background radioactivity in structures, 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 surveys of structures described in Section 9.4.5, sufficient data will be acquired to establish background levels in structures within the scope of the Phase 1 decommissioning activities.

**Origin of Site Radioactivity**

Radioactivity associated with the project premises originated from 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 shows the location of WMAs on the project premises and Figures 4-2, 4-3, 4-4, and 4-5 show the locations of the facilities of interest. This list does not include facilities existing in 2009 that will be removed before the Phase 1 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 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)

**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

[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 within the scope of this plan.

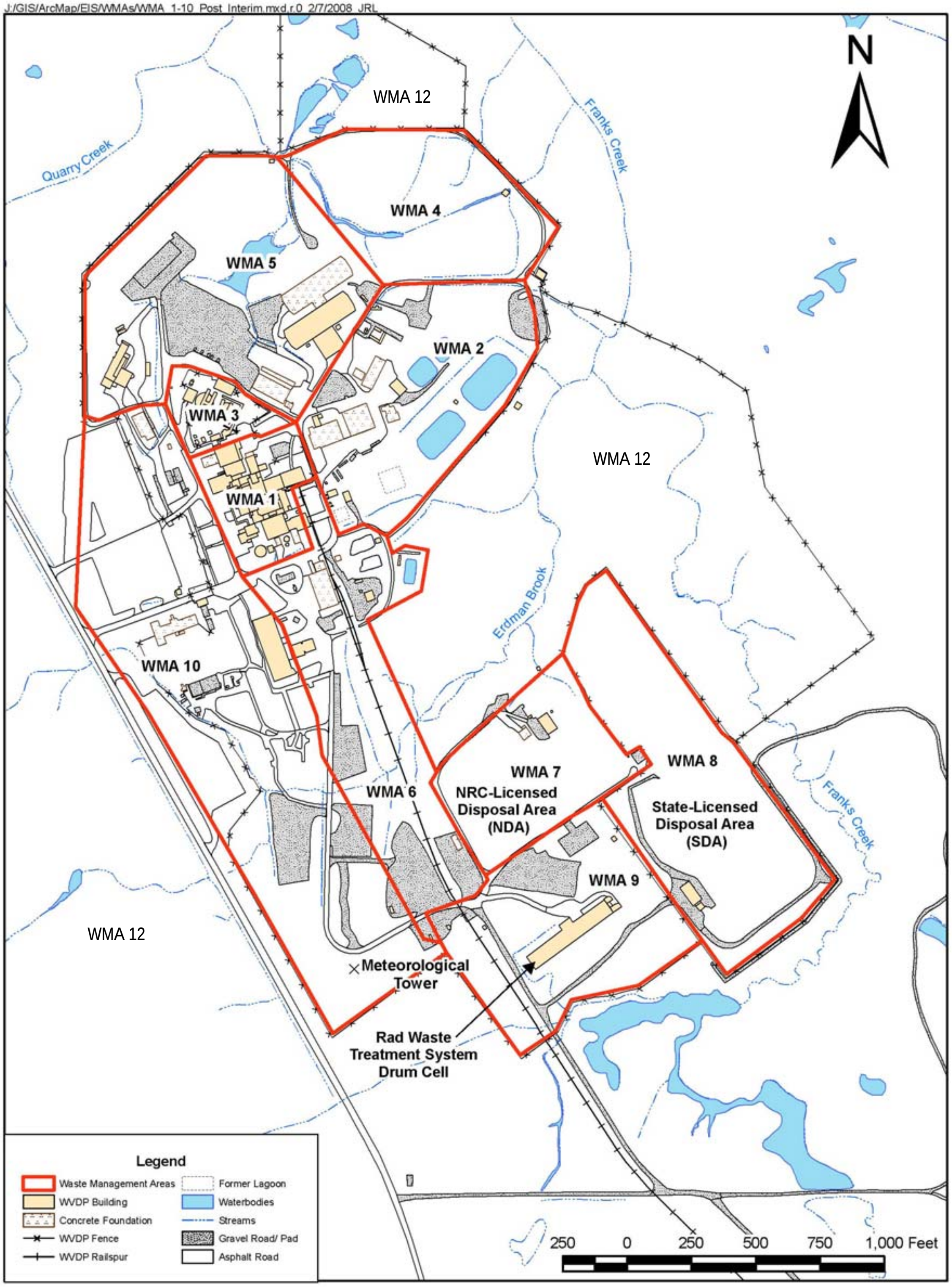


**Figure 4-1. Location of WMAs on the Project Premises**

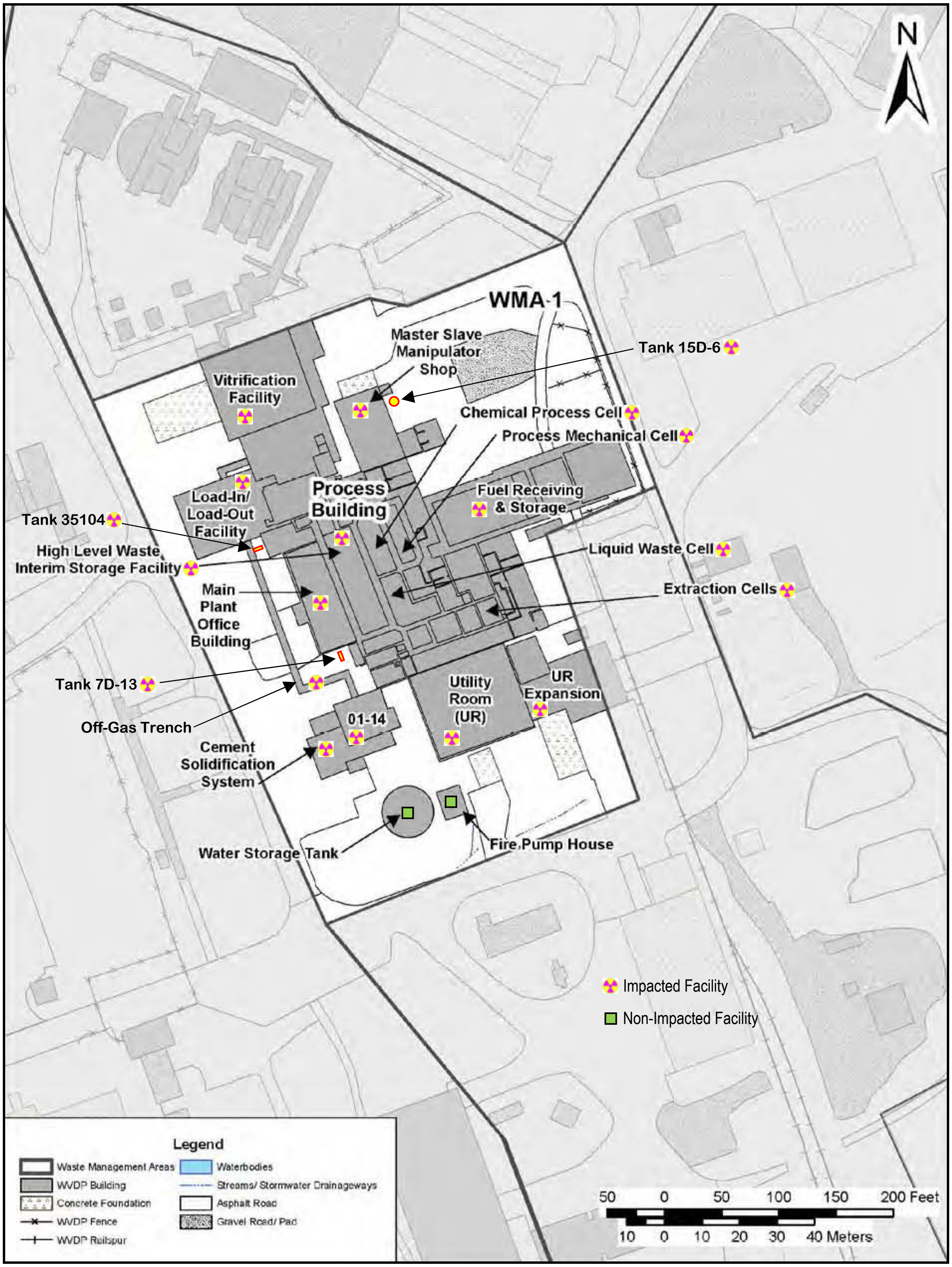


**Figure 4-2. Impacted and Non-Impacted Facilities in WMA 1**

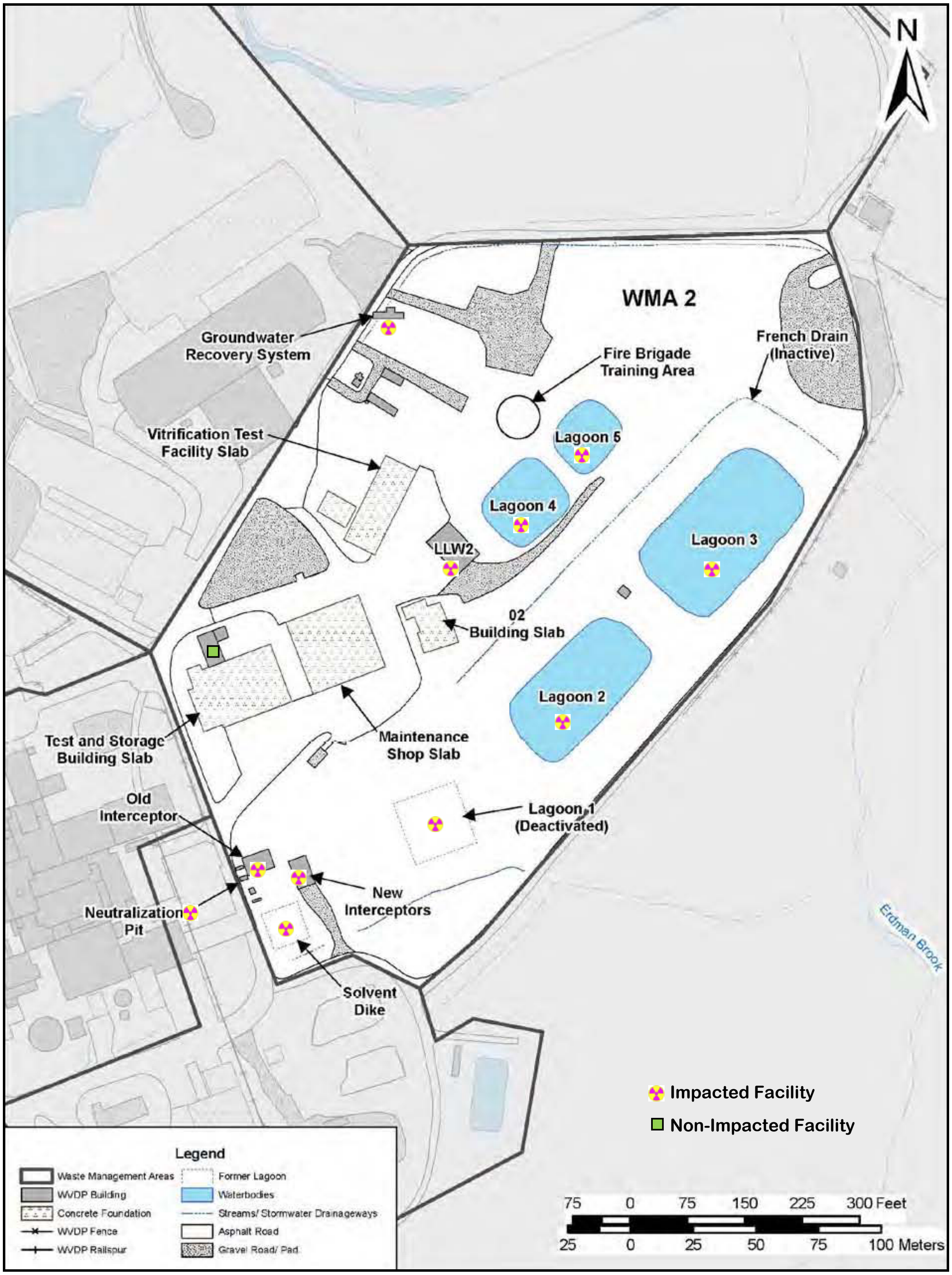


**Figure 4-3. Impacted and Non-Impacted Facilities in WMA 2**

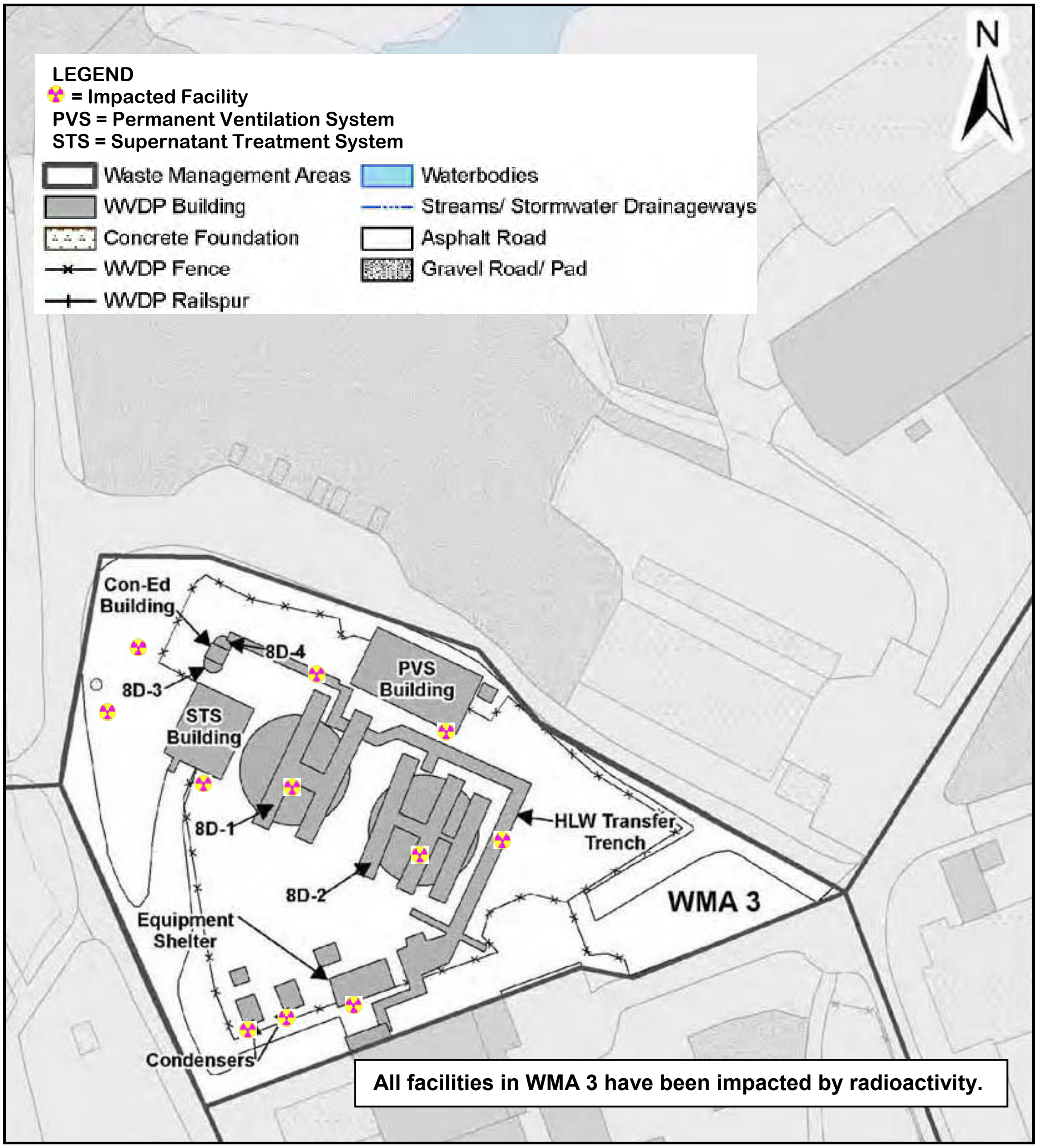


**Figure 4-4. Impacted Facilities in WMA 3**

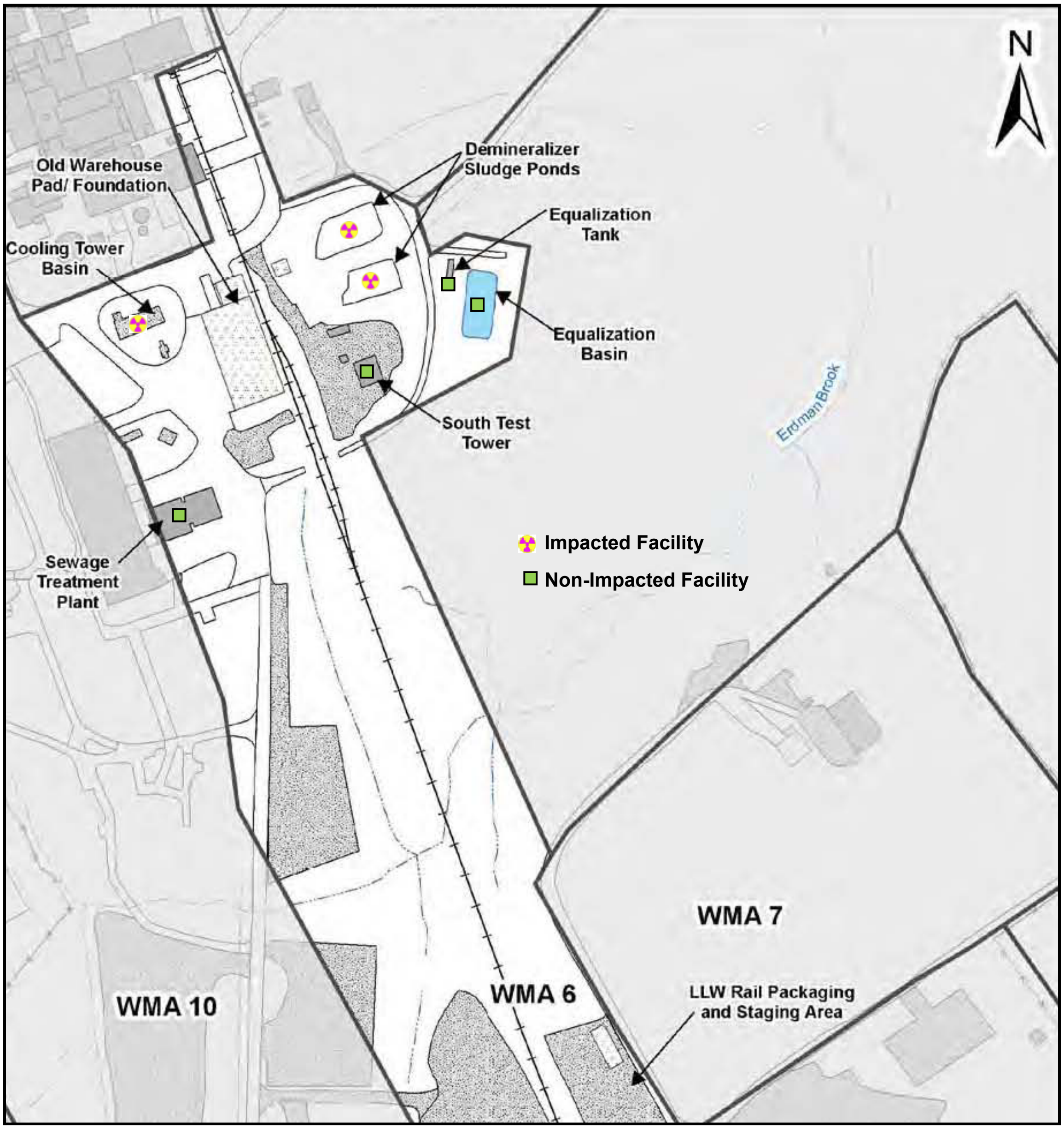


**Figure 4-5. Impacted and Non-Impacted Facilities in WMA 6**

### 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 2009, based on process history, the results of routine radiological surveys, and the results of the WVDP environmental monitoring program (WVES and URS 2009). 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 2009, the excavation dug for its removal will 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 will 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 will 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 between different areas of the Process Building and in other facilities of interest such as the Vitrification Facility depending on the point in the reprocessing cycle where the contamination originated. Other factors discussed below also influenced radionuclide distributions inside the Process Building and the Vitrification Facility.

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 and their application to portions of site facilities 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).

The method used to establish the ratios for U-232, U-233, U-234 and U-236, which involved use of analytical data from a sample collected in the Acid Recovery Pump Room (Mahoney 2002), may have somewhat underestimated the amounts of these radionuclides with respect to the amount of U-238 in areas of the Process Building further downstream. However, as the uranium isotopes are only a small fraction of the alpha-emitting radionuclides in the residual radioactivity (Am-241 and Pu-239 are ~1000 times greater), the impact of underestimating uranium inventory (U-232, U-233, U-234, and U-236) is not significant.

**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 Supernatant Treatment System 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 | µCi/g | Nuclide | µCi/g | Nuclide | µCi/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 | NA(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) Not available. No analysis was performed for U-232.

**Process Building Distributions**

During the Facility Characterization Project, the spent fuel distribution, the Batch 10 distribution, and area-specific radionuclides distributions were used in conjunction with sample analytical data to determine the appropriate radionuclide activity inventory for various representative areas of the Process Building. For example, in the calculation of the bounding radionuclide activity inventory of the Product Purification Cell, where uranium concentrations would be expected to be highest, the radionuclide distribution was determined from six samples obtained from the floor and walls of this cell, rather than using the spent fuel distribution, which would be more representative of radionuclides in earlier steps of the process stream (Choroser 2003).

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 spent fuel distribution in Table 4-1.

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. There are significant variations from these relative fractions in the different areas for which data were compiled.

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

**Vitrification Facility Distributions**

The other facility with a significant amount of residual radioactivity is the Vitrification Facility. 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 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(2) | 0.75 |
| Cm-243 | 0.27 | Pu-239 | 63 | U-233(2) | 0.41 |
| Cm-244 | 6.3 | Pu-240 | 47 | U-234(2) | 0.19 |

**Table 4-5. Estimated Process Building Residual Activity at Start of Decommissioning**[1]

| Nuclide | Estimate (Ci) | Nuclide | Estimate (Ci) | Nuclide | Estimate (Ci) |
|---|---|---|---|---|---|
| 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).

(2) The estimated amounts of these radionuclides could be somewhat low due to the manner in which their scaling ratios to U-238 were initially developed (Mahoney 2002). However, as the uranium isotopes are only a small fraction of the alpha-emitting radionuclides in the residual radioactivity (Am-241 and Pu-239 are ~1000 times greater), the impact of underestimating Uranium inventory (U-232, U-233, U-234, and U-236) is not significant.

Table 4-6 shows the total estimated residual radioactivity inventory 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 |
| 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 have been 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 |
| 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 (WVES 2008f)[6]. 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 decommissioning activities. Essentially all of this radioactivity is in the Vitrification Cell.

[6] The estimated amounts of other radionuclides are as follows: 35 curies of Ni-63, 189 curies of Sm-151, 19 curies of Pu-238, 5 curies of Pu-239, 4 curies of Pu-240, 175 curies of Pu-241, 153 curies of Am-241, 10 curies of Cm-242, and 35 curies of Cm-244 (WVES 2008f).

**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 filters 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.[7] 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.

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 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)[8].

[7] This conclusion is based primarily on records showing that 22,400 cubic feet of sediment were 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.

[8] 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

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

| Nuclide | Estimate (Ci) | Nuclide | Estimate (Ci) | Nuclide | Estimate (Ci) |
|---|---|---|---|---|---|
| Am-241 | 391 | Np-237 | 0.55 | Tc-99 | 12 |
| C-14 | 0.036 | Pu-238 | 164 | 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 | 301,000 | Pu-241 | 578 | U-235 | 0.005 |
| I-129 | 0.018 | Sr-90 | 35,400 | 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.

In October 2009, the liquid levels in inches from the tank bottoms were as follows:

Tank 8D-1 – 7.75 inches    Tank 8D-2 - <2.5 inches

Tank 8D-3 – 28.0 inches    Tank 8D-4 – 78.8 inches.

Preparations were being made in late 2009 to remove and process liquid from Tank 8D-4.

A Sampling and Analysis Plan for the Waste Tank Farm was prepared in 2009 (WVES 2009a). This plan provides for additional characterization of each of the four underground waste tanks. If this plan were to be fully implemented, it would provide additional data on residual radioactivity within each tank, including in the Supernatant Treatment System equipment that is inside Tank 8D-1.

Note that conditions in the underground waste tanks will change after the Waste Tank Farm and Vault Drying System described in Section 3 begins operation. This system is designed to dry (remove) 2000-4000 gallons of liquid from Tank 8D-1 per year and the same amount from Tank 8D-2, along with 100-400 gallons of liquid from Tank 8D-3 per year and this amount from Tank 8D-4 (WVES 2009c). This system may be operational when Phase 1 decommissioning activities begin. The amounts of residual activity listed in Table 4-9 will diminish slightly as the liquid evaporates during drying system operation.

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

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.

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.

As explained in Section 3, the transfer pumps in Tanks 8D-3 and 8D-4 will be removed before Phase 1 of the 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 2009 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.[9]

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

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

NOTE: (1) From 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 decommissioning activities begin had been impacted by site radioactivity as of 2009.

[9] 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 was performed in 2008. This characterization focused on hazardous contaminants and radionuclides in the source area of the north plateau groundwater plume (Michalczak 2007). The results have been incorporated into this plan.
>
> 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 northwest 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[10] within the project premises.
- Section 4.2.4 provides the same information on subsurface soil.

[10] Sediment in this context includes stream sediment, lagoon sediment, and drainage ditch sediment.

- 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 of 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 2008 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, URS 2002, Klenk 2009, Michalczak 2009b, and WVES and URS 2009). 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 Laboratory Information Management System.

### 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 radionuclides 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 radionuclide 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 radionuclides, 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.[11] 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 (pCi/g dry) | Subsurface soil (pCi/g dry)[(2)] | Sediment (pCi/g dry) | Surface water (pCi/L) | Groundwater (pCi/L) |
|---|---|---|---|---|---|
| Gross alpha | 1.3E+01 (2.7E+01) | 1.3E+01 (1.7E+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) | 2.9E+01 (6.1E+01) | 1.6E+01 (2.7E+01) | 2.3E+00 (2.0E+01) | 4.6E+00 (2.8E+01) |
| H-3 | NA | NA | NA | <8.2E+01 (6.3E+02) | <8.6E+01 (9.4E+02) |
| C-14 | NA | NA | NA | <1.3E+01 (4.1E+02) | <2.7E+01 (7.4E+00) |
| Sr-90 | 9.5E-02 (3.1E+00) | <2.3E-02 (1.2E-01) | <3.4E-02 (1.6E-01) | 9.0E-01 (1.2E+01) | 2.4E+00 (7.4E+00) |
| Tc-99 | NA | NA | NA | <1.8E+00 (7.3E+00) | <1.8E+00 (4.0E+00) |
| I-129 | NA | NA | NA | <7.9E-01 | <6.0E-01 |

[11] 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 (pCi/g dry) | Subsurface soil (pCi/g dry)[2] | 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) | <2.4E-02 (1.5E-01) | 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) | <2.4E-02 (<4.2E-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) | 7.9E-01 (1.1E+00) | 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) | 4.2E-02 (1.2E-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) | 8.6E-01 (1.1E+00) | 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) | <1.2E-02 (<2.4E-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) | <1.0E-02 (<1.9E-02) | <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) | <1.1E-02 (<1.3E-02) | <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) This column was added after sufficient background soil samples were collected in 2008 to allow for comparison purposes.

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 2008, the earliest was from a sample collected in 1967.[12] 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

[12] 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.

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.

**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).[13] 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.)

### 4.2.3 Radiological Status of Surface Soil and Sediment

Since the facility has operated, numerous soil sampling studies have been conducted onsite, 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

[13] 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.

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.[14]

- 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 will 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.
- 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

[14] 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.

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[15] 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) alpha-emitting radionuclides and (2) beta-gamma emitting radionuclides.

NOTE

The information provided below does not include data from characterization measurements for Sr-90 in subsurface soil, surface water, and groundwater collected during a 2008-2009 investigation to support design of mitigation measures for the leading edge of the north plateau groundwater plume. The results of this characterization can be found in report WVDP-500 (WVES 2009b).

This characterization program focused on conditions in the northern part of WMA 2 and in WMA 4. Direct-push soil borings and groundwater samples were obtained using a Geoprobe® unit. A total of 63 soil samples were analyzed for Sr-90. In addition, 74 microwells were installed to collect groundwater in the sand and gravel unit.

Data from this characterization program has redefined the leading edge of the north plateau groundwater plume, which is now known to form three small lobes as shown in Figure 4-14. This configuration appears to result from the uneven distribution of coarse and fine sediment within the sand and gravel unit, which affects local groundwater flow rates.

[15] Near-surface in this context means a few feet from the surface.

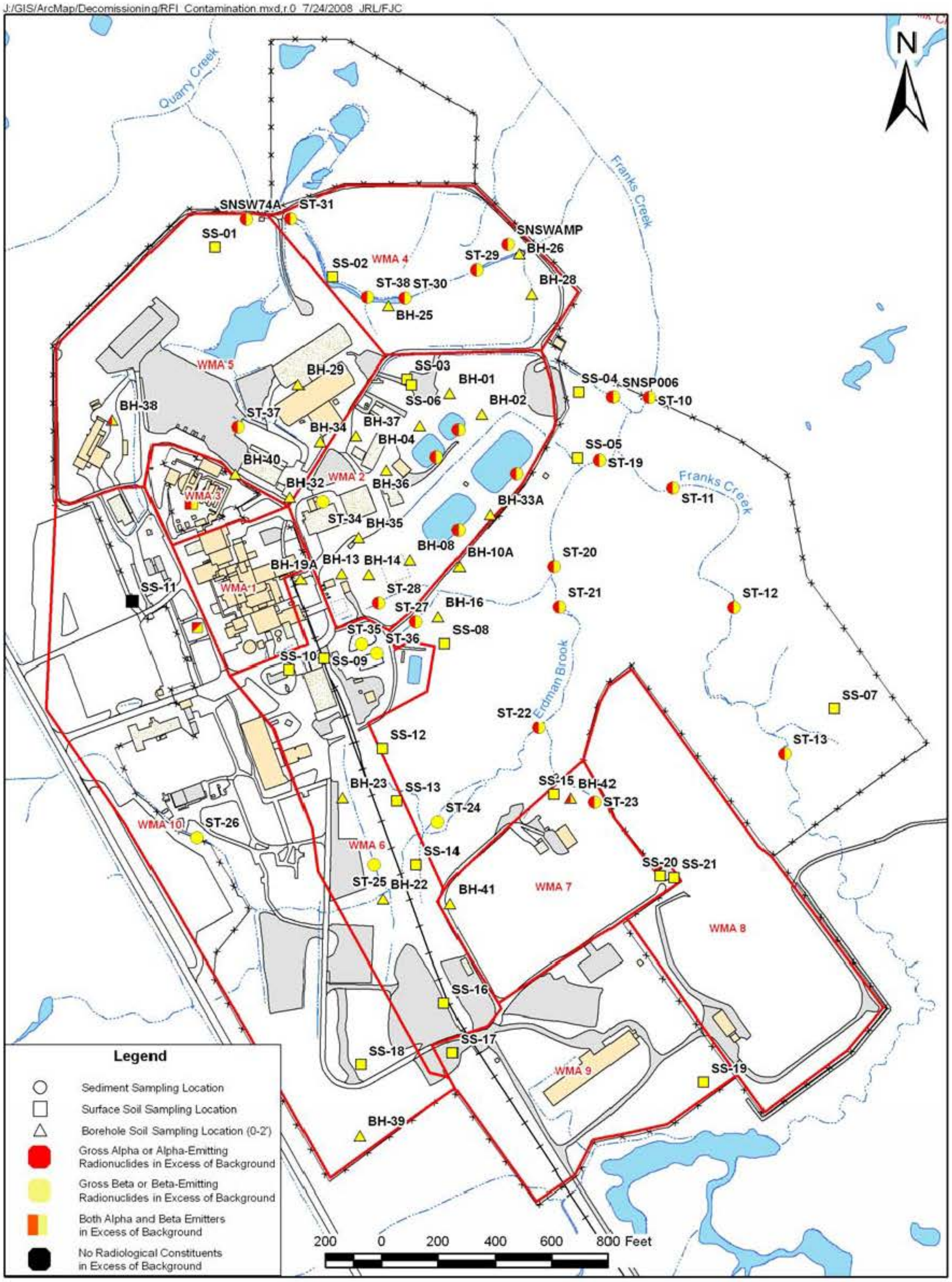


**Figure 4-6. Surface Soil and Sediment Locations With Radionuclide Concentrations in Excess of Background**

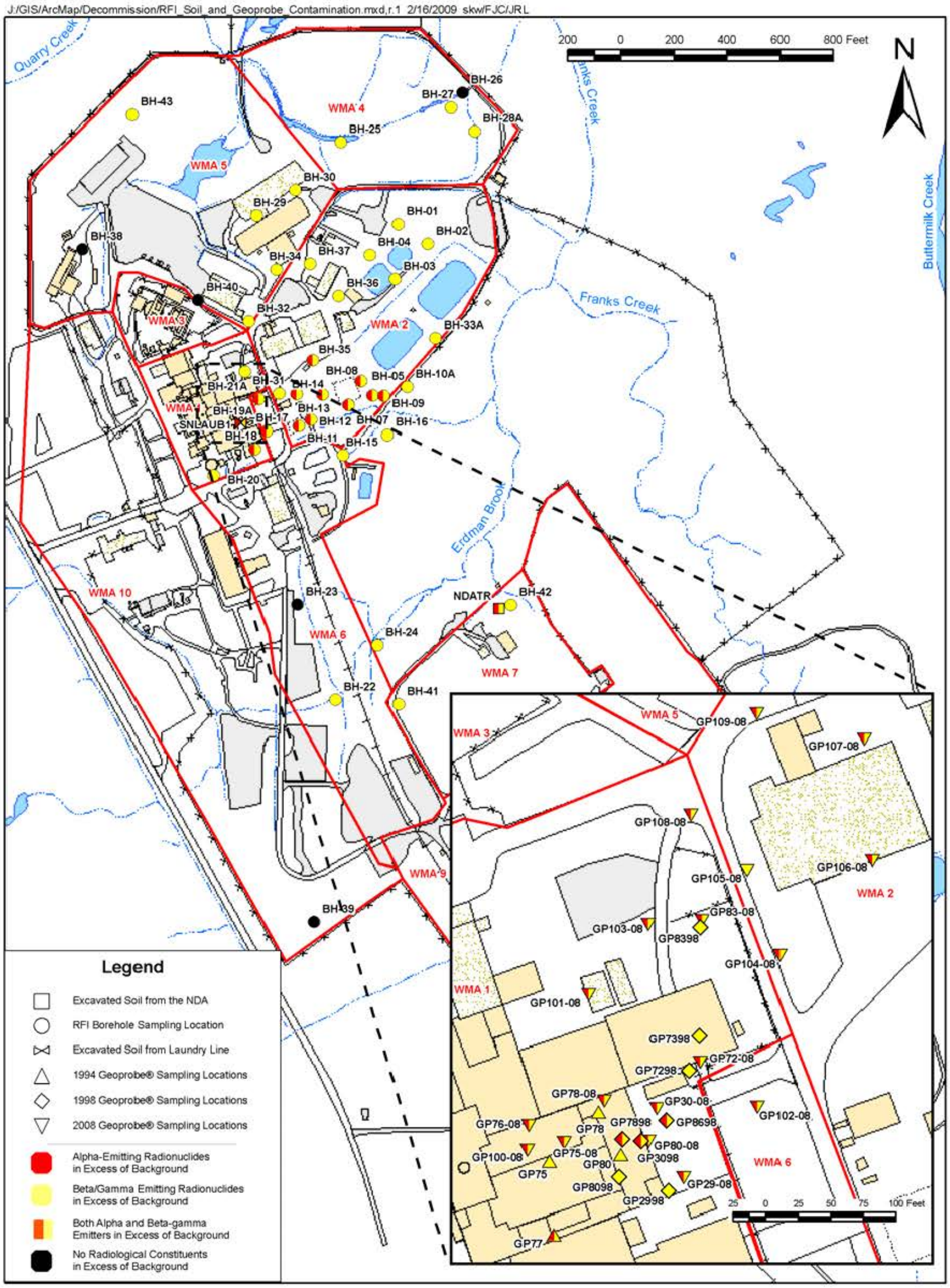


**Figure 4-7. Subsurface Soil Locations With Radionuclide Concentrations in Excess of Background**

Most subsurface soil data were taken from the 1993 RCRA Facility Investigation sampling program and three Geoprobe® sampling efforts (1994, 1998, and 2008) 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, 1998, and 2008 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 containing excavated soil generated at or near the NDA. Soil was largely from the Interceptor Trench, immediately downgradient of the NDA, when it was installed in 1990, and from nonspecific "special holes" (WVNSCO 1997c). Although the packaged soil has since been shipped offsite, it is likely that radionuclide concentrations in subsurface soil remaining in the NDA will 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.

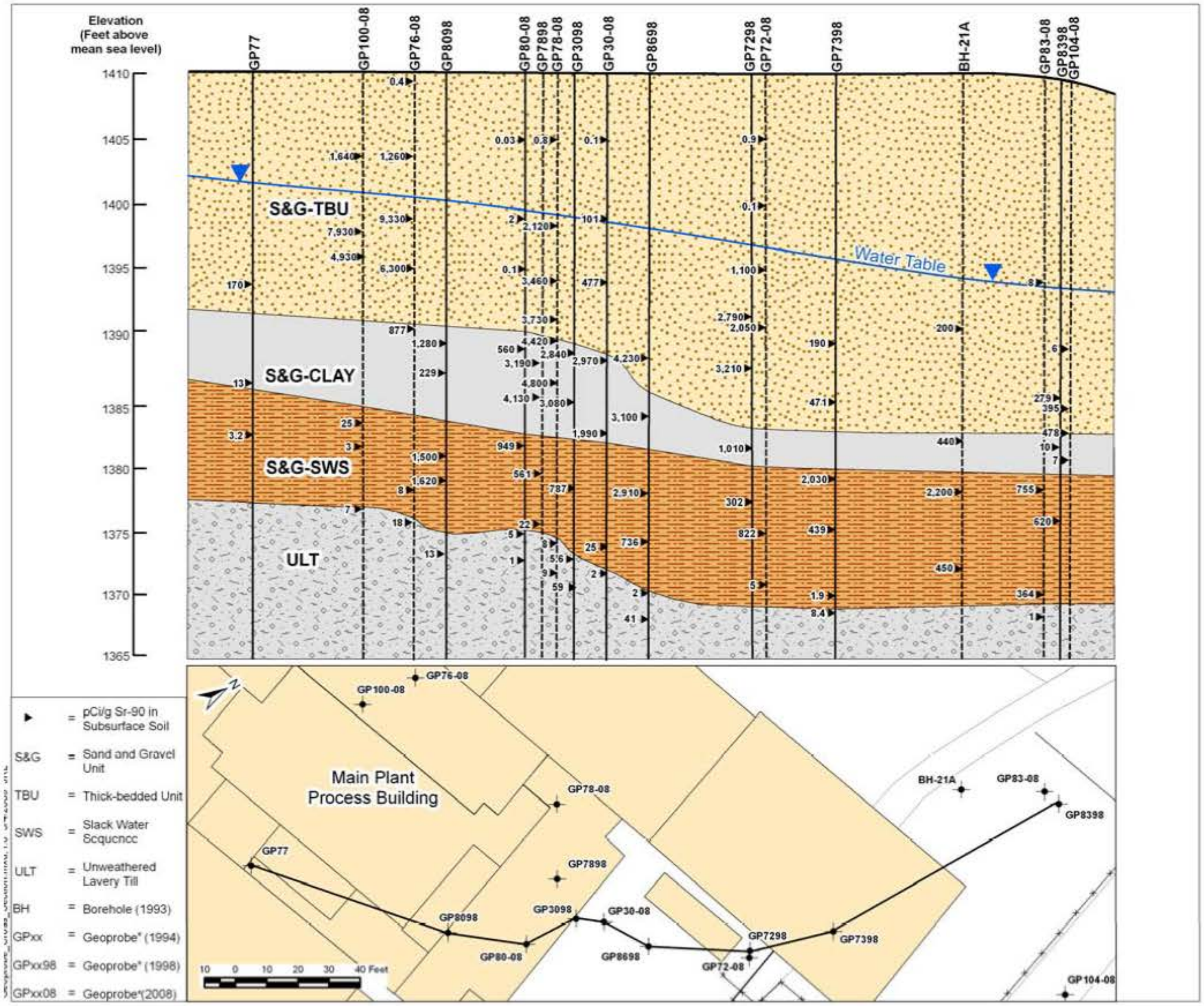


**Figure 4-8. Cross-section of Sr-90 Concentrations Versus Depth in Subsurface Soil in WMA 1**

### 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-"). Subsurface Geoprobe® soil sample locations are designated "GP." 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 Geoprobe® Investigations in 1994, 1998, and 2008, and from three borehole locations from the 1993 RCRA Facility Investigation. Additional data were taken from one sample collected in 2004 near a breach in an underground wastewater line near the laundry.

Above-background concentrations in subsurface soil from WMA 1 were noted for Sr-90, Cs-137, U-232, U-233/234, U-235/236, U-238, Pu-238, Pu-239/240, and Am-241. Maximum radionuclide concentrations are listed in Table 4-12. Except for the Cs-137 and Am-241 maxima observed from the sample near the laundry line breach, all maxima were from samples taken in 2008 under the Process Building. Maxima from Geoprobe® locations were found at depths of 14 to 42 feet in the saturated layer of the sand and gravel unit. High ratios of Sr-90 to Cs-137 observed in WMA 1 (with a median ratio of about 300 to 1 and a maximum ratio of over 63,000 to 1 [see Table B-11 in Appendix B]) reflect the influence of the north plateau groundwater plume. Maximum ratios of other radionuclides to Cs-137 in WMA 1 were: U-232 (0.023 to 1), U-233/234 (12 to 1), U-235/236 (1.1 to 1), U-238 (18 to 1), Pu-238 (0.18 to 1), Pu-239/240 (0.80 to 1) and Am-241 (2.7 to 1).

**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 | U-232 | U-233/234 | U-235/236 | U-238 | Pu-238 | Pu-239/240 | Am-241 |
| Note (2) | 3.3E+03 | 9.3E+03 | 5.0E-02 | 1.9E+00 | 2.2E-01 | 1.7E+00 | 5.6E-01 | 3.7E+00 | 8.7E+01 |

NOTES: (1) See Figure 4-2 for a map of facilities in WMA 1.

(2) Sampling related to laundry line breach in 2004 (Cs-137, Am-241); Geoprobe® sampling underneath Process Building in 2008 (GP7608 at 15-17' depth [Sr-90]; GP10408 at 20-22' depth [U-232]; GP7608 at 38-40' depth [U-233/234]; GP8308 at 40-42' depth [U-235/236]; GP2908 at 14-16' depth [U-238], GP7608 at 19-21' depth [Pu-238, Pu-239/240].

**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 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 |
| Surface soil north of Lagoons 4 and 5 (BH-04) | 8.5E-01 | 1.2E+00 |

**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 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 |
| 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 |

**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 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.

Above-background concentrations in subsurface soil from WMA 2 were noted for Sr-90, Cs-137, U-232, U-233/234, U-235/236, U-238, Pu-238, Pu-239/240, and Am-241. Maximum radionuclide concentrations at various points in WMA 2 are listed in Table 4-15. The highest concentrations of all radionuclides were found in saturated soil six-to-eight feet deep from one location (BH-8) downgradient of Lagoon 1. Other maxima were also found in samples taken under the Solvent Dike and downgradient of the interceptors in saturated soil in the sand and gravel unit.

As noted in the WMA 1 discussion, ratios of Sr-90 to Cs-137 were also elevated in WMA 2, downgradient of the source of the north plateau plume. However, ratios were much lower than in WMA 1 (i.e., a median ratio of 1.9 to 1 and a maximum of 750 to 1 [as compared with the median of about 300 to 1 and the maximum of over 63,000 to 1 in WMA 1). Maximum ratios of other radionuclides to Cs-137 in WMA 2, as summarized in Table B-11, were: U-232 (1 to 1), U-233/234 (7 to 1), U-235/236 (1.1 to 1), U-238 (4.4 to 1), Pu-238 (0.089 to 1), Pu-239/240 (0.11 to 1) and Am-241 (0.23 to 1).

**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, U-238 max at 6-8' 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 (BH-14 at 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 |
| East of the former TSB (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 |
| Downgradient of MPPB, near the former TSB [GP10508, 28-30' depth) | ≤Bkg | 7.6E+02 | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg |
| Downgradient of MPPB, south of the former Maintenance Shop (GP10608, at 20-22' depth [Sr-90] and at 22-24' depth [Am-241, U isotopes]) | ≤Bkg | 6.6E+01 | ≤Bkg | 9.0E-01 | 2.2E-01 | ≤Bkg | ≤Bkg | ≤Bkg | 3.4E-02 |
| Downgradient of MPPB, near Vit Test Facility (GP10708, at 30-32' depth [Sr-90] and at 12-14' depth [U-235/236]) | ≤Bkg | 3.8E+02 | ≤Bkg | ≤Bkg | 1.9E-01 | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg |
| Downgradient of MPPB, near area of the leach field for the former Maintenance Shop (GP10908, at 34-36' depth [Sr-90], and at 36-38' depth [U-232, U-238]) | ≤Bkg | 2.3E+02 | 1.3E-01 | ≤Bkg | ≤Bkg | 1.0E+00 | ≤Bkg | ≤Bkg | ≤Bkg |

NOTE: (1) See Figure 4-3 for a map of facilities in WMA 2. Facilities not labeled in Figure 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 4-7 for a map with the above sampling locations.

LEGEND: "≤Bkg" = Background was not exceeded. MPPB = Main Plant Process Building. TSB = Test and Storage Building.

**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 radionuclides 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 may be 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.

Sampling of subsurface soil by Geoprobe® in 2008 south of the Fuel Receiving and Storage Area, close to 1993 sampling locations BH-17 and BH-19A, continued to show above-background concentrations of most radionuclides. See Figure 4-7. As with WMA 1 and WMA 2, elevated ratios of Sr-90 to Cs-137 in the portion of WMA 6 lying between WMAs 1 and 2 (with a median of 174 to 1 and a maximum of 1115 to 1) reflected the influence of the north plateau groundwater plume. However, maximum concentrations of Cs-137 and Sr-90 in the subsurface saturated layer were lower than those observed in BH-17 and BH-19A in 1993.

**Table 4-19. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil From WMA 6(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 |
| Surface soil along rail spur south of STP (BH-23, SS-13) | 1.8E+00 | 3.2E-01 | NA | NA | NA | NA | NA | NA | NA |
| Sediment along drainage by rail spur south of STP (ST-25) | 2.1E+00 | 1.3E-01 | NA | NA | NA | NA | NA | NA | NA |

**Table 4-19. Above-Background Concentrations of Radionuclides in Surface Soil, Sediment, and Subsurface Soil From WMA 6(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 |
| Surface soil by FRS (1994 sampling near rail spur) | 1.6E+02 | 1.2E+01 | NA | NA | NA | NA | NA | NA | NA |
| Surface soil by Cooling Tower (SS-10) | 1.3E+01 | 1.4E+00 | NA | NA | NA | NA | NA | NA | NA |
| Surface soil by Old Incinerator (WVNSCO 1994, Table 3-2 [1990 data]) | 1.9E+01 | 2.3E+00 | NA | NA | NA | NA | NA | NA | NA |
| Surface soil by Old Warehouse (SS-09) | 1.3E+01 | 9.3E-01 | NA | NA | NA | NA | 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 | NA | NA | NA | NA |
| Sediment from South Demineralizer Sludge Pond (WVNSCO 1994 Table 3-19 [1988 data], ST-36) | 3.8E+01 | 3.5E-01 | NA | NA | NA | NA | NA | NA | NA |
| Subsurface soil near the Utility Room (BH-17, 14-16' depth) | 2.4E+00 | 2.7E+00 | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg | 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 | ≤Bkg | ≤Bkg | ≤Bkg | ≤Bkg | 1.5E-01 | 2.0E-01 | 8.0E-01 |
| Subsurface soil near rail spur south of the FRS (GP10208, 14-16' depth) | 1.1E+00 | 2.2E+02 | 9.1E-02 | 1.3E+00 | 3.5E-01 | 1.4E+00 | ≤Bkg | 4.9E-02 | 1.4E-01 |

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.

LEGEND: NA = Not analyzed. "≤Bkg" = Background was not exceeded. FRS = Fuel Receiving and Storage Building, STP = Sewage Treatment Plant |

**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 during construction of the Interceptor Trench or from nonspecific "special holes" 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 Interceptor Trench or “special holes” (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)(2) | 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.

(2) A total of 15 samples were collected in 1998 near Trailer City. Two samples showed approximately 1.0 pCi/g Cs-137, with Cs-137 in the other samples less than this concentration.

**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 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 radionuclides) 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.[16] 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. (Permitted effluent water discharged from lagoon 3 through WNSP001 often contains small but 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 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 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.

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[16] 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)

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 Table 4-23 may not be indicative of radionuclides in soil, but of radiation from the wastes being processed and/or stored nearby.

The onsite 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).

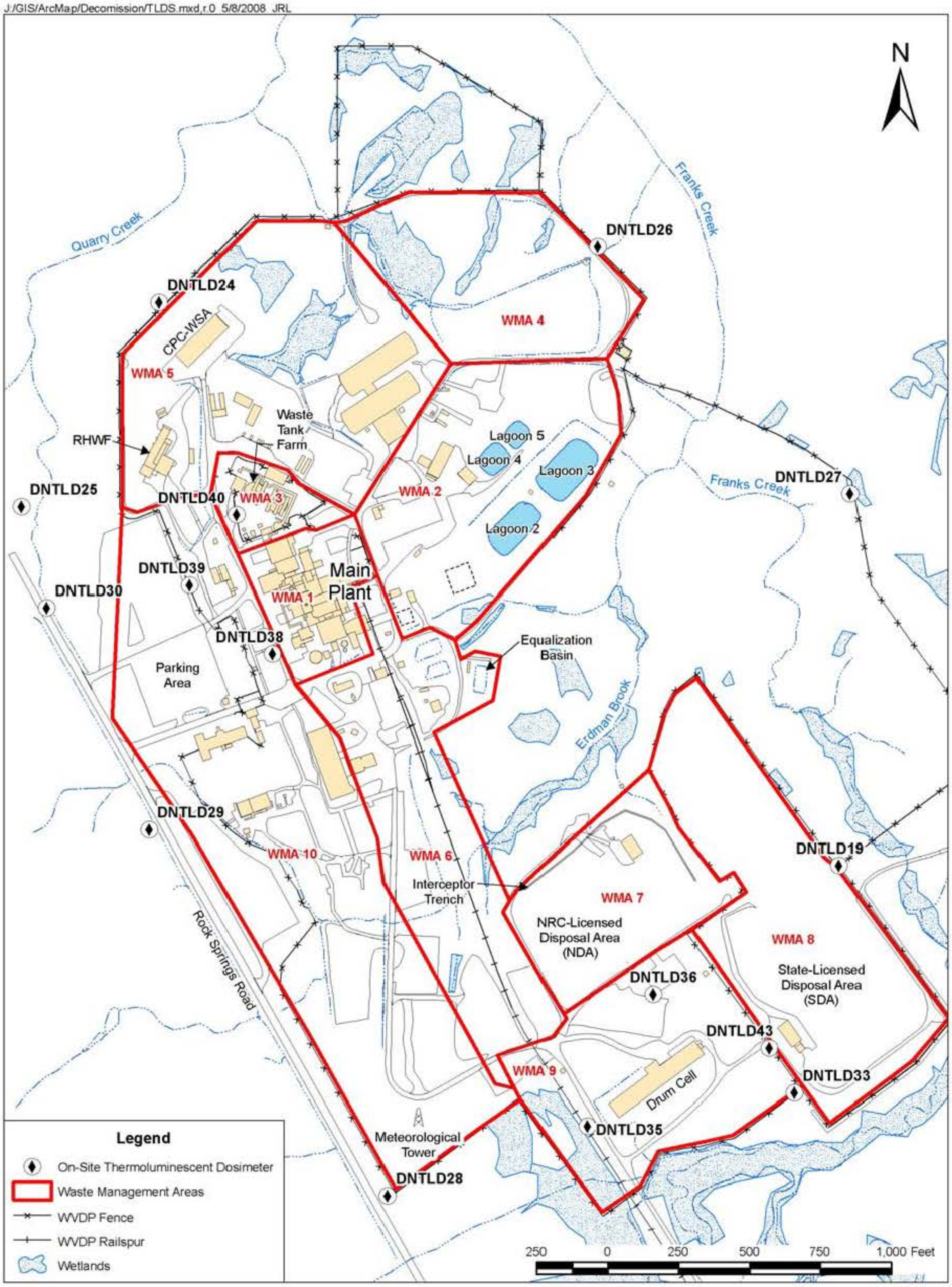


**Figure 4-9. Onsite Environmental TLD Locations**

**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 Process Building 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 Process Building 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 permitted 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*.

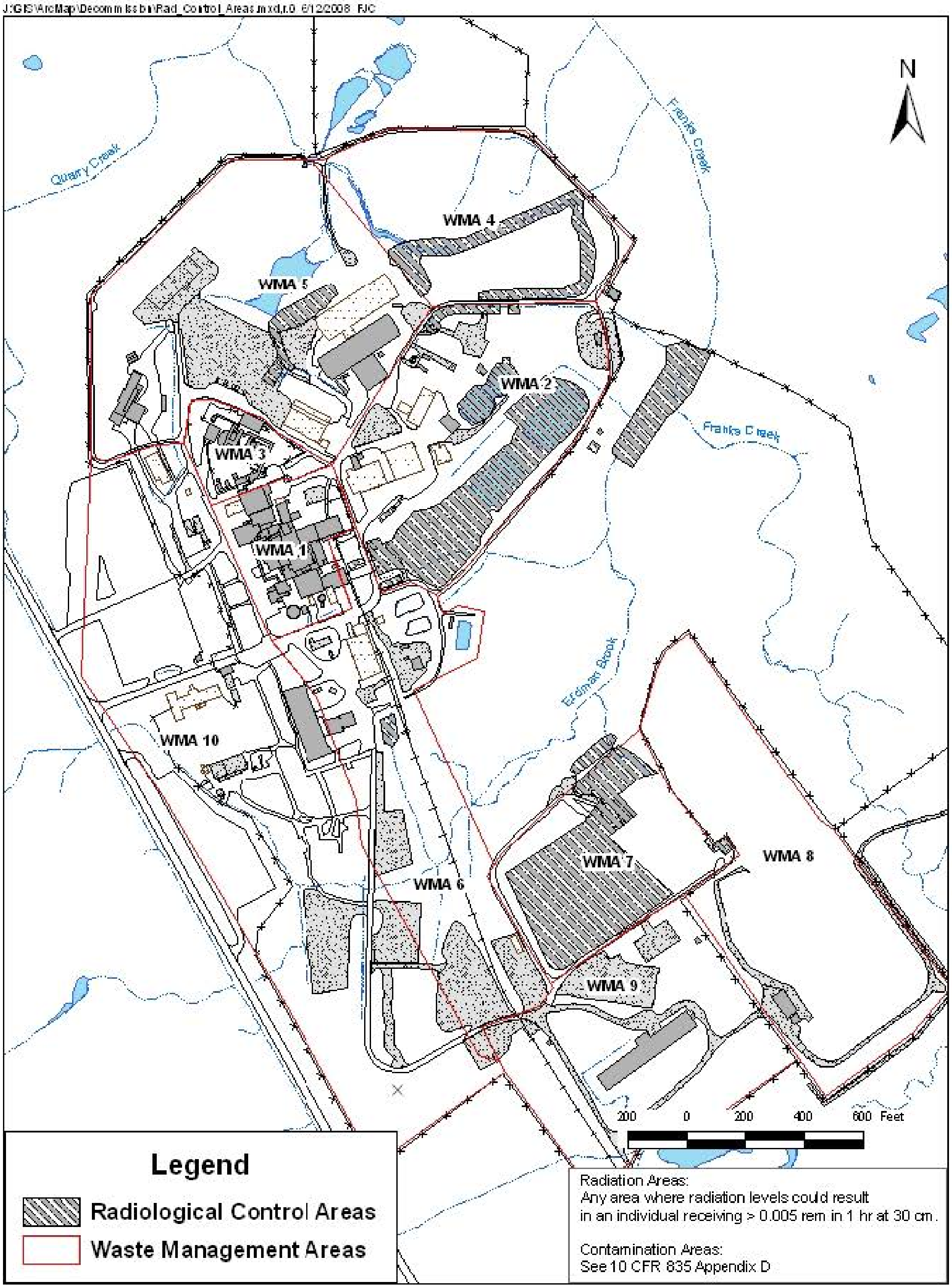


**Figure 4-10. WVDP Radiological Control Areas. (**Facilities with radiological controlled areas are outlined in black. Radiological Control Areas are current as of June 2008. )

### 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 permitted 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 permitted 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) permitted discharge points.

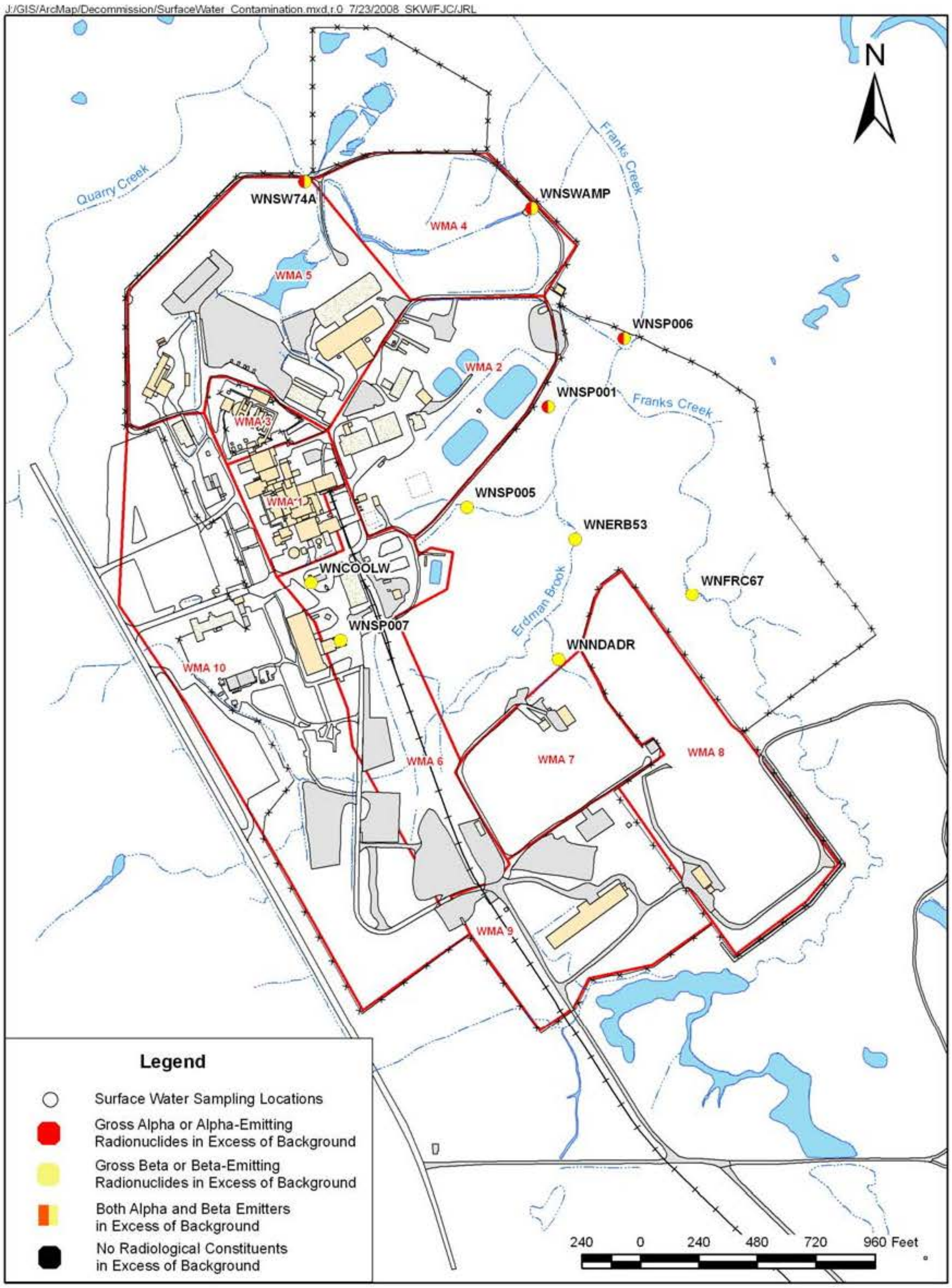


**Figure 4-11. Surface Water Locations with Radionuclide Concentrations in Excess of Background**

**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 |
| Pu-238 | < 3.4E-02 | 2.1E-02 | ± | 3.4E-02 | 1.4E-01 |

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

| Location | Median | Average | | | Maximum |
|---|---|---|---|---|---|
| | | Result | ± | Uncertainty | |
| **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

NOTE

The information provided below does not include data from characterization measurements for Sr-90 in subsurface soil and groundwater collected during a 2008-2009 investigation to support design of mitigation measures for the leading edge of the north plateau groundwater plume. However, results from this investigation were used to redefine the leading edge of the plume as shown in Figure 4-14. Complete results of this characterization can be found in report WVDP-500 (WVES 2009b).

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 |

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

| WMA | Monitoring Point | Constituent | Median | Average Result | ± | Uncertainty | Maximum |
|---|---|---|---|---|---|---|---|
| WMA 2 | | 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 |
| | | 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 |

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

| WMA | Monitoring Point | Constituent | Median | Average Result | ± | Uncertainty | Maximum |
|---|---|---|---|---|---|---|---|
| | | 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 |
| | | 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).

Groundwater was collected in 2008 in accordance with a sampling and analysis plan (Michalczak 2007) for a Geoprobe® characterization of the north plateau. Data from this sampling program have been included in the tables and figures for this section.

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.

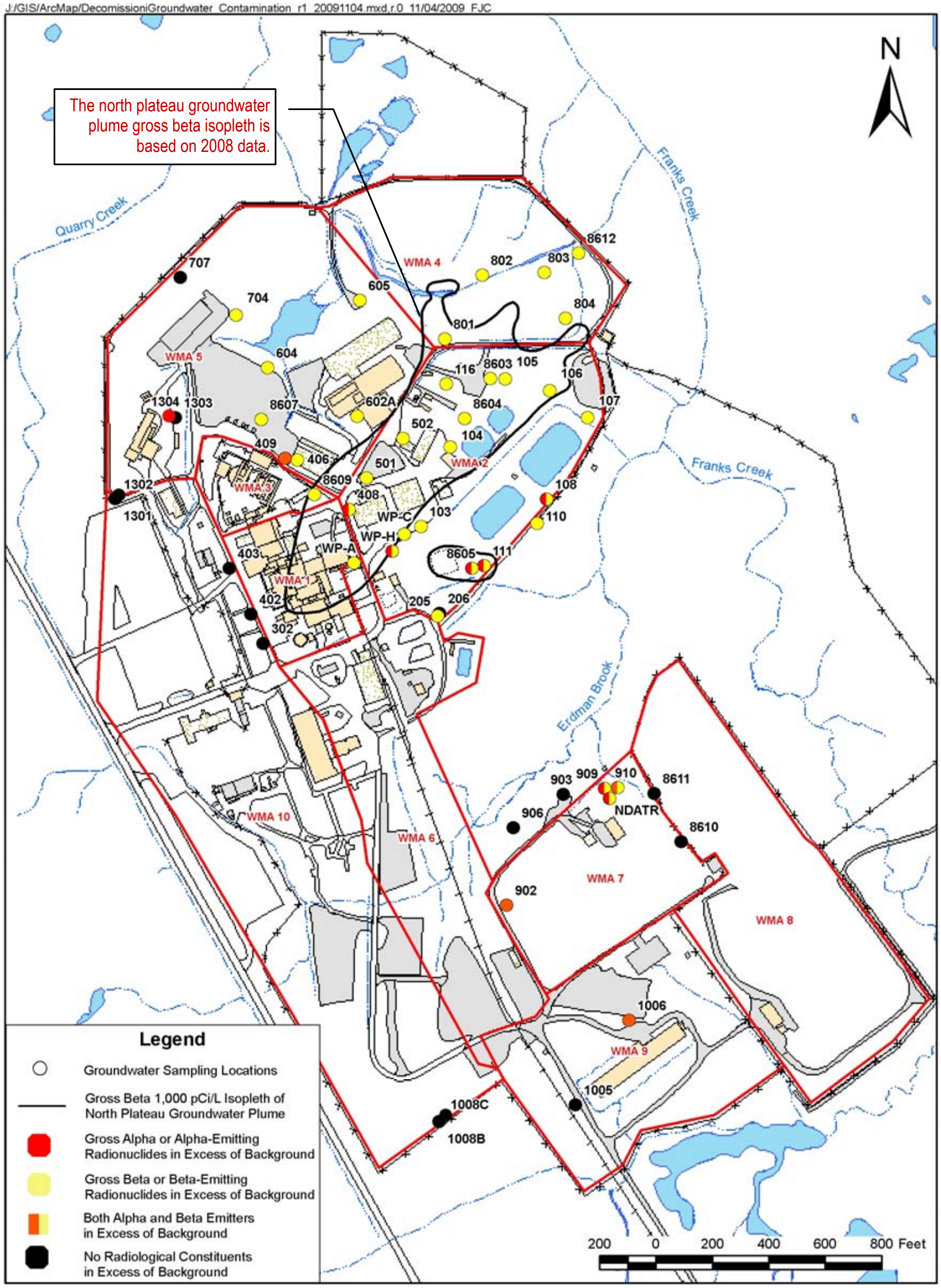


**Figure 4-12. Routine Groundwater Monitoring Locations with Radionuclide Concentrations in Excess of Background**

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 alpha-emitting radionuclides 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, which could be affected by naturally occurring radionuclides, were not included in Table 4-26 or Figure 4-13).

As can be seen in Figure 4-13, concentrations of beta/gamma-emitting radionuclides exceeding background are evident at most locations downgradient of the Process Building. Most non-impacted points were noted in WMA 5 northwest of the north plateau groundwater plume. Alpha-emitting radionuclide concentrations exceeding background were found immediately downgradient of the Process Building and downgradient of the Interceptors.

**Table 4-26. Maximum Above-Background Radionuclide Concentrations (pCi/L) at Groundwater Geoprobe® Points by WMA, Location, and Depth(1)**

| WMA | Point | Constituent | Maximum | Point | Constituent | Maximum |
|---|---|---|---|---|---|---|
| WMA 1 | GP8098 (22-24') | H-3 | 6.4E+04 | GP2908 (17-19') | U-232 | 1.0E+00 |
| | GP29 (27-29') | C-14 | 2.3E+03 | GP2908 (17-19') | U-233/234 | 1.1E+01 |
| | GP30 (18-20') | Sr-90 | 1.2E+06 | GP2908 (17-19') | U-235/236 | 4.6E-01 |
| | GP72 (30-32') | Tc-99 | 1.2E+04 | GP2908 (17-19') | U-238 | 1.2E+01 |
| | GP29 (21-23') | I-129 | 3.0E+01 | GP7608 (20-22') | Pu-239/240 | 4.5E-01 |
| | GP7608 (20-22') | Cs-137 | 1.2E+02 | GP76 (27-29') | Am-241 | 4.7E-01 |
| WMA 2 | GP47 (11-13') | H-3 | 3.4E+04 | GP44 (14-16') | U-233/234 | 3.7E+01 |
| | GP66 (30-32') | C-14 | 4.0E+02 | GP44 (14-16') | U-235/236 | 6.2E-01 |
| | GP8298 (20-24') | Sr-90 | 2.8E+05 | GP60 (12-14') | U-238 | 1.5E+01 |
| | GP68 (25-27') | Tc-99 | 5.8E+01 | GP59 (17-19') | Pu-238 | 4.5E+00 |
| | GP47 (11-13') | I-129 | 8.2E+01 | GP59 (17-19') | Pu-239/240 | 7.9E+00 |
| | GP46 (12-14') | Cs-137 | 1.5E+02 | GP59 (17-19') | Am-241 | 5.9E+00 |
| | GP44 (14-16') | U-232 | 7.8E+01 | - | - | - |
| WMA 3 | GP20 (15-17') | H-3 | 1.5E+03 | GP20 (15-17') | I-129 | 2.5E+00 |
| | GP20 (15-17') | Sr-90 | 5.2E+01 | - | - | - |
| WMA 4 | GP32A (5-7') | H-3 | 1.3E+03 | GP8998 (16-18') | Sr-90 | 6.5E+03 |
| WMA 5 | GP43 (12-14') | H-3 | 2.0E+04 | GP53 (14-16') | Tc-99 | 8.0E+01 |
| WMA 5 | GP40 (13-15') | Sr-90 | 3.8E+03 | GP43 (12-14') | I-129 | 4.6E+00 |

**Table 4-26. Maximum Above-Background Radionuclide Concentrations (pCi/L) at Groundwater Geoprobe® Points by WMA, Location, and Depth[(1)]**

| WMA | Point | Constituent | Maximum | Point | Constituent | Maximum |
|---|---|---|---|---|---|---|
| WMA 6 | GP70 (26-28') | H-3 | 6.8E+03 | GP70 (21-23') | Tc-99 | 3.1E+01 |
| | GP70 (16-18') | C-14 | 1.4E+02 | GP70 (21-28') | I-129 | 1.1E+01 |
| | GP70 (16-18') | Sr-90 | 2.8E+04 | - | - | - |
| WMA 12 | GP48 (7-9') | H-3 | 1.5E+03 | GP50 (8-10') | U-238 | 7.2E-01 |
| | GP50 (8-10') | Sr-90 | 1.3E+01 | - | - | - |

. NOTE: (1) Points ending with "97," "98," or "08" were collected in 1997, 1998, or 2008, 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 gross beta isopleth, was approximately 300 feet wide and 800 feet long in 1994. By 2002, the plume area had expanded to approximately 350 feet by 1050 feet, and by early 2009 to about 600 feet (at its widest point near the leading edge) by 1400 feet (WVES and URS 2009). See Figure 4-14. Additional data from investigations performed in recent years have better defined the extent of the plume.

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, 2002, and 2008. (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 11 routine groundwater monitoring locations that define the plume as of December 2008. Contour lines show a gradual lengthening and expansion of the plume toward the northeast, with the highest concentration (i.e., well 408 at 3.17E+05 pCi/L) near the Process Building and lower concentrations near the leading edge. Characterization sampling in 2008 has better defined the leading edge of the plume (WVES 2009b). The most recent delineation, as defined by the 1000 pCi/L gross beta isopleth, indicated that the leading edge was split into three lobes, and that the northern lobe is beginning to encroach on the Construction and Demolition Debris Landfill. 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.

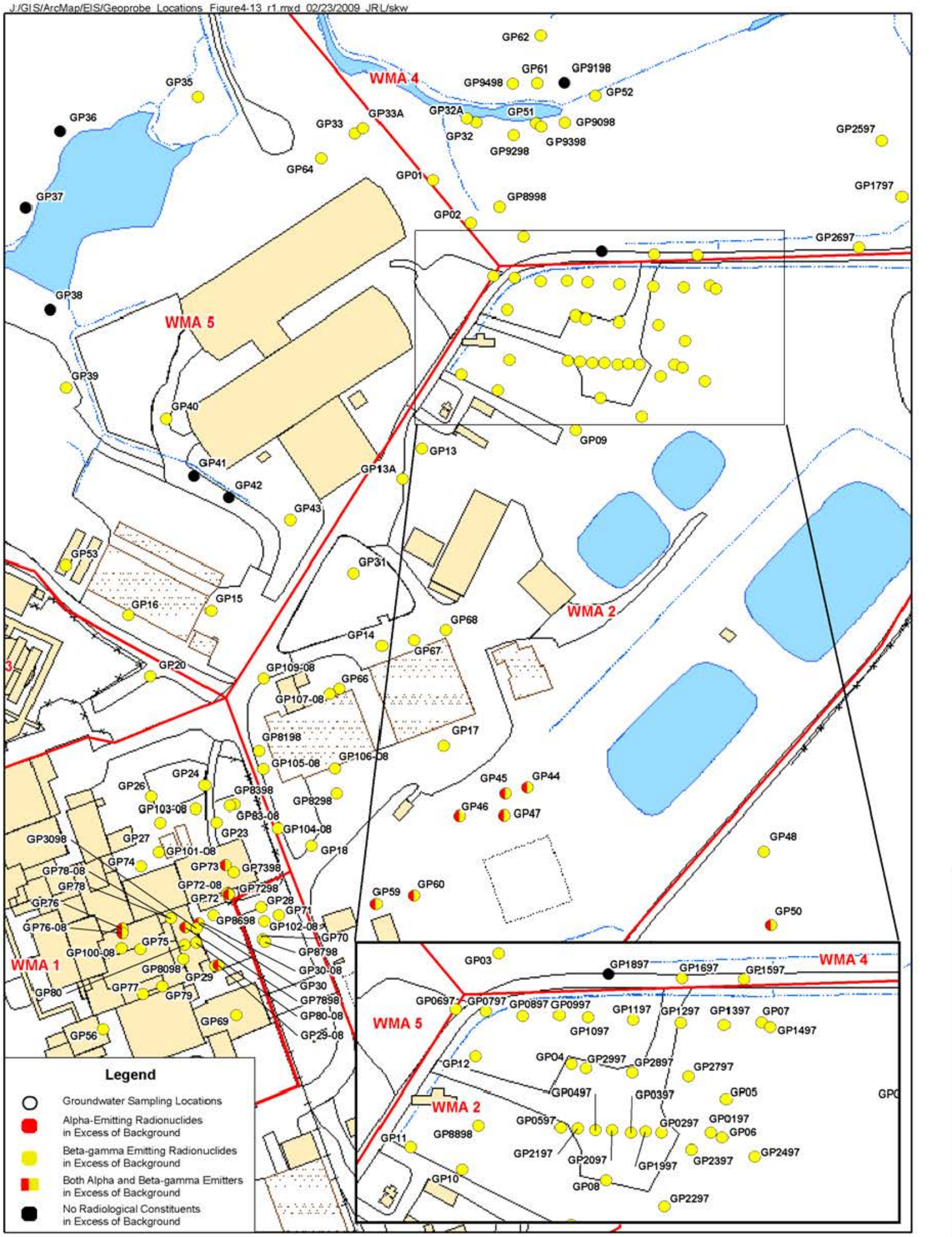


**Figure 4-13. Geoprobe® Groundwater Locations with Radionuclide Concentrations in Excess of Background**

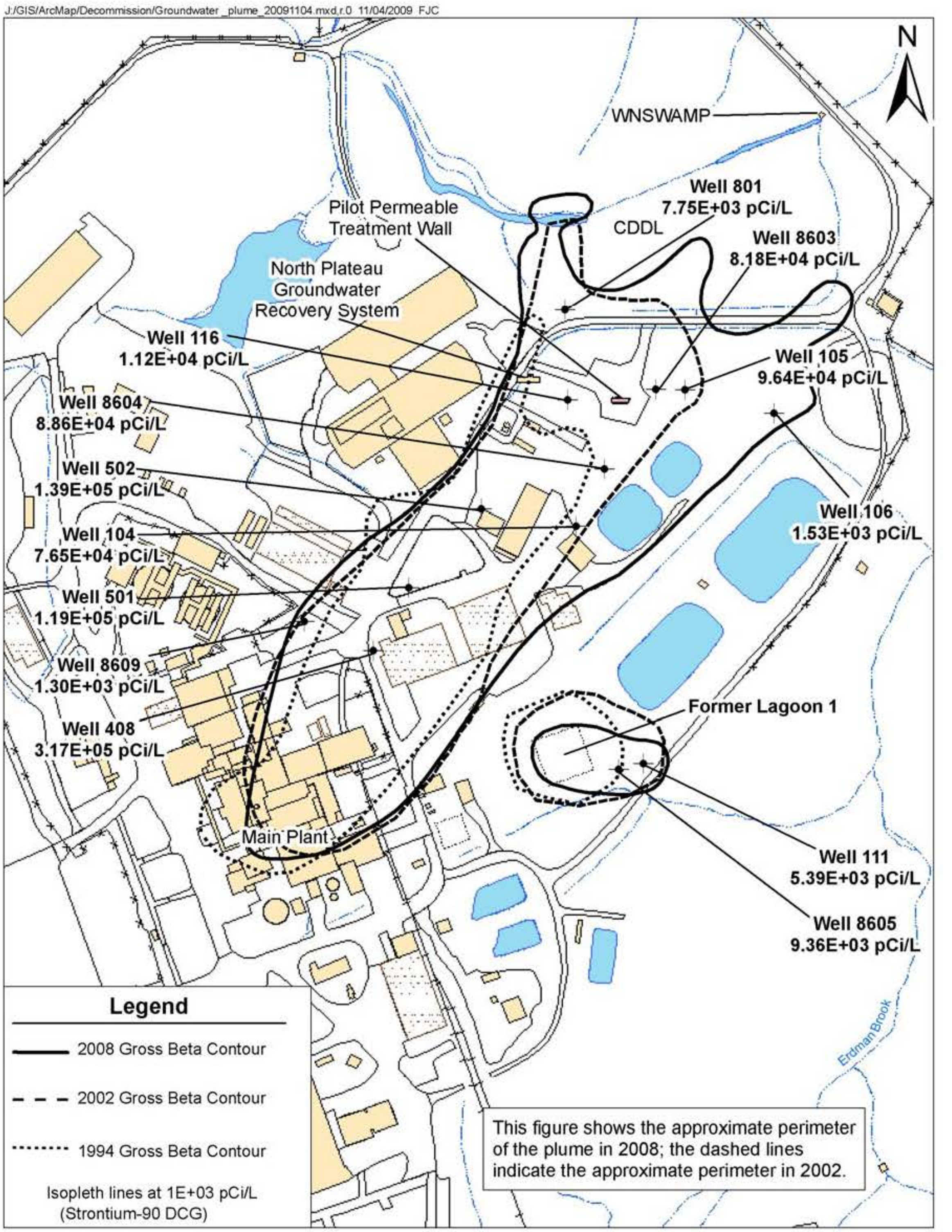


**Figure 4-14. North Plateau Groundwater Plume**

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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 decommissioning to establish cleanup criteria that will 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 base-case and alternative 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. It discusses the deterministic sensitivity analyses of model input parameters. It also describes the probabilistic uncertainty analysis and the multi-source model for subsurface soil DCGLs that was found to be limiting for many radionuclides of interest.
- Section 5.3 discusses considerations related to dose integration and describes analyses performed to ensure that cleanup criteria used in Phase 1 will 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 preliminary, order-of-magnitude dose assessments related to remediation of subsurface soil; and provides for final dose assessments 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 radiation surveys,
- The information in Appendix C that supplements the content of this section,
- The information in Appendix D on engineered barriers and groundwater flow fields, and
- The information in Appendix E on details of the probabilistic uncertainty analysis.

## 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 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 decommissioning and then, after remediation of subsurface soil in the two major areas of interest, will 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 will 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 area is discussed below.

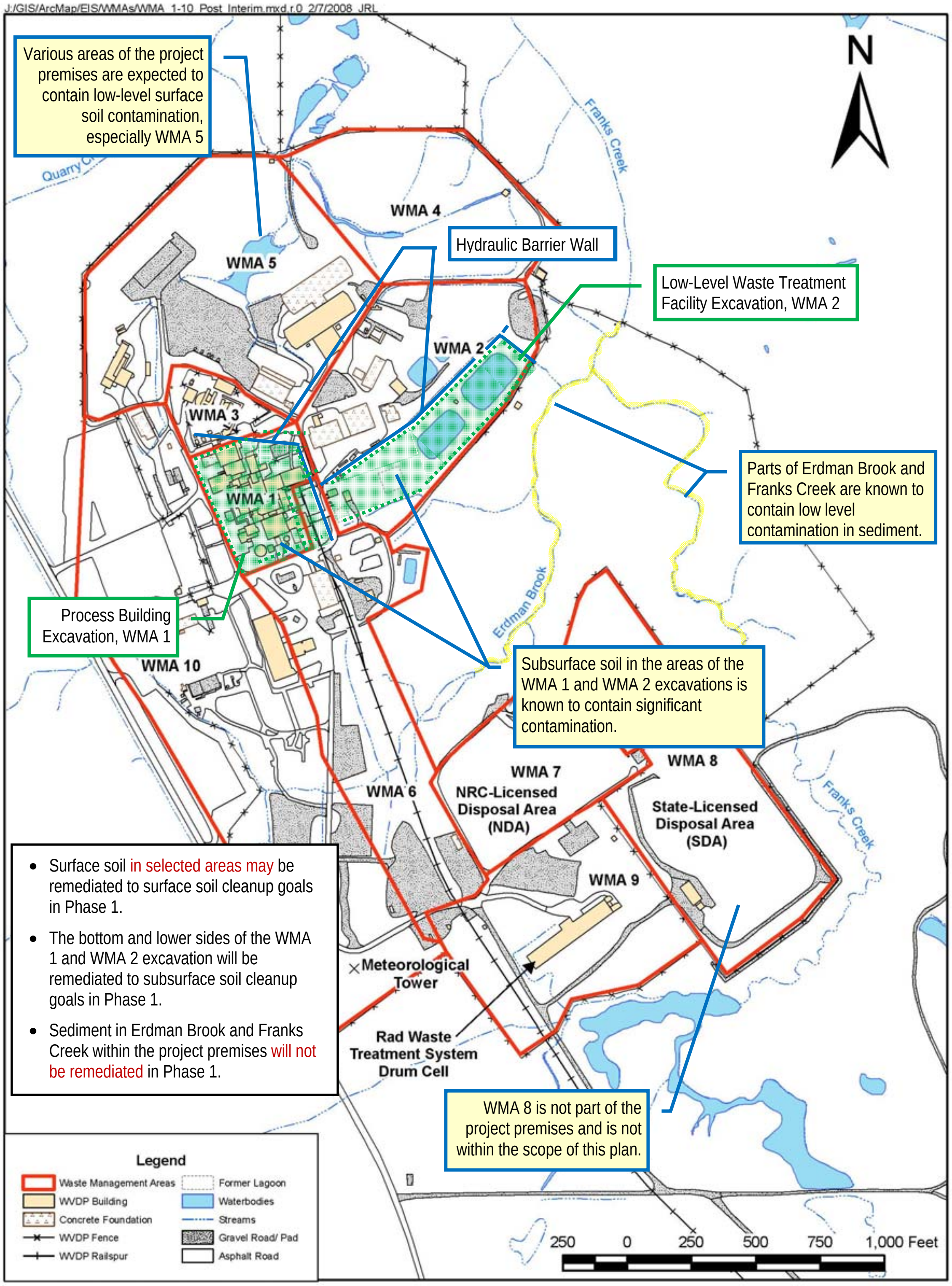



**Figure 5-1. Areas of Interest – Surface Soil, Subsurface Soil, and Streambed Sediment Within the Project Premises**

**Surface Soil**

As explained in Section 1 of this plan, surface soil and sediment in drainage ditches on the project premises will be characterized for radioactivity 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 are 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 release criteria in 10 CFR 20.1402 serve two purposes:

- They will support remediation of surface soil on selected portions of the project premises in Phase 1 of the decommissioning, and
- They will support decision-making for Phase 2 of the decommissioning.

The surface soil DCGLs and cleanup goals apply only to areas where there is no subsurface contamination, i.e., contamination below a depth of one meter.

**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 will 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 will be completely removed during Phase 1 of the decommissioning, along with the source area of the north plateau groundwater plume. The excavation for this purpose will be approximately 2.8 acres in size and extend more than 40 feet below the ground into the unweathered Lavery till. Figure 5-1 shows the approximate location of this excavation.

[1] The subsurface soil DCGLs will 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 models described in this section are representative of the Phase 2 conditions.

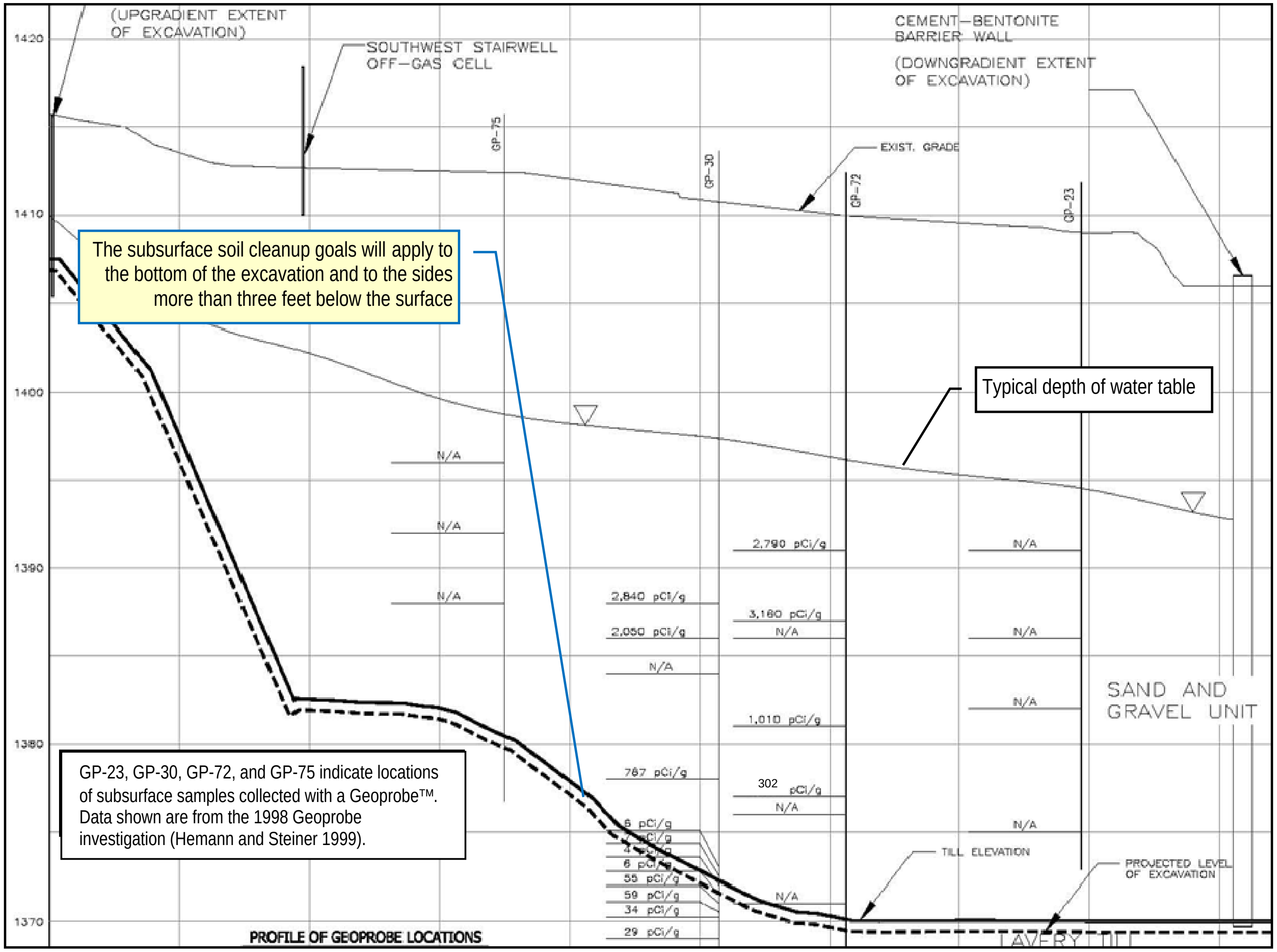



**Figure 5-2. Conceptual Cross Section View of WMA 1 Excavation With Representative Soil Data on Sr-90 Concentrations** (See Section 4.2 for more data and Section 7 for the excavation details.)

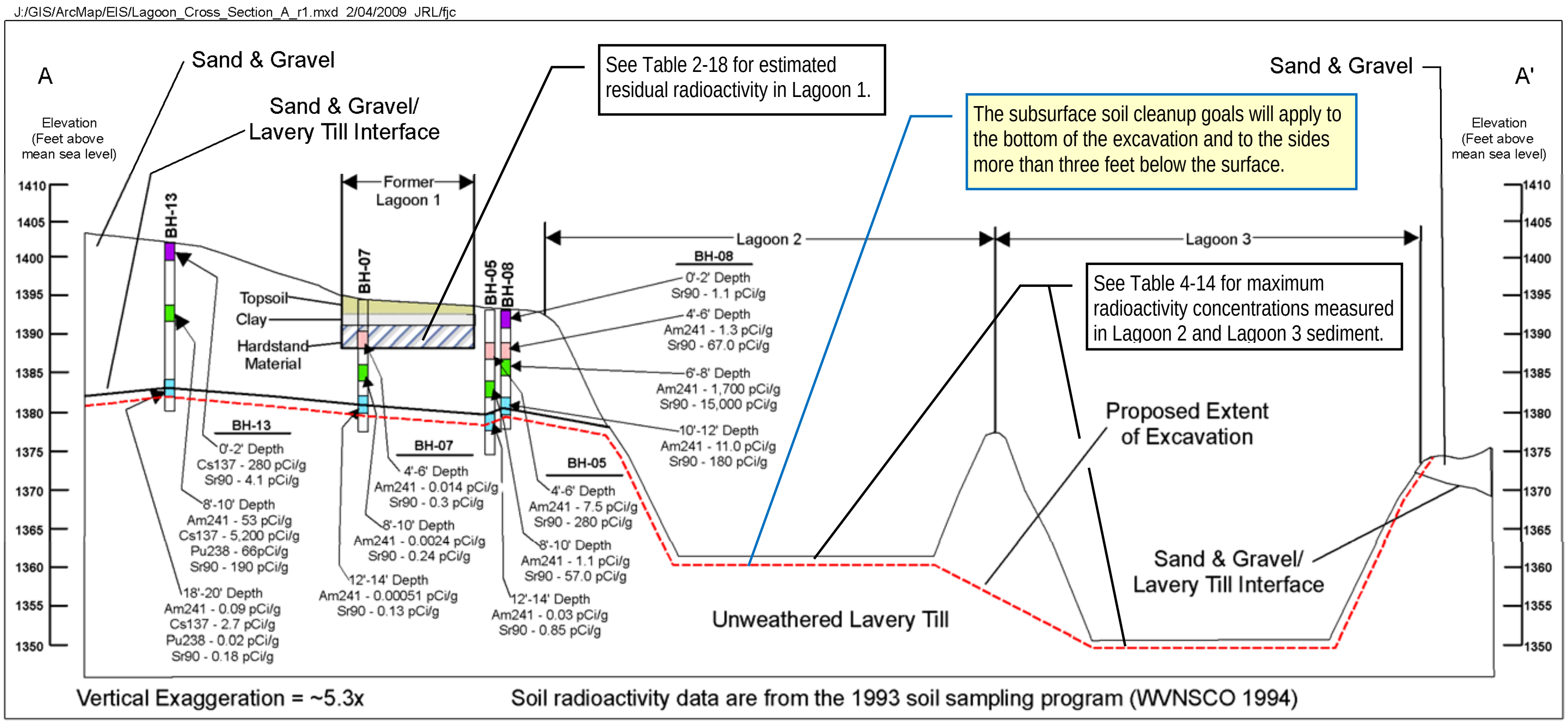



**Figure 5-3. Conceptual Cross Section View of WMA 2 Excavation With Representative Data on Subsurface Soil Contamination** (See Section 4.2 for more data and 7 for excavation details.

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, 41 pCi/g, and 59 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 | 1.1E-01(2) | 38-40 | none | none |
| Sr-90 | 5.9E+01(4) | 38.5-39 | 8.5E-01 | 12-14 |
| Tc-99 | <5.5E-01(2) | 37-39 | none | none |
| I-129 | <2.9E-01(2) | 38-40 | none | none |
| Cs-137 | 3.9E+00(2) | 38-40 | 4.5E-01 | 12-14 |
| U-232 | 4.1E-02 | 24-26 | 1.2E-02 | 12-14 |
| U-233/234 | 2.3E+00(2) | 38-40 | 1.8E-01 | 12-14 |
| U-235 | 1.4E-01(3)(5) | 24-26 | <5.9E-03 | 12-14 |
| Np-237 | <2.1E-02(2) | 37-39 | none | none |
| U-238 | 1.4E+00 | 41-43 | 1.1E-01 | 12-14 |
| Pu-238 | <2.3E-02(2) | 38-40 | 1.0E-02 | 12-14 |
| Pu-239/240 | <6.4E-02(2) | 38-40 | <5.9E-03 | 12-14 |
| Pu-241 | <5.7E-01(2) | 38-40 | <1.3E+00 | 12-14 |
| Am-241 | <1.3E-01(2) | 38-40 | 3.0E-02 | 12-14 |
| Cm-243/244 | <2.3E-02(2) | 38-40 | none | none |

NOTES: (1) See Table C-4 for the complete data set, which includes samples at nine locations entirely within the unweathered Lavery till within the WMA 1 excavation area. Based on boring log data, only one sample (BH-05) taken within the WMA 2 excavation area contained only unweathered Lavery till soil; the others contained some soil from the sand and gravel layer.

(2) Data are from the 2008 north plateau groundwater plume Geoprobe® investigation described in Section 4, with the highest non-detection values recorded (with amended sample 7608 results).

(3) Data are from sample BH-05 collected during the 1993 RCRA facility investigation described in Section 4. .

(4) Data are from point GP3098 from the 1998 north plateau Geoprobe® sampling described in Section 4.

(5) U-235/U-236 result.

**Additional Characterization Planned**

The characterization program described in Section 9 will provide additional data on radioactivity in subsurface soil in WMA 1 and WMA 2 and lagoon sediment in WMA 2.

The actual depth of the WMA 1 excavation will extend at least one foot into the unweathered Lavery, and this is where the subsurface soil cleanup goals will apply, as explained in Section 7. The configuration of the residual source will 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 will be made to remove Lagoons, 1, 2, and 3, the interceptors, the Neutralization Pit, and the Solvent Dike. The area of this excavation will be approximately 4.2 acres and its depth will 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 will extend at least one foot into the unweathered Lavery, and this is where the subsurface soil cleanup goals will apply, as explained in Section 7. In the cases of Lagoon 2 and Lagoon 3, the excavation will extend approximately two feet below the bottom the lagoons, which extend into the Lavery till. The configuration of the residual source will 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. 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.

[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 the cleanup goals.

**Streambed Sediment**

Streambed sediment refers only to sediment in Erdman Brook and the portion of Franks Creek running through the project premises. Figure 5-12 in Section 5.2 below shows precisely where streambed sediment DCGLs apply.

Surface soil DCGLs will be applied to sediment in ditches, in tributaries to Erdman Brook and Franks Creek, and in other parts of the project premises, with the subsurface soil 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. (However, 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 described in Section 9 will provide additional data on radioactivity in the sediment of the two streams.

DCGLs (cleanup goals) for streambed sediment based on the unrestricted use criteria in 10 CFR 20.1402 will support decision-making for Phase 2 of the decommissioning, and remediation of contaminated sediment in Erdman Brook and the portion of Franks Creek on the project premises is this were to be accomplished in Phase 2.

### 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 for sediment in drainage ditches on the project premises and in tributaries to 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 shown in Figure 5-12.

Each set of DCGLs was developed as if the area of interest remediated to the applicable DCGLs were to be the only area to which a hypothetical future resident or recreationist might be exposed. However, it is more likely that a variety of receptors will 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 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 decommissioning: mainly the WMA 1 area and the large area in WMA 2 to be excavated. Surface soil in selected areas 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.

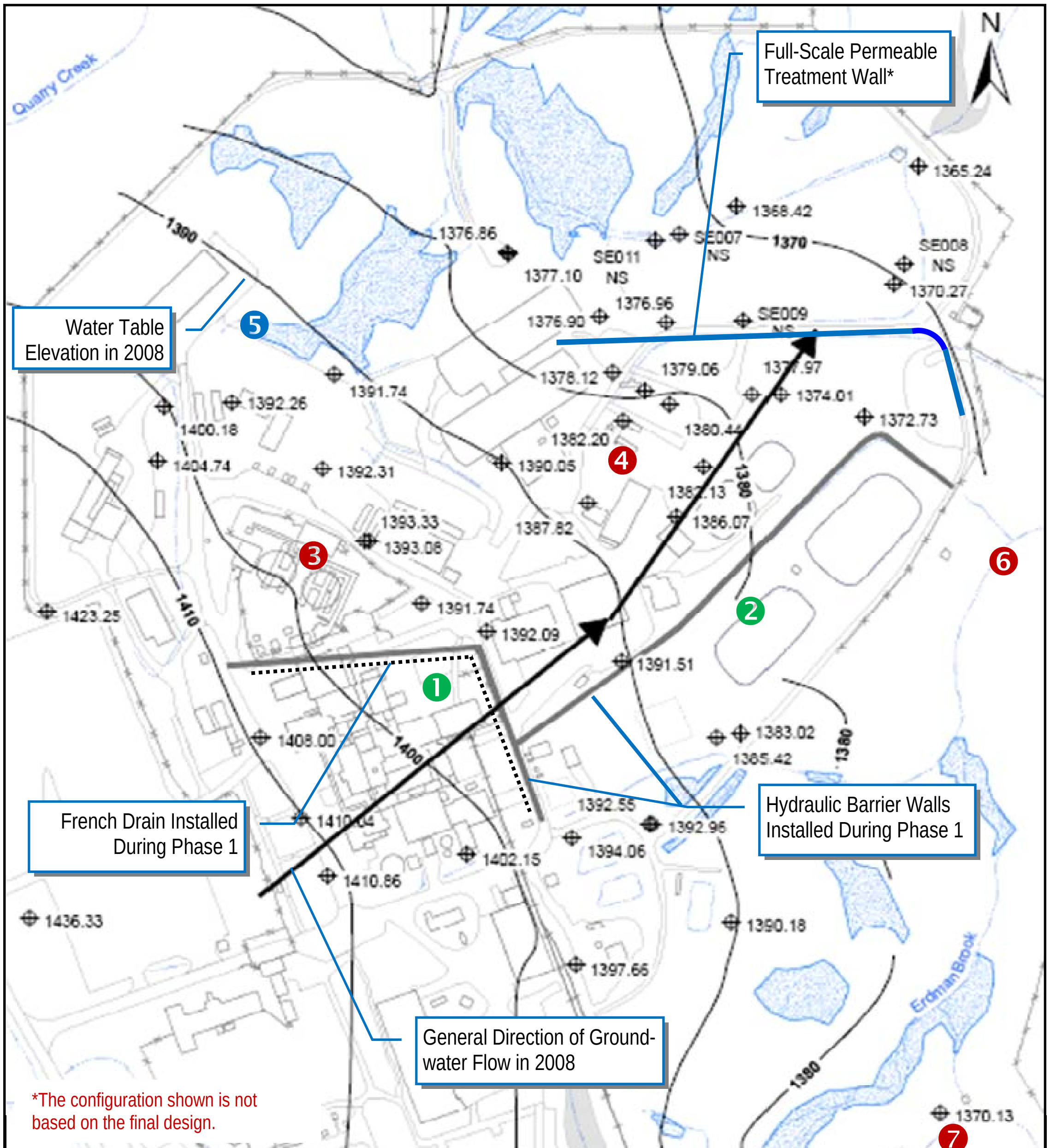
The table at the bottom of the Figure 5-4 shows the approximate amounts of total radioactivity in the different source areas based on estimates provided in Section 4. In this illustration, the remediated WMA 1 and WMA 2 excavated areas are the Phase 1 sources. The Waste Tank Farm, the non-source area of the north plateau groundwater plume, and the NDA are the Phase 2 sources, as is low-level contamination in streambed sediment. Low-level contamination in surface soil – 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 this sources much smaller than for the others (with the exception of streambed sediment).

Figure 5-4 shows other features of the project premises at the conclusion of the Phase 1 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 full-scale Permeable Treatment Wall; and
- The hydraulic barrier walls to be installed during Phase 1 of the decommissioning as described in Section 7 and the French drain to be emplaced upgradient of the WMA 1 hydraulic barrier wall.

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

[3] As noted in Section 7.11, surface soil in selected areas of the project premises may be remediated during the Phase 1 decommissioning activities to ensure that surface soil cleanup goals are achieved in these areas.




| | | |
|---|---|---|
| 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 Decommissioning**

**Potential Conditions at the Conclusion of the WVDP Decommissioning**

To determine whether criteria used in Phase 1 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 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 likely remain under NRC license. With Phase 1 of the decommissioning being accomplished, the contamination remaining in the WMA 1 and WMA 2 excavations will be residual radioactivity at concentrations below the subsurface soil cleanup goals 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 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 will have no impact on the site-wide close-in place-alternative and will 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 will be disposed of offsite. However, while the remediation standards will be the same, the critical group for potential future exposures will not be the same for all parts of the site. Because remediation to unrestricted release standards under Phase 1 of the 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. (Alternative scenario analyses described in Section 5.2 also evaluate exposure to a residential gardener.) 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 a single receptor) on the DCGLs and the associated cleanup goals to be used to guide remediation during Phase 1 of the 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 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 the CHILD landscape evolution model, which was calibrated for the site using a probabilistic process.

The CHILD model was used for 26 forward-in-time simulations to predict erosion rates at the WVDP over a 10,000-year time period. The models generally predicted minimal erosion on the central portion of the north plateau, gully development along the north plateau rim, and active erosion along the steep valley sides of Erdman Brook and Franks Creek. In the more erosive north plateau scenarios, gullies were predicted to advance within 328 to 656 feet of the Process Building area within the 10,000 year simulation 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 proceeds 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.

These projections formed the basis for the alternate conceptual models involving erosion that are described in Section 5.2.

### 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 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 will 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).

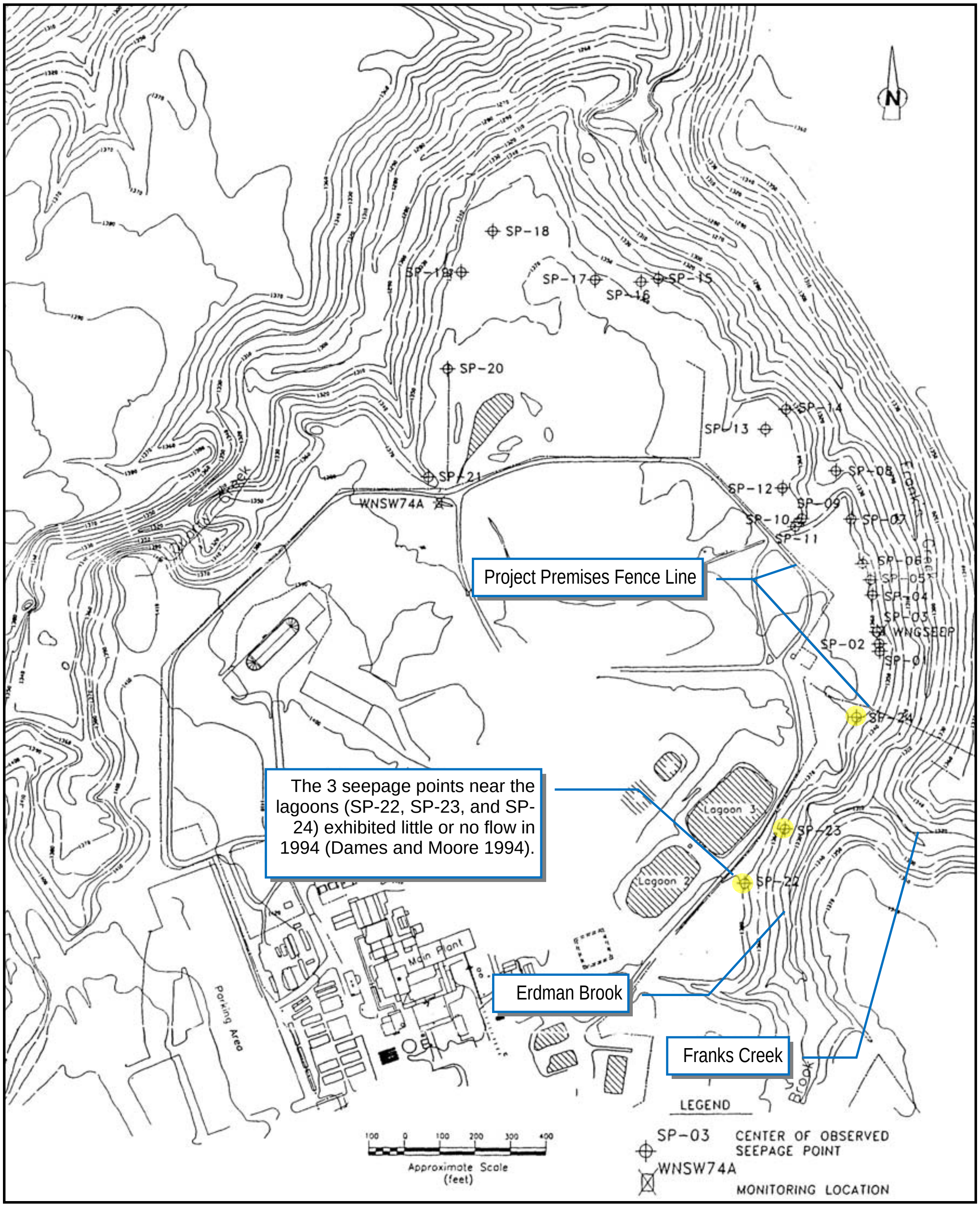



**Figure 5-5. Locations of Perimeter Seeps on the North Plateau** (From Dames and Moore 1994)

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

As noted previously, streambed sediment will not be remediated during Phase 1 of the decommissioning. The presence of groundwater seeps in the Erdman Brook area was one factor taken into account in the decision not to proceed with this remediation during Phase 1, since these seeps could possibly result in recontaminating the sediment in Erdman Brook.[4]

However, the potential for significant radioactivity in seeps in this area following Phase 1 of the decommissioning will 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 will be at very low concentrations; and
- Groundwater flow changes with the Phase 1 vertical hydraulic barriers in place, as described in Appendix D, will 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 was taken into account in the decision to not proceed with remediation of sediment in Erdman Brook and in the portion of Franks Creek on the project premises during Phase 1 of the decommissioning was 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 remain intact.

Note that Table D-4 in Appendix D provides flow balance estimates for post-Phase 1 conditions. These estimates do not show an increase in downward groundwater flow to the Kent Recessional Sequence following Phase 1 of the decommissioning.

### 5.1.7 Potential Impacts on the Kent Recessional Sequence

The potential for impacts on groundwater in the Kent Recessional Sequence from 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).

[4] Seeps could also release contamination into Quarry Creek. Quarry Creek lies outside of the project premises and is not within the scope of Phase 1 decommissioning activities.

It will 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 will take even longer considering the sorptive properties of the Lavery till (Table 3-20).

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 will 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 will 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 will 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 will not significantly impact the Lavery till sand.

Note that Section 9 provides for characterization surveys around selected Process Building foundation pilings to determine whether there might be evidence of contaminant migration along some of the pilings downward towards the Kent Recessional Sequence.

### 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 using RESRAD in the deterministic mode to produce the initial base cases;
- Performing parameter sensitivity analyses and refining the conceptual models and the DCGLs as appropriate based on the results;
- Performing a probabilistic uncertainty analysis to evaluate the degree of conservatism in model input parameters, producing probabilistic peak-of-the-mean and 95$^{th}$ percentile DCGLs;

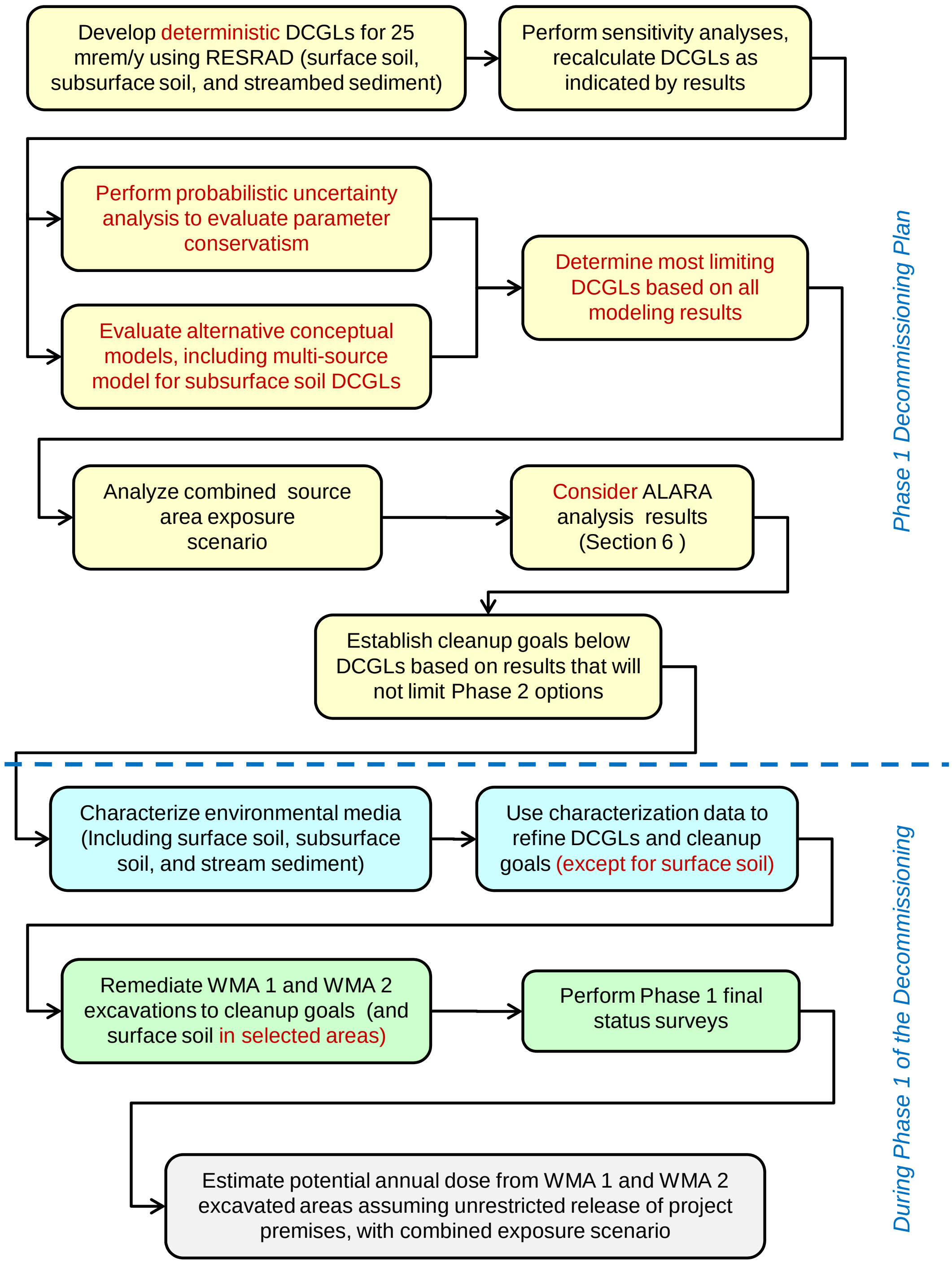



**Figure 5-6. General Dose Modeling Process**

- Evaluating alternate conceptual models, including a residential gardener and a multi-source conceptual model for subsurface soil DCGLs, for comparison with the initial base-case models;

- Evaluating the DCGLs produced by all of the modeling and determining the most limiting DCGLs for each radionuclide of interest; Analyzing combined source area exposure scenarios;
- Considering 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 decommissioning will not limit Phase 2 options;
- Characterizing surface soil, subsurface soil, and streambed sediment as specified in Section 9;
- Refining the DCGLs and cleanup goals based on the resulting data[5];
- Completing remediation of the WMA 1 and WMA 2 excavations and selected surface soil areas to the cleanup goals;
- Performing Phase 1 final status surveys in the remediated Phase 1 areas, and
- Making estimates of the potential future doses for the remediated WMA 1 and WMA 2 deep excavation areas using these data.

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 provides the following information:

- Subsection 5.2.1 describes the conceptual models used for developing DCGLs for surface soil.
- Subsection 5.2.2 describes the conceptual models used for developing DCGLs for subsurface soil.
- Subsection 5.2.3 describes the conceptual model used for developing DCGLs for streambed sediment.
- Subsection 5.2.4 describes the mathematical model (RESRAD) used to calculate deterministic DCGLs for the various conceptual models.

[5] The characterization to be performed as described in Section 9 will provide data on the depth and lateral extent of contamination that may be useful in better defining source geometry in the conceptual model. For example, if the actual streambed and stream bank source geometry were found to be substantially different from that assumed in the conceptual model, then the conceptual model would be revised accordingly and the DCGLs recalculated. The same approach would be used for the subsurface soil DCGLs. However, there are no plans to recalculate surface soil DCGLs for this reason because the assumed one meter source thickness is generally conservative and it is important to avoid changes to surface soil DCGLs that would impact the design of the Phase 1 final status surveys. 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.

- Subsection 5.2.5 provides the modeling results – the deterministic DCGLs – along with a discussion of these results.
- Subsection 5.2.6 describes sensitivity analyses performed.
- Subsection 5.2.7 describes the probabilistic uncertainty analysis.
- Subsection 5.2.8 describes the multi-source analysis for subsurface soil DCGLs that takes into account releases of radioactivity from the bottoms of the deep excavations by diffusion.

The DCGL development 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 will 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 will 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 will be expanded if necessary following completion of soil and sediment characterization described in Section 9. If other radionuclides show up in concentrations significantly above the minimum detectable concentrations, additional DCGLs will 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 will ensure that dose criteria of 10 CFR 20.1402 will be met in

2041.[6] 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 Surface Soil DCGL Development

The initial base-case conceptual model for development of surface soil DCGLs is described first.

**Surface Soil Conceptual Model (Base-Case)**

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.

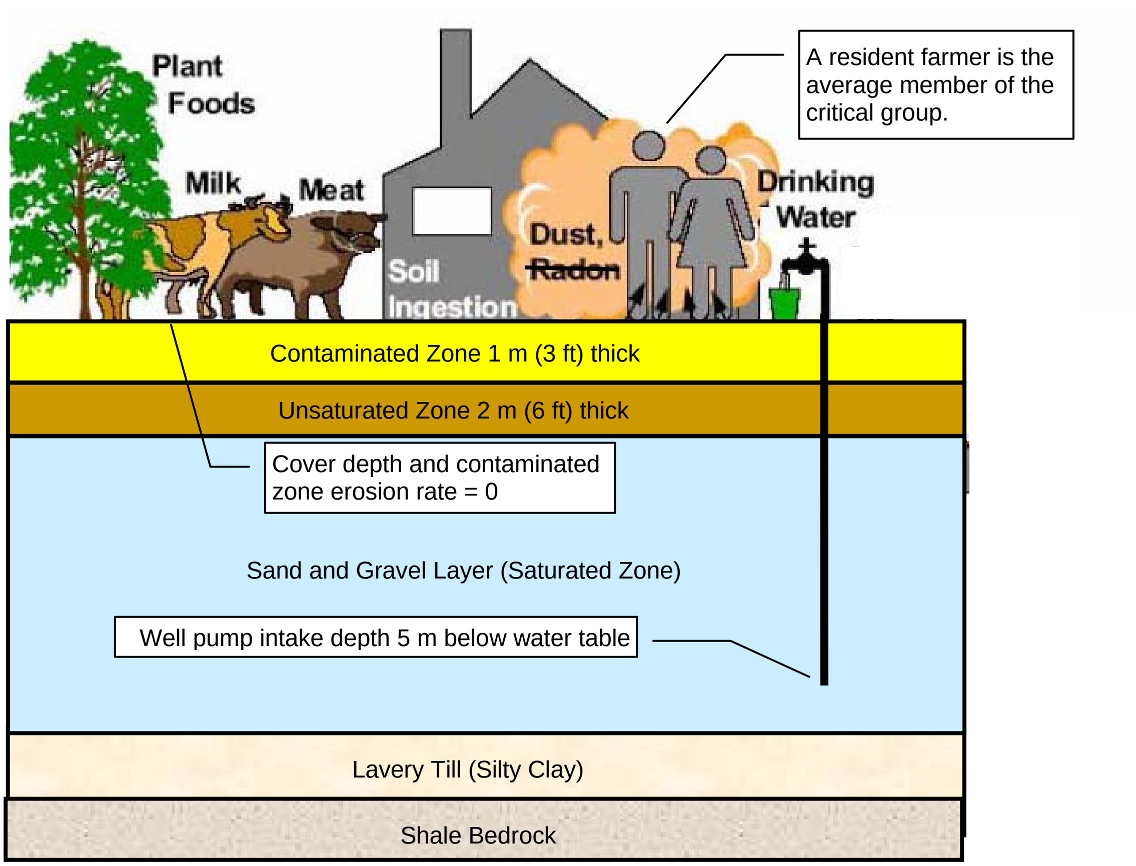



**Figure 5-7. Conceptual Model for Surface Soil DCGL Development**

[6] This approach will support any license termination actions that may take place in Phase 2 of the decommissioning. As noted previously, the decision on the Phase 2 decommissioning approach could be made within 10 years of the Record of Decision and Findings Statement documenting the Phase 1 decisions. If this approach were to involve unrestricted release of the site, achieving this condition would be expected to take at least another 20 years due to the large scope of effort to exhume the underground waste tanks and the NDA. It is therefore highly unlikely that conditions for unrestricted release of the project premises could be established before 2041. If Phase 2 were to involve closing radioactive facilities in place, then institutional controls would remain in place.

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 assumed 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. (This assumption was confirmed by evaluation of alternate conceptual models involving erosion and a residential gardener as discussed below.)

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 will 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 less resistance to radionuclide movement though soil, allowing less time for radioactive decay to take place.[7]

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

[7] 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 shows the basis for these values. The use of lower $K_d$ values than those in south plateaus soil is conservative for water pathways, but may not be conservative for plant uptake and direct exposure pathways. However, the model would be conservative for south plateau conditions for most radionuclides.

**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 |
| 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) | 5.0E+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. The one meter thickness is an appropriate compromise for the set of radionuclides of interest whose primary dose pathways range from direct exposure, to groundwater ingestion, to plant uptake.

(3) This assumption is conservative because it results in no depletion of the source through erosion.[8]

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;
- Because the model considers only surface contamination, the resulting DCGLs and cleanup goals are applicable only to portions of the project premises where there is no subsurface contamination (i.e., contamination does not extend beyond a depth of 1 meter);
- All water use (e.g., household, crop irrigation, and livestock watering) is from contaminated groundwater;
- Adequate productivity from a well pumping from the aquifer will 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.

**Alternate Conceptual Model for Surface Soil DCGLs (Erosion, Offsite Receptor)**

Other conceptual models were considered, even though the resident farmer model with its many exposure pathways is generally considered to be the most conservative model. To

[8] The conservative nature of the assumption can be demonstrated by assuming that erosion takes place and evaluating potential doses to a receptor located in a gully where radioactivity has been displaced by erosion. As explained in the discussion of alternate conceptual models below, the receptor in the area of the gully would receive less dose on an annual basis than would the resident farmer due to factors such as source dilution, spending less time in the contaminated area, and receiving exposure through fewer pathways. Consideration of potential doses to an offsite receptor from radioactivity displaced to the stream through erosion indicates that there is a reasonable expectation that offsite doses would not be significant either.

confirm that the assumption of no erosion in the contamination zone (one of the key parameters in Table 5-3) is conservative, an analysis was performed to estimate the potential doses to an offsite receptor from radioactivity that could be released from the hypothetical garden used in the base-case model through erosion.

In this analysis, eroded soil was assumed to be transported in surface water to a receptor located on Cattaraugus Creek near the confluence with Buttermilk Creek who ingested both the water and fish harvested from the water and used the water to irrigate a garden. The results showed that doses to this receptor would be insignificant.

**Alternate Conceptual Model for Surface Soil DCGLs (Residential Gardener)**

Another alternative exposure scenario was evaluated to confirm that the base-case resident farmer scenario is bounding for development of surface soil DCGLs. This alternative scenario involved a residential gardener scenario.

The receptor in the residential gardener scenario is a hypothetical person who resides in the area and grows a vegetable garden. This scenario differs from the resident farmer scenario in that the person of interest does not consume meat or milk produced on the property and spends less time outdoors in the hypothetical garden. The well pumping rate used in this scenario was lower than that used in the resident farmer model (1140 cubic meters per year compared to 5720 meters per year) to reflect the smaller garden being used and the lower well water usage.

This alternative exposure scenario produced DCGLs that were slightly higher than those produced by the base-case resident farmer model for all 18 radionuclides. Consequently, the base-case model is bounding for surface soil DCGL development when compared to the residential gardener scenario. (See Section 5.2.7 for the results of the probabilistic uncertainty analysis.)

### 5.2.2 Subsurface Soil Conceptual Models

**Evaluation of Various Subsurface Soil Conceptual Models**

The analyses described in Revision 0 and Revision 1 to this plan made use of the base-case conceptual model for subsurface soil DCGL development described below and illustrated in Figure 5-8. Minor changes were made to this conceptual model in Revision 2 that produced DCGLs that were slightly higher for most radionuclides.

Additional analyses were also performed to determine whether this conceptual model, which makes use of the resident farmer scenario, represented the bounding case for potential future doses from the remediated deep excavations. These additional analyses, which are described below, involved:

- Evaluating the potential acute dose to the hypothetical individual drilling the well (the two meter diameter cistern) used in the original base case model,
- Evaluating potential acute dose to a hypothetical individual who might drill a natural gas well in the area of one of the deep excavations,
- Evaluating potential doses to a recreational hiker in the area of the lagoons in WMA 2 assuming that unchecked erosion would eventually produce deep gullies in this area,
- Evaluating potential doses to an offsite receptor from residual radioactivity at the bottom of the deep excavation in WMA 2 that might be released to Erdman Brook if deep gullies were to eventually cut into this area, and
- Evaluating a residential gardener scenario.

Of these five alternate conceptual models, one, the residential gardener model, was found to be more limiting for some radionuclides than the original base-case resident farmer scenario.

To help determine whether the input parameters used in the original base-case model were sufficiently conservative, a comprehensive probabilistic uncertainty analysis was performed (similar analyses were also performed for surface soil and streambed sediment DCGL development). Section 5.2.7 describes this analysis. The resulting peak-of-the-mean DCGLs were somewhat lower for most radionuclides than the DCGLs produced by the deterministic resident farmer and residential gardener scenarios.

Another analysis was performed to evaluate whether continuing release of residual radioactivity from the bottom of the deep excavations would influence potential future doses from the remediated deep excavations. Section 5.2.8 describes this analysis. The original base-case conceptual model was modified to add a secondary source of radioactivity from residual contamination at the bottom of the deep excavation that moves upward by diffusion and is drawn into the hypothetical well, resulting in additional dose to the resident primarily from the drinking water pathway.

This multi-source model was analyzed using the resident farmer scenario and also the residential gardener scenario, the latter with three different upper contamination zone geometries to evaluate the sensitivity of the model to the contamination zone area and thickness. The results showed that this model was more limiting for nine of the 18 radionuclides of interest than the other subsurface soil DCGL conceptual models that were evaluated.

Consideration of the results of all of this subsurface soil dose modeling led to the decision to use the lowest DCGLs among all of the modeling results as the basis for the subsurface soil cleanup goals in the interest of conservatism.

### Initial Base-Case Conceptual Model

Figure 5-8 illustrates the initial base-case 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.

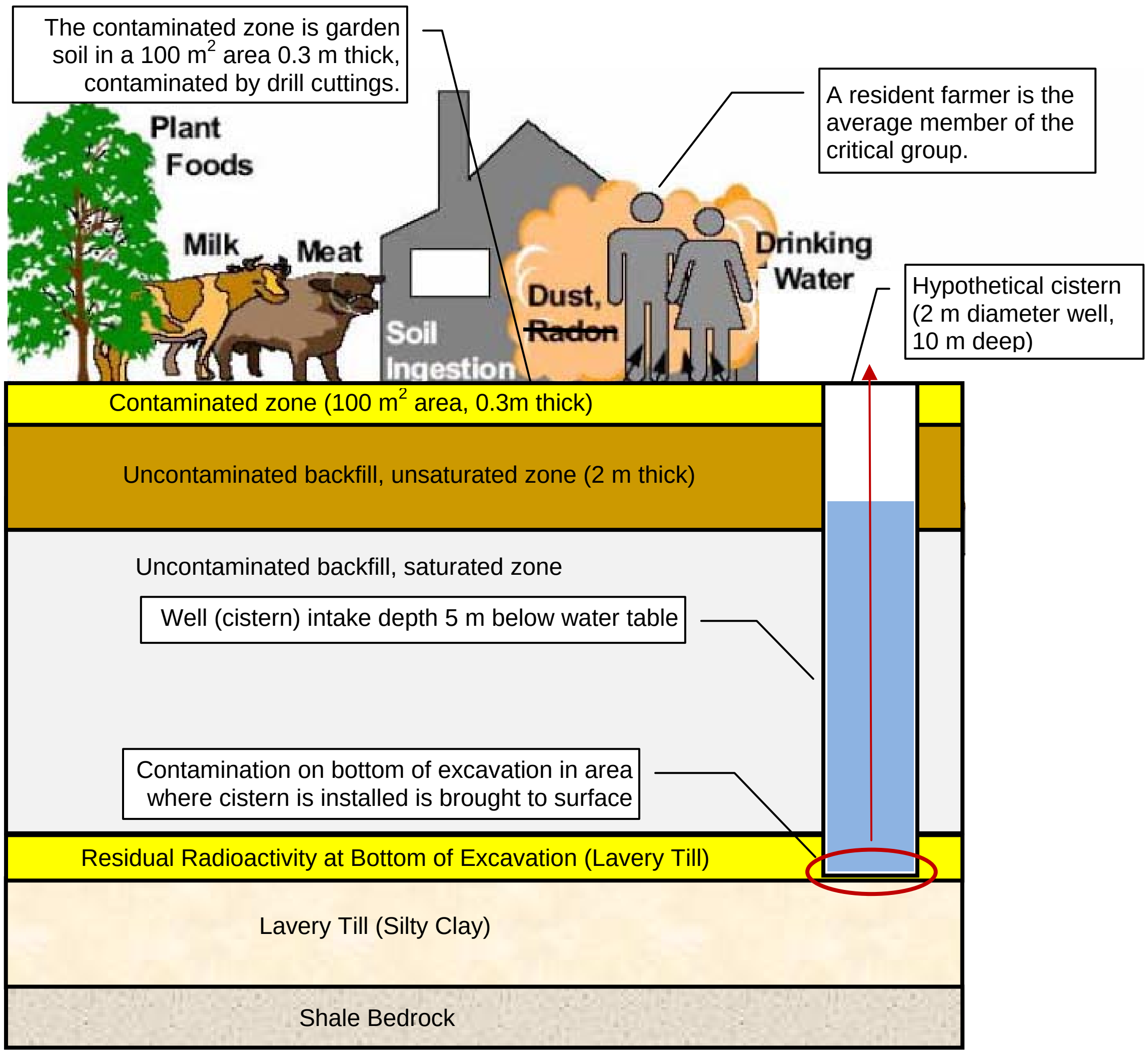



**Figure 5-8. Conceptual Model for Subsurface Soil DCGL Development**

Other possible critical groups were considered as with the conceptual model for surface soil DCGLs. However, a resident farmer was initially assumed 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 will be more than 10 feet below the surface in the remediated WMA 1 and WMA 2 areas (the bottoms of the excavations will be more than 10 feet below the surface and uncontaminated soil will be used to backfill the excavations).

Note that Section 7 specifies that the uncontaminated backfill as shown in the figure will 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 will 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.

Note that a combination of inadvertent intrusion and continuing releases from the bottoms of the remediated deep excavations was also evaluated in the multi-source conceptual model as described in Section 5.2.8, 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).[9]
- 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 was not expected to leach upward into the source of water available to the resident farmer. (However, additional analysis showed that doses from continuing releases from the contamination at the bottom of the excavation would be significant for some radionuclides as described in Section 5.2.8.)

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)] |

[9] Note that larger contamination zone areas were evaluated in the multi-source conceptual model described in Section 5.2.8

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

| Exposure Pathways | Active |
|---|---|
| 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 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.[10]

[10] The conservative nature of the assumption can be demonstrated by assuming that erosion takes place and evaluating potential doses to a receptor located in a gully where radioactivity has been exposed by erosion. As explained in the discussion of alternate conceptual models below, the receptor in the area of the gully would receive less dose on an annual basis than would the resident farmer due to factors such as spending less time in the contaminated area and receiving exposure through fewer pathways. Consideration of potential doses to an offsite receptor from radioactivity displaced to the stream through erosion indicates that there is a reasonable expectation that offsite doses would not be significant either, as discussed below.

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, as noted previously.

**Alternate Conceptual Model for Subsurface Soil DCGLs (Cistern Well Driller)**

A drilling worker scenario evaluates 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 using RESRAD with conservative assumptions. Key elements in the model included:

- The drilling worker being exposed to excavated Lavery till material from the bottom of the excavation that was deposited on top of uncontaminated soil in the vicinity of the cistern for a 40 hour period, even though the actual exposure period would likely be much shorter;
- The contamination zone being nine square meters in area and 0.333 meters thick, based on an excavated volume of three cubic meters of contaminated Lavery till material; and
- An assumption of no water shielding, even though water in a cuttings pond would typically provide shielding from direct radiation.

The exposure pathways considered included inadvertent ingestion of contaminated soil, inhalation of contaminated dust, and direct exposure to contaminated soil brought to the surface during the drilling. The resulting DCGLs, which are shown in Table 5-11c in Section 5.2.8, were greater than the subsurface soil DCGLs for all radionuclides developed for the resident farmer scenario, indicating the well driller scenario is less limiting that the resident farmer scenario used in developing the subsurface soil DCGLs.

**Alternate Conceptual Model for Subsurface Soil DCGLs (Erosion, Onsite Receptor)**

An alternate conceptual model was evaluated involving the potential impact of unchecked erosion in WMA 2 to an onsite receptor. The model assumed that gully erosion would produce narrow, deep steep-sided gullies, conditions where building a home and growing crops would not be practical. A plausible scenario for these conditions would involve a recreationist spending time hiking in the area, which is assumed to be rent by deep gullies that extend to the bottom of the WMA 2 excavation. Figure 5-9 illustrates the basic conceptual model. This scenario was analyzed using RESRAD in the deterministic mode.

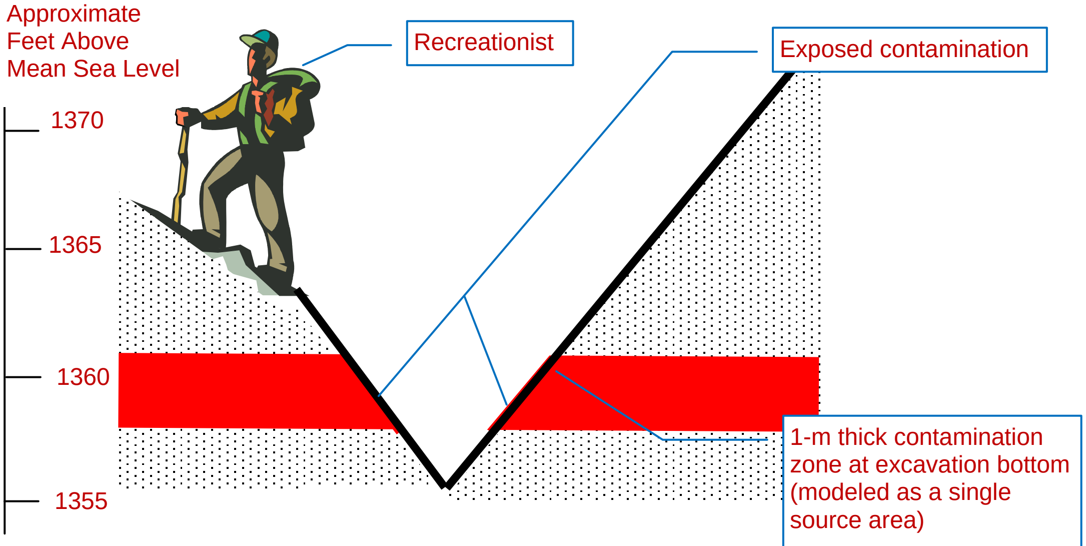



**Figure 5-9 Recreationist Conceptual Model Cross Section**

The modeling of this recreationist scenario produced DCGLs for 25 mrem per year that were more than one order of magnitude greater than the DCGLs produced with the initial base-case resident farmer/cistern scenario for all 18 radionuclides of interest as shown in Table 5-11c in Section 5.2.8. These results demonstrate that the resident farmer/cistern scenario is more limiting for an onsite receptor.

**Alternate Conceptual Model for Subsurface Soil DCGLs (Erosion, Offsite Receptor)**

Another alternative scenario was evaluated to determine the potential impact of long-term erosion in WMA 2 to an offsite receptor. This analysis estimated the potential doses to an offsite receptor from radioactivity that could be released from the bottom of the remediated WMA 2 excavation due to formation of a gully that eventually cut through the bottom of the backfilled excavation.

In this analysis, radioactivity in eroded soil from the bottom of the WMA 2 backfilled excavation was assumed to be transported in surface water to a receptor located on Cattaraugus Creek near the confluence with Buttermilk Creek who ingested both the water and fish harvested from the water and used the water to irrigate a garden. Both the area of Lagoon 1 and the area of Lagoon 3 were considered using conservative erosion rates. The results showed that doses to this receptor would be insignificant compared to the onsite receptor doses estimated in the base-case resident farmer model. Table 5-11c below shows the DCGLs calculated for the Lagoon 3 area.

**Alternate Conceptual Model for Subsurface Soil DCGLs (Natural Gas Well Driller)**

Installation of a natural gas well was also evaluated. Installation of this type of well would take longer than installation of a cistern because the well would be much deeper, would require well/formation development by hydrofracturing, and would require the installation of conveyance piping and valving. The analysis focused on exposure to the drilling worker. Key elements in the model included:

- The natural gas well being 0.5 meter (20 inches) in diameter and 100 meters (330 feet) deep (a conservative estimate given typical depths in excess of 1,000 meters); and
- The drilling worker being exposed to excavated Lavery till material from the bottom of the excavation that was deposited in a cuttings pit near the worker's location for 500 hours.

The exposure pathways considered included inadvertent ingestion of contaminated soil, inhalation of contaminated dust, and direct exposure to contaminated soil brought to the surface during the drilling. RESRAD version 6.4 in the deterministic mode was used to perform the calculations. The resulting DCGLs shown in Table 5-11c below were one or more orders of magnitude greater than the deterministic base-case resident farmer subsurface soil DCGLs for all radionuclides, demonstrating that the base-case resident farmer-cistern installation scenario is more limiting.

**Alternate Conceptual Model for Subsurface Soil DCGLs (Residential Gardener)**

Another alternative exposure scenario was evaluated to determine whether the base-case resident farmer-cistern installation scenario was bounding for development of subsurface soil DCGLs. This alternative scenario involved a residential gardener scenario.

The receptor in the residential gardener scenario is a hypothetical person who resides in the area and grows a vegetable garden. This scenario differs from the resident farmer scenario in that the person of interest does not consume meat or milk produced on the property and spends less time outdoors in the hypothetical garden. The well pumping rate used in this scenario was lower than the rate used in the resident farmer model (1140 cubic meters per year compared to 5720 meters per year) to reflect the smaller area being used and the lower well water usage.

This analysis was performed using three models which differed with respect to the area of the contamination zone and its thickness:

- Model 1 used a 100 square meter area and 0.3 meter depth, the base-case values in the base-case resident farmer deterministic analysis;
- Model 2 used a 300 square meter area and 0.1 meter depth; and
- Model 3 used a 50 square meter area and 0.6 meter depth;

This alternative exposure scenario produced DCGLs for some radionuclides that were lower than those produced by the base-case resident farmer model. In most cases, Model 2 with the largest contamination zone area produced the lowest DCGLs due to higher groundwater concentrations from reduced dilution and larger contaminated fractions from ingestion pathways. The results appear in Section 5.2.8 and were taken into account in establishing revised cleanup goals.

### 5.2.3 Streambed Sediment Conceptual Model

Figure 5-10 illustrates the conceptual model for development of streambed sediment DCGLs. Table 5-6 identifies the exposure pathways considered.

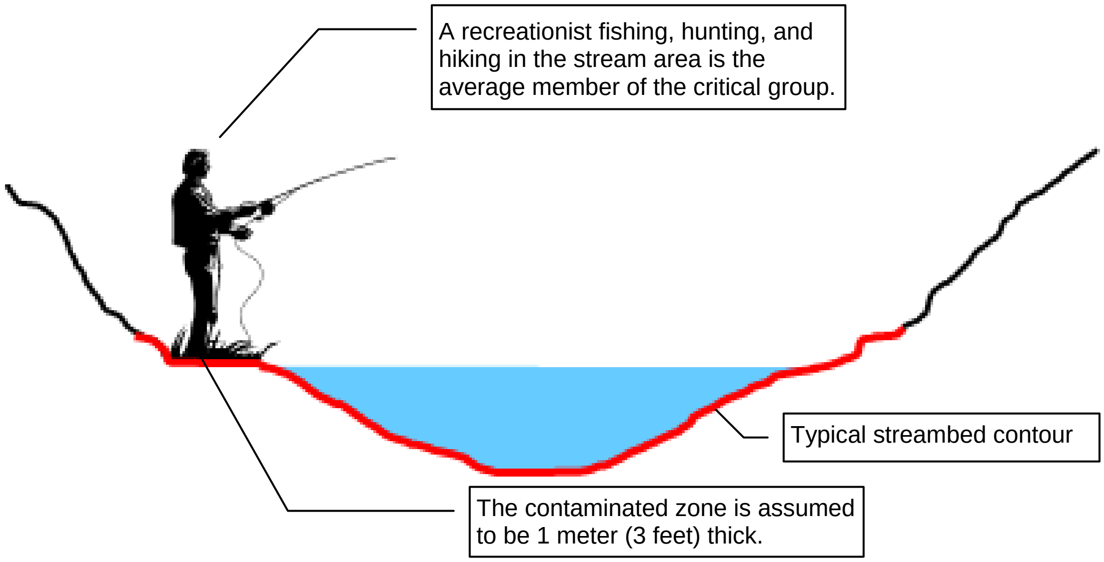



**Figure 5-10. Conceptual Model for Streambed DCGLs Development**

**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, which would minimize inhalation exposure. Additionally, vegetation along the streambed will likely preclude significant wind scour and subsequent inhalation. To confirm these conclusions, the model was revised to include the inhalation pathway as well as to make other minor refinements; these changes did not produce a significant difference in the results.

(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.

The conceptual model for streambed sediment was developed after consideration of how residual radioactivity enters and moves though the streams, plausible future land uses for the stream valleys, how humans might be exposed to residual contamination in the streams or on the banks, and plausible habits of a person who might spend time at the streams in the future. Such considerations led to selection of a conceptual model compatible with RESRAD. The RESRAD code was determined to be an appropriate mathematical model based on its extensive use in evaluating potential doses from radioactivity in surface soil and its use in the surface soil DCGL and subsurface soil DCGL models for this project.

As shown in Figure 5-10, the contamination zone was assumed to be on the stream bank rather than in the stream itself. This model is consistent with typical conditions observed along Frank's Creek downstream of the Lagoon 3 outfall as shown by the radiological control area in Figure 5-11 represented by the roped-off area. It is conservative compared to having the contamination zone in the stream itself where water would act as shielding to reduce the direct radiation dose.

The photograph in Figure 5-11 was taken from just inside the project premises security fence looking upstream toward the southwest. The confluence with Erdman Brook lies about 200 feet upstream from where the people are standing and the Lagoon 3 outfall is about 500 feet from where the people are standing.

**Figure 5-11. Franks Creek Looking Upstream** (2008 WVDP photo)

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.[11]
- 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.
- 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 will 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:

[11] 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, factors which are appropriately not considered at this time.

- 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 will be necessary for this purpose.[12]

### 5.2.4 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 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

[12] 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 2009). 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.

- The conservative assumption of no erosion for soil and sediment sources in the development of DCGLs, so there will be no source depletion from erosion.

The RESRAD model has limitations in this application in that it was developed for soil exposures and therefore does not specifically address certain transport mechanisms associated with sediment, such as:

- Periodic saturation of the contaminated zone located along a stream bank flood zone;
- Erosion/scour of stream bank material and subsequent downstream deposition to the stream-bottom;
- Deposition of clean material onto the stream bank, transported downstream from unimpacted upstream locations;
- Variability in surface water concentrations due to fluctuation in flow rates during storm events;
- Partitioning of contaminants between the surface water and stream-bottom sediment; and
- Variability of airborne dust loads due to varying stream bank sediment moisture content.

To address the simplifications of the conceptual model, and still retain conservatism in the results, the following assumptions were made for the sediment model:

- The model will not allow the contaminated zone to be below the water table (as may periodically happen to the stream bank), therefore it was assumed that there was no unsaturated zone, and that the water table exists immediately below the source;
- The inhalation parameter values were conservatively selected to reflect soil on a farm, although stream bank sediment is likely to result in lower respirable dust loadings;
- Contaminated groundwater is assumed to discharge to the stream, where it is impounded and contributes to fish bioaccumulation;
- Fish ingested from the stream are large enough to provide a significant number of meals each year, but are assumed to only be exposed to contaminated water and never swim to uncontaminated sections of the stream; and
- In addition to assuming the fish are never in clean water, the recreationist is assumed to eat only fish that are contaminated when, in actuality, the stream will not support fish at all at the present time owning to the small amount of water typically present as shown in Figure 5-11.

The conceptual model just described represents plausible conditions on the stream banks and in the streambeds. It is considered to be a valid model for the long term in support of a Phase 2 strategy involving unrestricted release, that is, the site-wide removal alternative in the Decommissioning EIS. However, it would not necessarily serve as a valid model if the Phase 2 sources were to be closed in place, as with the site-wide close-in-place alternative.

This limitation results from the model not accounting for processes that could impact the streams in the future under the site-wide close-in-place alternative. For example, impacts on the streams could occur in the long term from unchecked erosion in the radioactive waste disposal areas, surface water runoff from eroded areas, and increased seepage of contaminated groundwater into the streams. Such impacts could include increases in radionuclide concentrations in water in the streams as well as increases in contamination in the sediment.

This limitation would be considered in any decision made by DOE to remediate sediment in the streams and on the stream banks. Such remediation during Phase 1 decommissioning activities would require a revision to this plan.

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).

### 5.2.5 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 will not exceed 25 mrem per year when considering the dose contribution from each radionuclide individually. Note that the surface soil DCGLs apply only to areas of the project premises where there is no subsurface soil contamination and that the subsurface soil DCGLs apply only to the bottoms and lower sides (extending from a depth of three feet and greater) of the large excavations in WMA 1 and WMA 2.

**Table 5-8. DCGLs For 25 mrem Per Year ($DCGL_W$ Values in pCi/g)(1)**

| Nuclide | Surface Soil | Subsurface Soil(3) | Streambed Sediment |
|---|---|---|---|
| Am-241 | 4.3E+01 | 7.1E+03 | 1.6E+04 |
| C-14 | 2.0E+01 | 3.7E+05 | 3.4E+03 |

**Table 5-8. DCGLs For 25 mrem Per Year ($DCGL_W$ Values in pCi/g)(1)**

| Nuclide | Surface Soil | Subsurface Soil(3) | Streambed Sediment |
|---|---|---|---|
| Cm-243 | 4.1E+01 | 1.2E+03 | 3.6E+03 |
| Cm-244 | 8.2E+01 | 2.3E+04 | 4.8E+04 |
| Cs-137(2) | 2.4E+01 | 4.4E+02 | 1.3E+03 |
| I-129 | 3.5E-01 | 5.2E+01 | 3.7E+03 |
| Np-237 | 9.4E-02 | 4.3E+00 | 5.2E+02 |
| Pu-238 | 5.0E+01 | 1.5E+04 | 2.0E+04 |
| Pu-239 | 4.5E+01 | 1.3E+04 | 1.8E+04 |
| Pu-240 | 4.5E+01 | 1.3E+04 | 1.8E+04 |
| Pu-241 | 1.4E+03 | 2.4E+05 | 5.1E+05 |
| Sr-90(2) | 6.3E+00 | 3.2E+03 | 9.5E+03 |
| Tc-99 | 2.4E+01 | 1.1E+04 | 2.2E+06 |
| U-232 | 5.8E+00 | 1.0E+02 | 2.6E+02 |
| U-233 | 1.9E+01 | 1.9E+02 | 5.7E+04 |
| U-234 | 2.0E+01 | 2.0E+02 | 6.0E+04 |
| U-235 | 1.9E+01 | 2.1E+02 | 2.9E+03 |
| U-238 | 2.1E+01 | 2.1E+02 | 1.2E+04 |

NOTES: (1) Refer to Sections 5.2.7 and 5.2.8 for discussions about how this set of DCGLs was considered in establishing cleanup goals.
(2) Sr-90 and Cs-137 DCGLs reflect 30 years of decay and apply to the year 2041 and later.
(3) The lower deterministic DCGL of the resident farmer and residential gardener conceptual models.

As noted previously, the sum-of-fractions rule will 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 water, 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 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 DCGLs calculated using the initial base-case model conservative. For the surface soil DCGLs, these factors include, for example, the relatively short local growing season, which makes it likely that crop and forage yields will 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) used in the initial base-case model at two meters (about 6.6 feet) is much larger than the diameter of a typical water well (eight inches)[13].
- 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 will 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 will be impacted by streambed sediment sources exclusively. It is more likely that exposure will be from both impacted and non-impacted areas.

[13] 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.

- 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.

**Applicability of Streambed Sediment DCGLs**

The conceptual model used for developing DCGLs for stream bed sediment in Erdman Brook and the portion of Franks Creek on the project premises assumed that these streams have steep banks. This condition exists in most parts of the streams but not all parts. Consequently, it is necessary to define where the streambed sediment DCGLs and cleanup goals apply.

Figure 5-12 shows the points where the streambed sediment DCGLs and cleanup goals apply. As indicated on the figure, the surface soil DCGLs and cleanup goals apply upstream of these points and to the small tributaries to the streams.

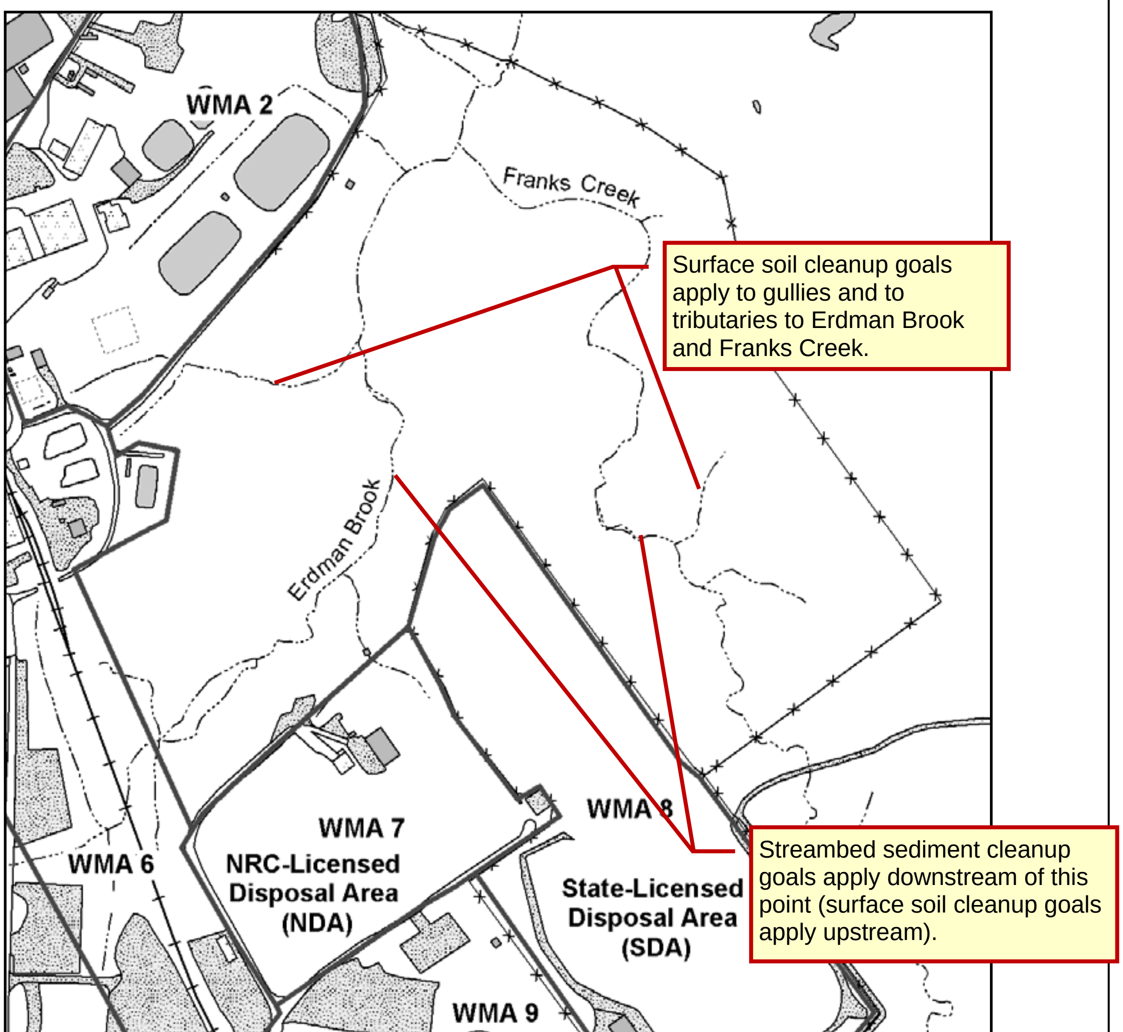



**Figure 5-12. Areas Where Streambed Sediment DCGLs and Cleanup Goals Apply**

### 5.2.6 Discussion of Sensitivity Analyses

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

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

| Parameter | Run | Change in Sensitivity Parameter | Minimum DCGL Change | | Maximum DCGL Change | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Indoor/Outdoor Fraction | 1 | -32% | -22% | U-232 | 0% | I-129 |
| | 2 | 21% | 0% | I-129 U-234 | 28% | U-232 |
| Contamination Zone Thickness | 3 | -50% | 9% | U-232 | 81% | Sr-90 |
| | 4 | 200% | -28% | U-235 | 0% | Cs-137 |
| Unsaturated Zone Thickness | 5 | -50% | -3% | U-235 | 0% | Cs-137 Sr-90 U-232 |
| | 6 | 150% | 0% | Cs-137 Sr-90 U-232 | 12% | U-235 |
| Irrigation/Pump Rate | 7 | -57% | -1% | U-232 | 65% | I-129 |
| | 8 | 70% | -36% | I-129 | 1% | U-232 |
| Soil/Water Distribution Coefficients ($K_d$) | 9 | lower | -71% | U-234 | 0% | Cs-137 |
| | 10 | higher | -3% | U-232 | 867% | U-234 |
| Hydraulic Conductivity | 11 | -55% | -36% | I-129 | 0% | Cs-137 Sr-90 U-232 |
| | 12 | 57% | 0% | Cs-137 Sr-90 U-232 | 40% | I-129 |
| Runoff/ Evaporation Coefficient | 13 | -23% | -29% | U-234 | 2% | U-232 |
| | 14 | 15% | -2% | U-232 | 79% | I-129 |
| Depth of Well Intake | 15 | -40% | -40% | I-129 | 0.0% | Cs-137 Sr-90 U-232 |
| | 16 | 100% | 0% | Cs-137 Sr-90 U-232 | 99% | I-129 |
| Length Parallel to Aquifer Flow | 17 | -30% | 0% | Cs-137 Sr-90 U-232 | 30% | I-129 |
| | 18 | 21% | -12% | I-129 | 0.0% | Cs-137 Sr-90 U-232 |
| Hydraulic Gradient | 19 | -33% | -23% | I-129 | 0.0% | Cs-137 Sr-90 U-232 |
| | 20 | 33% | 0% | Cs-137 Sr-90 U-232 | 23.3% | I-129 |
| Gamma Shielding Factor | 21 | -38% | 0% | Cs-137 I-129 Sr-90 U-232 U-233 U-234 U-235 U-238 | 0.0% | Cs-137 I-129 Sr-90 U-232 U-233 U-234 U-235 U-238 |

**Table 5-9 Summary of Parameter Sensitivity Analyses – Surface Soil DCGLs[1]**

| Parameter | Run | Change in Sensitivity Parameter | Minimum DCGL Change | | Maximum DCGL Change | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| | 22 | 87% | -24% | U-232 | 0.0% | I-129 |
| Indoor Dust Filtration Factor | 23 | -60% | 0% | Cs-137 I-129 Sr-90 U-234 | 0.2% | U-232 |
| | 24 | -25% | 0% | Cs-137 I-129 Sr-90 U-233 U-234 | 0.1% | U-232 |
| Dust Loading Factor | 25 | -70% | 0% | Cs-137 I-129 Sr-90 U-234 | 0.3% | U-232 |
| | 26 | 67% | 0% | U-232 | 0.0% | Cs-137 I-129 Sr-90 U-235 U-238 |
| Root Depth | 27 | -67% | 0% | Cs-137 I-129 Sr-90 U-232 U-233 U-234 U-235 U-238 | 0.0% | Cs-137 I-129 Sr-90 U-232 U-233 U-234 U-235 U-238 |
| | 28 | 233% | 0% | I-129 | 193.7% | Sr-90 |
| Food Transfer Factors | 29 | lower | -38% | U-235 | 875% | Sr-90 |
| | 30 | higher | -97% | Sr-90 | -42% | U-238 |
| Mass Balance Model | 31 | NA | -67% | U-234 | 0.0% | Cs-137 Sr-90 U-232 |

NOTES: (1) 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 sensitivity analysis 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. 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 a decreased DCGL for U-235 and produced no change for Cs-137, I-129, and U-232. Similarly, increasing the unsaturated zone thickness increased the U-235 DCGL and produced no change for Cs-137, I-129, and U-232. 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 U-234, which ranged from a decrease of 71 percent when lowering the $K_d$, to an increase of 867 percent when increasing the $K_d$.
- Decreasing the hydraulic conductivity significantly reduced the DCGL for I-129 due to reduced dilution and larger groundwater dose relative to other pathways at the time of peak dose. Similarly, increasing the hydraulic conductivity significantly increased the DCGL for I-129.
- Variations in the runoff/evapotranspiration coefficients had the greatest effect on U-234 and I-129, 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 I-129 DCGL, due to increased/decreased dilution in the aquifer.
- Changes to the hydraulic gradient most significantly impacted I-129, due to the large water dependent pathway contributions.
- Decreasing the gamma shielding factor had no impact; however, increasing the shielding factor decreased the U-232 DCGL.
- Changes to the indoor dust filtration factor had minimal impact on DCGLs, due to relatively larger contribution to dose from other pathways.
- Similarly, changes to the dust loading factor had minimal impact on DCGLs, due to relatively larger contribution to dose from other pathways.
- Decreases in root depth did not significantly impact the DCGLs; however, increased root depths impacted Sr-90 most significantly due to relatively large plant pathway doses.

- 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 Sr-90, Cs-137, or 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 initial base-case model DCGLs, which are detailed in Appendix C.

**Table 5-10 Summary of Sensitivity Analyses – Subsurface Soil DCGLs**

| Parameter | Run | Change in Sensitivity Parameter | Minimum DCGL Change | | Maximum DCGL Change | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Indoor/Outdoor Fraction | 1 | -32% | -25% | Cs-137 | 0.3% | U-238 |
| | 2 | 21% | 0% | I-129 | 35% | U-232 |
| Contamination Zone Thickness | 3 | -67% | -65% | U-238 | 170% | Sr-90 |
| | 4 | 233% | -4% | U-232 | 98% | U-234 |
| Unsaturated Zone Thickness | 5 | -50% | -1% | I-129 | 58% | U-238 |
| | 6 | 150% | 0% | Cs-137 Sr-90 U-232 U-235 | 2218% | U-234 |
| Irrigation/Pump Rate | 7 | -57% | -39% | I-129 | 57% | U-238 |
| | 8 | 70% | 0% | Cs-137 | 20% | I-129 |
| Soil/Water Distribution Coefficients ($K_d$) | 9 | lower | -86% | U-238 | 116% | U-232 |
| | 10 | higher | -20% | U-232 | 2168% | U-234 |
| Hydraulic Conductivity | 11 | -55% | 0% | no change | 0% | no change |
| | 12 | 57% | 0% | no change | 0% | no change |
| Runoff/ Evaporation Coefficient | 13 | -23% | -44% | U-234 | 61% | U-238 |
| | 14 | 15% | -11% | U-232 | 117% | U-234 |
| Indoor Gamma Shielding Factor | 15 | -38% | 0% | U-238 | 19% | U-232 |
| | 16 | 87% | -27% | Cs-137 | 1% | U-238 |
| Indoor Dust Filtration Factor | 17 | -60% | 0% | U-238 | 0% | U-235 |
| | 18 | -25% | 0% | Cs-137 I-129 Sr-90 U-233 U-234 U-238 | 0% | U-235 |
| Inhalation Dust Loading | 19 | -70% | 0% | U-238 | 1% | U-233 |
| | 20 | 67% | 0% | U-235 | 0% | Cs-137 I-129 Sr-90 |
| Root Depth | 21 | -67% | -65% | Sr-90 | 1% | U-233 |
| | 22 | 233% | 0% | U-238 | 181% | Sr-90 |
| Food Transfer Factors | 23 | lower | -0.1% | U-238 | 522% | Sr-90 |
| | 24 | higher | -93% | Sr-90 | 0% | U-234 |

**Discussion of Subsurface Soil Results**

The sensitivity analysis results for the subsurface soil source initial base-case model were 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 significant change for U-238. 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, U-234, and U-238. 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 significant changes for U-234 and U-238.
- 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.
- The hydraulic conductivity changes had no impact on DCGLs because the mass balance groundwater model was used.
- 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.
- Changes to the gamma shielding factor most significantly impacted Cs-137 and U-232, based on a relatively large external exposure dose.
- The indoor dust filtration factor variations had no impact on DCGLs, due to relatively large dose contributions from other pathways.
- Changes to the dust loading factor had a minimal impact on DCGLs, due to relatively large dose contributions from other pathways.
- Varying the root zone depth impacted the Sr-90 DCGL most significantly.

- The plant transfer factor is most sensitive for Sr-90, as the dose is mainly due to ingestion via plant uptake.

**Table 5-11 Summary of Sediment DCGL Sensitivity Analysis**

| Parameter | Run | Change in Sensitivity Parameter | Minimum DCGL Change | | Maximum DCGL Change | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Outdoor Fraction | 1 | -50% | 2% | I-129 | 97% | U-232 |
| | 2 | 100% | -50% | U-232 | -3% | I-129 |
| Source Thickness | 3 | -50% | 0% | U-235 | 29% | Sr-90 |
| | 4 | 200% | -23% | U-233 | 0% | Cs-137 |
| Soil/Water Distribution Coefficients ($K_d$) | 5 | lower | -76.5% | U-234 | 26% | U-232 |
| | 6 | higher | -64.5% | U-233 | 52% | U-234 |
| Runoff/Evaporation Coefficient | 7 | -23% | 0% | Cs-137 | 4% | U-232 |
| | 8 | 15% | -3% | I-129 | 0% | Cs-137 |
| Mass Loading for Inhalation | 9 | -70% | 0% | Cs-137 I-129 Sr-90 U-232 | 1% | U-233 |
| | 10 | 67% | -3% | U-234 | 0% | Cs-137 I-129 Sr-90 |
| Root Depth | 11 | -67% | 0% | no change | 0% | no change |
| | 12 | 233% | 0% | U-232 U-235 | 50% | Sr-90 |
| Food Transfer Factors | 13 | lower | 1% | U-232 | 852% | Sr-90 |
| | 14 | higher | -98% | Sr-90 | -13% | U-232 |

**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 external exposure 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.
- Varying the $K_d$ values had a minimal effect on the Cs-137 DCGL, but decreasing the $K_d$ decreased the Sr-90 DCGL 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.
- Changes to the mass loading factor had minimal impact on DCGLs.
- Decreasing the root zone depth did not impact DCGLs; however, increasing the depth increased the Sr-90 DCGL significantly.
- 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.

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

The probabilistic uncertainty analysis discussed in the next subsection provided insight into the degree of conservatism in model input parameters, producing DCGLs that were generally lower than those from the deterministic analyses.

### 5.2.7 Probabilistic Uncertainty Analysis

The probabilistic uncertainty analysis has been performed for each of the three conceptual models to supplement the deterministic sensitivity analyses just described. These probabilistic analyses generated results that quantify the total uncertainty in the

[14] 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."

DCGLs resulting from the variability of key input parameters, and also provide perspective regarding the relative importance of the contributions of different input parameters to the total uncertainty in the DCGLs. This information supports a risk-informed approach to establishing cleanup goals for Phase 1 of the decommissioning.

These analyses were performed using the probabilistic modules of RESRAD version 6.4, which utilize Latin hypercube sampling, a modified Monte Carlo method, allowing for the generation of representative input parameter values from all segments of the input distributions. Input variables for the models were selected randomly from probability distribution functions for each parameter of interest. The number of parameters treated probabilistically for each conceptual model was as follows: surface soil 102, subsurface soil 67, and streambed sediment 63, with these figures including the biotransfer factors and the $K_d$ values for the 18 radionuclides of interest for each zone (contaminated, saturated, unsaturated) and media each model. Appendix E provides details of the analyses.

Table 5-11a summarizes the results of the analyses.

**Table 5-11a. Summary of Results of Probabilistic Uncertainty Analyses(1)**

| Nuclide | Surface Soil DCGLs (pCi/g) | | Subsurface Soil DCGLs (pCi/g) | | Streambed Sediment DCGLs (pCi/g) | |
|---|---|---|---|---|---|---|
| | Determ(2) | Peak-of-the-Mean(3) | Limiting Determ(4) | Peak-of-the-Mean(3) | Determ(5) | Peak-of-the-Mean(3) |
| Am-241 | 4.3E+01 | **2.9E+01** | 7.1E+03 | **6.8E+03** | 1.6E+04 | **1.0E+04** |
| C-14 | 2.0E+01 | **1.6E+01** | **3.7E+05** | 7.2E+05 | 3.4E+03 | **1.8E+03** |
| Cm-243 | 4.1E+01 | **3.5E+01** | 1.2E+03 | **1.1E+03** | 3.6E+03 | **3.1E+03** |
| Cm-244 | 8.2E+01 | **6.5E+01** | 2.3E+04 | **2.2E+04** | 4.8E+04 | **3.8E+03** |
| Cs-137(6) | 2.4E+01 | **1.5E+01** | 4.4E+02 | **3.0E+02** | 1.3E+03 | **1.0E+03** |
| I-129 | 3.5E-01 | **3.3E-01** | **5.2E+01** | 6.7E+02 | 3.7E+03 | **7.9E+02** |
| Np-237 | **9.4E-02** | 2.6E-01 | **4.3E+00** | 9.3E+01 | 5.2E+02 | **3.3E+02** |
| Pu-238 | 5.0E+01 | **4.0E+01** | 1.5E+04 | **1.4E+04** | 2.0E+04 | **1.2E+04** |
| Pu-239 | 4.5E+01 | **2.5E+01** | 1.3E+04 | **1.2E+04** | 1.8E+04 | **1.2E+04** |
| Pu-240 | 4.5E+01 | **2.6E+01** | 1.3E+04 | **1.2E+04** | 1.8E+04 | **1.2E+04** |
| Pu-241 | 1.4E+03 | **1.2E+03** | **2.4E+05** | 2.5E+05 | 5.1E+05 | **3.4E+05** |
| Sr-90(6) | 6.3E+00 | **4.1E+00** | **3.2E+03** | 3.4E+03 | 9.5E+03 | **4.7E+03** |
| Tc-99 | 2.4E+01 | **2.1E+01** | **1.1E+04** | 1.4E+04 | 2.2E+06 | **6.6E+05** |
| U-232 | 5.8E+00 | **1.5E+00** | 1.0E+02 | **7.4E+01** | 2.6E+02 | **2.2E+02** |
| U-233 | 1.9E+01 | **8.3E+00** | **1.9E+02** | 9.9E+03 | 5.7E+04 | **2.2E+04** |
| U-234 | 2.0E+01 | **8.5E+00** | **2.0E+02** | 1.3E+04 | 6.0E+04 | **2.2E+04** |

**Table 5-11a. Summary of Results of Probabilistic Uncertainty Analyses(1)**

| Nuclide | Surface Soil DCGLs (pCi/g) | | Subsurface Soil DCGLs (pCi/g) | | Streambed Sediment DCGLs (pCi/g) | |
|---|---|---|---|---|---|---|
| | Determ(2) | Peak-of-the-Mean(3) | Limiting Determ(4) | Peak-of-the-Mean(3) | Determ(5) | Peak-of-the-Mean(3) |
| U-235 | 1.9E+01 | **3.5E+00** | **2.1E+02** | 9.3E+02 | 2.9E+03 | **2.3E+03** |
| U-238 | 2.1E+01 | **9.8E+00** | **2.1E+02** | 4.6E+03 | 1.2E+04 | **8.2E+03** |

NOTES: (1) Values shown in boldface are lower of the pair of values being compared.
(2) Revised deterministic DCGLs based on parameter changes described in Appendix C.
(3) Probabilistic peak-of-the-mean DCGLs based on analyses described in Appendix E.
(4) These values are the limiting DCGLs for subsurface soil from the residential gardener alternate scenario analysis discussed above. Subsurface soil DCGLs are discussed further in Section 5.2.8, which describes the results of an analysis that takes into account continuing releases from the bottoms of the remediated deep excavations.
(5) These are the revised DCGLs based on parameter changes described in Appendix C.
(6) These values take into account 30 years decay.

Table 5-11a shows that:

- For surface soil, the peak-of-the-mean probabilistic DCGLs are lower than the revised deterministic DCGLs for all radionuclides except Np-237.
- For subsurface soil, the limiting deterministic analysis results from the residential gardener alternative scenario described above are more limiting than the peak-of-the-mean DCGLs for 10 of the 18 radionuclides. (However, the additional deterministic multi-source analysis that includes continuing releases from the bottoms of the remediated deep excavations as discussed in Section 5.2.8 results in even lower DCGLs for many of the radionuclides of interest.)
- For streambed sediment, the peak-of-the-mean DCGLs are more limiting than the revised deterministic DCGLs.

For most radionuclides, the 95$^{th}$ percentile probabilistic DCGLs are lower than the peak-of-the-mean DCGLs as shown in Appendix E. The peak-of-the-mean DCGLs are considered to be appropriate to compare with the deterministic DCGLs because NRC indicates that when using probabilistic dose modeling, the peak-of-the-mean dose distribution should be used for demonstrating compliance with its License Termination Rule in 10 CFR 20, Subpart E (NRC 2006).

After consideration of the results of the probabilistic uncertainty analysis and the analyses of alternate exposures discussed previously, DOE has determined that it is appropriate to use the peak-of-the-mean DCGLs for surface soil and for streambed sediment and the lowest DCGLs of the various subsurface soil evaluations. Subsurface soil DCGLs are addressed in Section 5.2.8.

### 5.2.8 Subsurface Soil DCGL Multi-Source Analysis

As noted in Section 5.2.1, the original base-case conceptual model used in developing the subsurface soil DCGLs recognizes one source of contamination – the Lavery till from the bottom of one of the deep excavations that is brought to the surface during construction of the hypothetical cistern. This model does not consider potential impacts to groundwater in the backfilled excavation from continuing release of remaining residual radioactivity at the bottom of the deep excavations.

To address this limitation, analyses were performed that take into account the impacts of releases of this other residual radioactivity on both a hypothetical residential gardener and a resident farmer with a modified model that accounts for a surface and a subsurface source of radiation. Figure 5-13 illustrates the modified conceptual model used in these analyses.

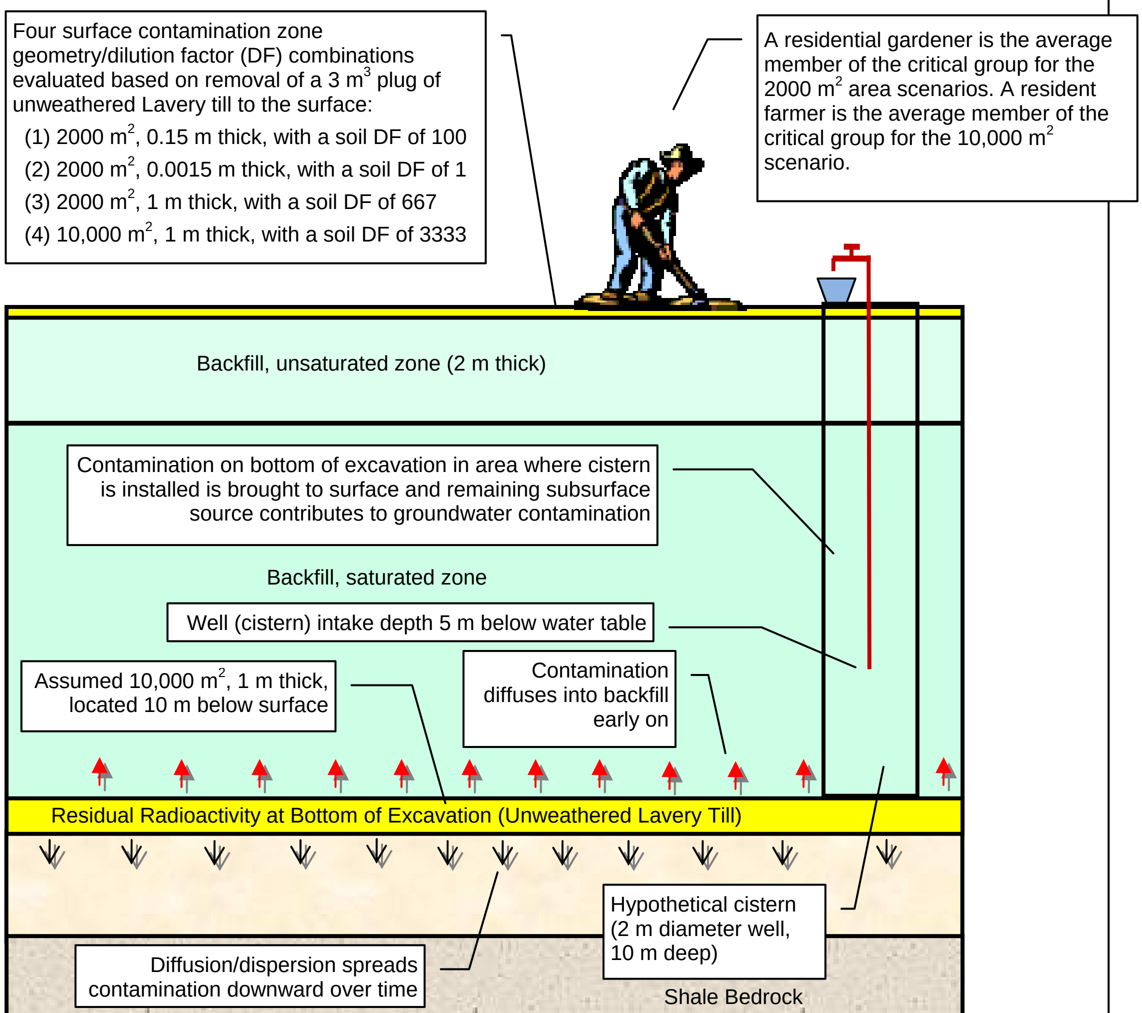



**Figure 5-13. Modified Conceptual Model for Subsurface Soil DCGL Development**

With this model, the subsurface soil DCGLs are based on exposure to residual radioactivity associated with the bottom of the deep excavation in the unweathered Lavery till, with (1) soil from this area assumed to be relocated to the surface during installation of a cistern and (2) with the remaining contaminated Lavery till in the excavation bottom serving as a continuing source of contaminants to groundwater. These sources and the exposure pathways considered are described below.

**Excavation Bottom Treated as Two Sources of Contamination**

The excavation bottom is treated as two distinct sources: (1) a plug of contaminated soil from the excavation bottom that is brought to the surface during installation of the cistern and spread over the entire surface of the hypothetical garden, and (2) the remaining contaminated Lavery till at the excavation bottom from which residual radioactivity moves upward by diffusion and enters groundwater being drawn into the well. Both the residential gardener scenario and the resident farmer scenario were considered as indicated in Figure 5-13.

The surface source that results from the contribution of contamination in soil being removed from the bottom of the excavation and brought to the surface and the contribution of contamination in irrigation water has the following characteristics:

- It is assumed that the contaminated material is evenly spread across the entire hypothetical garden and mixed uniformly in the soil to varying depths (the surface contamination zone),
- Exposure occurs from direct exposure and soil pathways associated with contaminated soil brought to the ground surface, and
- Exposure occurs from groundwater pathways as contaminated water is drawn into the well and used as irrigation water resulting in plant contamination and animal contamination where these plants are used as feed. As a result, the resident is exposed to radioactivity from the plants being consumed and, in the case of the resident farmer scenario, from meat and milk produced from cattle that have been raised on the contaminated feedstock.

The subsurface source remaining at the bottom of the excavation is assumed to have the following characteristics:

- The diffusive movement of contamination from the excavation bottom (the subsurface contamination zone) begins immediately after the excavation is backfilled and results in contaminating the aquifer,
- Contaminated groundwater entering the well is a source to soil in the surface contamination zone because well water is used to irrigate the garden, and
- Drinking water exposure occurs from contaminated well water being used as a source of drinking water.

Table 5-11b shows the exposure pathways evaluated.

**Table 5-11b. Exposure Pathways for Modified Subsurface Soil DCGL Model**

| Exposure Pathways | Residential Gardener | Resident Farmer |
|---|---|---|
| External gamma radiation from contaminated soil | Yes | Yes |
| Inhalation of airborne radioactivity from re-suspended | Yes | Yes |

**Table 5-11b. Exposure Pathways for Modified Subsurface Soil DCGL Model**

| Exposure Pathways | Residential Gardener | Resident Farmer |
|---|---|---|
| contaminated soil | | |
| Plant ingestion (produce impacted by contaminated soil and groundwater contaminated by primary and secondary sources) | Yes | Yes |
| Meat ingestion (beef impacted by contaminated soil and groundwater contaminated by primary and secondary sources) | No | Yes |
| Milk ingestion (impacted by contaminated soil and groundwater contaminated by primary and secondary sources) | No | Yes |
| Aquatic food ingestion | No | No |
| Ingestion of drinking water (from groundwater contaminated by primary and secondary sources) | Yes | Yes |
| Soil ingestion | Yes | Yes |
| Radon inhalation | No | No |

Details of the modeling including values of input parameters such as distribution coefficients appear in the calculation package (Price 2009).

**Mathematical Models**

Calculation of the combined dose utilized information from the three-dimensional near field STOMP finite difference model of the north plateau for groundwater transport, a model that estimated the drinking water dose associated with contamination from the subsurface source diffusing into the aquifer, and RESRAD dose to source ratios associated with unit soil concentrations to determine the total dose from all pathways. The calculations were implemented with a FORTRAN language computer program that estimates time dependent human health impacts.[15]

The model performs mass balance calculations and develops concentrations over time for three distinct areas (1) the remaining subsurface source, (2) the backfilled saturated zone, and (3) the surface which has been contaminated with material excavated from the subsurface source and radionuclides in irrigation water.

In order to identify controlling scenarios, the area of the contaminated zone at the surface and the degree of mixing into the soil of the garden were varied.

The STOMP model was executed with parameter values for the contaminated area and well pumping rates that corresponded with assumptions used in the RESRAD model for the exposure scenarios under consideration. A contaminated area of 10,000 $m^2$ and pumping rate of 5720 $m^3$/y were used to evaluate the resident farmer, and a contaminated area of 2,000 $m^2$ and well pumping rate of 1140 $m^3$/y were used to evaluate the residential gardener scenario. The residential gardener scenario assumed several source

[15] These analyses were deterministic analyses. Consideration was given to performing probabilistic analyses instead. However, the complexity of the multi-source model made a probabilistic analysis impractical.

configurations within the contaminated area for the three $m^3$ of contaminated Lavery till assumed to be excavated to the surface:

- Contamination is spread over the surface in a thin layer (1.5 mm thick) of undiluted till,
- Contamination is spread over the surface and then tilled into the soil to a depth of 15 cm, and
- Contamination is spread over the surface and then tilled into the soil to a depth of 1 m.

The source configuration determined to be most limiting for each radionuclide was used as the basis for the development of the subsurface DCGLs.

**Results**

Table 5-11c shows the results of the analyses compared to DGCLs developed using other conceptual models.

**Table 5-11c. Subsurface Soil DCGL Comparison (pCi/g)**[1]

| Nuclide | Multi-Source | Cistern Well Driller | Recreat. Hiker | Lagoon 3 Erosion | Natural Gas Well Driller | Basic Deterministic Models[2] | Probabilistic Peak of the-Mean |
|---|---|---|---|---|---|---|---|
| Am-241 | **6.3E+03** | 1.7E+04 | 2.7E+05 | 2.9E+05 | 1.4E+05 | 7.1E+03 | 6.8E+03 |
| C-14 | **9.9E+02** | 2.3E+09 | 3.3E+08 | 6.4E+06 | 4.9E+09 | 3.7E+05 | 7.2E+05 |
| Cm-243 | 3.6E+03 | 1.1E+04 | 5.0E+04 | 1.8E+05 | 1.2E+05 | 1.2E+03 | **1.1E+03** |
| Cm-244 | 3.4E+04 | 3.3E+04 | 1.0E+09 | 3.9E+05 | 2.6E+05 | 2.3E+04 | **2.2E+04** |
| Cs-137[3] | 2.8E+03 | 6.7E+03 | 9.8E+05 | 7.4E+05 | 9.2E+04 | 4.4E+02 | **3.0E+02** |
| I-129 | **7.5E+00** | 8.0E+05 | 1.9E+06 | 3.5E+05 | 9.2E+06 | 5.2E+01 | 6.7E+02 |
| Np-237 | **1.0E+00** | 6.6E+03 | 2.7E+04 | 5.9E+05 | 6.6E+04 | 4.3E+00 | 9.3E+01 |
| Pu-238 | **1.3E+04** | 2.0E+04 | 1.5E+06 | 2.7E+05 | 1.6E+05 | 1.5E+04 | 1.4E+04 |
| Pu-239 | **3.1E+03** | 1.9E+04 | 2.8E+05 | 2.4E+05 | 1.5E+05 | 1.3E+04 | 1.2E+04 |
| Pu-240 | **3.4E+03** | 1.9E+04 | 2.8E+05 | 2.4E+05 | 1.5E+05 | 1.3E+04 | 1.2E+04 |
| Pu-241 | 5.5E+05 | 5.5E+05 | 1.7E+07 | 1.2E+07 | 4.5E+06 | **2.4E+05** | 2.5E+05 |
| Sr-90[3] | **2.8E+02** | 8.7E+05 | 1.6E+08 | 9.2E+06 | 1.1E+07 | 3.2E+03 | 3.4E+03 |
| Tc-99 | **5.9E+02** | 7.9E+07 | 2.2E+08 | 4.7E+07 | 9.4E+08 | 1.1E+04 | 1.4E+04 |
| U-232 | 8.8E+01 | 1.6E+03 | 2.8E+04 | 4.5E+05 | 1.6E+04 | 1.0E+02 | **7.4E+01** |
| U-233 | 2.7E+02 | 6.2E+04 | 1.3E+06 | 2.9E+06 | 4.9E+05 | **1.9E+02** | 9.9E+03 |
| U-234 | 2.8E+02 | 6.4E+04 | 1.4E+06 | 3.1E+06 | 5.0E+05 | **2.0E+02** | 1.3E+04 |
| U-235 | 2.9E+02 | 1.2E+04 | 4.2E+04 | 3.2E+06 | 1.4E+05 | **2.1E+02** | 9.3E+02 |
| U-238 | 3.0E+02 | 3.7E+04 | 1.9E+05 | 3.3E+06 | 3.6E+05 | **2.1E+02** | 4.6E+03 |

NOTES: (1) The lowest DCGLs are shown in boldface.
(2) The lower value of the deterministic resident farmer and residential gardener DCGLs.
(3) These values take into account 30 years decay.

In nine cases, the DCGLs developed using other conceptual models are lower than the DCGLs developed by the multi-source model that accounts for continuing releases from the bottom of the deep excavations:

- The peak-of-the-mean probabilistic DCGLs, which did not take into account continuing releases from the bottom of the deep excavations, are lower for Cm-243, Cm-244, Cs-137, and U-232; and
- The limiting deterministic DCGL from the deterministic resident farmer and residential gardener conceptual models, which did not take into account continuing releases from the bottom of the excavations, was lower for Pu-241, U-233, U-234, U-235, and U-238.

This situation can be attributed to conceptual model differences such as different contamination zone geometry.

### 5.2.9 Overall Conclusions

Development of DCGLs proved to be an iterative process.

For surface soil DCGLs, the initial-base case conceptual model was determined to be more conservative than an alternate conceptual model involving erosion and the resulting potential doses to an offsite receptor. However, the probabilistic peak-of-the-mean DCGLs were lower than the base-case deterministic DCGLs for all radionuclides except Np-237. The peak-of-the-mean DCGLs were therefore selected as the basis for the surface soil cleanup goals to be conservative.

For subsurface soil DCGLs, analysis of the residential gardener and the multisource alternate conceptual models showed that the initial base-case resident farmer model was not conservative. The probabilistic uncertainty analysis provided additional insight into potential future doses from residual radioactivity at the bottom of the deep excavations. In the interest of conservatism, the lowest DCGLs produced by the various models were selected as the basis for the subsurface soil cleanup goals.

For streambed sediment DCGLs, the refined base-case model produced essentially the same DCGLs as the initial base-case model. However, the probabilistic peak-of-the-mean DCGLs were lower and were therefore selected as the basis for the cleanup goals.

## 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 will not limit options for Phase 2 of the decommissioning.

### 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.[16] 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 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.

[16] 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.

**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 | 6.3E+03 | 5.7E+03 | 1.0E+04 | 1.0E+03 |
| C-14 | 9.9E+02 | 8.9E+02 | 1.8E+03 | 1.8E+02 |
| Cm-243 | 1.1E+03 | 9.9E+02 | 3.1E+03 | 3.1E+02 |
| Cm-244 | 2.2E+04 | 2.0E+04 | 3.8E+04 | 3.8E+03 |
| Cs-137(3) | 3.0E+02 | 2.7E+02 | 1.0E+03 | 1.0E+02 |
| I-129 | 7.5E+00 | 6.8E+00 | 7.9E+02 | 7.9E+01 |
| Np-237 | 1.0E+00 | 9.0E-01 | 3.2E+02 | 3.2E+01 |
| Pu-238 | 1.3E+04 | 1.2E+04 | 1.2E+04 | 1.2E+03 |
| Pu-239 | 3.1E+03 | 2.8E+03 | 1.2E+04 | 1.2E+03 |
| Pu-240 | 3.4E+03 | 3.1E+03 | 1.2E+04 | 1.2E+03 |
| Pu-241 | 2.4E+05 | 2.2E+05 | 3.4E+05 | 3.4E+04 |
| Sr-90(3) | 2.8E+02 | 2.5E+02 | 4.7E+03 | 4.7E+02 |
| Tc-99 | 5.9E+02 | 5.3E+02 | 6.6E+05 | 6.6E+04 |
| U-232 | 7.4E+01 | 6.7E+01 | 2.2E+02 | 2.2E+01 |
| U-233 | 1.9E+02 | 1.7E+02 | 2.2E+04 | 2.2E+03 |
| U-234 | 2.0E+02 | 1.8E+02 | 2.2E+04 | 2.2E+03 |
| U-235 | 2.1E+02 | 1.9E+02 | 2.3E+03 | 2.3E+02 |
| U-238 | 2.1E+02 | 1.9E+02 | 8.2E+03 | 8.2E+02 |

NOTES: (1) The base-case values for subsurface soil are the lowest values from Table 5-11c and the base-case values for streambed sediment are the lowest values from Table 5-11a.
(2) The results for the analysis of the combined base-case in this table (the lowest DCGLs in the various analyses for subsurface 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 | 2.9E+01 | 2.6E+01 | 1.0E+04 | 1.0E+03 |
| C-14 | 1.6E+01 | 1.5E+01 | 1.8E+03 | 1.8E+02 |
| Cm-243 | 3.5E+01 | 3.1E+01 | 3.1E+03 | 3.1E+02 |
| Cm-244 | 6.5E+01 | 5.8E+01 | 3.8E+04 | 3.8E+03 |
| Cs-137(3) | 1.5E+01 | 1.4E+01 | 1.0E+03 | 1.0E+02 |
| I-129 | 3.3E-01 | 2.9E-01 | 7.9E+02 | 7.9E+01 |
| Np-237 | 2.6E-01 | 2.3E-01 | 3.2E+02 | 3.2E+01 |
| Pu-238 | 4.0E+01 | 3.6E+01 | 1.2E+04 | 1.2E+03 |

**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) |
| Pu-239 | 2.5E+01 | 2.3E+01 | 1.2E+04 | 1.2E+03 |
| Pu-240 | 2.6E+01 | 2.4E+01 | 1.2E+04 | 1.2E+03 |
| Pu-241 | 1.2E+03 | 1.0E+03 | 3.4E+05 | 3.4E+04 |
| Sr-90(3) | 4.1E+00 | 3.7E+00 | 4.7E+03 | 4.7E+02 |
| Tc-99 | 2.1E+01 | 1.9E+01 | 6.6E+05 | 6.6E+04 |
| U-232 | 1.5E+00 | 1.4E+00 | 2.2E+02 | 2.2E+01 |
| U-233 | 8.3E+00 | 7.5E+00 | 2.2E+04 | 2.2E+03 |
| U-234 | 8.4E+00 | 7.6E+00 | 2.2E+04 | 2.2E+03 |
| U-235 | 3.5E+00 | 3.1E+00 | 2.3E+03 | 2.3E+02 |
| U-238 | 9.8E+00 | 8.9E+00 | 8.2E+03 | 8.2E+02 |

NOTES: (1) The base-case values are the lowest values from Table 5-11a.
(2) The results for the analysis of the combined base case in this table (the lowest DCGLs in the various analyses for subsurface 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 will 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 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 will outweigh the benefits. That is, areas where surface soil, subsurface soil, and stream 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 will be performed during the Phase 1 decommissioning activities will make use of updated information, such as actual rather than predicted waste disposal costs. However, the results will 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, will 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 decommissioning activities have achieved the desired dose-based criteria is through the process described in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000). This process is outlined in Section 9, which describes the general content of the Phase 1 Final Status Survey Plan. The Phase 1 Final Status Survey Plan provides the details.

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 will generally remove more material than necessary to meet the DCGL, so it is likely that the post-remediation average concentration will 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. Note that the surface soil cleanup goals apply only to areas of the project premises where there is no subsurface soil contamination and that the subsurface soil cleanup goals apply only to the bottoms and lower sides (extending from a depth of three feet and greater) of the large excavations in WMA 1 and WMA 2.

**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 | 2.6E+01 | 3.9E+03 | 2.8E+03 | 1.2E+04 | 1.0E+03 | 2.1E+04 |
| C-14 | 1.5E+01 | 1.6E+06 | 4.5E+02 | 8.0E+04 | 1.8E+02 | 5.9E+05 |
| Cm-243 | 3.1E+01 | 7.5E+02 | 5.0E+02 | 4.0E+03 | 3.1E+02 | 2.8E+03 |
| Cm-244 | 5.8E+01 | 1.2E+04 | 9.9E+03 | 4.5E+04 | 3.8E+03 | 3.6E+05 |
| Cs-137(4) | 1.4E+01 | 3.0E+02 | 1.4E+02 | 1.7E+03 | 1.0E+02 | 9.4E+02 |
| I-129 | 2.9E-01 | 6.0E+02 | 3.4E+00 | 3.4E+02 | 7.9E+01 | 2.0E+04 |
| Np-237 | 2.3E-01 | 7.5E+01 | 4.5E-01 | 4.3E+01 | 3.2E+01 | 1.1E+03 |
| Pu-238 | 3.6E+01 | 7.6E+03 | 5.9E+03 | 2.8E+04 | 1.2E+03 | 1.7E+05 |
| Pu-239 | 2.3E+01 | 6.9E+03 | 1.4E+03 | 2.6E+04 | 1.2E+03 | 1.7E+05 |
| Pu-240 | 2.4E+01 | 6.9E+03 | 1.5E+03 | 2.6E+04 | 1.2E+03 | 1.7E+05 |
| Pu-241 | 1.0E+03 | 1.3E+05 | 1.1E+05 | 6.8E+05 | 3.4E+04 | 7.5E+05 |
| Sr-90(4) | 3.7E+00 | 7.9E+03 | 1.3E+02 | 7.3E+03 | 4.7E+02 | 7.1E+04 |
| Tc-99 | 1.9E+01 | 2.6E+04 | 2.7E+02 | 1.5E+04 | 6.6E+04 | 4.2E+06 |
| U-232 | 1.4E+00 | 5.9E+01 | 3.3E+01 | 4.2E+02 | 2.2E+01 | 2.1E+02 |
| U-233 | 7.5E+00 | 8.0E+03 | 8.6E+01 | 9.4E+03 | 2.2E+03 | 4.4E+04 |
| U-234 | 7.6E+00 | 1.6E+04 | 9.0E+01 | 9.4E+03 | 2.2E+03 | 2.1E+05 |
| U-235 | 3.1E+00 | 6.1E+02 | 9.5E+01 | 3.3E+03 | 2.3E+02 | 2.0E+03 |
| U-238 | 8.9E+00 | 2.9E+03 | 9.5E+01 | 9.9E+03 | 8.2E+02 | 8.2E+03 |

NOTE: (1) These cleanup goals (CGs) are to be used as the criteria for the remediation activities described in Section 7 of this plan. Note that the streambed sediment cleanup goals will support unrestricted release of the project premises but will not necessarily support restricted release alternatives due to the continued presence of Phase 2 sources as discussed in Section 5.2.2.

(2) The $CG_W$ values for surface soil and streambed sediment are the same as the limited dose assessment DCGL values in the third and fifth columns of Table 5-13, respectively. The $CG_{EMC}$ values are based on the limiting case among the probabilistic analysis resident farmer analysis, the deterministic resident farmer analysis, and the deterministic residential gardener analysis.

(3) These $CG_W$ values are the assessment values in the third column of Table 5-12 reduced by a factor of 0.50 as discussed below. The $DCGL_{EMC}$ values are the limiting values from the multi-source analysis or the deterministic resident farmer/residential gardener deterministic analyses using the 1 $m^2$ area factor from Table 9-2. The subsurface soil cleanup goals apply only to the bottoms of the WMA 1 and WMA 2 deep excavations and to the sides of these excavations more than three feet below the ground surface.

(4) The cleanup goals for Sr-90 and Cs-137 apply to the year 2041 and later, that is, they incorporate a 30-year decay period from 2011. The 30-year decay period was selected for these key radionuclides because of their short half-life. As noted previously, the Phase 2 decision could be made within 10 years of issue of the Record of Decision and Findings Statement documenting the Phase 1 decision. If this approach were to involve unrestricted release of the site, achieving this condition would be expected to take more than 20 years due to the large scope of effort to exhume the underground waste tanks and the NDA. It is therefore highly unlikely that conditions for unrestricted release of the project premises could be established before 2041. If Phase 2 were to involve closing radioactive facilities in place, then institutional controls would remain in place after 2041. DOE will be responsible for maintaining institutional control of the project premises and providing for monitoring and maintenance of the project premises until completion of Phase 2 of the decommissioning.

The basis for these cleanup goals is as follows. Compliance with the cleanup goals used for remediation when mixtures of radionuclides are present will 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 decommissioning will 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.[17]

**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 will 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 will support all potential options for Phase 2 of the decommissioning. As indicated previously, these cleanup goals apply only to the bottom of the large WMA 1 and WMA 2 excavations and to the sides of these excavations three feet or more below the surface.

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

As noted in the discussion on the ALARA analysis results, DOE expects that the actual levels of residual radioactivity will 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 will be refined as appropriate after the data from the soil and sediment characterization program to be completed early in Phase 1 of the 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.

[17] 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.

If evaluation of the new data leads to refinement of the DCGLs and cleanup goals, then this plan will 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 will fall below the dose limit.[18]

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 will 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 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 and the multi-source analysis results as shown in Table 5-11c. The results were as follow:

WMA 1, a maximum of approximately 8 mrem a year

WMA 2, a maximum of approximately 0.2 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. Note that the primary dose driver for these estimates is Sr-90, which accounts for approximately 66 percent of the estimated dose for the WMA 1 excavation and approximately 61 percent of the estimate for the WMA 2 excavation.

NOTE

The use of maximum rather than average values in these dose estimates adds conservatism, as does including values that are simply the highest minimum detectable concentrations, especially in the case of Np-237. (There was a wide range of several orders of magnitude among the minimum detectable concentrations reported for the 2008 sample data.) As with the DCGLs, decay of Sr-90 and Cs-137 over 30 years is accounted for in the estimate.

[18] 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.

As noted previously, DOE will 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 will 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 will 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 decommissioning, depending on the approach selected for that phase.

## 5.5 Monitoring, Maintenance, and Institutional Controls

Inherent in the use of the 30-year decay period used in development of DCGLs and cleanup goals for Sr-90 and Cs-137 is the assumption that all or part of the project premises will not be released for unrestricted use before 2041. DOE will be responsible for monitoring and maintenance of the project premises and for maintaining institutional controls until completion of Phase 2 of the WVDP decommissioning, which is assumed to occur after 2041 if Phase 2 were to be designed to meet unrestricted release criteria. If a close-in-place approach was selected for Phase 2, then institutional controls are assumed to be required beyond 2041.

## 5.6 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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# 6.0 ALARA ANALYSIS

**PURPOSE OF THIS SECTION**

The purpose of this section is to describe how DOE will achieve a decommissioning goal below the 25 mrem per year dose limit in those areas remediated during Phase 1 of the decommissioning and describe quantitative cost-benefit analyses to demonstrate that potential future doses from residual radioactivity in surface soil, subsurface soil, and streambed sediment will 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 will achieve a 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 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 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 decommissioning. The ALARA process has been incorporated into the remediation strategy for the Phase 1 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 decommissioning activities and explains that a soil and sediment characterization program will be undertaken before the 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 will 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 decommissioning will focus on soil within these large excavations and that surface soil in selected areas of the project premises may also be remediated based on the results of the characterization program and on available funding.

Section 7 of this plan provides additional details of Phase 1 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 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 primary 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 decommissioning are ALARA include:

- A Phase 1 decommissioning strategy that promotes ALARA,
- Conservatism inherent in development of DCGLs and the lower cleanup goals that will guide the decontamination efforts, and

- Use of remediation processes that are conservative by nature.

Cost-benefit analyses will be performed during Phase 1 of the 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 decommissioning and in preparation for Phase 2, additional dose evaluations will 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 Decommissioning Strategy Promotes ALARA

As summarized in Section 1.10.2 and detailed in Section 7, DOE's Phase 1 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 will 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 will also be completely removed, including the Vitrification Facility, a process that will significantly reduce risk by reducing residual radioactivity on the project premises.
- The source area of the north plateau groundwater plume beneath the Process Building will be completely removed, a process that will 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 will be left in place after Phase 1 of the 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.[2]
- All radioactive waste generated in Phase 1 decommissioning activities will be disposed of offsite.
- Potentially contaminated soil and sediments within the project premises will be characterized to better define potential risk from residual radioactivity in these media, and surface soil exceeding DCGLs may be remediated in Phase 1, which will effectively eliminate the risk associated with this environmental media contamination.
- Essentially all radioactive material that will remain after the Phase 1 activities have been completed will be located underground, primarily in the underground waste

[2] If the site-wide removal alternative were to be selected for Phase 2 of the decommissioning, the hydraulic barrier walls would be removed during Phase 2.

tanks and in the NDA[3]. Controlled access to the WVDP will continue during the Phase 1 institutional control period, which will prevent access to this underground radioactivity.

### 6.2.2 Good Practices that Promote ALARA

The DOE radiological controls requirements identified in Section 1.7 and the supplemental technical standards associated with those requirements will be followed during the decommissioning activities as specified in Section 7. DOE Policy 441.1, *Department of Energy Radiological Health and Safety Policy*, and the associated implementation guidance, DOE Guide 441.1-2, *Occupational ALARA Program Guide*, include provisions for good practices that promote ALARA. Among these good practices will be:

- The use of spray fixatives or fog sprays during building demolition to reduce the potential spread of airborne contamination;
- The use of engineered surface water run-off controls during building demolition to reduce the potential spread of contamination by precipitation;
- The use of radiological containment to avoid spreading radioactive material during equipment removal, such as removal of piping in the HLW transfer trench and cutting and capping contaminated lines that remain when infrastructure such as the concrete floor slab of the LLW2 Building is removed;
- The use of airborne contamination controls to ensure that doses to workers will be below federally allowed limits;
- The use of personal protective equipment, such as respirators and anti-contamination clothing, in contaminated areas;
- Removal of all demolition debris that may fall within the footprints of removed infrastructure, such as the two-foot deep excavation made to remove the Equipment Shelter;
- Removal of debris remaining in the HLW transfer trench after contaminated piping removal and removal of any radioactive contamination spread to the trench during this work to the extent practicable[4];
- Requiring that the large excavations in WMA 1 and WMA 2 extend at least one foot into the unweathered Lavery till, a geologic unit that is relatively impervious to radionuclide migration;

[3] There may also be residual radioactivity above cleanup goals in surface soil and sediment, but this amount would be a small fraction of residual radioactivity below the surface.

[4] The HLW transfer trench is the only facility within the scope of the Phase 1 of the WVDP decommissioning that will remain in place. It is not expected to be radioactively contaminated when the piping removal begins. Even though radiological containment will be used in removal of the piping, spills during this work are possible.

- Removing easily removable contaminated soil in the large WMA 1 and WMA 2 excavations; and
- Installation of infiltration and surface water run-off controls such as liners, drainage collection systems, and berms below and around excavated soil laydown areas to prevent migration of contaminants into underlying groundwater and nearby surface waters.

Additional guidance in DOE-STD-ALARA1draft will also be considered.

#### 6.2.3 Conservatism in DCGL Development

The process for developing DCGLs for Phase 1 of the decommissioning as described in Section 5 was conservative in several respects. Section 5 provides examples of this conservatism. (As explained in Section 5, a probabilistic uncertainty analysis was performed to evaluate whether key input parameters used in DCGL development were sufficiently conservative and probabilistic peak-of-the-mean DCGLs are being used as the basis for surface soil and streambed sediment cleanup goals.)

#### 6.2.4 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 will 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 will 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 will 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 will 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.

**NOTE**

DOE has performed a preliminary ALARA analysis and provided for a later, more detailed ALARA analysis that will be performed during the remediation work. This approach is appropriate for Phase 1 of the decommissioning since information used in the analyses may change between the time of Decommissioning Plan issue and the time when remediation of the large excavations – the activity for which the analyses are most important – takes place. For example, waste disposal costs could increase significantly and possibly change the outcome of the analyses.

The results of the preliminary analysis provide useful information for planning purposes, even though it is possible that the later, more detailed analysis will produce different results. This two-step approach is consistent with guidance in Appendix N of NUREG-1757, Volume 2 (NRC 2006)

### 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:

| | | | |
|---|---|---|---|
| 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),[5] |
| r | = | monetary discount rate (per year), |
| $\lambda$ | = | 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:

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\text{Cost}_T}{\$2000 \times P_D \times 0.025 \times F \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}}$$

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 $DCGL_W$ 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$^2$).

[5] 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).

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.[6]

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

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

Therefore, since for subsurface soil: (1) the Conc to $DCGL_W$ ratio (R) will be the same as for surface soil, (2) the cost to remediate will likely be higher than for surface soil, and

[6] 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.

(3) surface soil at the $DCGL_W$ is ALARA, it is concluded that remediation below the subsurface 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.[7] 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 will 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 will 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 will likely be more costly than the remediation of an equivalent amount of surface soil.

Therefore, since for streambed sediments: (1) the Conc to $DCGL_W$ ratio (R) will be the same as for surface soil, (2) the cost to remediate will 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.

### 6.3.6 Addressing Intergenerational Concerns

The consequences (i.e., doses) of the remediation that will take place during Phase 1 of the decommissioning could occur over a lengthy period, especially if Phase 2 of the decommissioning were to involve a site-wide removal approach resulting in unrestricted release of the property. (In a Phase 2 site-wide close-in-place approach, the potential future doses from the remediated Phase 1 areas would be small compared to those from the Phase 2 source areas.) The impact of intergenerational doses on the cost-benefit analysis can be evaluated by considering the impact of lower discount rates.[8]

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[7] 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.

[8] Based on Office of Management and Budget guidance, present worth calculations are normally performed using both three and seven percent real discount rates. These discount rates are used to calculate the present worth of averted health effects regardless of when these effects are adverted. The three percent rate (as used in Section 6.3.3) approximates the real rate of return on long-term government debt, which serves as proxy for the real rate of return on savings. (NRC 2004)

Figure 6-1 shows the unit cost of remediation ($C_{Tu}$) as a function of the discount rate. It shows that with a discount rate of zero, the cost of remediation would be approximately $20/m$^2$. Because this unit cost is less than the $36.38/m$^2$ disposal component of the total remediation cost in the preliminary analysis (Section 6.3.3), the DCGLs result in intergenerational doses that are ALARA and further remediation would not be necessary.

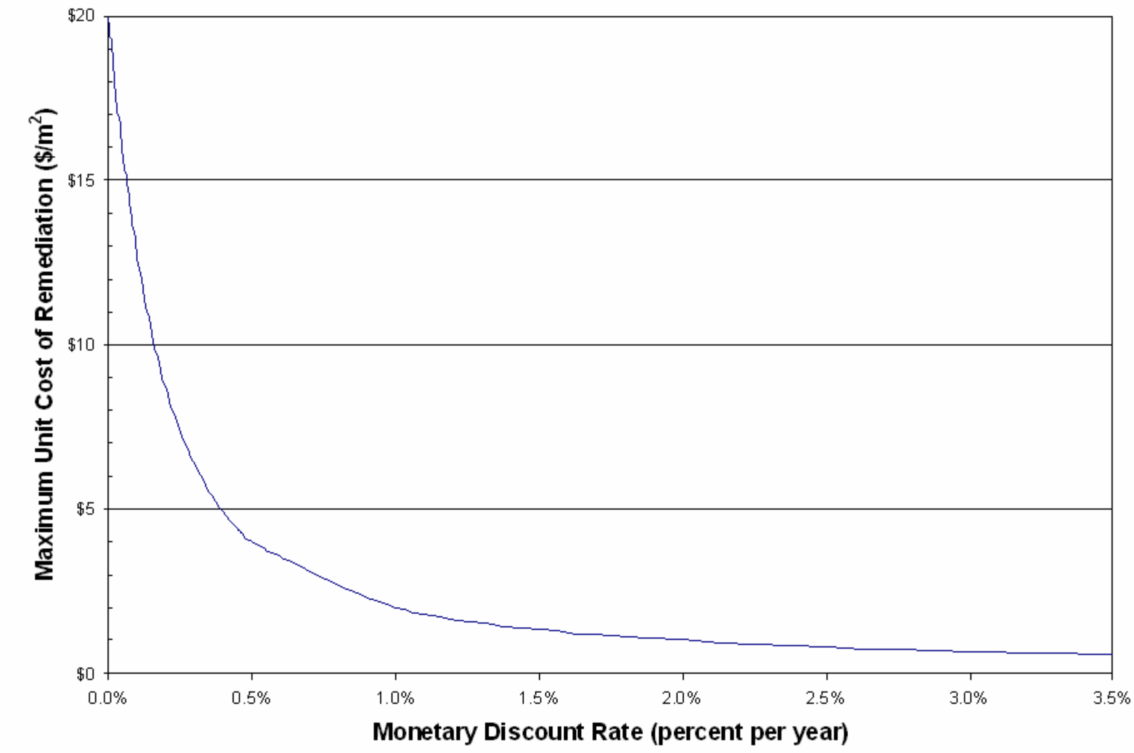


**Figure 6-1. Unit Remediation Costs vs. Monetary Discount Rate**

## 6.4 Additional Analyses

Additional ALARA analyses will be performed in connection with remediation of the WMA 1 and WMA 2 excavations. These analyses will 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 will be taken into account in the additional ALARA analyses. Consideration will 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, will be cost-effective. Consideration will be given as well to the effects of using lower discount rates on the estimated cost of remediation so that intergenerational concerns are taken into account.

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 decommissioning will support all potential options for Phase 2.

Also, as described in Section 5, a final dose analysis will be performed using Phase 1 final status survey data from 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, Policies, Standards, and Guides**

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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 7.11, descriptions of the Phase 1 decommissioning activities;
- In Section 7.12, a summary of the types of remediation and demolition technologies to be employed; and
- In Section 7.13, 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 will 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 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 will be excavated, although some surface and subsurface soil contamination is expected to be present in other WMAs.

**Table 7-1. Facility and Area Conditions Expected 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. |
| | 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 Expected 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 decommissioning scope.] |
| 5 | Lag Storage Addition 4, Shipping 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 decommissioning scope.] |
| 9 | Drum Cell | No contamination above MDC. |
| 10 | New Warehouse | Not impacted by radioactivity. |

NOTES: (1) See also Table 2-13 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, the WVDP decommissioning will be accomplished in two phases. The following activities will take place in Phase 1.

**Facility and Equipment Removal**

The following facilities and equipment will 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 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/Shipping Depot 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 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;
- 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 will be characterized for radioactivity. Before the Process Building is removed, the new Canister Interim Storage Facility will be established 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 will be dug to remove the WMA 1 facilities and a second large excavation dug to remove key WMA 2 facilities. These excavations will extend down into the underlying Lavery till. Contaminated surface and subsurface soil in these excavations will be removed to achieve the cleanup goals for unrestricted release specified in Section

5[1]. The source area of the north plateau groundwater plume in WMA 1 will 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 will be sequenced for maximum efficiency. For example, the Low-Level Waste Treatment Facility will 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 will be used in management of the project.

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

### 7.2.2 General Requirements

The following general requirements will be adhered to during decommissioning activities described in Sections 7.3 through 7.11.

**Use of Approved Written Procedures**

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

**Remedial Technologies**

The decommissioning contractor will utilize efficient, proven technologies in accomplishment of the work. Section 7.12 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 potential importance in support of Phase 2 of the decommissioning. The Decommissioning Work Plan(s) will provide more-detailed information on remedial technologies to be used.

**Dealing With Unique Remediation Issues**

Given the complexities of the site, some remediation issues will be faced during Phase 1 of the WVDP decommissioning that are highly unusual, if not entirely unique. Two such issues are demolition of the Process Building and removal of the radioactive contamination in the source area of the north plateau groundwater plume that extends far below the building.

[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. The surface soil cleanup goals will be applied from the ground surface to a depth of three feet. The subsurface soil cleanup goals apply only to the large WMA 1 and WMA 2 excavations, including the excavation sides to within one meter (3.3 feet) of the surface.

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 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.

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 will be taken as necessary to eliminate or reduce potential impacts to human health and the environment during the Phase 1 decommissioning work and to prevent contamination of non-impacted areas of the project premises and recontamination of remediated areas.

The large excavations for WMA 1 and WMA 2 will be planned to minimize the impacts associated with handling of removed contaminated soil. Methods such as the following will be used to mitigate potential impacts from excavated contaminated soil:

- Arranging excavated soil in the laydown areas to facilitate radiological surveys and sampling of the soil for waste characterization purposes,
- Protecting laydown areas with a suitable covering material,
- Using water spray to minimize airborne radioactivity from piles of dry excavated contaminated soil;
- Placing suitable covering material over excavated soil to prevent the spread of contamination by precipitation;
- Establishing earthen berms equipped with runoff collection capability around the laydown areas to control surface water runoff; and
- Making provisions for sampling, removal, appropriate treatment, and disposal of water collected inside the berms, such as releasing it through a permitted outfall.

Such measures will also be used as practical in managing contaminated soil excavated during infrastructure removal, such as during the removal of foundations and floor slabs.

Fixatives and water spray will be used as necessary to minimize airborne radioactivity during demolition of contaminated structures and equipment. Suitable covering material will be placed over radioactive waste stored outdoors to help prevent the spread of contamination.

Confinement structures also will 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 will be taken also during demolition of noncontaminated concrete and steel and during transportation of waste generated in such work.

Mitigative measures will include as low as reasonably achievable (ALARA) considerations, such as removal of contaminated soil to concentrations below the cleanup goals in cases where this will be practical.

Special emphasis will be placed on measures to ensure that areas remediated during Phase 1 are not re-contaminated during subsequent Phase 1 decommissioning activities and that those areas not impacted by radioactivity are not inadvertently contaminated. Such measures will include use of suitable barriers, such as temporary fences, and warning signs.

Mitigative measures will also be taken to minimize impacts to areas where slurry will be mixed in connection with installing the hydraulic barriers as described in Section 7.3.8. Measures will also be taken to avoid damage to the hydraulic barriers after they are installed from subsequent Phase 1 decommissioning activities. These measures will include protecting the barriers from impacts associated with the movement of heavy equipment, such as by the use of temporary load-distributing or bridging spans at the ground surface in the locations where such equipment will cross the barriers.

Details will be provided in the Decommissioning Work Plan(s) or in a separate Mitigative Measures Plan.

**Radiological Controls**

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

**Worker Safety**

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

**Waste Management**

All waste generated during Phase 1 of the decommissioning will be disposed of offsite. The Waste Management Plan will implement DOE procedures identified in Section 1.9 and provide requirements and guidance for management of all types of waste.

In accordance with the Waste Management Plan, radioactive waste generated during proposed decommissioning activities will be characterized and disposed of offsite at appropriate government-owned or commercial disposal facilities. Hazardous and toxic waste will be managed and disposed of offsite in accordance with applicable requirements. Non-radioactive equipment and demolition debris will be disposed of offsite at a construction and demolition debris landfill.

DOE policies on waste minimization, pollution prevention, and recycling will be followed as specified in DOE Manual 435.1-1 *Radioactive Waste Management Manual*. Recycling of surplus equipment and metals such as radioactively contaminated lead in accordance with appropriate guidance will be considered.

Soil laydown areas will be located following guidance in the Waste Management Plan. Mitigative measures will be implemented for these areas as discussed previously. After the soil and ground covering material have been removed from these areas, they will be considered to have been impacted by radioactivity, even if there were no known spills. Phase 1 final status surveys will be performed in these areas as specified in Section 9 and the Phase 1 Final Status Survey Plan.[2]

**Backfill Soil**

Soil used as backfill in deep and shallow excavations associated with Phase 1 decommissioning activities will be obtained from outside the Center from an area that has not been impacted by site radioactivity. The properties of soil to be used as backfill in the deep excavations in WMA 1 and WMA 2 – especially the texture, hydraulic conductivity, and distribution coefficients – will be similar to those of the sand and gravel layer on the project premises as described in Section 3.

No soil removed during the excavation work will be used in filling an excavation, even if that soil were determined to be uncontaminated.

**Quality Assurance**

The quality assurance requirements of Section 8 will 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 will be implemented in other 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 will later be developed consistent with the DOE policy stated above. Likewise, more detailed designs will later be developed for engineered barriers and other engineered features of the decommissioning.

**Characterization**

As indicated in Section 4, the WVDP facilities and areas had not been completely characterized for radioactivity as of 2009. Additional characterization will be performed as necessary in accordance with the Characterization Sample and Analysis Plan, as explained in Section 9.

[2] Contamination found in excess of surface soil cleanup goals will be remediated as specified in Section 9.6. DOE may approve an exception to this requirement if the laydown area is located in a part of the project premises known to have subsurface radioactivity, or if surface soil contamination in excess of the cleanup goals was known to be present prior to establishing the laydown area.

The Characterization Sample and Analysis Plan will provide for characterizing soil and sediment. This characterization program will include the banks and streambeds of the portions of Erdman Brook and Franks Creek located on the project premises[3].

Characterization of subsurface soil in the area of the large WMA 1 and WMA 2 excavations will include collecting samples in the top portion of the Lavery till, as well as samples in the sand and gravel layer above the till. Samples of subsurface soil will also be collected along the upgradient and cross-gradient sides of the excavation footprint in WMA 1 and on the sides of the WMA 2 excavation footprint. Analytical data from these samples (1) will 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, (3) will help plan excavated soil management, and (4) will support planning Phase 1 final status surveys to be performed on the sides of the excavations.

Characterization measurements will include those necessary for waste management purposes. The Waste Management Plan will address characterizing excavated soil for waste management purposes, including surface and subsurface soil that is not likely to have been contaminated by by radioactivity.

Note that the specific 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 decommissioning activities will be changed as necessary to ensure that conditions at the conclusion of the Phase 1 decommissioning activities meet the DCGLs (i.e., the cleanup goals). This plan will 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 Table 5-14 in Section 5. The DCGLs and cleanup goals for Sr-90 and Cs-137 are based on a 30-year decay period, as discussed in Section 5.2.

**ALARA Analyses**

The results of the preliminary ALARA analysis are described in Section 6. As specified in Section 6, additional ALARA analyses will be performed during the WMA 1 and WMA 2 excavations using in-process survey data. These analyses will determine whether remediation to residual radioactivity concentrations below the cleanup goals will be cost-effective. If this is determined to be the case, then additional subsurface soil will be removed as indicated by the results of the analyses.

**Establishing Areas Where Surface Soil Meets Cleanup Goals**

DOE may elect to establish during Phase 1 of the decommissioning that certain areas of the project premises meet the surface soil cleanup goals, depending on factors such as

[3] 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.

characterization results and available project funding. Any such areas would be selected after evaluation of data from the characterization program; only areas with no subsurface contamination below one meter from the surface would be selected.

Surface soil in these selected areas would be remediated as necessary to achieve the surface soil cleanup goals although no remediation may be necessary in some areas. Phase 1 final status surveys would be performed in the selected areas, along with any confirmatory surveys desired by NRC. Details are provided in Section 7.11.

**In-Process Radiological Surveys**

In-process surveys will be performed in connection with the decommissioning activities for radiation protection and waste management purposes in accordance with the requirements of Section 9. These surveys will include sampling of excavated soil from both the deep and shallow excavations to support waste management.

**Radiological Status Surveys of Shallow Excavations**

Radiological status surveys will be performed in the shallow excavations resulting from removal of infrastructure such as concrete floor slabs and foundations and gravel pads. The Characterization Sample and Analysis Plan will provide the requirements for these surveys.

**Final Status Surveys and Confirmatory Surveys**

Phase 1 final status surveys will be accomplished in accordance with the Phase 1 Final Status Survey Plan, which will also address confirmatory surveys to be performed by NRC or its contractor, as explained in Section 9 of this plan. When Phase 1 final status surveys are specified below, inherent in the survey process will be any additional remediation necessary to achieve the cleanup criteria and resurveys of areas remediated to ensure that the criteria were achieved.[4]

The Phase 1 final status surveys focus primarily on areas to be made inaccessible by proposed decommissioning activities. Phase 1 final status surveys will 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 will be filled with radiologically uncontaminated earth imported from offsite only after the Phase 1 final status surveys and confirmatory surveys have been accomplished and the resulting data reviewed and accepted.

Phase 1 final status surveys will also be performed in excavated soil laydown areas and may also be performed in selected areas with no subsurface contamination that can meet surface soil cleanup criteria for unrestricted release.

For an excavated soil laydown area, Phase 1 final status surveys will be performed after the excavated soil and the ground covering is removed. The purpose of such surveys is generally to verify that the surface soil meets the cleanup goals. However, if the laydown

[4] Section 9 uses the term *Phase 1 final status surveys* to describe these surveys of excavations, which will follow the final status survey protocols of the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000).

area is known to have subsurface soil contamination, then the purpose of the surveys is to document the surface soil radiological conditions because such an area could not meet criteria for unrestricted release based only on surface soil contamination criteria.

Impacted areas that could be released for unrestricted use based on meeting surface soil cleanup goals may be identified during the characterization program, as noted previously. DOE will notify NRC at least 60 days before performing Phase 1 final status surveys to demonstrate that a particular area meets criteria for unrestricted release.

Surveys of excavations to remove infrastructure will be performed in accordance with the Characterization Sample and Analysis Plan, not the Phase 1 Final Status Survey Plan. An example of such a survey would be the shallow excavation made to remove the LLW2 Building floor slab and foundation.

**Monitoring, Maintenance, and Security**

DOE will be responsible for monitoring and maintenance of the project premises and for institutional controls until completion of Phase 2 of the WVDP decommissioning, which is assumed to occur after 2041. Details are provided in Appendix D.

## 7.3 WMA 1 Decommissioning Activities

This section describes the decommissioning activities in WMA 1, the Process Building and Vitrification Facility area, to be accomplished in Phase 1. Figure 7-1 shows WMA 1.

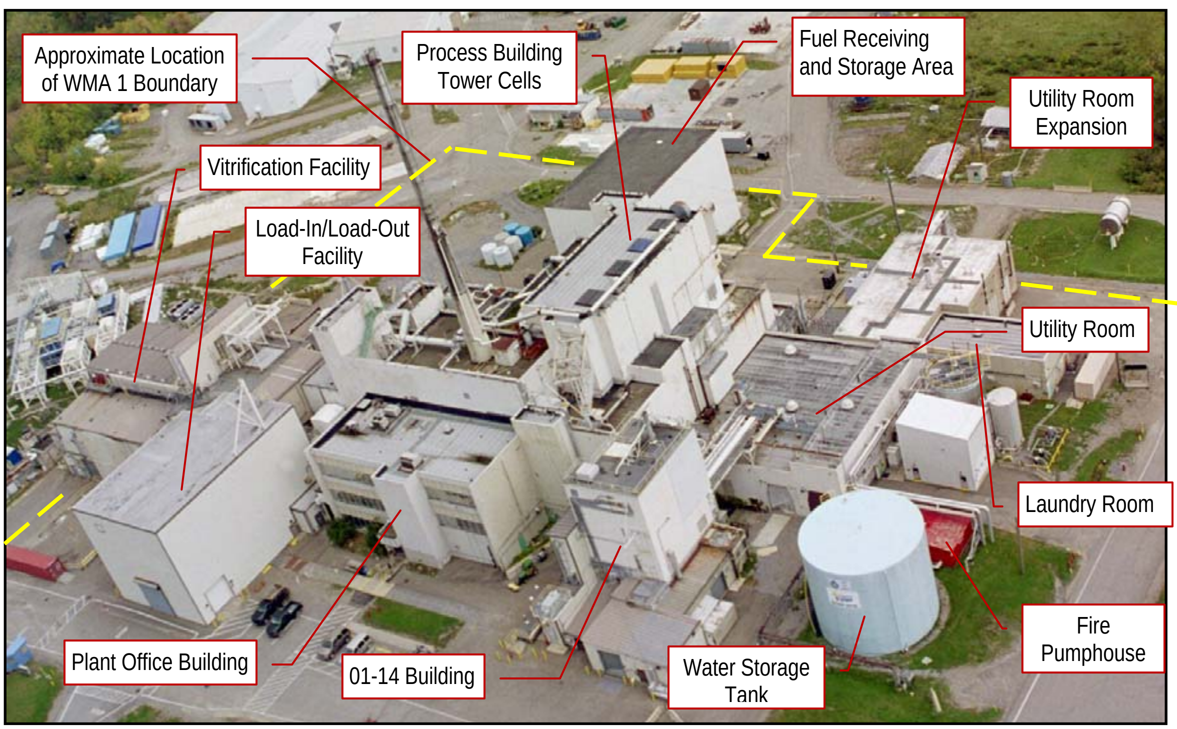



**Figure 7-1. WMA 1 in 2007**

### 7.3.1 Characterizing Soil and Streambed Sediment

Soil and sediment in WMA 1 will be characterized for residual radioactivity in accordance with the Characterization Sample and Analysis Plan described in Section 9. The results of this effort will be used in planning the excavation work described below.

### 7.3.2 Relocating the Vitrified HLW Canisters

The 275 vitrified HLW canisters will 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) will be set up on the south plateau. The Equipment Decontamination Room will be modified to support handling the vitrified HLW canisters and the Load-In Facility will be converted to a Load-Out Facility. The vitrified HLW canisters will then be moved from the HLW Interim Storage Facility (the former Chemical Process Cell) and loaded into shielded dry storage canisters. Each storage canister will be placed in a shielded onsite transport cask and moved by truck to the new Canister Interim Storage Facility. The storage canisters will be maintained there in protective storage until a decision is made and implemented with regard to disposal of the HLW canisters.

This approach is among several approaches described in conceptual engineering studies (WVNSCO and Scientech 2000, WVES 2009b) which are currently under evaluation by DOE. If this approach is selected by DOE, detailed designs based on the preliminary conceptual designs will be developed. These designs will 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 2008)[5]. 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 will include 69 shielded canisters and shielded modules made of reinforced concrete in which to store these shielded canisters. Each shielded canister will 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 offsite within a shielded transport cask by rail. The shielded canisters will be used for both onsite storage within the reinforced concrete storage modules and transport within a shielded transport cask.

The onsite shielded transport cask will be capable of (1) holding a single shielded canister, (2) loading and discharging the shielded canister in a horizontal position, and (3)

[5] 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 will be revised to reflect the appropriate changes.

**Modifications to the Equipment Decontamination Room**

These modifications will 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 will 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 will 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 will be constructed at the east wall of the facility between the shield door to the Equipment Decontamination Room and the air lock. This cell will 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 will 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 will 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 will 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 will be supported by jacks for stability, the open end of the onsite transport cask will be positioned within the outer part of the transfer cell to provide necessary radiation shielding, and the loaded shielded canister will be inserted into the cask and the cask shield plug installed. Figure 7-2 shows the conceptual arrangement.

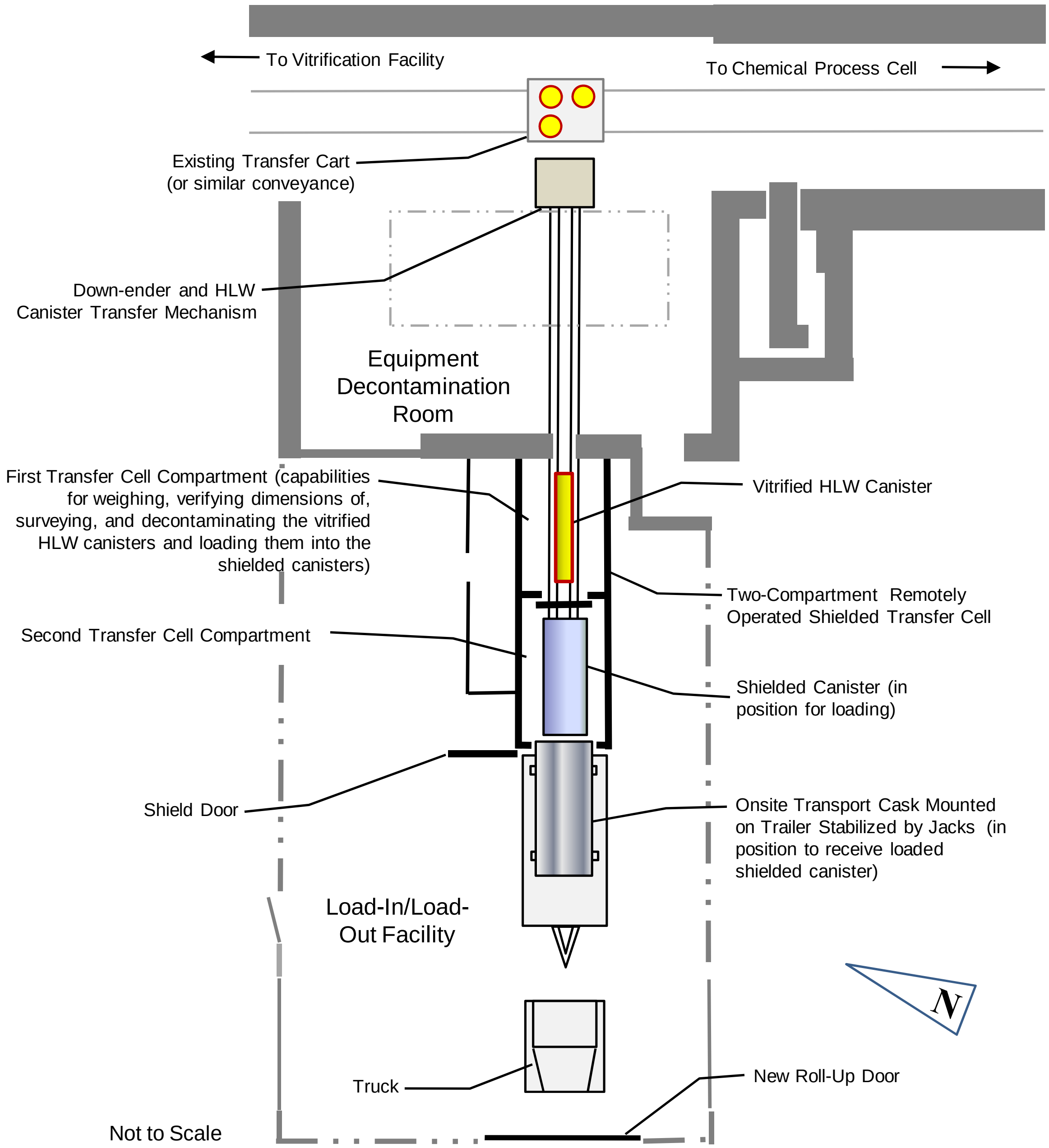



**Figure 7-2. Conceptual Arrangement for Transferring Vitrified HLW Canisters**

**Construction of the New Canister Interim Storage Facility**

The new Canister Interim Storage Facility will be constructed on the south plateau near the rail spur.[6] The facility will consist of a reinforced concrete pad with reinforced concrete storage modules to provide radiation shielding and mechanical protection. The concrete pad will be sufficient in size and load capacity to accommodate reinforced concrete storage modules for the 69 loaded shielded canisters. Soil in the area will be characterized for

[6] A report of a detailed evaluation of HLW canister storage options issued in September 2009 recommended this location (WVES 2009b).

geotechnical parameters to support the detailed design of the facility. The soil will also be characterized for radioactivity, either in connection with the geotechnical investigation or as provided for in the Characterization Sample and Analysis Plan.

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 will necessarily select this interim storage system, which is among a variety of systems approved by NRC for general use that will be considered by DOE.)

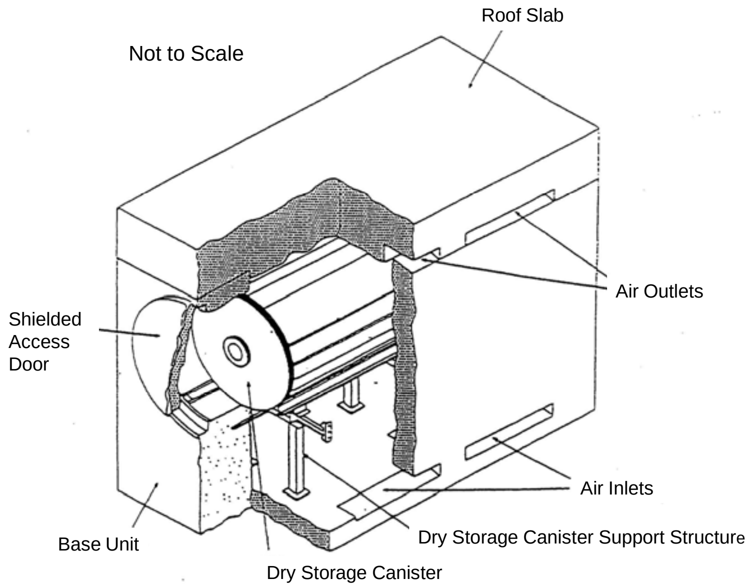



**Figure 7-3. Storage Module Conceptual Design** (from WVNSCO and Scientech 2000)

Appropriate fence(s), lighting, and remote monitoring equipment for security purposes will be provided. DOE will 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 will provide information on the detailed design of the facility to NRC and consult with NRC on preparation of the related documented safety analysis.

**Moving the Vitrified HLW Canisters to the New Canister Interim Storage Facility**

A process such as the following will be used to transport the vitrified HLW canisters to the new Canister Interim Storage Facility:

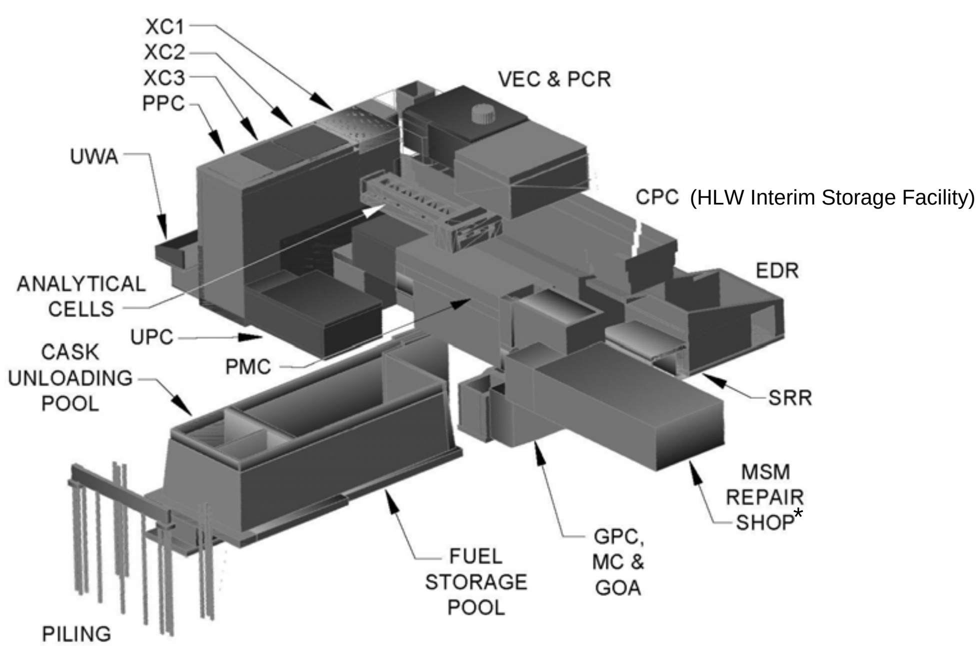
- Readiness reviews will be performed to ensure that all preparations for the move have been satisfactorily completed;
- The first shielded canister will be placed inside the shielded transfer cell;
- The onsite transport cask to receive the first shielded canister will be moved into the Load-In/Load-Out Facility and positioned next to the transfer cell;
- The first group of four vitrified HLW canisters will be moved into the Equipment Decontamination Room on the transfer cart or similar conveyance;

- The vitrified HLW canisters will be lifted from the cart one by one, lowered to a horizontal position, and moved into the transfer cell where appropriate measurements will be taken;
- After measurements and any necessary decontamination are completed, each of the four vitrified HLW canisters will be loaded into a shielded canister and the shielded canister will be loaded into the onsite transport cask; and
- The cask will be transported to the new Canister Interim Storage Facility where the shielded canister will be inserted into the designated reinforced concrete storage module and the module shielded access door installed.

This process will 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
EDR = Equipment Decontamination Room
GOA = General Purpose Cell Operating Aisle
GPC = General Purpose Cell
MC = Miniature Cell
MSM = Master-Slave Manipulator
PCR = Process Chemical Room
PMC = Process Mechanical Cell
PPC = Product Purification Cell
SRR = Scrap Removal Room
UPC = Uranium Product Cell
UWA = Upper Warm Aisle
VEC = Ventilation Exhaust Cell
XC1 = Extraction Cell 1
XC2 = Extraction Cell 2
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 will be performed as specified below. The below-ground portion of the building will be removed as specified in Section 7.3.8. As indicated previously, this work will be performed in accordance with the Decommissioning Work Plan, which will provide more details on the activities described below.

**Removing Equipment**

Equipment will be removed in connection with 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 – will also be removed. This equipment will be size reduced as necessary, characterized, packaged, and disposed of offsite. Size reduction will 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 will be removed to the extent practical before demolition. These materials will 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 mercury, such as switches, relays, and fluorescent lamps;
- Any electrical equipment known to contain polychlorinated biphenyls (PCBs); and
- Any remaining piping insulation known to contain asbestos.

These materials will be size reduced as necessary, characterized, packaged, and disposed of at an appropriate offsite disposal facility. Lead-based paint affixed to facility surfaces would not have to be removed from the demolition debris for the debris to be disposed of as radioactive waste at a disposal facility such as the EnergySolutions Clive, Utah facility or as non-radioactive waste at a construction and demolition debris landfill.

**Completing Process Building Decontamination**

Process Building areas known to have significant residual radioactivity will 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
- Equipment Decontamination Room
- Ventilation Exhaust Cell
- Head-End Ventilation Building

The process used will 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 results, verify that the Process Building can be demolished without exceeding 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 will be accomplished.

**Removing the Building to Grade Level**

The Process Building will be demolished to grade level using conventional demolition methods such as those described in Section 7.12. Fixatives will be applied to building surfaces with significant radioactive contamination before this is accomplished to help avoid the need for radiological containment. A fog spray will be used as appropriate during the demolition process. The resulting debris will 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 will 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 will be performed as specified below. The below-grade portion of the building will be removed as specified in Section 7.3.8.

**Preparing for Facility Removal**

Preparations to remove the Vitrification Facility to grade will be similar to those for the Process Building. Installed equipment will 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 will be evaluated to ensure compliance with NESHAP emission limits during demolition. Fixatives will be applied to surfaces with significant radioactive contamination levels.

**Removal of the Facility to Grade Level**

After such preparations are completed, the Vitrification Facility will be removed to grade level using conventional demolition methods such as those described in Section 7.12. The thick reinforced concrete walls and roof structures will be segmented as necessary using a technique such as diamond wire cutting.

The resulting debris will 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 will 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. Figure 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 will 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 will remain in place temporarily and will 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 will be cut into segments, removed from the concrete trench, characterized, packaged for disposal, and disposed of at an offsite disposal facility. The trench itself will remain in place temporarily and will 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 will be similar to the process used for the 01-14 Building and will 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 will remain in place temporarily and will 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 will be removed to their concrete floor slabs and foundations, which will removed during excavation of the subgrade structures and the plume source area. (Note that some WMA 1 facilities and parts of others lie outside of footprint of the large excavation; these are discussed below.)

**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 decommissioning process for these facilities will 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 will 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 late 2009, this 20 feet by 24 feet by 10 feet high steel building was not known to have been impacted by radioactivity. Decommissioning this structure will 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 2009. Decommissioning will 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 2009. Decommissioning will 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.

**Removal of Floor Slabs and Foundations Outside of the Large Excavation**

The floor slabs and foundations of those WMA 1 structures that lie outside of the footprint of the large excavation will be removed. These structures include portions of the Utility Room and Utility Room Expansion, the Fire Pumphouse, and the Water Storage Tank. This work, which will be coordinated with the activities in Sections 7.3.8 and 7.3.9, will entail using a process such as the following:

- Removing the concrete and the underlying soil to approximately two feet below grade;
- Disposing of the demolition debris and soil at appropriate offsite facilities;
- Performing radiological status surveys in the shallow excavations;
- Evaluating the resulting data and performing Phase 1 final status surveys as applicable in accordance with Section 7.11;
- Arranging for any confirmatory surveys to be performed; and
- After all of the surveys are completed and any issues resolved, backfilling the shallow excavations with clean earthen backfill.

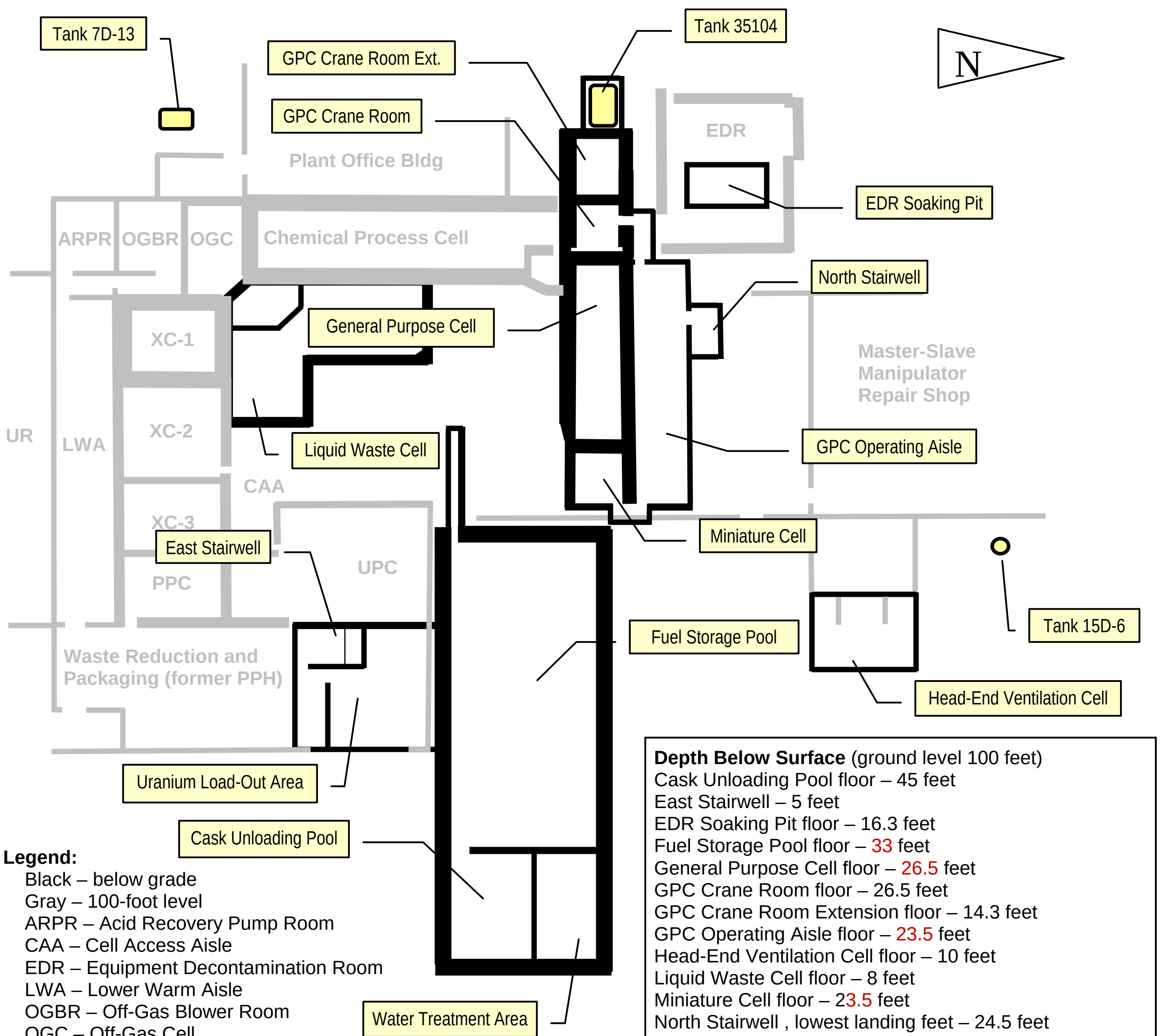
### 7.3.8 Removing the Underground Structures and Equipment and the Plume Source Area

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 will 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 assumed 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).

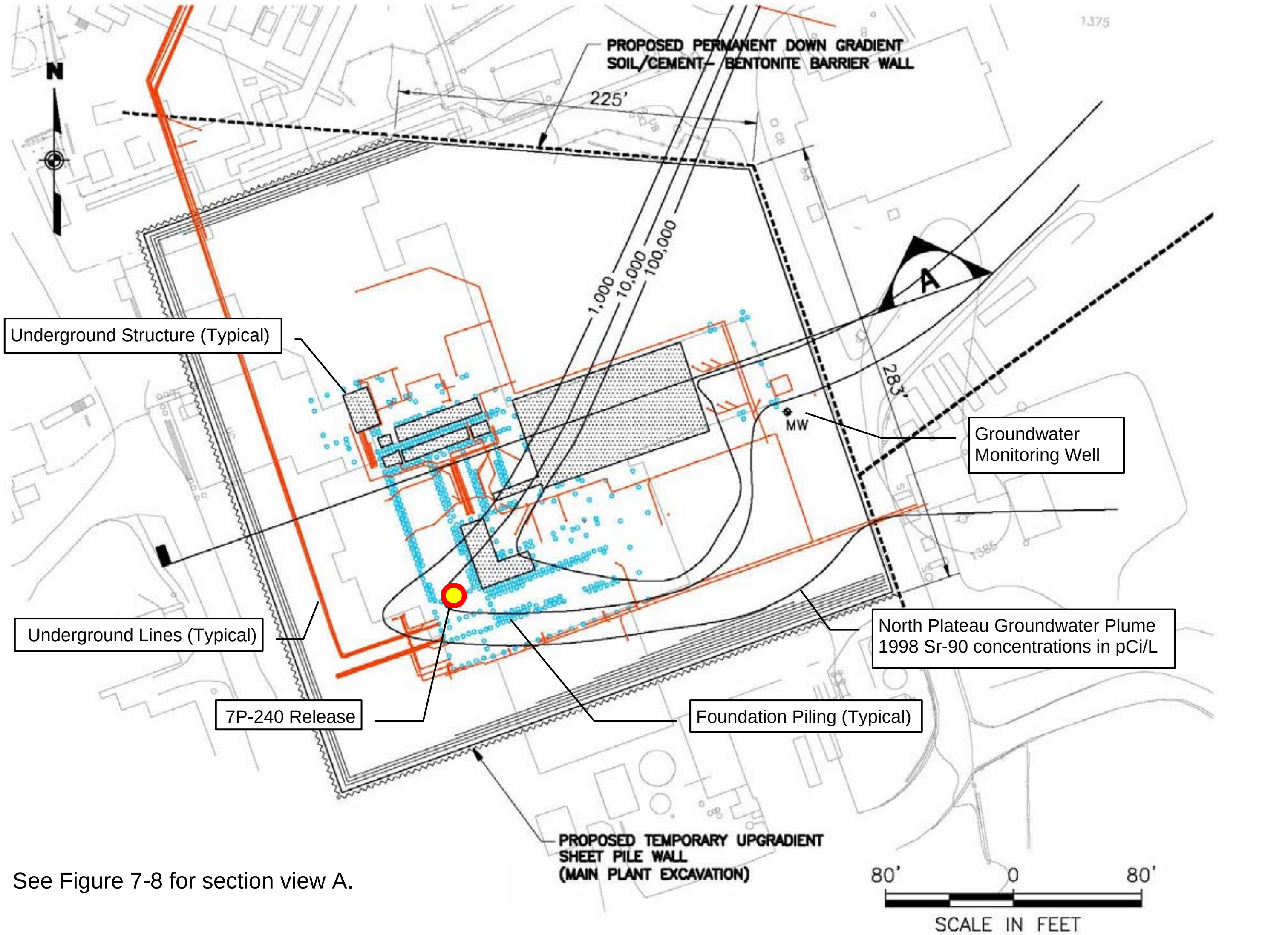



**Legend:**

Black – below grade
Gray – 100-foot level
ARPR – Acid Recovery Pump Room
CAA – Cell Access Aisle
EDR – Equipment Decontamination Room
LWA – Lower Warm Aisle
OGBR – Off-Gas Blower Room
OGC – Off-Gas Cell
PPC – Product Purification Cell
PPH – Product Packaging and Handling Area
UPC – Uranium Product Cell
UR – Utility Room
XC-1 – Extraction Cell 1
XC-2 – Extraction Cell 2
XC-3 – Extraction Cell 3

**Depth Below Surface** (ground level 100 feet)
Cask Unloading Pool floor – 45 feet
East Stairwell – 5 feet
EDR Soaking Pit floor – 16.3 feet
Fuel Storage Pool floor – 33 feet
General Purpose Cell floor – 26.5 feet
GPC Crane Room floor – 26.5 feet
GPC Crane Room Extension floor – 14.3 feet
GPC Operating Aisle floor – 23.5 feet
Head-End Ventilation Cell floor – 10 feet
Liquid Waste Cell floor – 8 feet
Miniature Cell floor – 23.5 feet
North Stairwell , lowest landing feet – 24.5 feet
Tank 7D-13 bottom – 14 feet
Tank 35104 vault floor – 13.3 feet
Tank 15D-6 bottom – 15.5 feet
Uranium Load-Out Area – 5 feet (level with outside grade)
Water Treatment Area floor – 48 feet

**Figure 7-5. Layout of Process Building Underground Structures**




See Figure 7-8 for section view A.

**Figure 7-6. Conceptual Layout of WMA 1 Excavation**

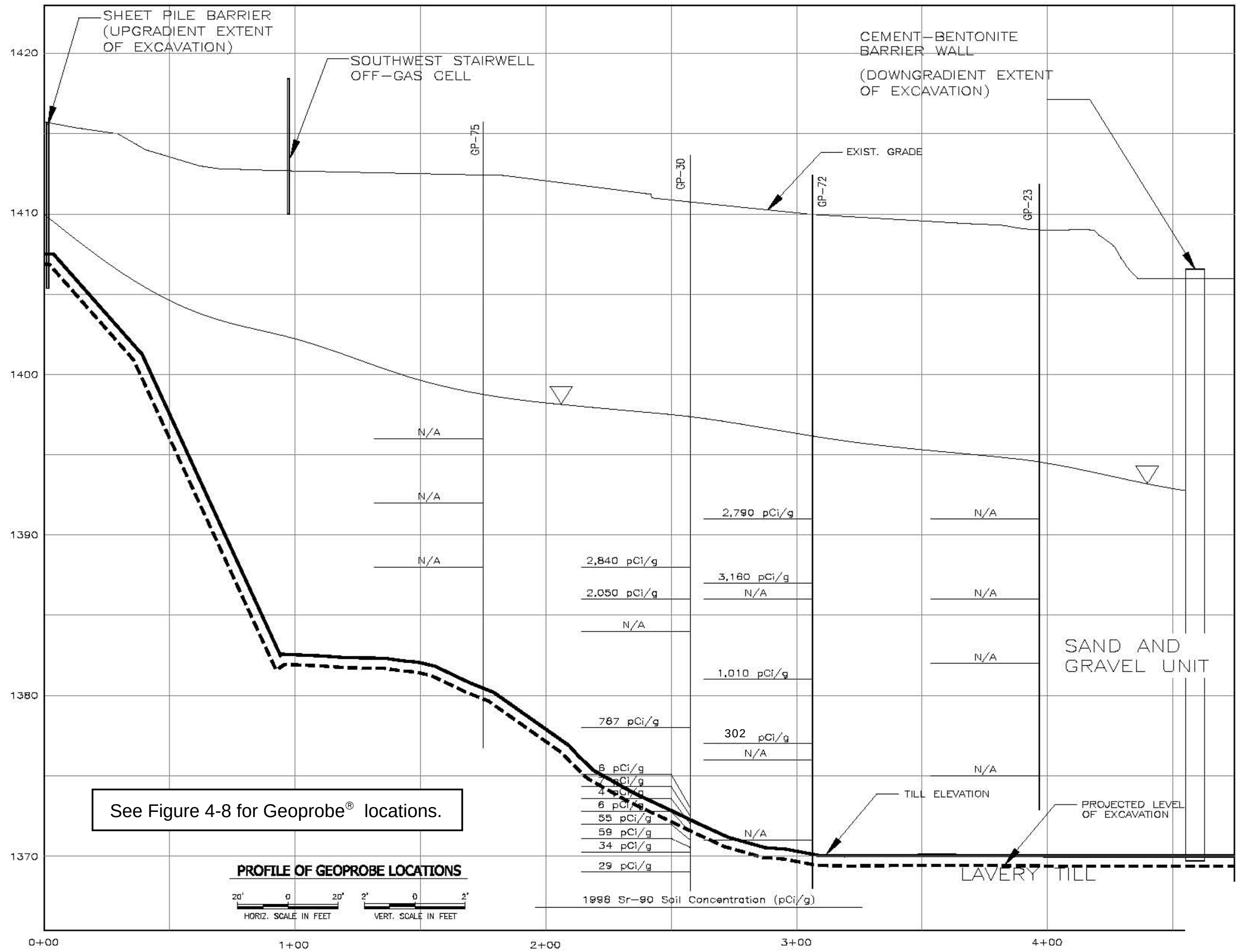



**Figure 7-7. Conceptual WMA 1 Excavation Contour, With Selected Subsurface Soil Data**

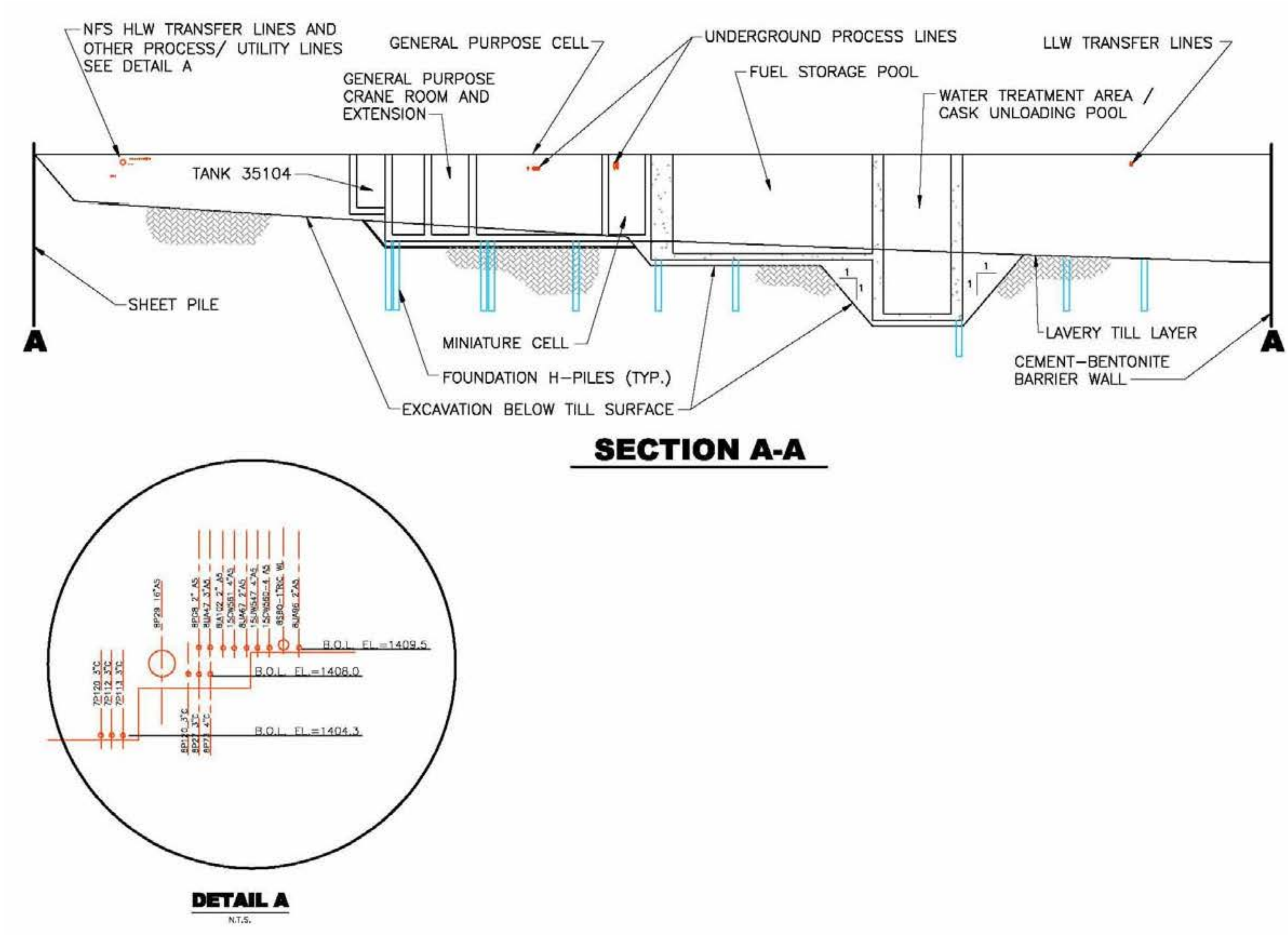



**Figure 7-8. Excavation Cross Section**

**Excavation Conceptual Design**

The horizontal limits of the excavation will be based primarily on physical considerations, although consideration will also be given to analytical data on subsurface soil contamination at the planned excavation boundary acquired during the characterization program. As can be seen in Figure 7-6, the western edge of the excavation will lie near the road in front of the Plant Office Building. The northern edge of the excavation will follow the walkway between the Vitrification Facility and the Waste Tank Farm. The eastern edge will follow the road between the Process Building area and the interceptors. The southern edge will 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 will comprise approximately three acres.

The depth of the excavation will vary depending on the subsurface structures. Figure 7-8 shows a representative cross section (which is identified on Figure 7-6).

**Hydraulic Barrier Wall Installation**

To control groundwater, a vertical hydraulic barrier will be installed around the area to be excavated as shown in Figure 7-6 and Figure 7-7. The upgradient portion will be built of sheet pile. The downgradient portion will consist of a soil-cement-bentonite slurry wall. Both will extend at least two feet into the Lavery till and the slurry wall will remain in place after the excavation is backfilled.

Before the hydraulic barrier wall is installed, underground lines within its footprint that carried radioactive liquid will be located. Sections of these lines in the area where the barrier walls will be constructed will be removed in a controlled manner to avoid unnecessary release of contamination. During this process, characterization measurements will be taken in the end of each line that will remain in place outside of the excavation and the line capped.

The total length of the slurry wall will 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 will 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 will provide the stability necessary to accommodate the nearby excavation.[7]

The sheet pile section of the hydraulic barrier wall will be installed using a conventional pile driver. Construction of the soil-cement-bentonite slurry wall will 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 will be constructed near the slurry wall;

[7] 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 will perform as planned in controlling groundwater intrusion and supporting the excavation design. The extra thickness will accommodate some excavation into the upper portion of the barrier wall with sufficient thickness remaining to ensure satisfactory performance.

- Keeping the trench filled with slurry during the excavation process to help support the trench walls during the excavation;
- Backfilling the trench with a mixture of clean soil, cement, and bentonite to displace the slurry, which will 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 at an appropriate offsite disposal facility; and
- Disposing of the uncontaminated soil at an appropriate offsite disposal facility.

The resulting slurry wall will have a maximum in-place saturated hydraulic conductivity of 6.0E-06 cm/s. It will 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 will be coordinated with soil removal since the north plateau plume source area lies beneath the Process Building. Detailed planning for the excavation will 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 – will also be taken into account.

Preparations, in addition to installation of the hydraulic barrier wall, will include installation of extraction wells to dewater the excavation. The removed water will 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 will 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 will be installed, the sheet piles installed, the dewatering wells installed and placed in operation, and appropriate radiological controls established. The excavation process will be accomplished in two phases using conventional excavation equipment. It is expected that approximately 75 percent of the soil to be excavated will be saturated. All soil removed from the excavation will be disposed of at appropriate offsite disposal facilities.

The first phase will involve removal of soil in the vadose zone. Characterization data will be used to determine that portion of the soil that is likely to be uncontaminated. These data, supplemented by in-process survey data collected as described in Section 9, will be used to segregate excavated soil that is unlikely to be contaminated from excavated soil that is determined to be contaminated. Guidance provided in the Waste Management Plan will be followed during these activities.

Excavation of soil in the saturated zone will begin after the dewatering wells have removed groundwater in the confined area to the extent practical. The groundwater will 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 will be removed during the excavation. Excavated soil in the saturated zone will be segregated like excavated soil removed from the vadose zone to the extent practicable in accordance with the Waste Management Plan.

Groundwater accumulating in the excavation will be pumped out, treated as necessary at the Low-Level Waste Treatment Facility or using a portable treatment system containing ion exchangers and filters, and discharged to Erdman Brook through an SPDES-permitted outfall.

Soil will 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 will 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 will be removed and disposed of offsite as radioactive waste.[8] Contaminated soil with radioactivity concentrations below cleanup goals will be removed where practical, consistent with the ALARA process as described in Section 6 and Section 7.2.2. Soil will be excavated as close to the hydraulic barrier wall as practical. The other sides of the excavation will have a slope of approximately 45 degrees.

**Removal of Underground Structures, Floor Slabs, and Foundations**

The demolition of below-grade cells and structures shown in Figure 7-5 will 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, will be removed early in the process to facilitate the excavation work. After soil is excavated to expose their structures, the below-grade cells will be demolished with conventional demolition equipment such as diamond wire saws.

The foundation pilings supporting the Process Building will 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 will be characterized and disposed of offsite. In connection with this work, samples of soil will be collected around representative pilings, including at points several feet below the surface in accordance with the Characterization Sample and Analysis Plan and the Phase 1 Final Status Survey Plan. Analytical data from the samples will 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 and Appendix G.

If sampling were to identify elevated activity exceeding cleanup goals in the till material adjacent to the pilings, actions will be taken to ensure that these exceedences are addressed (e.g., continued soil excavation around the pilings).

[8] 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 will be limited by the excavation boundaries. That is, any soil found to exceed the cleanup goals will be removed only within the confines of the downgradient hydraulic barrier wall and the sheet piles installed on the other sides of the excavation.

**Removal of Underground Tanks and Piping**

The three underground tanks and all radioactively contaminated underground piping within the excavated area will be removed and disposed of as radioactive waste. Planning for underground line removal will 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 will 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, will be cut near the joints, with pieces containing the joints being disposed of as mixed waste. The remainder of this piping will be disposed of as LLW.

This process will apply to radioactive lines and also to nonradioactive sanitary lines and utility lines, which will be removed during the course of the work because it is unlikely that it will be practicable to leave them in place. Underground piping outside of the excavation will 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 will 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 will be paid to areas around the remaining sections of the Process Building support pilings. Surveys performed around the support pilings will extend to sufficient depth to evaluate the extent, if any, of the downward migration of contamination along the pilings. Arrangements will also be made for an independent confirmatory survey to be performed on behalf of the regulatory agencies.

After the confirmatory survey is completed and regulatory concurrence is received, the area will be backfilled with uncontaminated earth and graded as necessary to restore to it a near natural appearance. The backfill material will be obtained from similar offsite geologic deposits. The properties of this material (especially the texture, hydraulic conductivity, and distribution coefficients) will be similar to those of the sand and gravel layer on the project premises as described in Section 3.

A French drain will be emplaced during backfilling of the excavation to prevent groundwater from mounding near the hydraulic barrier wall. Water from the French drain will 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 will be removed after the excavation is backfilled. The piling and any confinement structure used will be disposed of offsite at appropriate waste disposal facilities. Figure 7-9 provides a conceptual cross-section view of the backfilled excavation. The cross section in this figure is similar to Section 7-8, that is, Section A-A on Figure 7-6,

Appendix D addresses monitoring and maintenance of the WMA 1 area after completion of Phase 1 of the decommissioning. Appendix D also provides information on expected changes to the groundwater flow field that will occur with completion of the Phase 1 decommissioning activities in WMA 1.

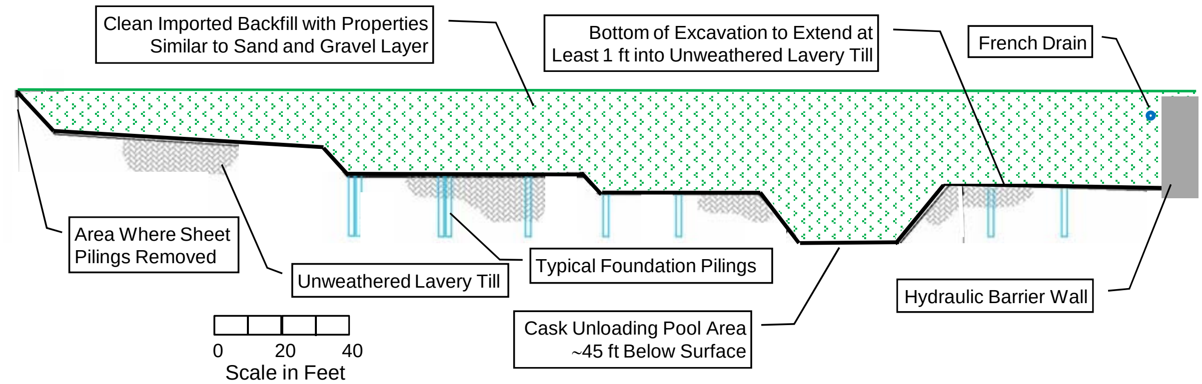



**Figure 7-9. Conceptual Cross-Section View of the Backfilled WMA 1 Excavation**

## 7.4 WMA 2 Decommissioning Activities

This section addresses decommissioning of the Low-Level Waste Treatment Facility area, WMA 2, which is shown in Figure 7-10.

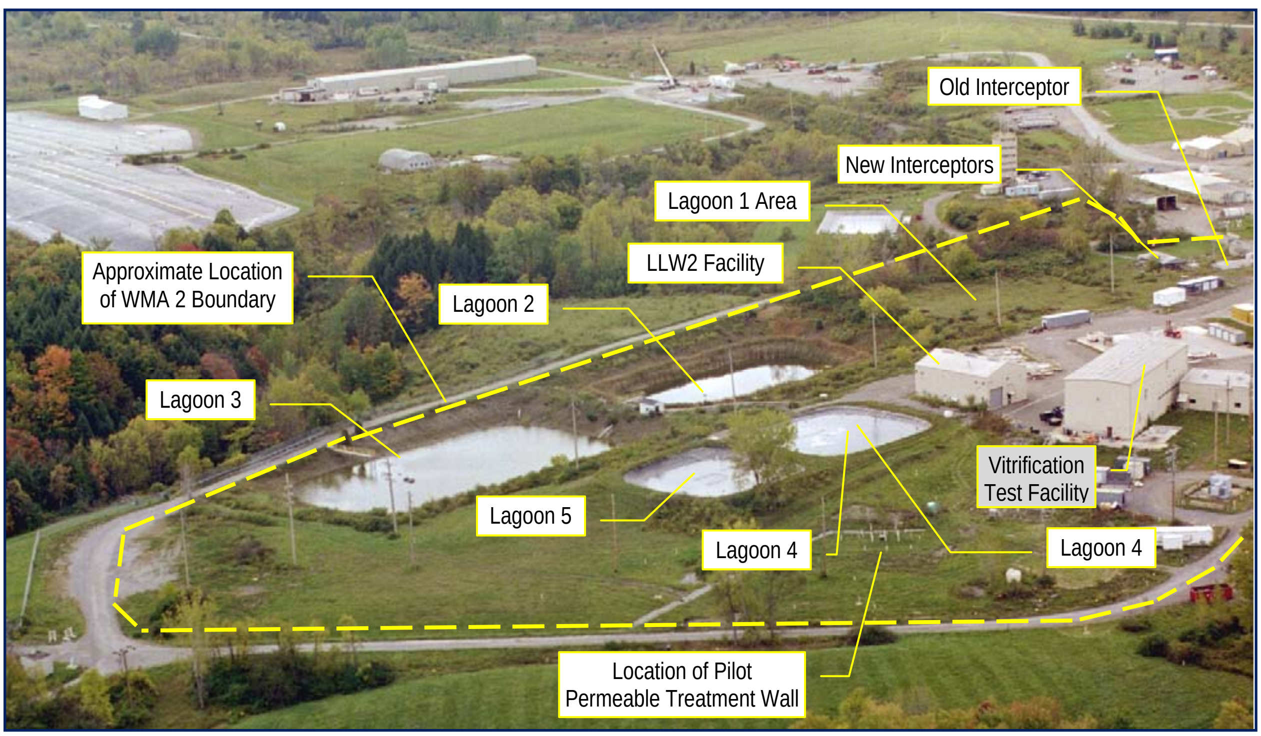



**Figure 7-10. WMA 2 in 2007**

The sequence for the Phase 1 decommissioning activities in WMA 2 will be developed during detailed planning. The LLW2 facility will 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 will be characterized for waste management purposes and disposed of at appropriate offsite waste disposal facilities. Absorbents will 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 will be characterized for residual radioactivity in accordance with the Characterization Sample and Analysis Plan described in Section 9. The results of this effort will be used in planning the excavation work described below. (This characterization will not include subsurface soil in areas impacted by the north plateau groundwater plume except in the portion of WMA 2 where the excavation will be located.)

### 7.4.2 Removing Structures

The structures in WMA 2 will 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 will be coordinated with digging the WMA 2 excavation addressed in Section 7.4.3, which will encompass the area of these facilities as well as the Solvent Dike. During this process, characterization measurements will be taken in the end of each underground line that will 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 will 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[9];
- Performing radiological 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.

[9] 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 will be among the sources considered in Phase 2 of the decommissioning.

**Neutralization Pit**

The Neutralization Pit will 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 will 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 will 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 will 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 will be removed and the building footprints excavated approximately two feet below grade. Radiological status surveys will be performed in the excavated areas, and arrangements made for an independent confirmatory survey if desired by the regulators. After the surveys have been completed, the excavations will be filled with clean earthen backfill.

### 7.4.3 Decommissioning the Lagoons

Decommissioning of Lagoons 1, 2, and 3 will involve constructing a vertical hydraulic barrier on the northwest and northeast sides of the lagoons and digging a single large excavation. Lagoons 4 and 5 will be removed separately. Figure 7-11 shows the conceptual plan view of the large excavation and the location of the hydraulic barrier wall. Figure 7-12 shows the conceptual cross section.

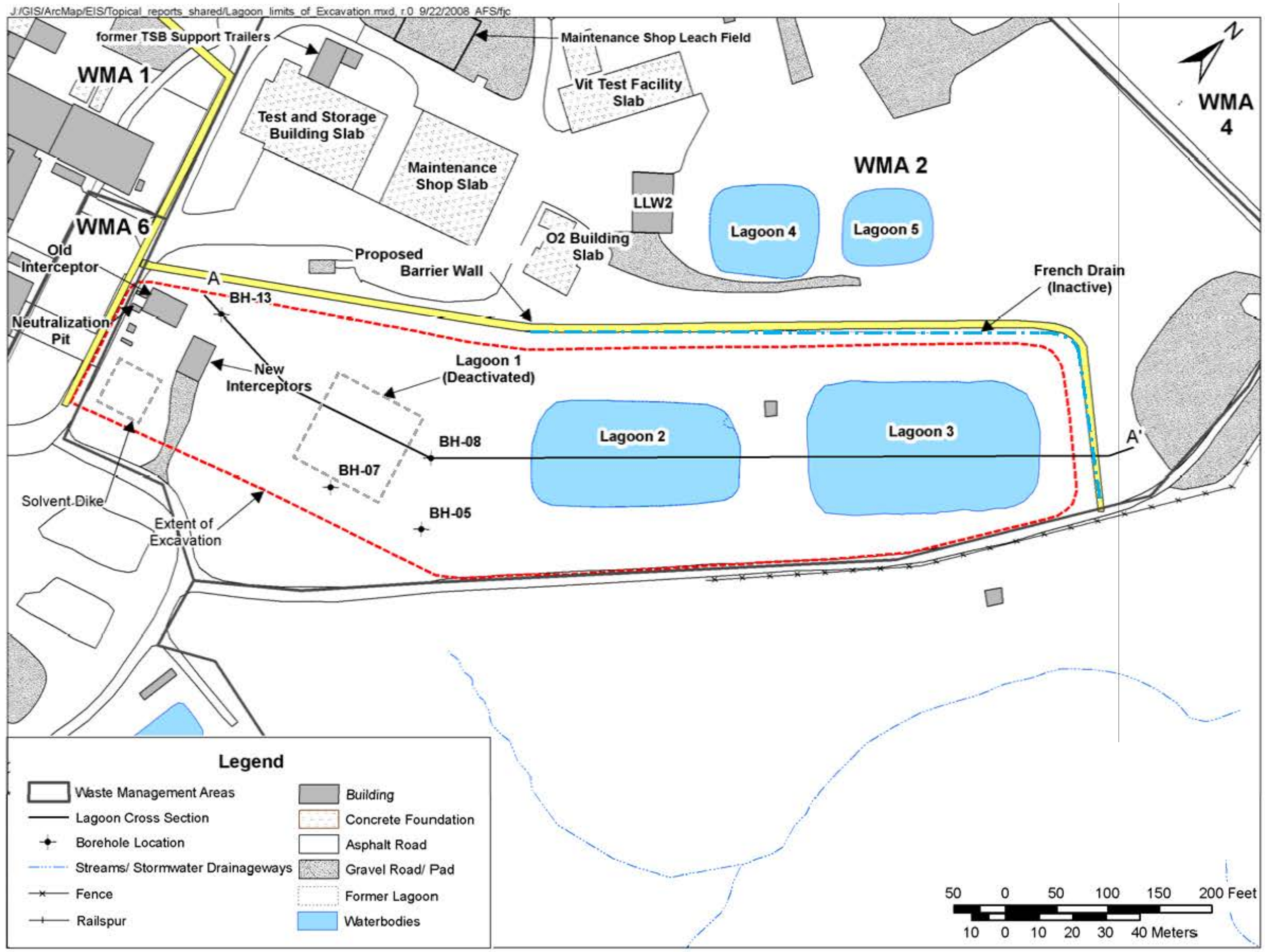



**Figure 7-11. Conceptual Arrangement of WMA 2 Excavation, Plan View**

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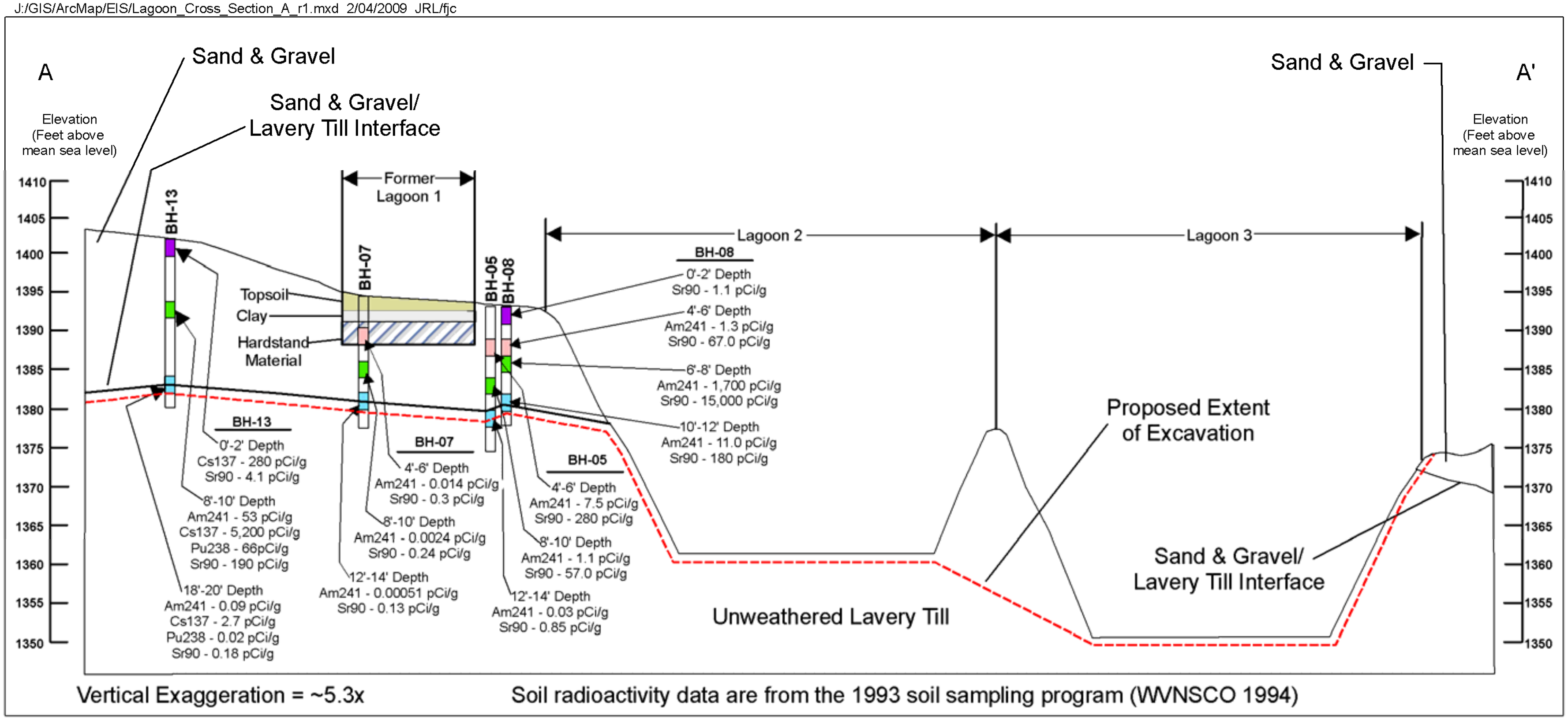


**Figure 7-12. Conceptual Arrangement of WMA 2 Excavation, Cross Section**

**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 will be installed as shown in Figure 7-11. This hydraulic barrier will consist of a soil-cement-bentonite barrier wall that will extend approximately two feet into the Lavery till. It will remain in place after the excavation is backfilled.

Before the hydraulic barrier wall is installed, underground lines in its footprint that carried radioactive liquid will be located. Sections of these lines in the area where the wall will be constructed will be removed in a controlled manner to avoid unnecessary release of contamination. During this process, characterization measurements will be taken in the end of each line that will remain in place and the line capped.

The total length of the barrier wall will be approximate 1100 feet. It will be sufficiently wide to provide the stability necessary to permit excavation up to the base of the wall. This barrier wall will connect with the WMA 1 hydraulic barrier wall as shown in Figure 7-11. It will 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 will extend to within about three feet of grade and be topped with clean earthen backfill. 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 will 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 and sediment characterization program. The depth of the water table in the area – typically about seven feet below the surface – will also be taken into account.

Preparations, in addition to installation of the hydraulic barrier wall, will 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 of the large excavation as described below will be coordinated with removal of soil in other parts of the excavation. Before excavation begins, the hydraulic barrier wall will be installed. Extraction wells to dewater the excavation will be installed as with the WMA 1 excavation. Water removed by the wells will be treated as necessary – such as by use of a portable wastewater treatment system with ion exchangers and filters – and discharged to Erdman Brook through a SPDES-permitted outfall.

The excavation process will be accomplished in two phases using conventional excavation equipment. The first phase will involve removal of soil in the vadose zone. It is expected that approximately one-half of the total amount of soil to be removed will be unsaturated. As with the WMA 1 excavation, characterization data will be used to determine the portion of the excavated soil that it likely to be uncontaminated and these data, supplemented by in-process survey data collected as specified in Section 9, will be used to segregate the excavated soil that is unlikely to be contaminated from that which has been determined to be contaminated in accordance with the

Waste Management Plan. All soil removed from the excavation will be disposed of at appropriate offsite disposal facilities.

The second phase will involve removal of soil in the saturated zone. Wastewater piping within the excavated area will be removed. Groundwater accumulating in the excavation will be pumped out, treated as necessary using a portable treatment system containing ion exchangers and filters, and discharged to Erdman Brook through an SPDES-permitted outfall.

Figure 7-12 shows the planned depth of excavation. The excavation will 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.[10] Remedial action surveys will 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 will be removed. Soil with radioactivity concentrations exceeding cleanup goals will be excavated up to and into the hydraulic barrier as practicable. However, the lateral extent of the remediation will not exceed the boundary shown in Figure 7-11 during Phase 1. All removed equipment and excavated soil will be disposed of offsite at appropriate disposal facilities.

**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 will be installed around Lagoon 1 as necessary to control groundwater flow into the area to be excavated. The excavation will 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 will be excavated, along with the underlying sediment. The excavation will 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 will be performed in the excavated area and arrangements will be made for independent confirmatory surveys before the excavation is filled in, as described below. (These surveys will be performed when the entire WMA 2 excavation has been completed.)

Any sheet piles installed to facilitate excavation of Lagoon 1 will be removed after the lagoon is excavated and disposed of offsite at appropriate disposal facilities.

**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 will 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 will be removed. Contaminated lagoon sediment will be removed along with at least one

[10] Note that Figure 7-12 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.

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 will be performed in the excavated area and arrangements will 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 will be decommissioned using the same process as Lagoon 2.

**Solvent Dike**

Radioactively contaminated soil in the Solvent Dike area will be removed before the large excavation is dug. This sequence will facilitate management of any unexpected wastes that might be present.

**Other Parts of the Excavation**

Removal of soil in between the facilities in the area to be excavated will be coordinated with excavation of the facilities themselves so that the entire area is excavated as indicated in Figures 7-11 and 7-12, with the excavation extending at least one foot into the Lavery till. Any sheet piles installed to facilitate excavation of Lagoon 1 will be removed after that lagoon is excavated.

**Surveying and Backfilling the Excavation**

Phase 1 final status surveys will 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 will be filled with clean earthen backfill and the surface leveled with the surrounding area. The backfill material will be obtained from similar offsite geologic deposits. The properties of this material will 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 residual water, the geomembranes, and the concrete and clay liners in Lagoons 4 and 5 will be removed and underlying soil excavated to a maximum depth of two feet. Water pumped out of the lagoons will be treated and discharged through an SPDES-permitted outfall. The geomembranes, liners, and excavated soil will be disposed of at appropriate offsite disposal facilities. After completion of this work, a radiological status survey will be performed in the area, and arrangements made for any independent confirmatory surveys required by the regulators. The excavated area will be filled with clean earth after the surveys.

**Final Condition of the Backfilled Excavation**

Figure 7-13 shows a conceptual cross-section view of the backfilled excavation. The location of this cross section is similar to the A-A section shown on Figure 7-12, except that it passes through the area of the interceptors on the southwest side of the excavation.

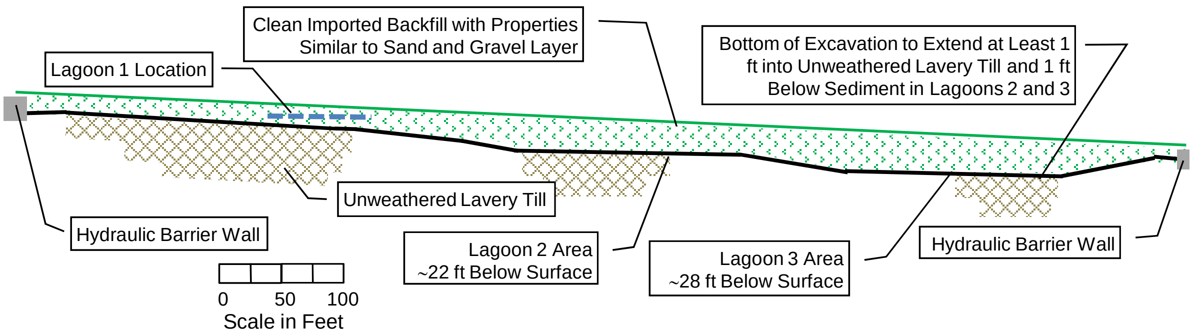


**Figure 7-13. Conceptual Cross-Section View of the Backfilled WMA 2 Excavation**

Appendix D addresses monitoring and maintenance of the WMA 2 area after completion of Phase 1 of the decommissioning. Appendix D also provides information on expected changes to the groundwater flow field that will occur with completion of the Phase 1 decommissioning activities in WMA 2.

## 7.5 WMA 3 Decommissioning Activities

This section addresses 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 pumps, transfer pumps, and submersible 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 will be removed with appropriate radiological controls and the resulting demolition debris characterized and disposed of at an appropriate offsite disposal facility. These facilities are expected to have low levels of residual radioactivity, mostly inside installed equipment.

**Con-Ed Building**

This small concrete block building located over the Tank 8D-3/8D-4 vault will be removed by removing the installed equipment, demolishing the structure to grade level, and performing radiological 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 – will be removed using a process similar to that used for the Con-Ed Building. The condensers will also be removed and disposed of at an offsite waste disposal facility. Soil in the footprints of the building and condenser foundations will be removed to a maximum depth of two feet below grade. Radiological status surveys will be performed in the excavated areas and arrangements made for any independent confirmatory surveys required by the regulators. Afterwards, the excavated areas will 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 will 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 will be removed and disposed of offsite using a process such as the following:

- Preparations will 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 will be removed using appropriate radiological controls, decontaminated as necessary, cut into sections during removal, and packaged for disposal;
- The pump support structures will be removed in conjunction with removal of the pumps; and
- The pump segments and the support structures will be disposed of offsite at appropriate waste disposal facilities.

The submersible pumps in Tanks 8D-3 and 8D-4 will 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 will 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 and may contain significant amounts of residual radioactivity. DOE will follow the applicable provisions of DOE Manual 435.1-1, *Radioactive Waste Management Manual* concerning the piping and the other equipment.

The piping and other equipment will be removed using a process such as the following:

- Preparations will 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 will be cut into sections and packaged for disposal;
- The other equipment will 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 will be disposed of offsite at an appropriate waste disposal facility.

After the piping has been removed, radiological status surveys will 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 will continue during Phase 1 of the decommissioning and until such time that Phase 2 of the decommissioning begins. The tank and vault drying system installed during the work to establish the interim end state described in Section 3 will remain in operation.

The existing dewatering well will 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 will be collected, sampled, treated if necessary using a portable wastewater treatment system, and released through a SPDES-permitted outfall.

Appendix D provides additional information on these matters.

## 7.6 WMA 5 Decommissioning Activities

This section addresses removal of Lag Storage Addition 4 and the associated Shipping Depot, 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. These structures are not expected to have any contamination above the minimum detectable concentrations, except for the Remote-Handled Waste Facility, which may have low-level contamination is some areas and possibly higher levels of contamination in the Work Cell.

### 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 will 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 radiological status surveys in the area excavated;
- Evaluating the resulting data and determining whether to perform Phase 1 final status surveys to establish that these areas meet the surface soil cleanup criteria;
- If Phase 1 final status surveys are to be performed, notifying NRC to this effect;
- Performing the Phase 1 status surveys, if they are to be accomplished;
- Arranging for any confirmatory surveys required by the regulators; 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 will be coordinated with the demolition under this plan.

The Remote-Handled Waste Facility will 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 radiological status surveys in the area excavated;
- Evaluating the resulting data and determining whether to perform Phase 1 final status surveys to establish that these areas meet the surface soil cleanup criteria;
- If Phase 1 final status surveys are to be performed, notifying NRC to this effect;
- Performing the Phase 1 status surveys, if they are to be accomplished;
- Arranging for any confirmatory surveys required by the regulators; 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 will be removed and disposed of as LLW if they have been exposed to radioactivity; otherwise, they will remain in place.

#### 7.6.3 Removing Remaining Floor Slabs and Foundations and Gravel Pads

All remaining concrete floor slabs and foundations will 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 will 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 will 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 will be followed by radiological status surveys of the excavated areas. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After all of the surveys have been completed, the excavations will be filled with clean earthen backfill.

### 7.6.4 Establishing that Surface Soil Meets Cleanup Goals

Characterization data on surface soil and subsurface soil collected within WMA 5 will be evaluated. Based on this evaluation, parts of WMA 5 may be selected as appropriate for any necessary remediation and for Phase 1 final status surveys. These activities will be performed as specified in Section 7.11.

## 7.7 WMA 6 Decommissioning Activities

This section addresses 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 will 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 will be completely removed, including the underground concrete tanks, with the concrete foundation and underlying soil removed approximately two feet below grade. It is not expected to be radioactively contaminated.

After completion of this work, a radiological status survey will be performed in the excavated area. The resulting data will be evaluated, taking into account experience with buildup of natural and manmade radioactivity in sewage sludge (ISCORS 2005), and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable, and arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the excavated area will be filled with clean earthen backfill.

### 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. It is not expected to be radioactively contaminated.

The liner and approximately two feet of underlying soil will be removed and disposed of offsite. After completion of this work, a radiological status survey will be performed in the area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable and arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the area will be filled with clean earthen backfill.

### 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. It is not expected to be radioactively contaminated.

The tank will be demolished and approximately two feet of underlying soil removed, with this material disposed of offsite. After completion of this work, a radiological status survey will be performed in the area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the area will be filled with clean earthen backfill.

### 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 will 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 radiological status survey will be performed in the area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the area will be filled with clean earthen backfill.

### 7.7.5 Removing the South Waste Tank Farm Test Tower

This test tower is not expected to be radioactively contaminated. It will 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 radiological status survey will be performed in the area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the area will be filled with clean earthen backfill.

### 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– will be removed, with underlying soil removed to a maximum depth of two feet below grade. After completion of this work, a radiological status survey will be performed in the area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the area will be filled with clean earthen backfill.

#### 7.7.7 Establishing that Surface Soil Meets Cleanup Goals

Characterization data on surface soil and subsurface soil collected within WMA 6 will be evaluated. Based on this evaluation, parts of WMA 6 may be selected as appropriate for any necessary remediation and for Phase 1 final status surveys. These activities will be performed as specified in Section 7.11.

### 7.8 WMA 7 Decommissioning Activities

WMA 7, the NDA area, is shown in Figure 3-41. The NDA will continue to be monitored and maintained during Phase 1 and no decommissioning actions related to the NDA itself will take place in this phase of the decommissioning. The only Phase 1 decommissioning actions will involve removal of the remaining concrete slabs and gravel pads associated with the NDA hardstand.

These concrete slabs and gravel pads will be removed and the footprints of these areas will be excavated to a maximum of depth two feet below grade, with the debris and excavated materials disposed of at appropriate offsite disposal facilities. Radiological status surveys will be performed in the excavated areas and arrangements made for any independent confirmatory surveys required by the regulators. After completion of the surveys, these areas will be filled with clean earthen backfill.

### 7.9 WMA 9 Decommissioning Activities

This section describes decommissioning activities in the Integrated Radwaste Treatment System Drum Cell area, which is shown in Figure 3-42. Phase 1 decommissioning activities in this area will involve removal of the Drum Cell, the trench soil container area, and the subcontractor maintenance area. The Drum Cell is not expected to have any contamination above minimum detectable concentrations, nor are the other areas.

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 will 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 radiological status survey will be performed in the excavated area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the excavated area will be filled with clean earthen backfill.

The trench soil container area is located northwest of the Drum Cell. The material in this area will be removed and its footprint excavated to a maximum depth of approximately two feet below grade, with the excavated materials disposed of offsite. Radiological status surveys will be performed in the excavated area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the area will be filled with clean earth.

The subcontractor maintenance area, a gravel pad near the rail spur, will be removed using the process used for the trench soil container area.

Characterization data on surface soil, near surface soil, and subsurface soil collected within WMA 9 will also be evaluated. Based on this evaluation, parts of WMA 9 may be selected as appropriate for any necessary remediation and for Phase 1 final status surveys. These activities will be performed as specified in Section 7.11.

### 7.10 WMA 10 Decommissioning Activities

The Phase 1 decommissioning activities in this WMA, the support and services area, will 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 will 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 is not expected to be radiologically contaminated.

The New Warehouse will 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 radiological status survey will be performed in the excavated area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the excavated area will be filled with clean earthen backfill.

The remaining floor slabs and foundations in the area – including those for the Administration Building, the Expanded Environmental Laboratory, and the Fabrication Shop – will also be removed, with underlying soil removed to a maximum depth of approximately two feet below grade. The former Waste Management Storage Area will also be removed in the same manner. After completion of this work, a radiological status survey will be performed in each excavated area. The resulting data will be evaluated and if it is determined to be appropriate, NRC will be notified that Phase 1 final status surveys will be performed in the area. These surveys will be performed as applicable. Arrangements will be made for any independent confirmatory surveys required by the regulators. After completion of all of the surveys, the excavated areas will be filled with earthen backfill.

Characterization data on surface soil, near surface soil, and subsurface soil collected within WMA 10 will also be evaluated. Based on this evaluation, parts of WMA 10 may be selected as appropriate for any necessary remediation and for Phase 1 final status surveys. These activities will be performed as specified in Section 7.11.

The Meteorological Tower and the Security Gatehouse and fences will remain in place.

## 7.11 Establishing Areas Where Surface Soil Meets Cleanup Goals

As discussed in the previous subsections, DOE may elect to establish that surface soil in selected areas of the project premises meets the surface soil cleanup goals. The areas of interest will be identified based on the results of the characterization program.

### 7.11.1 Areas of Interest

These areas will have no subsurface radioactive contamination deeper than one meter (3.3 feet) from the surface based on process knowledge and characterization results and would be required to meet the surface soil cleanup goals. They may include areas where foundations and floor slabs are removed, areas where characterization shows residual radioactivity concentrations below the surface soil cleanup goals, and other areas where radioactive contamination exceeding the cleanup goals could be removed with relatively minor effort. They may also include areas to be used for temporary storage of excavated contaminated soil. Given these factors, not all of the areas of interest can be identified early in Phase 1 and in some cases any required remediation and the required Phase 1 final status surveys will necessarily take place late in Phase 1.

### 7.11.2 Process to be Followed

The process would include steps such as the following if this effort is undertaken:

- The initial characterization program would be completed and the resulting data reviewed,
- The excavated soil laydown areas would be identified,
- The primary areas of interest would be selected and NRC notified,
- Each area of interest would be remediated as necessary to meet the surface soil cleanup goals,
- Phase 1 final status surveys would l be performed,
- The applicable portions of the Phase 1 Final Status Survey Report would be prepared, and
- Arrangements will be made for any independent confirmatory surveys required by the regulators.

A similar process would be followed when building foundations and floor slabs are removed. The radiological status survey data would be evaluated and the decision made as to whether to establish that the area – in this case a shallow excavation about two feet deep – meets the surface soil cleanup goals. If the decision were to be made to do this, NRC would be notified, the Phase 1 final status surveys performed, the applicable portions of the Phase 1 Final Status Survey Report prepared, and arrangements made for any confirmatory surveys to be performed. The shallow excavation would be backfilled with clean earthen backfill only after all the surveys have been completed and any related issues resolved.

The Phase 1 Final Status Survey Report would document the final status surveys performed in these areas. In addition, one or more maps would be prepared to document the precise locations of all such areas and copies of these maps provided to NRC, NYSERDA, and NYSDEC. The areas would also be identified in an appropriate manner, such as by the use of fences and signs.

#### 7.11.3 Additional Area of Interest

As shown in Figure 3-8, a small portion of WMA 12 lies within the project premises security fence. Any characterization data on surface soil and subsurface soil collected within this portion of WMA 12 will be evaluated. Based on this evaluation, parts of the portion of WMA 12 within the project premises may be selected for any necessary remediation and for Phase 1 final status surveys. These activities would be performed as specified above.

### 7.12 Remedial Technologies

A combination of conventional technologies and proven innovative technologies will be used to accomplish the 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 will be provided in the Decommissioning Work Plan(s).

#### 7.12.1 Pipe Cutting and Other Metal Cutting

The following methods will be used as applicable for cutting radioactively contaminated piping and metal liners, equipment, and structural components. Methods will 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.

**Liquid Nitrogen Cutting**

A liquid nitrogen cutting and cleaning system such as that offered by Nitrocision® can be used to cut metal and decontaminate concrete without producing a secondary waste stream. This system can be used either manually or robotically and can be equipped with a vacuum capture system to collect decontamination debris. A Nitrocision® liquid nitrogen cutting and cleaning system is expected to be in operation in support of facility deactivation work at the WVDP in late 2009 or early 2010.

**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.

### 7.12.2 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 feet 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.

### 7.12.3 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 7-14 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 7-14 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.

### 7.12.4 Concrete Decontamination

Contaminated concrete surfaces will be decontaminated using conventional means such as vacuuming and wiping with cloths dampened with water or non-hazardous decontamination agents. The following technologies will also be considered and used as appropriate:

**Concrete Shaver**

Marcrist Industries and Demolition Technologies manufactures 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 7-15). 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 7-15. 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 7-14, can be used effectively in concrete demolition where radiological conditions make use of remote-controlled equipment preferable.

**Decontamination Using Liquid Nitrogen Cutting**

As noted previously, a liquid nitrogen cutting and cleaning system such as that offered by Nitrocision® can be used to decontaminate concrete without producing a secondary waste stream.

### 7.12.5 Demolition of Structures

Structures will 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 will 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 7-16)

### 7.12.6 Excavation and Grading

DOE will use conventional equipment to remove soil, equipment, and portions of concrete structures, such as tracked excavators. Backhoes and bulldozers will be used as needed. Similar equipment will be used for grading the site.

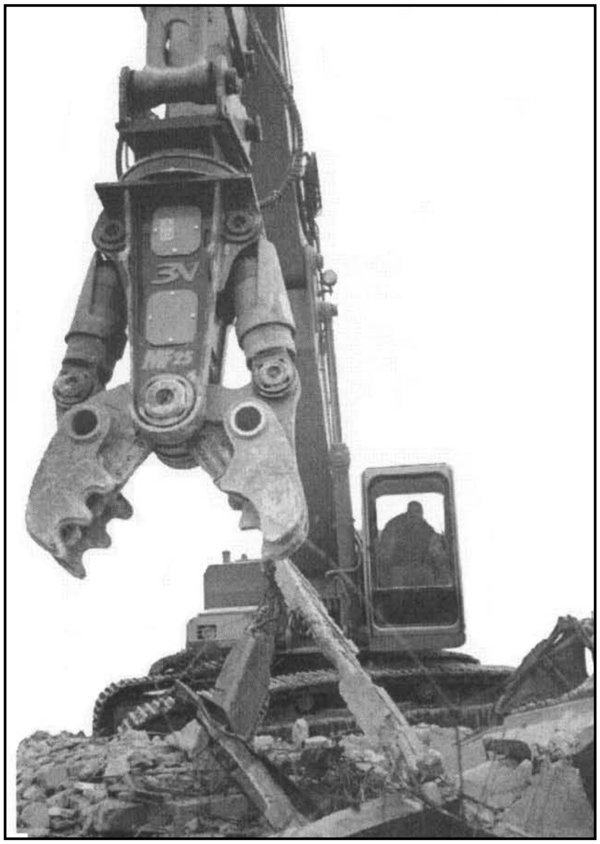

**Figure 7-16. Universal Demolition Processor**

### 7.13 Schedule

Due to the circumstances of the 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 7-17 provides a conceptual schedule for the project, with the basic sequence and order-of-magnitude time frames for major actions.

Work related to removal of the Process Building and the source area of the north plateau groundwater plume is expected to form the project critical path. The decommissioning contractor will develop an optimum sequence after completion of detailed planning. One necessary restraint involves installation of the WMA 1 hydraulic barrier wall before beginning the WMA 2 excavation to reduce infiltration of groundwater into the WMA 2 excavation. The total schedule duration will depend largely on available funding.

The dates on the schedule are contingent upon completion of the NRC review process related to this plan. Before the decommissioning begins, DOE will provide a more detailed schedule to NRC for information. DOE also recognizes that circumstances can change during the decommissioning so that the 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.

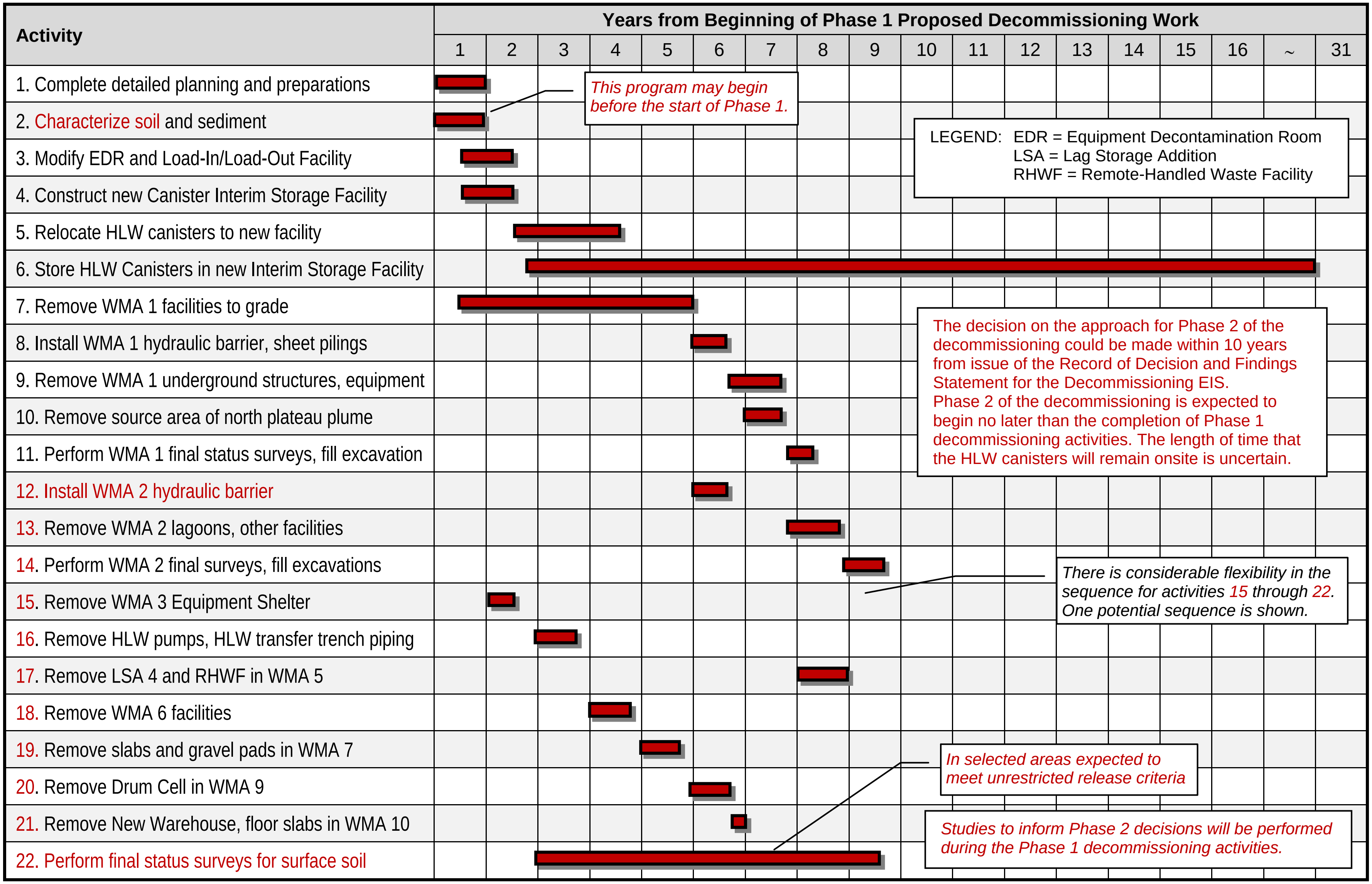


**Figure 7-17. Conceptual Schedule of Phase 1 Decommissioning Activities**

## 7.13 References

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

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# 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 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 will be managed and implemented. It is also intended to show NRC staff how accurate, high-quality information will be provided to support Phase 1 of the decommissioning.

**INFORMATION IN THIS SECTION**

The focus of this section is appropriate because the 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 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 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 will be completed before 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 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 Canister Interim Storage Facility for the vitrified HLW canisters and the hydraulic barrier walls that will 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 decommissioning project.

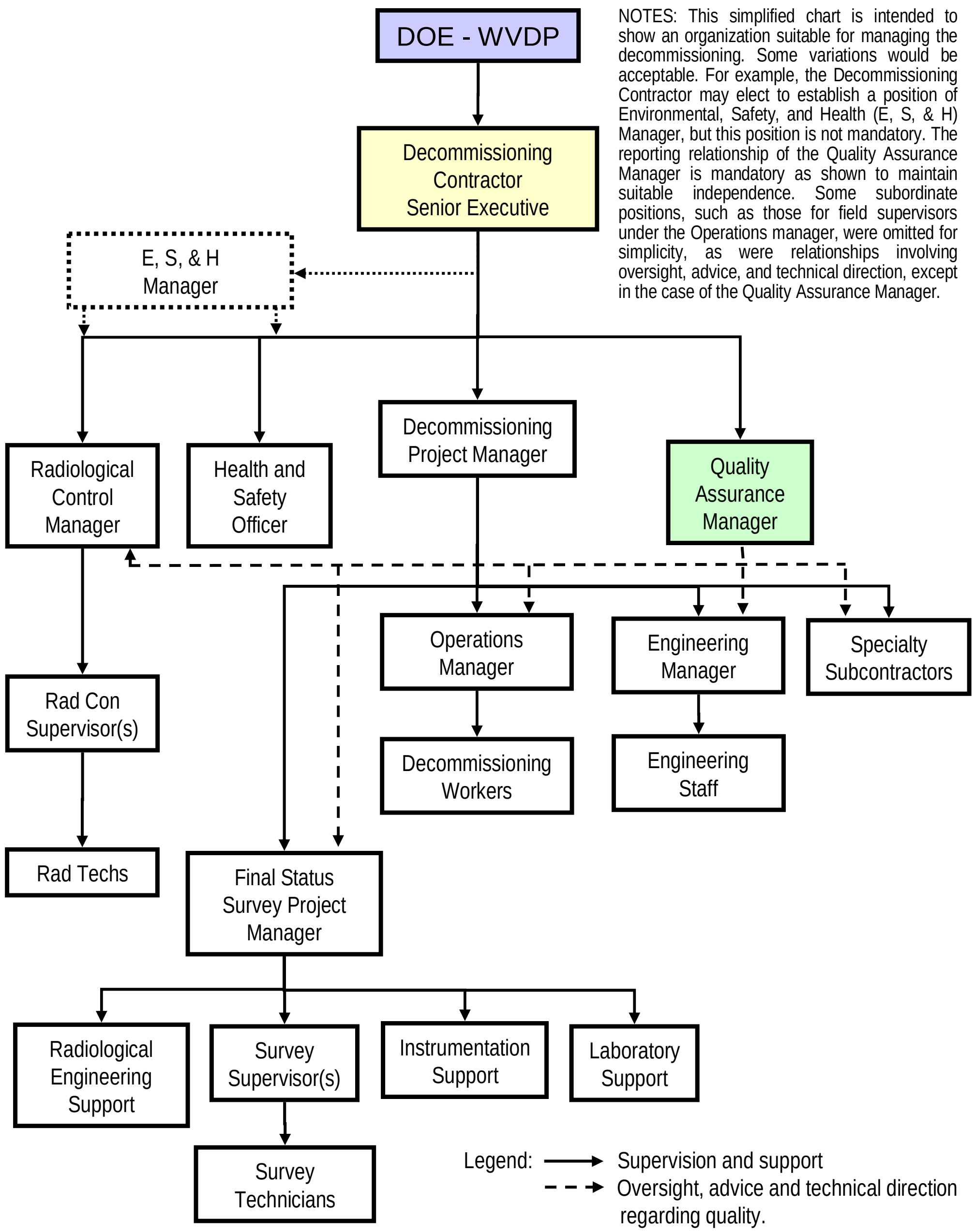


**Figure 8-1. Decommissioning Organization Quality Assurance Relationships**

Quality must be built into the 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 will be developed by the decommissioning organization. QA will 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 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 decommissioning has already been performed by DOE contractors and more will 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 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 2009a), along with four supporting QA procedures (SAIC 2008a, 2008b, 2009b, and 2009c) 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) under the requirements of the WSMS QA Plan (WSMS 2009a)[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 SAIC required WSMS to comply with applicable requirements of:

- The SAIC QA Project Plan that applies to decommissioning preparations (SAIC 2008a), and
- The WSMS procedure for preparing technical documents and performing engineering calculations for the EIS and this plan (WSMS 2009b).

**Oversight and Review**

WSMS review of subcontracted work related to this plan is carried out in accordance with the WSMS QA Plan (WSMS 2009a) and the related procedure (WSMS 2009b). In addition, DOE provides independent oversight of the work performed by site contractors.

### 8.2.3 Other Engineering Work

DOE will ensure that other engineering data and engineering work, calculations, and modeling provided by DOE contractors in support of Phase 1 of the 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 will 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 2009b).

## 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 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 decommissioning.

Specifically, the QA Program will 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 decommissioning will be subject to the applicable controls of the QA Program and activities covered by the QA Program will be identified in program-defining documents.

The Decommissioning QA Program will 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 will include the following elements:

- It will 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 will 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 decommissioning in accordance with those procedures. The QA procedures will 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 will be implementing.
- It shall utilize this plan and appropriate implementing QA procedures to meet its objectives.
- It will require training and qualification of workers and quality verification personnel in accordance with DOE Order 414.1C, with instruction on implementing quality assurance in decommissioning activities and documentation of the objectives and content of the training or qualification, attendees, and dates of attendance.
- NRC will 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 will ensure that the data sets are of the type and quality needed to demonstrate with sufficient confidence that decommissioning activities can be carried out in accordance with applicable requirements. The objective will be met through the use of the data quality control processes for data collection design, analysis, and evaluation.

The data quality control processes will ensure that: (1) the elements of the facility characterization and Phase 1 final status survey plans will be implemented in accordance with the approved procedures; (2) surveys will be conducted by trained personnel using calibrated instrumentation; (3) the quality of the data collected will be adequate; (4) all phases of facility characterization and final survey data acquisition and evaluation will be properly reviewed, and oversight provided; and (5) corrective actions, when identified, will be implemented in a timely manner and determined to be effective. This portion of the QA Program will be applied to all aspects of final facility characterization and Phase 1 final status survey activities. Basic elements of the QA Program as they will 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 Phase 1 final status survey measurements will be trained and qualified in accordance with DOE Order 414.1C. Training will 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 will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity. Records of training will 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 will identify each measurement. Generation, handling, and storage of the original Phase 1 final status survey and facility characterization plans and data packages will be controlled. Records will be designated as quality documents and they will be maintained as such in accordance with WVDP procedures.

**Survey and Sampling Methods**

Areas or facilities to be characterized or surveyed will 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 will be performed in accordance with approved, written procedures that describe the methods and techniques used for the facility characterization or Phase 1 final 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 will be established by procedure. When control is to be transferred, chain of custody forms will accompany the sample for tracking purposes. Secure storage will be provided for archived samples.

**Quality Assurance Project Plans**

Quality assurance for each major task associated with characterization and the Phase 1 final status survey will be described in a QA Project Plan that provides a blueprint for how the quality system of this section will be applied to the particular task. Such plans will 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) will also be considered.

**Quality Control**

Procedures will establish built-in QC verification in the processes for both field and laboratory measurements. The QC verifications will duplicate the original measurements where possible. Acceptance criteria will be established to ensure data are within appropriate bounding conditions. Laboratory analysis verification testing will make use of blank, spiked duplicate and replicate samples and measurements in addition to duplicates. If the acceptance criteria are not met, an investigation will be conducted to determine the cause and corrective action.

**Selection, Calibration and Operation of Instrumentation**

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as well as assure the validity of the data. Instrument calibration will be performed with traceable sources using approved procedures. Issuance, control and operation of instruments will be conducted in accordance with WVDP procedures. Instrument operability will be verified using background and check sources as specified in Section 9.

**Control of Tools and Sample Containers**

New sample containers will be used for each individual sample taken. This practice will ensure the data obtained from each sample will meet QA requirements. Tools will 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, will 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 will be developed within the framework of applicable engineering requirements. The adequacy of these engineering products will be verified or validated by individuals or groups other than those who performed the work. Verification and validation work will be completed before approval and implementation.

A control process that meets the intent of the appropriate requirements of ASME NQA-1 (ASME 2000) will be implemented. Controls will be determined through a controlled process that considers environmental and quality impact.

Basic elements of the QA Program as they will 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) will be correctly translated into design outputs (such as specifications, drawings, procedures, and instructions). Calculations and associated design decisions will be checked for correctness during the design process. Design outputs will be verified to confirm that they will be suitable for their intended use.

Changes to final designs (including field changes and modifications and nonconforming items that will be dispositioned "use as is" or "repair") will 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 – will be verified as appropriate. Computer programs will 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 decommissioning activities will 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 decommissioning activities will be completed, checked, reviewed, and approved prior to using their results. The process for developing calculations that support decommissioning activities will require that calculations define the input data, assumptions, analytical methods, results, and conclusions. An independent reviewer will perform the verification of the correctness of the calculations including the validity of the input data and assumptions. The reviewer also will verify that any modeling of engineering barriers correctly models the design as described in the design documents. As stated above, computer programs will 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 will describe acceptable methods used for engineering tasks associated with decommissioning and contain acceptance criteria to ensure that these tasks will 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 will assure that documents, including changes, will be reviewed for adequacy and approved for release by authorized personnel, and will 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 will 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 will 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 will be formally retired after their use has ended or after they are superceded by another project document. The QA Program will specify details of how this will 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 decommissioning activities important to health and safety will 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 will be used for decommissioning surveys and measurements. Documentation will be maintained to demonstrate that only properly calibrated and maintained equipment will be used; details of how this will be accomplished will 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 will be satisfied.

Materials, parts, or components that will 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 will 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 will perform assessments of decommissioning work processes and operations through the WVDP decommissioning project organization self-assessments, audits, and surveillances. These may include, but will not be limited to, inspections/surveillances, tests, and QA audits.

The assessments will be provided by designated decommissioning project or qualified QA personnel who have sufficient authority and organizational independence to perform these assessments. These personnel will not have direct responsibilities in the areas they will be assessing. The assessments will 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 decommissioning project work processes, operations, procedures, and the QA Program from trending information;
- Audit and surveillance reports which document the items identified above, that will be managed and controlled by 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 will 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 will 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., will be retained.
- Measures shall be established which assure that quality-related procurement documents will be retained.
- Measures shall be established which assure that appropriate records pertaining to audits will be retained.
- Measures shall be established which assure that records associated with radioactive material and personnel exposure controls will 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 will specify in particular where QA records will be stored during the decommissioning and after the 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, *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 2008b, *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 2009a, *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 2. SAIC, Germantown, Maryland, June 29, 2009.

SAIC 2009b, *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 2. SAIC, Germantown, Maryland, June 29, 2009.

SAIC 2009c, *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 2. SAIC, Germantown, Maryland, June 29, 2009.

WSMS 2009a, *Washington Safety Management Solutions LLC Quality Assurance Plan*, WSMS QA 100, Revision 1. Washington Safety Management Solutions, Aiken, South Carolina, August 10, 2009.

WSMS 2009b, *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, Revision 2. Washington Safety Management Solutions, West Valley, New York, September 29, 2009.

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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 decommissioning.

**INFORMATION IN THIS SECTION**

This section first refers to the cleanup criteria for surface soil, subsurface soil, and streambed sediment that will be used to ensure that the level of remediation achieved during Phase 1 will 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 Final Status Survey Report.

While this section addresses all applicable requirements for facility radiation surveys, it does so in general terms because two supplemental documents will provide additional details: a Characterization Sample and Analysis Plan and a Phase 1 Final Status Survey Plan.

**RELATIONSHIP TO OTHER PLAN SECTIONS**

To put into perspective the information in this section, one must consider:

- The information in Section 1 on the project background and those facilities and areas within the scope of the DP;
- The information in Section 2 on facilities to be removed before the Phase 1 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 decommissioning activities and related characterization activities described in Section 7.

The characterization survey process described in this section applies to characterization surveys performed in connection with 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 will 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 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 cleanup activities will support a range of approaches that might be used for Phase 2 of the decommissioning, cleanup goals lower than the DCGLs will be used as indicated in Section 7. These goals are identified in Table 5-14 of Section 5. <u>The cleanup goals are referred to in this section simply as the DCGLs for consistency in terminology</u>.

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 9-3 provide ranges of area factors. The DCGLs apply to the following areas:

- The surface soil DCGLs apply to surface soil throughout the project premises where there is no subsurface contamination below one meter and to the sides of the WMA 1 and WMA 2 large excavations from the ground surface to one meter below the ground surface,
- The subsurface soil DCGLs apply only to the bottoms of the WMA 1 and WMA 2 large excavations and to the sides of these excavations from the bottoms up to one meter below the ground surface, and
- The streambed sediment DCGLs apply only to the streambeds and banks of the portions of Erdman Brook and Franks Creek shown in Figure 5-12.

**Table 9-1 Surface Soil Cleanup Goal Area Factors**(1)

| Nuclide | $DCGL_W$ 10,000 $m^2$ (pCi/g) | Area Factors ($DCGL_{EMC}/DCGL_W$) | |
|---|---|---|---|
| | | 100 $m^2$ | 1 $m^2$ |
| Am-241 | 2.6E+01 | 1.5E+01 | 1.5E+02 |
| C-14 | 1.5E+01 | 2.8E+02 | 1.1E+05 |
| Cm-243 | 3.1E+01 | 3.0E+00 | 2.4E+01 |
| Cm-244 | 5.8E+01 | 1.8E+01 | 2.1E+02 |
| Cs-137 | 1.4E+01 | 2.8E+00 | 2.2E+01 |
| I-129 | 2.9E-01 | 3.8E+01 | 2.1E+03 |
| Np-237 | 2.3E-01 | 6.0E+00 | 3.2E+02 |
| Pu-238 | 3.6E+01 | 1.7E+01 | 2.1E+02 |
| Pu-239 | 2.3E+01 | 2.5E+01 | 3.0E+02 |
| Pu-240 | 2.4E+01 | 2.4E+01 | 2.9E+02 |
| Pu-241 | 1.0E+03 | 1.3E+01 | 1.3E+02 |
| Sr-90 | 3.7E+00 | 2.6E+01 | 2.1E+03 |
| Tc-99 | 1.9E+01 | 2.2E+01 | 1.4E+03 |
| U-232 | 1.4E+00 | 5.4E+00 | 4.4E+01 |
| U-233 | 7.5E+00 | 3.7E+01 | 1.1E+03 |
| U-234 | 7.6E+00 | 4.1E+01 | 2.1E+03 |
| U-235 | 3.1E+00 | 2.6E+01 | 1.9E+02 |
| U-238 | 8.9E+00 | 2.9E+01 | 3.3E+02 |

NOTE: (1) The values in the second column are the cleanup goals ($CG_W$) from Table 5-14 and are based on the probabilistic peak-of-the-mean $DCGL_W$ values for combined soil-streambed sediment exposure assuming 22.5 mrem/y from surface soil. The area factors are based on the limiting case among the probabilistic analysis resident farmer analysis, the deterministic resident farmer analysis, and the deterministic residential gardener analysis.

**Table 9-2. Subsurface Soil Cleanup Goal Area Factors**(1)

| Nuclide | $DCGL_W$ 2,000 $m^2$ (pCi/g) | Area Factors ($DCGL_{EMC}/DCGL_W$) | |
|---|---|---|---|
| | | 92 $m^2$ (2) | 1 $m^2$ |
| Am-241 | 2.8E+03 | 1.1E+00 | 4.3E+00 |
| C-14 | 4.5E+02 | 1.2E+01 | 1.8E+02 |
| Cm-243 | 5.0E+02 | 3.2E+00 | 8.0E+00 |
| Cm-244 | 9.9E+03 | 1.5E+00 | 4.5E+00 |
| Cs-137 | 1.4E+02 | 9.3E+00 | 1.2E+01 |
| I-129 | 3.4E+00 | 4.7E+00 | 9.9E+01 |
| Np-237 | 4.5E-01 | 4.2E+00 | 9.6E+01 |
| Pu-238 | 5.9E+03 | 1.0E+00 | 4.8E+00 |
| Pu-239 | 1.4E+03 | 1.6E+00 | 1.9E+01 |
| Pu-240 | 1.5E+03 | 1.5E+00 | 1.7E+01 |

**Table 9-2. Subsurface Soil Cleanup Goal Area Factors**[1]

| Nuclide | $DCGL_W$ 2,000 $m^2$ (pCi/g) | Area Factors ($DCGL_{EMC}/DCGL_W$) | |
|---|---|---|---|
| | | 92 $m^2$ (2) | 1 $m^2$ |
| Pu-241 | 1.1E+05 | 2.3E+00 | 6.2E+00 |
| Sr-90 | 1.3E+02 | 2.6E+00 | 5.6E+01 |
| Tc-99 | 2.7E+02 | 8.1E+00 | 5.7E+01 |
| U-232 | 3.3E+01 | 2.1E+00 | 1.3E+01 |
| U-233 | 8.6E+01 | 3.6E+00 | 1.1E+02 |
| U-234 | 9.0E+01 | 3.6E+00 | 1.0E+02 |
| U-235 | 9.5E+01 | 3.5E+00 | 3.5E+01 |
| U-238 | 9.5E+01 | 3.6E+00 | 1.0E+02 |

NOTE: (1) The values in the second column are the cleanup goals ($CG_W$) from Table 5-14. The area factors are based on the multi-source analysis or the resident farmer analysis.

(2) The 92 $m^2$ area results from the grid spacing of the STOMP model, which is described in Section 5.

**Table 9-3. Streambed Sediment Cleanup Goal Area Factors**[1]

| Nuclide | $DCGL_W$ 1,000 $m^2$ (pCi/g) | Area Factors ($DCGL_{EMC}/DCGL_W$) | |
|---|---|---|---|
| | | 100 $m^2$ | 1 $m^2$ |
| Am-241 | 1.0E+03 | 2.9E+00 | 2.1E+01 |
| C-14 | 1.8E+02 | 2.0E+01 | 3.3E+03 |
| Cm-243 | 3.1E+02 | 1.2E+00 | 9.0E+00 |
| Cm-244 | 3.8E+03 | 8.7E+00 | 9.4E+01 |
| Cs-137 | 1.0E+02 | 1.2E+00 | 9.4E+00 |
| I-129 | 7.9E+01 | 8.6E+00 | 2.5E+02 |
| Np-237 | 3.2E+01 | 3.4E+00 | 3.3E+01 |
| Pu-238 | 1.2E+03 | 9.1E+00 | 1.4E+02 |
| Pu-239 | 1.2E+03 | 9.1E+00 | 1.4E+02 |
| Pu-240 | 1.2E+03 | 9.1E+00 | 1.4E+02 |
| Pu-241 | 3.4E+04 | 2.9E+00 | 2.2E+01 |
| Sr-90 | 4.7E+02 | 7.2E+00 | 1.5E+02 |
| Tc-99 | 6.6E+04 | 5.1E+00 | 6.3E+01 |
| U-232 | 2.2E+01 | 1.2E+00 | 9.5E+00 |
| U-233 | 2.2E+03 | 2.7E+00 | 2.0E+01 |
| U-234 | 2.2E+03 | 8.5E+00 | 9.7E+01 |
| U-235 | 2.3E+02 | 1.2E+00 | 8.6E+00 |
| U-238 | 8.2E+02 | 1.4E+00 | 1.0E+01 |

NOTE: (1) The values in the second column are the cleanup goals ($CG_W$) from Table 5-14 and are based on the probabilistic peak-of-the-mean $DCGL_W$ values for combined soil-streambed sediment exposure assuming 2.5 mrem/y from streambed sediment. The area factors are based on the deterministic evaluation of the recreationist scenario.

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 will 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 will 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 will 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 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 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 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 work to be accomplished before the Phase 1 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 usually be performed to define the resulting radiological conditions.

Such surveys are not within the scope of this plan since they will be completed before 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 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 provide 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 will be performed. 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 was accomplished in 2008. The results of this program are summarized in Section 4. Also, a sample and analysis plan for additional characterization of the Waste Tank Farm was developed in 2008 and 2009 (Michalczak and Hadden-Carter 2009). This plan is expected to be implemented for additional characterization of Tank 8D-4 and possibly Tanks 8D-1 and 8D-2 as well.

### 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 decommissioning, this plan uses the terminology "Phase 1 final status" and "radiological status" rather than "final status".

Because the decision to release or a final decision on status of the Phase 1 decommissioned areas will 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 1 decommissioning activities. The intent is that Phase 1 final status surveys will 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 their status. Requirements for the Phase 1 final status surveys are addressed in Section 9.6 below.

Note that surveys of shallow excavations to remove infrastructure such as floor slabs, foundations, and hardstands are being performed in accordance with the Characterization Sample and Analysis Plan, rather than the Phase 1 Final Status Survey Plan. These surveys – which are similar to the Phase 1 final status surveys – are simply called "radiological status" surveys in recognition of the difference in the requirements document.

### 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 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 will:

- 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 2009) and that described in Section 4. Table 4-11 shows background concentrations in various environmental media for most radionuclides of interest.

Additional background measurements will be taken in connection with characterization surveys outlined in Section 9.4. The characterization surveys for environmental media will be described in a separate Characterization Sample and Analysis Plan to be developed and submitted for NRC review. The additional measurements will include exposure rates and samples from non-impacted soil in suitable non-impacted (background) reference areas. These additional samples will be subjected to appropriate radionuclide–specific analyses to address all 18 radionuclides of interest.

Applicable guidance for background surveys in the MARSSIM (NRC 2000) and in NUREG-1505 (Gogolak, et al. 1997) will be incorporated. Guidance provided in NUREG-1757, Vol. 2, (NRC, 2006) will be considered to ensure that quality objectives and survey execution, controls, and results are consistent with those of the characterization and Phase 1 final status surveys.

The surveys and sampling in non-impacted offsite areas to establish a basis for background radioactivity levels will be described in detail in the Characterization Sample and Analysis Plan. The application of the background data during assessment and use of the data obtained in the characterization and Phase 1 final status surveys will be based on guidance in Chapter 8 of the MARSSIM (NRC 2000) and will be described in each of the respective plans.

Since all surface soil in areas of interest on the project premises will be treated as impacted for Phase 1 final status surveys purposes, it is anticipated that the Sign test will be used in the Phase 1 final status surveys to show $DCGL_W$ compliance and application of a background reference area will not be necessary. (If the DCGLs were to be revised in a manner that results in lower values for naturally occurring radionuclides, the Wilcoxon Rank Sum test would be required instead, and a background reference area would become necessary.)

### 9.4 Characterization Surveys

As noted above, four formal characterization survey programs have been completed for portions of the project premises, additional characterization for the Waste Tank Farm is

[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.

planned, and routine sampling and analysis of environmental media are performed in connection with the WVDP environmental monitoring program.

The Characterization Sample and Analysis Plan, the contents of which are described below, will provide for additional characterization of soil, sediment, and groundwater on the project premises. This plan is expected to be issued before the start of Phase 1 decommissioning activities. The characterization performed will be consistent with the following objectives and guidance.

### 9.4.1 Characterization Sample and Analysis Plan Content

This plan will provide details of characterization surveys to be performed to more precisely determine the extent and the amount of residual radioactivity in environmental media as Phase 1 decommissioning activities begin.

**Scope of the Plan**

The plan will focus primarily on radioactivity in soils, sediment, and groundwater on the project premises. It will also address the following matters:

- Identifying the presence of buried infrastructure,
- Collecting geotechnical data to support hydraulic barrier wall design,
- Determining the radiological status of soil around representative Process Building foundation pilings when they become accessible during demolition of the Process Building,
- Determining the radiological status of the HLW transfer trench after piping and equipment in the trench is removed,
- Determining the radiological status of soil in the bottom of shallow excavations after removal of infrastructure such as concrete slabs and foundations, and
- Collecting data to guide soil removal and to verify that remediation goals for a particular location have been achieved.

The plan will not address characterization of structures or characterization of materials (equipment, demolition debris, or excavated soil) for waste management purposes. Section 9.4.5 describes additional characterization surveys to support facility removal that will be performed in connection with Phase 1 decommissioning activities.

For characterizing materials for waste management purposes, the decommissioning contractor will provide a procedure and obtain DOE approval of this procedure. This procedure will 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 will apply to, among other materials, surface and subsurface soil not known to have been impacted by radioactivity. (As an alternative, these matters may be addressed in the Waste Management Plan.)

**Requirements and Guidance to be Followed**

This plan will 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 will identify the radionuclides of interest. It will 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. An additional 12 radionuclides have been identified as possibly (but unlikely to be) present at the site. In addition, the presence of progeny not in equilibrium with the 18 radionuclides of interest has also been identified as a possible concern. Both issues have the potential for requiring changes to the radionuclides of interest list. Data collected in implementation of the Characterization Sample and Analysis Plan will determine whether this is necessary.

**Data Quality Objectives**

This plan will 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 will 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 will be addressed in the Characterization Sample and Analysis Plan.

**Background Radiation and Radioactivity**

The Characterization Sample and Analysis Plan will 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 will specify the methods to be used to collect the necessary characterization data. Among the methods considered will be:

- Exposure rate measurements
- Surface contamination scans
- Surface contamination direct measurements
- Smear surveys for removable contamination
- Soil samples
- Groundwater samples
- Sediment samples

Note that surface contamination scans, surface contamination direct measurements, and smear surveys for removable contamination apply only to surveys of the empty HLW transfer trench.

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 will 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 will 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.(3) |
| Exposure rate | Eberline RO-2(1) | Ion Chamber | 0.1 mrem/h | For low-range readings. (3) |
| 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 $cm^2$<br>85 dpm/100 $cm^2$ | Scans 100 dpm, direct measurements 85 dpm. (3) |
| Beta | Ludlum 43-89(1) | ZnS (Ag) scintillator, 100 $cm^2$ probe | 2,500 dpm/100 $cm^2$<br>800 dpm/100 $cm^2$ | Scans 2,500 dpm, direct measurements 800 dpm. (3) |
| Beta-gamma | Ludlum 44-40(1) | Geiger-Mueller (G-M) shielded pancake probe | 3,300 cpm/mrem/h | For scanning in tight areas. (3) |
| Beta-gamma | Ludlum 44-9(1) | G-M unshielded pancake probe | 3,300 cpm/mrem/h | For scanning in tight areas(3) |
| Beta-gamma | Ludlum 44-6(1) | G-M sidewall detector | 1,200 cpm/mrem/h | For use as a pipe probe.(4) |

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 will be specified in the Characterization Sample and Analysis Plan.

(3) Suitable for surveys of empty HLW transfer trench but not of soil areas.

(4) For use in surveys of underground lines on the edges of excavations as specified in Section 9.7.

Samples may be analyzed onsite or shipped to an offsite contract laboratory for analysis. Laboratory methods, instruments and sensitivities will be in accordance with New York State protocols for environmental analysis. Any laboratory used for environmental sample analysis will have appropriate New York State Department of Health Environmental Laboratory Approval Program certification, or equivalent.

Table 9-5 indicates the target minimum detectable concentrations for radionuclides in laboratory analyses of soil samples. Minimum detectable concentration requirements are set to whichever is lower: (1) approximately 10 percent of the most restrictive radionuclide-specific cleanup goal identified in Table 5-14, (2) 25 percent of nominal background for naturally occurring radionuclides, or (3) standard laboratory minimum detectable concentrations.

**Table 9-5. Radionuclide Target Sensitivity for Laboratory Sample Analysis**

| Nuclide | Instrument/Method | Target Sensitivity pCi/g(1) |
|---|---|---|
| Am-241 | Alpha and/or gamma spectrometry | 1(4) |
| C-14 | Sample oxidizer and liquid scintillation | 2(4) |
| Cm-243/244(5) | Alpha and/or gamma spectrometry | 1(4) |
| Cs-137 | Gamma spectrometry | 0.1(4) |
| I-129 | Gamma spectrometry and/or gas flow proportional counting | 0.06(2) |
| Np-237 | Alpha and/or gamma spec | 0.01(2) |
| Pu-238 | Alpha spectrometry | 1(3) |
| Pu-239/240(5) | Alpha spectrometry | 1(3) |
| Pu-241 | Liquid scintillation | 15(3) |
| Sr-90 | Liquid scintillation | 0.9(2) |
| Tc-99 | Gas flow proportional counting | 3(2) |
| U-232 | Alpha spectrometry | 0.5(2) |
| U-233/234(5) | Alpha spectrometry | 0.2(3) |
| U-235 (-236)(5) | Alpha spectrometry | 0.1(3) |
| U-238 | Alpha spectrometry | 0.2(3) |

NOTES: (1) Dependent on sample size, counting time, etc.
(2) Approximately 10 percent of the most restrictive radionuclide-specific cleanup goal identified in Table 5-14.
(3) 25 percent of background for naturally occurring radionuclides.
(4) Standard laboratory minimum detectable concentrations.
(5) When analytical results cannot be identified to a single isotope, the results will be applied to the isotope with the more restrictive DCGL.

### Survey Locations

This plan will 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 will 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 will specify the type and extent of characterization measurements in different facilities and areas.

**Handling Waste Generated During Characterization**

The Characterization Sample and Analysis Plan will specify how to minimize and manage investigative derived waste.

**Health and Safety**

This plan will 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 will 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 will specify necessary supporting procedures, such as those for obtaining, handling, preserving, and packaging samples, as well as chain of custody procedures.

**Documentation**

This plan will 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 will 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 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 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 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 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 depths of six inches (15 cm) and 3.3 feet (one meter) 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 decommissioning activities in the facility or areas involved, so long as sufficient time is allowed to make preparations based on the data. Measurements and sampling in outside areas are dependent on the weather.

- Scale of decision-making: Areas of interest will generally conform to particular areas to undergo decommissioning, i.e., decisions will 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 will be addressed in the Characterization Sample and Analysis Plan. It will 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) will be addressed.

**Optimizing the Design**

The content of the Characterization Sample and Analysis Plan will 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 will apply to characterization. The following matters will 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) will be addressed.

**Field Instruments**

Field instruments will be calibrated in accordance with written procedures using standards traceable to the National Institute of Standards and Technology. They will be calibrated every six months and following any substantial repair. Battery status, check source response, and background measurements will 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 will 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 will 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 will be performed before use. Control charts with specified limits of acceptability will be used to document and trend source and background checks.

Offsite analytical laboratories will be required to meet all applicable quality requirements; the laboratory Quality Assurance Plan will be reviewed to ensure that applicable requirements are included. Offsite laboratories will be audited to assure quality performance.

**Sample Chain of Custody**

Sample chain of custody procedures will be established and followed to ensure that sample accountability and integrity are maintained. This process will 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 will 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 will be recorded in a legible manner and reviewed for matters such as accuracy of recording and transcription, procedure compliance, completeness, and consistency. Data recorded on the location of field measurements and samples in excavations will include the depth of the measurement point with respect to the nearby ground surface or the elevation above mean sea level. Calculations will be checked and conclusions will be peer reviewed. Problems identified will be resolved before the data are utilized. Data reports and documents will be archived and maintained to comply with the Project Quality Assurance Program described in Section 8.

### 9.4.4 Applying DQOs for Characterization Surveys

The following example illustrates how DQOs will be applied to characterize a particular area of interest in a manner supportive of final status survey information needs.

The example is the footprint of the old hardstand, which was located on the west side of the Lag Storage Additions 3 and 4. The old hardstand footprint has the potential for radioactive contamination below the surface due to the major spill described in Table 2-17. The footprint of the old hardstand will undergo characterization as part of the planned Characterization Sample and Analysis Plan activities.

The Characterization Sample and Analysis Plan includes a set of characterization objectives that form the basis for DQO planning process. Of this set, the following are pertinent to the old hardstand area:

- Evaluate appropriateness of the current list of radionuclides of interest,
- Verify absence of additional radionuclides of interest,
- Identify the presence/absence of buried contamination,
- Determine extent of surface contamination,
- Identify soil waste stream characteristics, and
- Obtain data to support Phase 2 planning.

Data collection requirements specific to the old hardstand for each of these objectives will be developed as part of the DQO evaluation contained in the Characterization Sample and Analysis Plan. Characterization Sample and Analysis Plan decision-making (and consequently, Characterization Sample and Analysis Plan data collection activities) will be sequential with respect to these objectives. For example, data collection to verify the absence of additional radionuclides of interest may result in changes to the list of analytes for the balance of sampling work conducted for the old hardstand. As another example, if sampling identifies contamination likely to require remediation (either as a discretionary Phase 1 activity or during Phase 2), analyses would be conducted to determine waste stream characteristics.

Characterization Sample and Analysis Plan data collection will support final status survey requirements in a number of ways. The first two Characterization Sample and Analysis Plan objectives listed above will determine the list of radionuclides that final status survey activities will need to address. If contamination is encountered deeper than one meter (third objective), then the old hardstand area will not be a candidate for Phase 1 final status survey data collection, and instead data collection will focus on identifying the nature and extent of surface and subsurface contamination that is present. Alternatively, if contamination is present above Phase 1 DCGL requirements but not at depths greater than one meter, DOE may defer remediation until Phase 2.

If initial Characterization Sample and Analysis Plan data collection indicates the old hardstand area is a candidate for Phase 1 final status surveys (i.e., there is no evidence of contamination exceeding DCGL requirements in surface soils and no evidence of contamination deeper than one meter), then the balance of soil sampling from the former hardstand area would be conducted in a manner consistent with final status survey DQO requirements. Final status survey sampling requirements are described in detail in the Phase 1 Final Status Survey Plan. In general, these would include biased surface soil samples (representative of a 0 to 15 cm depth and representative of a 0 to 1 m depth) that targeted specific locations of concern (e.g., historical locations where contamination was present, gamma walkover survey anomalies, etc.) to determine $DCGL_{EMC}$ compliance, and systematic surface soil sampling (representative of a 0 to 15 cm depth and representative of a 0 to 1 m depth) to determine $DCGL_W$ compliance.

### 9.4.5 Characterization Surveys of Structures

Because the structures within the scope of Phase 1 decommissioning activities will be entirely removed, the additional characterization of these structures will focus on data necessary to support worker protection, waste management, and minimizing airborne radioactivity releases during demolition, this last factor being especially important for the Process Building. This characterization will take into account available radiological data.

> NOTE
>
> Because the additional characterization of structures within the scope of Phase 1 of the decommissioning is being performed for the purposes of health and safety, environmental control, and radioactive waste management – activities which are based on DOE procedures as explained in the Section 1 and will have no bearing on conditions at the conclusion of Phase 1 decommissioning activities – the additional characterization is briefly summarized below, rather than being described in more detail.

**Available Radiological Data**

Available characterization data on structures to be removed during Phase 1 of the decommissioning are summarized in Section 4.1.5. A large body of data is available on different areas of the Process Building and the Vitrification Building from the Facility Characterization Project undertaken during the 2002 to 2005 period. The Facility Characterization Project produced a total of 33 radioisotope inventory reports with bounding estimates of residual radioactivity in different building areas. Characterization data on structures other than the Process Building and Vitrification Facility are more limited.

A substantial amount of data on radionuclide distributions in different parts of the Process Building was developed during the Facility Characterization Project. These data have been used for waste management purposes during deactivation work and the characteristics of many of the waste streams associated with Process Building equipment have been well defined.

Due to the continuing deactivation work, radiological conditions in parts of the Process Building and some other areas will have changed by the time the interim end state is reached. Updated characterization data may be available for some areas before Phase 1 decommissioning activities begin.

Given this situation, the initial step in additional characterization of the structures of interest will entail review of available data to determine the additional data that will be needed.

**Review of Available Radiological Data**

Radiological conditions within the structures can vary widely. In the Process Building, some areas have never been entered since plant operations began due to high radiation levels while others have virtually no contamination.

Available radiological data for the facility or area of interest will be reviewed, considering activities that may have taken place since those data were collected. Such data will include data collected in the Facility Characterization Project and in end of task surveys discussed in Section 9.2.3. Conditions in the facility or area of interest will be taken into account in evaluating these data, such as cases where concrete grout has been poured over contamination on a floor or a fixative applied to a contaminated wall.

Because radiological conditions in different areas vary widely, the scope of additional characterization will be tailored to the potential hazards involved.

**Exposure Rate Measurements**

A clear understanding of the general area dose rates and any significant hot spots is necessary for all controlled areas where people will be working.

Before work begins in such an area, exposure rate measurements will be taken to identify the general area dose rates and the hot spots in those areas with hot spot potential, unless such data reflecting current conditions are already available. Table 9-6 identifies field instruments suitable for a wide range of exposure rate measurements inside structures.

**Table 9-6. Radiological Field Instruments for Facility Characterization**

| 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 low potential areas. |
| Exposure rate | Ludlum 44-10(1) | 2-inch NaI scintillator | 900 cpm/μR/h | For scanning low potential areas. |
| Alpha | Ludlum 43-89(1) | ZnS (Ag) scintillator, 100 $cm^2$ probe | 100 dpm/100 $cm^2$<br>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 $cm^2$<br>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 |

### Contamination Measurements

A general understanding of accessible removable contamination is necessary for all controlled areas where people will be working. Before work begins in such an area, smears will be taken on representative surfaces and counted for removable beta and alpha radioactivity unless sufficient data on removable contamination are already available. Airborne radioactivity measurements will be made as necessary to support radiation protection planning. Surface scans for total alpha and/or beta contamination will be performed only in cases where the resulting data would be useful for planning purposes.

Because the facilities of interest in Phase 1 of the decommissioning will be entirely removed, surveys of inaccessible areas are not expected to be necessary.

### Characterization Data for Waste Management Purposes

Where necessary for waste stream characterization, exposure rate measurements will be made and smears or other physical samples of materials will be analyzed to determine radionuclide distributions.

### Documentation of Additional Characterization Surveys

These characterization data will be formally documented and reviewed. Information recorded for exposure rate and contamination measurements will include the date, location, type of measurement, instrument type, instrument serial number, and mode of operation.

Samples will be controlled and laboratory analyses performed and documented using the processes for environmental media samples described above.

**Quality Assurance and Quality Control**

All facility characterization measurements will be accomplished and documented in accordance with the provisions of Section 8 of this plan and the Quality Assurance Project Plan.

## 9.5 In-Process Surveys

In-process or remedial action support surveys will be performed while remediation is in progress. The primary purposes of these surveys are to guide decontamination and determine when remediation to the cleanup goals specified in Section 5 has been attained. In-process surveys also support radiation protection.

### 9.5.1 Measurement Methods and Instrumentation

Measurement methods and instruments used will be identical to those utilized during the characterization surveys described in Section 9.4 and the Phase 1 final status surveys. Survey quality objectives during in-process survey activities for soil and sediment will be aligned with the quality objectives of the Phase 1 final status surveys, to ensure that decisions and interpretations of data have the same confidence as those based on the Phase 1 final status survey results. Data quality objectives and quality control parameters will be consistent with those identified for the Characterization Sample and Analysis Plan, in Section 9.4, above. Media-specific and instrument/method-specific background levels developed by measurements and sampling in the Characterization Sample and Analysis Plan will be applied during the remediation, usually through subtraction from onsite analysis of samples.

The Characterization Sample and Analysis Plan will specify the sampling, instruments and data objectives for surveys in the area around the Process Building foundation pilings, an area that will not be readily accessible until late in the excavation in WMA 1 when overlaying structures are removed. In-process surveys in this area will be used to guide remediation and to identify locations for biased sampling.

Because surveys performed in the deep excavations are expected to be dominated by Sr-90, nuclide-specific measurements by onsite sample analysis will be used to guide the excavation. Where practicable, correlations between gamma exposure rates and soil radioactivity concentrations will 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.

### 9.5.2 Scan Surveys and Direct Measurements

Investigation levels for scanning surveys to identify areas of elevated activity will be determined in the implementation of the Characterization and Sampling Plan. Scanning surveys will be performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action. Areas of elevated activity typically represent a small portion of the site or survey unit. Thus, random or systematic direct measurements or sampling on a grid spacing may have a low probability of identifying these areas, so that scanning surveys are typically performed before direct measurements or sampling. Because of the inability to detect certain radionuclides of interest in scanning surveys as discussed below, collection and analysis of soil samples will be necessary using protocols specified in the Characterization Sample and Analysis Plan and the Phase 1 Final Status Survey Plan.

**Scan Minimum Detectable Concentrations**

Procedures are provided in the MARSSIM for calculating scan minimum detectable concentrations (MDCs) for particular survey instruments. More detail on signal detection theory and instrument response is provided in NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions* (NRC 2007). These procedures will be followed to obtain appropriate scan MDCs for the specific instruments to be used at the site. These calculations will take into account site-specific factors such as soil properties, the expected distribution of radionuclides in soil, and the scanning speed. This information will be developed as part of future planning activities for the project and will be available for NRC review.

To assist with current planning activities, estimated scanning MDCs in soil for the radionuclides of interest were obtained for field survey instruments by reviewing available information, and these values are shown in Table 9-7. Information is only provided for 14 of the 18 radionuclides, as four have no or minimal photon (gamma ray and X-ray) emissions making them impractical to detect with field scanning instruments. Field survey instruments for soil contamination are generally limited to those that can detect photons, given the uneven terrain and conditions encountered in the field. This is in contrast to survey instruments that can be used for buildings, many of which allow for the detection of alpha and beta contamination as well as gamma emissions.

**Table 9-7. Estimated Scanning Minimum Detectable Concentrations (MDCs) of Radionuclides in Soil**

| Radionuclide | Type of detector | Scan MDC (pCi/g) |
|---|---|---|
| Am-241 | FIDLER | 30 |
| C-14 | NA(1) | - |
| Cm-243 | 2” by 2” NaI | 50 |
| Cm-244 | FIDLER | 300 |
| Cs-137 | 2” by 2” NaI | 7(2) |
| I-129 | FIDLER | 60 |
| Np-237 | FIDLER | 30 |
| Pu-238 | FIDLER | 100(3) |
| Pu-239 | FIDLER | 200(3) |
| Pu-240 | FIDLER | 100 |
| Pu-241 | NA(1) | - |
| Sr-90 | NA(1) | - |

**Table 9-7. Estimated Scanning Minimum Detectable Concentrations (MDCs) of Radionuclides in Soil**

| Radionuclide | Type of detector | Scan MDC (pCi/g) |
|---|---|---|
| Tc-99 | NA(1) | - |
| U-232 | FIDLER | 60 |
| U-233 | FIDLER | 500 |
| U-234 | FIDLER | 60 |
| U-235 | FIDLER | 30 |
| U-238 | FIDLER | 60 |

NOTES: (1) NA means not applicable; either there are no photons associated with the radionuclide or the photon yield is too low to allow for detection by field scanning instruments.

(2) A specific calculation of scanning minimum detectable count rate for Cs-137 in soil performed in connection with preparation of the Phase 1 Decommissioning Plan yielded a value equivalent to 7 pCi/g Cs-137. A comparable value of 6.4 pCi/g is given in Table 6.7 of the MARSSIM when units are given in pCi/g.

(3) While scan MDCs of 10 and 20 pCi/g are reported for Pu-238 and Pu-239, respectively, in Appendix H of MARSSIM, much larger values were reported elsewhere. The values given here are those expected to be reasonably achievable under field conditions.

The scanning MDCs given in Table 9-7 are representative of those that reasonably can be expected to be obtained with currently available instruments under conditions encountered in the field. These values were obtained from reported values and scanning experience at other radioactively contaminated sites.[3]

Experience for the Shallow Land Disposal Area site in Pennsylvania indicated that the calculated values were much lower than was actually obtainable under field conditions, which is reflected in the values given in Table 9-7. For some radionuclides (such as Pu-238 and Pu-239), a wide range of values was reported. In this case, a midpoint value is given in the table.

Information for scan MDCs was not available for about half of the radionuclides. In these cases, the energy spectrum and yields of the gamma rays and X-rays were reviewed along with the relative detector response (by photon energy). This allowed for an estimate to be made of the scan MDCs for those radionuclides without published information.

The scan MDCs for some radionuclides exceed surface and subsurface soil $DCGL_W$ values (cleanup goals) given in Table 5-14 of this plan. Also, the general approach used to calculate scan MDCs assumes flat terrain and does not account for situations where scans may be occurring on the sides of excavations. Experience has shown that it is difficult to obtain scan MDCs at the levels calculated using conditions that are more ideal than

[3] Calculations of scan MDCs are provided in a number of gamma walkover plans including the *Site Radiological Survey Plan* for the CWM Chemical Services site in Model City, New York (CWM 2006) and the *Final Gamma Walkover Survey Sampling and Analysis Plan* for the Shallow Land Disposal Area FUSRAP site in Pennsylvania (USACE 2003). Additional information reviewed included the article *Detection of Depleted Uranium in Soil Using Portable Hand-Held Instruments* (Coleman and Murray 1999) and *Ask an Expert Question and Answer Page on Survey Instruments (conventional)* (ORAU 2009). These sources provided a range of scan MDCs for several different detectors.

generally occur at the site. The values given in Table 9-7 account for expected field conditions.

Because there may be multiple radionuclides present at many locations, it will be necessary to achieve soil concentrations at some relatively small fraction of the DCGLs to arrive at definitive conclusions as to the need to conduct further remedial action. This typically cannot be done using scanning instruments. Rather, scanning techniques are generally used to indicate the presence of elevated radioactivity (above background) and the radionuclides that may have elevated concentrations. Definitive conclusions as to the presence or absence of contamination above radionuclide-specific DCGLs will be made by making direct static measurements or by collecting samples for analysis.

**Direct Measurements**

Direct measurements may be collected at random locations in the area of interest. Alternatively, direct measurements may be collected at systematic locations to supplement scanning surveys for the identification of small areas of elevated activity. Direct measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify locations for direct measurements to further define the areal extent of residual radioactivity and to determine maximum radiation levels within an area, although these types of direct measurements are usually associated with preliminary surveys (i.e., scoping, characterization, remedial action support). All direct measurement locations and results shall be documented.

For those radionuclides that cannot be effectively measured directly in the field, samples of the soil in the area under investigation will be collected and then analyzed with a laboratory-based procedure including gamma spectrometry, beta analysis using liquid scintillation counting, or alpha spectrometry following separation chemistry.

### 9.5.3 Documentation

Data collected during in-process survey field measurements and sampling will be formally controlled and documented as specified in Section 8. Data recorded on the location of field measurements and samples in excavations will include the depth of the measurement point with respect to the nearby ground surface or the elevation above mean sea level. Data reports and documents will 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 will be accomplished in accordance with a Phase 1 Final Status Survey Plan(s). Because the 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 will 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) will 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.

Appendix G describes the conceptual framework for the Phase 1 Final Status Survey Plan.

### 9.6.1 Phase 1 Final Status Survey Plan Content

The Phase 1 Final Status Survey Plan(s) will 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 will 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 will follow provisions in NUREG-1757 Volume 2 (NRC 2006) and guidance of the MARSSIM (NRC 2000).

**Overview of Survey Design**

This plan will provide a brief overview of the survey design. This design will 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 will 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 will identify the cleanup criteria specified in Section 5. It will also identify investigative levels and how they were established.

**Use of Characterization Data for Phase 1 Final 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 will be addressed in the Phase 1 Final Status Survey Plan.

**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 decommissioning activities, this will be identified in the Phase 1 Final Status Survey Plan.

**Classification of Areas**

Different areas of the project premises facilities and areas of interest will be classified based on potential for radioactive contamination. Four classifications will 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 2009. Preliminary classification will 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 will be made as to whether or not that area exceeds the regulatory limit. Areas within a survey unit will have a similar usage history and contamination potential and be contiguous areas of the same area classification.

Survey units will be specified in the Phase 1 Final Status Survey Plan. They will be identified in tables or drawings or a combination of the two. Among areas considered in designating survey units will be:

- The bottoms and sides of the WMA 1 and WMA 2 excavations before they are back-filled;
- Laydown areas for excavated soil after the soil has been removed; and
- Areas where Phase 1 final status surveys are to be performed for surface soil.

In some survey units, data from characterization will be sufficient for Phase 1 final status survey purposes; this matter will be addressed in the Phase 1 Final Status Survey Plan.

**Background Radiation and Radioactivity**

Appropriate measurements will 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 will be subtracted from Phase 1 final status survey results.

**Data Quality Objectives**

Data Quality Objectives for the Phase 1 final status survey will 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 will 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 will incorporate performance-based measurement systems, specifying the analytical sensitivity goal of each survey method. Individual methods (i.e., static surface counts) will 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, will 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 will specify the field and laboratory instruments to be used and the sensitivity of these instruments and methods. Table 9-8 shows typical field instruments to be addressed in the plan.

**Table 9-8. 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 gammas |

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 will be specified in the Phase 1 Final Status Survey Plan.

The Phase 1 Final Status Survey Plan will specify how the minimum detectable concentration (MDC) for media samples and the MDC for scanning surveys ($MDC_{scan}$) will be determined for each instrument and technique using methods specified in NUREG-1757, Volume 2 (NRC 2006). It will 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 will 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 9-5.

### Scan Surveys

Scan surveys of survey units of the different classifications will be performed as indicated in Table 9-9 below. The purpose of such scan surveys is to identify small areas of elevated activity.

**Table 9-9. 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<br>>$MDC_{scan}$ if $MDC_{scan}$ is greater than $DCGL_W$. |
| Class 3 | Judgmental | >$DCGL_W$ or<br>>$MDC_{scan}$ if $MDC_{scan}$ is greater than $DCGL_W$. |
| Non-impacted | None | Not applicable. |

NOTES: (1) Entire surface of accessible soil areas.
(2) Surveys will be both systematic and judgmental.

The derivation of scan and fixed MDCs will 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 will be provided, with reference to the detailed procedure(s) to be used for this purpose.

### Survey Grids

Survey grids of appropriate size will be laid out and marked on excavations and land areas. Where practicable, grids established for characterization surveys will 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 will 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 will specify how this will be done in particular areas.

### Surveys and Sampling of Individual Facilities and Areas

This plan will specify the process to determine the number of samples required in different areas following MARSSIM protocols. This process will 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 will 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 will be reclassified correctly, and a survey appropriate to the new classification will be performed.

Whether the measurement results demonstrate that the survey unit meets the release criteria will then be determined. The process for this and the statistical tests to be used will 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 will be performed until the release criteria are achieved.

One radionuclide (I-129) in surface soil will be treated as a special case because its cleanup goal is the same order of magnitude as the minimum detectable concentration in typical laboratory sample analyses.[4] 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 this radionuclide should not appear in background soil samples, analysis at a detection limit near the DCGL will be sufficient to flag results should a sample indicate the presence of either radionuclide above its detection limit.

[4] In Revision 1 of this plan, both I-129 and Np-237 were identified as special cases because of low cleanup goals. The revised surface soil cleanup goal for Np-237 is higher than the Revision 1 value (0.23 vs. 0.096 pCi/g). Typical laboratory detection limits for Np-237 in soil samples are around 0.01 pCi/g as shown in the 2008 data in Table C-4. However, typical laboratory detection limits for I-129 are in the 0.1 to 0.3 pCi/g range as shown in Table C-4, so the laboratory detection limit may exceed 50 percent of the cleanup goal for this radionuclide. Although Table 9-5 specifies a target detection limit of 0.06 pCi/g for I-129, it is unlikely that this value can be consistently achieved in practice without special efforts.

The Phase 1 Final Status Survey Plan will provide an alternate method for evaluating analytical results for I-129 that do not exceed the minimum detectable concentration. 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. Scaling factors for spent fuel reprocessed specified in Table 4-1 will 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 well below the cleanup goal.

The amounts of I-129 that might be found in surface soil contamination, if any, will likely be small. This conclusion is based on comparisons between the estimated amounts of this radionuclide 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 inventory of I-129 to be more than seven orders of magnitude less than the estimated Cs-137 present.

**Health and Safety**

This plan will 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 will 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 will specify necessary supporting procedures, such as those for obtaining and managing samples.

**Documentation**

This plan will detail the requirements for formally documenting and archiving Phase 1 final status survey data, in accordance with the requirements of Section 9.8. Data recorded on the location of field measurements and all sample locations in excavations will include the depth of the measurement point with respect to the elevation above mean sea level.

### 9.6.2 Data Quality Objectives for the Phase 1 Final Status Survey

The DQOs will be detailed in the Phase 1 Final Status Survey Plan; they will 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 will produce necessary data.

- Defining the study boundaries. Radionuclides of interest, areas of interest, necessity to obtain data to support the 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 will 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 will 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 will apply, along with the quality requirements for the characterization survey as identified in Section 9.4.3. These matters will 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 decommissioning activities in adjacent areas and the impact of routes for transportation of radioactive materials on survey units and area classification will 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 will change during deactivation work performed before the start of the decommissioning, additional surveys will be performed as 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 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 9-10 shows laboratory instruments and methods, along with their sensitivities.

**Table 9-10. 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 decommissioning activities in each area, characterization measurements will be taken as specified in Section 9.4.5. The measurements will 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 will 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 will typically include exposure rates, removable contamination, and total contamination. Samples will 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 will be exposed so surveys can be made only in cases where this is essential for radiation protection purposes.

***Justification for Planned Survey Measurements***. These are the appropriate measurements necessary to facilitate radiation protection and support planning decommissioning activities and waste management.

**Characterization of Other WMA 1 Facilities**

The other facilities to remain within WMA 1 after 2009 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 will be entirely or partially within the bounds of the planned excavation, characterization measurements will 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-2009 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 on the third and fourth floor in the 01-14 Building that contain ventilation system equipment, are not routinely surveyed.

***Description of Planned Survey Measurements***. Measurements will typically include exposure rates, removable contamination, and total contamination. Representative embedded piping in the 01-14 Building floor slab, except for sealed floor drains, will 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 will 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.

**Characterization of Subsurface Piping in WMA 1**

DOE has evaluated contaminated underground piping as described in Appendix F. 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 will be removed, packaged and disposed of at offsite disposal facilities. There is no intent in Phase 1 of the decommissioning to trace or excavate underground piping outside the bounds of the excavation.

When these lines become exposed during the course of decommissioning work, measurements will 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 will be taken after the interior surfaces of the lines are exposed during the course of decommissioning work. Three types of measurements will 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 will be obtained and analyzed to determine the radionuclide distributions.

***Justification for Survey Measurements***. These measurements will provide information on interior contamination levels that will 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 will 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 in WMA 1 Facilities**

In-process surveys will be performed in the Process Building and Vitrification Facility during remediation as specified in Section 9.5. In-process surveys in other WMA 1 facilities will also be performed during remediation as described in Section 9.5. However, the scope of such surveys will be minimal because of the relative low potential for contamination, except in some areas of the 01-14 Building which may contain significant contamination.

**In-Process Surveys in the WMA 1 Excavation**

In-process surveys will be performed in connection with removing soil during the large WMA 1 excavation as specified in Section 7 and Section 9.5. They will be coordinated with surveys performed around Process Building foundation pilings that are specified in the Characterization Sample and Analysis Plan.

When the excavation has reached the planned depth of at least one foot into the unweathered Lavery till, a systematic in-process survey will be performed as specified in the Characterization Sample and Analysis Plan. Survey grids will be laid out. A complete gamma scan of both the floor and the sides of the excavation will be performed to identify areas of elevated activity as evidenced by above-background measurements. Biased soil samples will be collected from areas of elevated activity and analyzed onsite for Sr-90 and Cs-137. Systematic soil samples will also be collected and analyzed onsite for Sr-90.

The survey results and sample analytical data will be used to determine if additional soil removal is necessary. If additional soil removal is necessary, an additional in-process survey will be performed in the area of interest after the soil is removed using the protocols described in the Characterization Sample and Analysis Plan.

**In-Process Surveys Related to Subsurface Piping in WMA 1**

In-process surveys will be performed during removal of piping as described in Section 9.5. Some characterization surveys will effectively be in-process surveys since they will be performed in conjunction with piping removal activities.

**Phase 1 Final Status Surveys in the WMA 1 Excavation**

As explained previously, the final end-state of the Process Building and Vitrification Facility will involve total removal including excavation of the subsurface portions, backfilling with soil, and installing a vertical hydraulic barrier wall on the down-gradient side of the excavation footprint. Phase 1 final status surveys will be performed for exposed subsurface areas before they are backfilled in accordance with the Phase 1 Final Status Survey Plan, which will provide details of the surveys required.

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 in the WMA 1 Excavation**

After Phase 1 final status surveys are completed, arrangements will be made to have any desired confirmatory surveys performed.

**Radiological Status Surveys Outside of the Large Excavation**

After all facilities within WMA 1 have been removed, radiological status surveys of the areas outside of the large excavation will be performed. These areas will consist of the shallow excavations for removal of infrastructure not within the large excavation footprint, that is, the footprints of the portions of the Utility Room, Utility Room Expansion, and the Laundry Room floor slabs and foundations and the floor slabs and foundations for the Fire Pumphouse and Water Storage Tank. These surveys will be performed in accordance with the Characterization Sample and Analysis Plan.

**Confirmatory Surveys in Areas Outside of the Large Excavation**

After these radiological status surveys are completed, arrangements will be made to have any desired confirmatory surveys of these areas performed before they are backfilled.

### 9.7.2 WMA 2 Low-Level Waste Treatment Facility Area

Of the facilities to remain within WMA 2 after 2009 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)[5].

***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**

[5] Although no radioisotope inventory report was issued for the Old Interceptor, these radiation surveys were taken for characterization purposes for the Facility Characterization Project.

Characterization measurements will 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 will typically include exposure rates, removable contamination, total contamination, and core samples of facility surfaces in cases where they will produce information of value. Smears or samples of building materials will 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 decommissioning activities and waste management in cases where such information is not already available.

**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 will be removed, packaged and disposed of at offsite disposal facilities. There is no intent in Phase 1 of the 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 will 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 will be taken after the interior surfaces of the lines are exposed when the lines are cut. Two types of measurements will 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 will provide information to support waste characterization purposes and to support decision-making for Phase 2 of the decommissioning.

**In-Process Surveys of WMA 2 Area**

In-process surveys will be performed during remediation as described in Section 9.5. These surveys will include the surface of the soil in excavations made during removal of the interceptors, the Neutralization Pit, and the associated valve pits.

**In-Process Surveys in the WMA 2 Excavation**

In-process surveys of the completed large excavation will be performed in a manner similar to those for the WMA 1 large excavation described in Section 9.7.1, except that there are no foundation pilings involved. In-process surveys will be performed on all sides of the excavation. The different conditions in WMA 2 will be taken into account, especially the situation where Lagoon 2 and Lagoon 3 extend within the Lavery till and require only limited excavation to reach the point where all of the sediment and at least one foot of the underlying Lavery till has been removed, as specified in Section 7.

**In-Process Surveys Related to Subsurface Piping in WMA 2**

In-process surveys as subsurface piping is encountered during remediation will be performed as specified in Section 9.5.

**Phase 1 Final Status Surveys in WMA 2 Areas**

After decommissioning activities are completed in these areas, Phase 1 final status surveys will be performed in each survey unit in accordance with the Phase 1 Final Status Survey Plan. These surveys will focus on the exposed soil in the large excavation made to remove Lagoons 1-3, the interceptors, the Neutralization Pit, and Solvent Dike.

Radiological status surveys will be performed in other areas of interest in accordance with the Characterization Sample and Analysis Plan. These surveys will include 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. Similar surveys will also be performed in the excavation to remove the Maintenance Shop leach field equipment and in the areas where Lagoons 4 and 5 were removed.

**Confirmatory Surveys in WMA 2 Areas**

After the Phase 1 final status surveys are completed, arrangements will be made to have confirmatory surveys performed. NRC or its contractor will be afforded an opportunity to perform confirmatory surveys in excavations before they are filled in.

**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 will 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 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 will be performed in connection with decommissioning activities.

***Description of Planned Survey Measurements***. Measurements will 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 decommissioning activities and waste management.

**WMA 3 Facility In-Process Surveys**

In-process surveys will be performed during remediation as specified in Section 9.5.

**WMA 3 Facility Radiological Status Surveys**

After decommissioning activities are completed in this area, radiological status surveys will be performed in accordance with the Characterization Sample and Analysis 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 will 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

The primary 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 will be performed in this building commensurate with plans for its disposition, which is removal.

***Description of Planned Survey Measurements.*** Measurements will typically include exposure rates, removable contamination, and total contamination. Representative smears will be analyzed for radionuclides of interest.

***Justification for Planned Survey Measurements***. These are the appropriate measurements necessary to facilitate radiation protection and support decommissioning activities and waste management.

**Characterization of Lag Storage Addition 4/Shipping Depot**

Characterization measurements will be performed in this building commensurate with plans for its disposition, which is removal.

***Description of Planned Survey Measurements.*** Measurements will 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.

**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 will 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 will 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 will 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 will be taken after the interior surfaces of the lines are exposed when the lines are cut. Two types of measurements will 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 will provide information to support for waste characterization purposes and to support decision-making for Phase 2 of the decommissioning.

**In-Process Surveys**

In-process surveys will be performed during remediation of the Remote-Handled Waste Facility and the Lag Storage Addition 4/Shipping Depot as specified in Section 9.5. In-process surveys of subsurface piping will also be performed as specified in Section 9.5 as this piping is encountered during remediation of the Remote-Handled Waste Facility.

**Radiological Status Surveys of the Excavations Where Facilities Are Removed**

As explained previously, the Remote-Handled Waste Facility will be completely removed. After decommissioning activities are completed, including demolition and removal of the floor slab and foundation and removal of the empty underground tank vault, radiological status surveys of the exposed excavation surface will be performed in accordance with the Characterization Sample and Analysis Plan. Similar surveys will be performed in the shallow excavation where the Lag Storage Addition 4/Shipping Depot is removed.

**Confirmatory Surveys of the Excavations Where Facilities Are Removed**

After the radiological status surveys are completed in the areas where the Remote-Handled Waste Facility and the Lag Storage Addition 4/Shipping Depot were removed, arrangements will be made to have any desired confirmatory surveys accomplished by the NRC or its contractor. Arrangements will also be made for any confirmatory surveys NRC desires to be accomplished at the time when the piping ends in the Remote-Handled Waste Facility excavation are accessible, prior to the excavation being filled in.

**Radiological Status and Confirmatory Surveys of Other Floor Slabs and Foundations**

Also considered in the radiological status surveys and confirmatory surveys will 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 Characterization Sample and Analysis Plan are completed, the areas of interest will 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 will be performed in this structure commensurate with plans for its disposition, which is removal.

***Description of Planned Survey Measurements***. Measurements will typically include exposure rates, removable contamination, and total contamination. Representative smears will be analyzed for radionuclides of interest.

***Justification for Planned Survey Measurements***. These are the appropriate measurements necessary to facilitate radiation protection and support decommissioning activities and waste management.

**Radiological 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 will be considered in the radiological status surveys. After surveys specified in the Characterization Sample and Analysis Plan are completed, the areas of interest will be made available to NRC or its contractor for any desired confirmatory surveys.

**Radiological Status Surveys of Equalization Tank Excavation**

Even though the equalization tank was not known to be impacted by radioactivity in mid-2009, as indicated in Section 7, radiological status surveys will be performed in the excavation made to remove the tank as a good practice. These surveys will be performed as specified in Characterization Sample and Analysis Plan and will typically include measurements with a sensitive gamma detector.

After surveys specified in the Characterization Sample and Analysis Plan are completed, the area will 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 will 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 is within the scope of this plan.

**WMA 7 Facility Characterization Surveys**

Characterization measurements of the hardstand will be performed in connection with decommissioning activities.

***Description of Planned Survey Measurements***. Measurements will 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 decommissioning activities and waste management.

**WMA 7 In-Process Surveys**

In-process surveys will be performed during remediation as specified in Section 9.5.

**WMA 7 Radiological Status Surveys**

Surveys of the resulting exposed excavation surfaces will be performed in accordance with the Characterization Sample and Analysis Plan.

**WMA 7 Confirmatory Surveys**

Arrangements will 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 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 will be performed in this building commensurate with plans for its disposition, which is removal. Characterization measurements will also be taken in the trench soil container area and the subcontractor maintenance area.

***Description of Planned Survey Measurements***. Measurements will 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.

**In-Process Surveys Related to the Radwaste Treatment System Drum Cell**

In-process surveys will be performed during removal activities as specified in Section 9.5.

**Radiological Status Surveys of the Radwaste Treatment System Drum Cell**

Following building demolition and removal of the floor slab and foundation, radiological status surveys on the exposed excavation surface will be performed in accordance with the Characterization Sample and Analysis Plan.

**Confirmatory Surveys of the Radwaste Treatment System Drum Cell Excavation**

After the radiological status surveys are completed, arrangements will be made to have any desired confirmatory surveys accomplished.

**The NDA Trench Soil Container Area and the Subcontractor Maintenance Area**

Characterization measurements will be performed in these areas commensurate with plans for their disposition, which is removal.

***Description of Planned Survey Measurements***. Measurements will 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 decommissioning activities and waste management.

Other surveys of this area will include in-process surveys in accordance with Section 9.5, radiological status survey of the excavations in accordance with the Characterization Sample and Analysis 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-2009.

#### WMA 10 Facility Characterization Surveys

Characterization measurements will be performed in these facilities, floor slabs, and foundations in connection with decommissioning activities.

***Description of Planned Survey Measurements***. Measurements will 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 will be performed during remediation as specified in Section 9.5.

#### WMA 10 Facility Radiological Final Status Surveys

Radiological status surveys on the exposed excavation surfaces will be performed in accordance with the Characterization Sample and Analysis Plan.

Radiological status surveys will be performed in the Security Gatehouse as a good practice because of the proximity of this facility to the Process Building. These surveys will 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 will 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. However, characterization surveys are planned for soil and for the banks and streambeds of Erdman Brook and Franks Creek in the portion of WMA 12 that lies within the project premises.

### 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 will be surveyed and sampled for laboratory analysis. However, subsurface soil in the non-source area of the plume and in other Phase 2 areas will not be addressed at this time.

***Description of Survey Measurements***. The process to be utilized will include:

- Consideration of available characterization data;
- Surface scans for gamma activity in areas likely to contain residual contamination;
- Surface and near-surface[6] soil samples, with the samples analyzed for the radionuclides of interest;
- 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 will be paid to the lagoons, basins, and discharge ponds, including the area of Lagoon 1 where previously buried radioactive debris will be removed. Details will appear in the Characterization Sample and Analysis Plan. To facilitate development

[6] Near-surface in this context means a few feet below the surface.

of the Characterization Sample and Analysis Plan, DOE had a set of goals developed for this plan and considered the input of other agencies on these goals as the Characterization Sample and Analysis Plan was prepared.

***Justification for Survey Measurements***. These measurements will 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**

***Description of Survey Measurements***.

Selected surface soil areas will undergo Phase 1 final status surveys, as explained in Section 7. The process to be utilized will be similar to that for characterization surveys, with details included in the Phase 1 Final Status Survey Plan. If grids were established for characterization surveys, the same grids will be reestablished and used where practicable. Characterization data will be considered in the survey design and used for Phase 1 final status survey purposes where practicable.

Also, radiological status surveys will be performed as specified in the Characterization Sample and Analysis Plan in the excavations made to remove the Equalization Basin and the two demineralizer sludge ponds.

***Justification for Survey Measurements***. These measurements will 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 will be made for confirmatory surveys by NRC or its contractor after the Phase 1 final status surveys and radiological status surveys are completed.

**Groundwater**

Radioactivity in groundwater will continue to be monitored during Phase 1 of the 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 decommissioning activities. Limited characterization surveys will be performed for groundwater.

**Surface Water/Stream Sediment**

Radioactivity in surface water and associated stream sediment will continue to be monitored during the decommissioning in connection with the environmental monitoring and control program outlined in Section 1.8 and Appendix D. The characterization program will includes surveys and sampling of the banks and beds of Erdman Brook and the portion of Franks Creek on the project premises, as noted previously.

## 9.8 Phase 1 Final Status Survey Report Requirements

The requirements for the Phase 1 Final Status Survey Report will 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 will be determined using the DQO Process in the project planning cycle. These report requirements will include the following.

#### 9.8.1 Overview of Results

The report will summarize the results of the surveys.

#### 9.8.2 Discussion of Changes

The report will include a discussion of any changes that were made in the Phase 1 final status survey from what was described in this plan or other prior submittals.

#### 9.8.3 Description of How Numbers of Samples Were Determined

The report will 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 will 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 will 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[7] 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[8]; 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

[7] This will include the location of "increment" samples used to form composite samples as described in Appendix G.

[8] This will include application of the as low as reasonably achievable (ALARA) principal as discussed in Section 6.

The report will include a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity.

#### 9.8.7 ALARA Practices

The report will include a description of how ALARA practices were employed to achieve final activity levels.

#### 9.8.8 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 will be included in the report.

#### 9.8.9 Impact of Survey Unit Failures

For any survey units that fail, the report will 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.

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Coleman and Murray 1999, “Detection of Depleted Uranium in Soil Using Portable Hand-Held Instruments,” IAEA-SM-359/P-5. Coleman, R.L. and M.E. Murray, Proceedings of the IAEA Annual Conference, Washington, D,C., November 1999.

CWM 2006, *Site Radiological Survey Plan, Model City, NY,* prepared by CWM Chemical Services, LLC., with assistance from Shaw Environmental, Inc. and URS Corporation, November 2006.

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, 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. Michalczak, L.M., West Valley Nuclear Services Company, West Valley, New York, January 16, 2004.

Michalczak and Hadden-Carter 2009, *Sample and Analysis Plan for the Waste Tank Farm Characterization Project*, WVDP-451, Revision 2 (or later revision). Michalczak, L.M. and P.J. Hadden-Carter, West Valley Environmental Services LLC, West Valley, New York, June 18, 2009.

ORAU 2009, *Ask an Expert Question and Answer Page on Survey Instruments (conventional),* at http://www.orau.gov/ddsc/expert/answers/instruments.htm, accessed on July 23, 2009.

NRC 1997, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, NUREG-1507. U.S. Nuclear Regulatory Commission, Washington, D.C., December 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 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.

USACE 2003, *Final Gamma Walkover Survey Sampling and Analysis Plan, Part 1 – Field Sampling Plan,* prepared for the U.S. Army Corps of Engineers, Buffalo District, by URS Corporation, April 21. 2003.

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 2009, *West Valley Demonstration Project Annual Site Environmental Report, Calendar Year 2008*, WVES and URS Group, Inc., West Valley, New York, September 2009.

# 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 |
| | | App E | Dose Modeling Probabilistic Uncertainty Analysis |
| | | App F | Estimated Radioactivity in Subsurface Piping |
| | | App G | Phase 1 Final Status Survey Conceptual Framework |

The annotated NUREG-1757 decommissioning plan evaluation checklist begins on the next page. Acronyms and abbreviations used in the checklist are as follows:

App = appendix 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-13 |
| ☐ The decommissioning objective proposed by the licensee (i.e., restricted or unrestricted use) | ES | ES-17 |
| ☐ 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<br>Table ES-2 | ES-19<br>ES-20 |
| ☐ A summary of the ALARA evaluations performed to support the decommissioning | ES | ES-21 |
| ~~☐ 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-21 |
| ☐ Any post-remediation activities (such as ground water monitoring) that the licensee proposes to undertake prior to requesting license termination | ES | ES-21 |
| ~~☐ A statement that the licensee is requesting that its license be amended to incorporate the DP~~ | NA | NA |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **1. Introduction**<br>*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 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 will be in effect at the time the decommissioning activities begin, i.e., the interim end state. It explains the relationship between Phase 1 and Phase 2.*<br>*The Introduction also briefly addresses the following matters covered by DOE procedures:*<br>• *Project management,*<br>• *Health and safety,*<br>• *Environmental monitoring and control, and*<br>• *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<br>Table 2-1<br>Table 2-2<br>Table 2-3 | 2-2<br>2-2<br>2-3<br>2-3 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ The chemical forms of the radionuclides authorized and used under all previous licenses | Table 2-1<br>Table 2-2<br>Table 2-6<br>Table 2-7<br>Table 2-8<br>Table 2-9 | 2-2<br>2-3<br>2-11<br>2-12<br>2-13<br>2-17 |
| ☐ A detailed description of how the radionuclides were used at the site | 2.1.1<br>2.1.2 | 2-5<br>2-14 |
| ☐ The location(s) of use and storage of the various radionuclides authorized under all previous licenses | 2.1.1<br>2.1.2 | 2-5<br>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<br>Figure 2-4 | 2-21<br>2-22 |
| **II.c. PREVIOUS DECOMMISSIONING ACTIVITIES** | | |
| ☐ A list or summary of areas at the site that were remediated in the past<br>*Also addresses additional remediation planned to achieve the interim end state.* | 2.2<br>Table 2-11<br>Table 2-13<br>Figure 2-5 | 2-18<br>2-19<br>2-25<br>2-23 |
| ☐ A summary of the types, forms, activities, and concentrations of radionuclides that were present in previously remediated areas | Table 2-11<br>Table 2-13 | 2-19<br>2-25 |
| ☐ The activities that caused the areas to become contaminated | 2.1.1<br>2.1.2 | 2-5<br>2-14 |
| ☐ The procedures used to remediate the areas, and the disposition of radioactive material generated during the remediation | 2.2.1<br>2.2.2 | 2-19<br>2-19 |
| ☐ A summary of the results of the final radiological evaluation of the previously remediated area | Table 2-13<br>2.2.2<br>Table 4-5<br>Table 4-6<br>Table 4-8 | 2-25<br>2-29<br>4-16<br>4-17<br>4-19 |
| ☐ A scale drawing or map of the site, facilities, and environs showing the locations of previous remedial activity | Figure 2-5 | 2-22 |
| **II.d. SPILLS**<br>*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 | 2-32 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ The types, forms, activities, and concentrations of radionuclides involved in the spill or uncontrolled release | Table 2-16<br>Table 2-17<br>Table 2-18 | 2-34<br>2-38<br>2-41 |
| ☐ A scale drawing or map of the site, facilities, and environs showing the locations of spills<br>*The locations of major spills are shown in the figures 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<br>Figure 2-4<br>Figure 2-6<br>Figure 2-7 | 2-21<br>2-22<br>2-33<br>2-37 |
| **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<br>Table 2-20<br>Table 2-21 | 2-43<br>2-44<br>2-45 |
| ☐ A scale drawing or map of the site, facilities, and environs showing the locations of former burials | Figure 2-3<br>Figure 2-4 | 2-21<br>2-22 |
| **III. FACILITY DESCRIPTION**<br>*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<br>3.2.2 | 3-2<br>3-32 |
| ☐ A description of the contours and features of the site | 3.1.2<br>Figure 3-3<br>Figure 3-4 | 3-2<br>3-95<br>3-96 |
| ☐ 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<br>3.2.1 | 3-27<br>3-29 |
| ☐ The location of the site relative to prominent features such as rivers and lakes | Figure 3-1<br>Figure 3-2 | 3-93<br>3-94 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ A map that shows the detailed topography of the site using a contour interval | Figure 3-3<br>Figure 3-4 | 3-95<br>3-96 |
| ☐ The location of the nearest residences and all significant facilities or activities near the site | 3.1.4 | 3-27 |
| ☐ 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<br>Figure 3-44 | 3-29<br>3-130 |
| ☐ A summary of the projected population in and around the site by compass vectors *[Projections not available by compass vector.]* | 3.2.2 | 3-32 |
| **III.c. CURRENT/FUTURE LAND USE** | | |
| ☐ A description of the current land uses in and around the site | 3.3.1<br>Figure 3-45 | 3-35<br>3-131 |
| ☐ A summary of anticipated land uses | 3.3.2 | 3-38 |
| **III.d. METEOROLOGY AND CLIMATOLOGY** | | |
| ☐ A description of the general climate of the region | 3.4.1 | 3-40 |
| ☐ Seasonal and annual frequencies of severe weather phenomena | 3.4.2 | 3-41 |
| ☐ Weather-related radionuclide transmission parameters | 3.4.3 | 3-41 |
| ☐ Routine weather-related site deterioration parameters | 3.4.4 | 3-42 |
| ☐ Extreme weather-related site deterioration parameters | 3.4.4 | 3-42 |
| ☐ A description of the local (site) meteorology | 3.4.5 | 3-42 |
| ☐ 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 | 3-47 |
| **III.e. GEOLOGY AND SEISMOLOGY** | | |
| ☐ A detailed description of the geologic characteristics of the site and the region around the site | 3.5 | 3-47 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ A discussion of the tectonic history of the region, regional geomorphology, physiography, stratigraphy, and geochronology | 3.5 | 3-47 |
| ☐ A regional tectonic map showing the site location and its proximity to tectonic structures | Figure 3-55 | 3-41 |
| ☐ A description of the structural geology of the region and its relationship to the site geologic structure | 3.5 | 3-47 |
| ☐ A description of any crustal tilting, subsidence, karst terrain, landsliding, and erosion | 3.5.3 | 3-52 |
| ☐ A description of the surface and subsurface geologic characteristics of the site and its vicinity | 3.5 | 3-47 |
| ☐ A description of the geomorphology of the site | 3.5.3 | 3-52 |
| ☐ A description of the location, attitude, and geometry of all known or inferred faults in the site and vicinity | 3.5.4 | 3-55 |
| ☐ A discussion of the nature and rates of deformation | 3.5.3 | 3-52 |
| ☐ 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 | 3-61 |
| ☐ 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<br>Table 3-15 | 3-61<br>3-61 |
| **III.f. SURFACE WATER HYDROLOGY** | | |
| ☐ A description of site drainage and surrounding watershed fluvial features | 3.6.1 | 3-65 |
| ☐ 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<br>Figure 3-3 | 3-65<br>3-95 |
| ☐ Topographic maps of the site that show natural drainages and man-made features | Figure 3-3<br>Figure 3-4 | 3-95<br>3-96 |
| ☐ A description of the surface water bodies at the site and surrounding areas | 3.6.1 | 3-65 |
| ☐ 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 | 3-67 |
| ☐ 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<br>Figure 3-4 | 3-95<br>3-96 |
| ☐ 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 | 3-68 |
| ☐ Topographic and/or aerial photographs that delineate the 100-year floodplain at the site | Figure 3-4 | 3-96 |
| ☐ 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 | 3-70 |
| ☐ Descriptions of monitoring wells | 3.7.2<br>4.2.8<br>Figure 4-12<br>Table B-15 | 3-72<br>4-58<br>4-63<br>B-41 |
| ☐ Physical parameters | 3.7.3 | 3-73 |
| ☐ A description of ground water flow directions and velocities | 3.7.1<br>Figure 3-62<br>Figure 3-63<br>Figure 3-64<br>Figure 3-65 | 3-71<br>3-148<br>3-149<br>3-150<br>3-151 |
| ☐ A description of the unsaturated zone | 3.7.4 | 3-73 |
| ☐ Information on all monitor stations including location and depth | Table B-15 | B-41 |
| ☐ A description of physical parameters | 3.7.3 | 3-73 |
| ☐ A description of the numerical analyses techniques used to characterize the unsaturated and saturated zones | 3.7.7 | 3-75 |
| ☐ The distribution coefficients of the radionuclides of interest at the site | 3.7.8<br>Table 3-20 | 3-77<br>3-80 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **III.h. NATURAL RESOURCES** | | |
| □ A description of the natural resources occurring at or near the site | 3.8 | 3-82 |
| □ A description of potable, agricultural, or industrial ground or surface waters | 3.8.3 | 3-84 |
| □ A description of economic, marginally economic, or subeconomic known or identified natural resources as defined in U.S. Geological Survey Circular 831 | 3.8 | 3-82 |
| □ 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 will be performed in connection with the decommissioning activities as specified in Section 9.*

| CONTENT | SECTION | PAGE |
|---|---|---|
| **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<br>Figure 4-1<br>Figure 4-2<br>Figure 4-3<br>Figure 4-4<br>Figure 4-5 | 4-5<br>4-7<br>4-8<br>4-9<br>4-10<br>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 cm², 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**<br>*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<br>Figure 4-6 | 4-29<br>4-32 |
| ☐ A summary of the background levels used during scoping or characterization surveys | 4.2.2<br>Table 4-11<br>Figure B-1<br>Table B-1 | 4-26<br>4-27<br>B-3<br>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<br>4.2.5 | 4-29<br>4-36 |
| ☐ The maximum and average radiation levels in mrem/hr at each location<br>*[Data are not available at sample locations.]* | 4.2.6 | 4-49 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ A scale drawing or map of the site showing the locations of radionuclide material contamination in surface soil | Figure 4-6 | 4-32 |
| **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<br>Figure 4-7<br>Figure 4-8 | 4-31<br>4-33<br>4-35 |
| ☐ A summary of the background levels used during scoping or characterization surveys | 4.2.2 | 4-26 |
| ☐ 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<br>4.2.5 | 4-31<br>4-36 |
| ☐ The depth of the subsurface soil contamination at each location | Figure 4-8<br>4.2.5 | 4-35<br>4-36 |
| ☐ A scale drawing or map of the site showing the locations of subsurface soil contamination | Figure 4-7<br>Figure 4-8 | 4-33<br>4-35 |
| **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<br>Figure 4-11 | 4-55<br>4-56 |
| ☐ A summary of the background levels used during scoping or characterization surveys | Table 4-11 | 4-27 |
| ☐ 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 | 4-57 |
| **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 | 4-58 |
| ☐ A summary of the background levels used during scoping or characterization surveys | Table 4-11 | 4-27 |

| 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 | 4-59 |
| **V. DOSE MODELING** | | |
| **V.a. UNRESTRICTED RELEASE USING SCREENING CRITERIA**<br>*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**<br>*Although no remediated areas will 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<br>5.2.2<br>5.2.3<br>5.2.8<br>Figure 5-7<br>Figure 5-8<br>Figure 5-9<br>Figure 5-10<br>Figure 5-13 | 5-21<br>5-26<br>5-34<br>5-52<br>5-21<br>5-27<br>5-32<br>5-34<br>5-53 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ 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<br>5.2.2<br>5.2.3<br>5.2.8<br>Figure 5-7<br>Figure 5-8<br>Figure 5-9<br>Figure 5-10<br>Figure 5-13 | 5-21<br>5-26<br>5-34<br>5-52<br>5-21<br>5-27<br>5-32<br>5-34<br>5-53 |
| ☐ Identification/description of the mathematical model used (e.g., hand calculations, DandD Screen v1.0, and RESRAD v5.81) | 5.2.4<br>5.2.8 | 5-38<br>5-55 |
| ☐ Description of the parameters used in the analysis | Table C-1<br>Table C-2<br>Table E-1<br>Table E-2<br>Table E-3<br>Table E-4<br>Table E-5<br>Table E-6 | C-3<br>C12<br>E-10<br>E-11<br>E-12<br>E-13<br>E-14<br>E-15 |
| ☐ Discussion about the effect of uncertainty on the results | 5.2.6 | 5-44 |
| ☐ Input and output files or printouts, if a computer program was used | App C<br>Related CD<br>App E<br>Related CD | C-1<br><br>E-1 |

**V.c. RESTRICTED RELEASE USING SITE-SPECIFIC INFORMATION**

*Although Phase 1 decommissioning activities will not result in a restricted release, this plan provides a limited site-wide integrated dose assessment to help place the Phase 1 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<br>5.2.2<br>5.2.3<br>5.2.8<br>Figure 5-7<br>Figure 5-8<br>Figure 5-9<br>Figure 5-10<br>Figure 5-13 | 5-21<br>5-26<br>5-34<br>5-52<br>5-21<br>5-27<br>5-32<br>5-34<br>5-53 |

| 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 | 5.2.1<br>5.2.2<br>5.2.3<br>5.2.8<br>Figure 5-7<br>Figure 5-8<br>Figure 5-9<br>Figure 5-10<br>Figure 5-13 | 5-21<br>5-26<br>5-34<br>5-52<br>5-21<br>5-27<br>5-32<br>5-34<br>5-53 |
| ☐ Identification/description of the mathematical model(s) used (e.g., hand calculations and RESRAD v5.81) | 5.2.4<br>5.2.8 | 5-38<br>5-55 |
| ☐ A summary of parameters used in the analysis | Table C-1<br>Table C-2<br>Table E-1<br>Table E-2<br>Table E-3<br>Table E-4<br>Table E-5<br>Table E-6 | C-3<br>C12<br>E-10<br>E-11<br>E-12<br>E-13<br>E-14<br>E-15 |
| ☐ A discussion about the effect of uncertainty on the results | 5.2.6 | 5-44 |
| ☐ Input and output files or printouts, if a computer program was used | App C<br>Related CD<br>App E<br>Related CD | C-1<br><br>E-1 |
| **V.d. RELEASE INVOLVING ALTERNATE CRITERIA**<br>*DOE will 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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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**<br>*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 | 6-3 |
| ☐ A quantitative cost benefit analysis | 6.3<br>6.4 | 6-6<br>6-12 |
| ☐ A description of how costs were estimated | 6.3.2 | 6-8 |
| ☐ A demonstration that the doses to the average member of the critical group are ALARA | 6.3<br>6.4 | 6-8<br>6-12 |
| **VIII. PLANNED DECOMMISSIONING ACTIVITIES**<br>*The remediation tasks are described in general terms. Every room and area is not addressed since decontamination will be limited and the facilities will be demolished. Typical remediation techniques to be used are described in Section 7.12, starting on page 7-48. More detail will be provided later in the Decommissioning Work Plan(s). Measures for preventing contamination or recontamination of the site due to decommissioning activities are addressed in Section 7.2.2 on page 7-6.*<br>**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 | 7-16 to 7-29 |
| ☐ A description of the remediation techniques that will be employed in each room or area of the contaminated structure | 7.12 | 7-47 |
| ☐ ~~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 |

[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 |
|---|---|---|
| ☐ 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 | 7-5 |
| ~~☐ 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 | 7-16 to 7-29 |
| ☐ A description of the techniques that will be employed to remediate each system in the facility or site | 7-12 | 7-47 |
| ~~☐ 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<br>7.4.2<br>7.5 | 7-16<br>7-31<br>7-38 |
| ~~☐ 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 |
| ~~☐ 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<br>7.4.3<br>7.7.4 | 7-21<br>7-32<br>7-43 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ A description the techniques that will be employed to remove or remediate surface and subsurface soil at the site | 7.3.8<br>7.4.3<br>7.7.4<br>7.12 | 7-21<br>7-32<br>7-43<br>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 will be remediated in Phase 1 of the decommissioning, and groundwater removed from the WMA 1 and WMA 2 excavations will 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<br>7.4.3 | 7-26<br>7-35 |
| ☐ A description of the remediation techniques that will be employed to remediate the ground or surface water | 7.3.8<br>7.4.3 | 7-26<br>7-32 |
| ~~☐ A description of the radiation protection methods and control procedures that will be employed during ground or surface water 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 remediation issues associated with remediating the ground or surface water | 7.2.2 | 7-6 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **VIII.e. SCHEDULES** | | |
| ☐ A Gantt or PERT chart detailing the proposed remediation tasks in the order in which they will occur | Figure 7-16 | 7-56 |
| ☐ A statement acknowledging that the dates in the schedule are contingent upon NRC approval of the DP | 7.13 | 7-55 |
| ☐ 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 | 7.13 | 7-55 |
| ☐ ~~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 |
| **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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| **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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| □ ~~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 |
| □ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| □ ~~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**<br>*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<br>Figure 8-1 | 8-2<br>8-2 |
| □ 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<br>8.1.2 | 8-3<br>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<br>8.1.2 | 8-3<br>8-4 |
| □ An organization chart of the QA program organization | Figure 8-1 | 8-2 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **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<br>8.2.1<br>8.2.2 | 8-3<br>8-5<br>8-6 |
| ☐ A description of the quality-affecting procedural controls of the principal contractors | 8.2.1<br>8.2.2<br>8.2.3<br>8.3.2 | 8-4<br>8-5<br>8-6<br>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<br>8.2.2<br>8.2.3<br>8.3.2 | 8-4<br>8-5<br>8-6<br>8-8 |
| ☐ 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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ 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<br>8.1.2 | 8-3<br>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<br>9.4.3 | 8-12<br>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 |
| **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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ 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**<br>*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 5-61 – 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 | 5-62 |
| 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<br>Table 9-2<br>Table 9-3 | 9-3<br>9-3<br>9-4 |
| ☐ If Class 1 survey units are present, the $DCGL_{EMC}$ values for each radionuclide and medium of concern | Table 5-14 | 5-62 |
| ☐ If multiple radionuclides are present, the appropriate $DCGL_W$ for the survey method to be used *[A $DCGL_W$ for a surrogate radionuclide will be developed if practicable after additional characterization data are obtain during Phase 1 decommissioning activities.]* | NA | NA |
| **XIV.b. CHARACTERIZATION SURVEYS** | | |
| ☐ A description and justification of the survey measurements for impacted media | 9.2.4<br>9.4<br>9.7 | 9-6<br>9-8<br>9-30 |
| ☐ A description of the field instruments and methods that were used for measuring concentrations and the sensitivities of those instruments and methods | 9.4<br>Table 9-4 | 9-11<br>9-11 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ 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<br>9.4.3<br>Table 9-5 | 9-11<br>9-15<br>9-12 |
| ☐ 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 decommissioning will present data in tables and figures similar to those in Section 2 and Section 4.]* | Table 2-10<br>Table 2-19<br>Table 4-3<br>Table 4-4<br>Table 4-5<br>Table 4-6<br>Table 4-8<br>Table 4-9 | 2-18<br>2-43<br>4-15<br>4-16<br>4-16<br>4-17<br>4-19<br>4-21 |
| ☐ Maps or drawings of the site, area, or building, showing areas classified as non-impacted or impacted *[The drawings provided in Section 4 will be confirmed or revised when additional characterization data become available early in Phase 1 of the decommissioning.]* | Figure 4-1<br>Figure 4-2<br>Figure 4-3<br>Figure 4-4<br>Figure 4-5 | 4-7<br>4-8<br>4-9<br>4-10<br>4-11 |
| ☐ Justification for considering areas to be non-impacted *[The justification provided in Section 4 will be confirmed or revised when additional characterization data become available early in Phase 1 of the 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 | 9-30 |
| ☐ 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 | 9.7.1 | 9-32 |
| ☐ 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 | 9-9 |
| **XIV.c. IN-PROCESS SURVEYS** | | |
| ☐ A description of field screening methods and instrumentation | 9.5 | 9-20 |
| ☐ 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 will be similar to those used during characterization and final status surveys. The field instruments suitable for scanning soil will not be able to detect non-gamma emitting radionuclides.]* | 9.5<br>Table 9-7 | 9-20<br>9-21 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| **XIV.d. FINAL STATUS SURVEY DESIGN** | | |
| *Phase 1 final status surveys will be performed in cases where the decommissioning activities will make an area inaccessible for later final status surveys and confirmatory surveys. These surveys will be managed as final status surveys although a potential for recontamination may exist in certain areas. Details will be provided in the Phase 1 Final Status Survey Plan. Appendix G describes the conceptual framework for the Phase 1 Final Status Survey Plan.* | | |
| ☐ A brief overview describing the final status survey design | 9.6.1 | 9-24 |
| ☐ 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 will be specified in the Final Status Survey Plan as indicated in Section 9.6.1 on page 9-17.]* | 9.6.1 | 9-24 |
| ☐ A description of the background reference areas and materials, if they will be used, and a justification for their selection *[Details will appear in the Final Status Survey Plan.]* | 9.6.1 | 9-25 |
| ☐ A summary of the statistical tests that will be used to evaluate the survey results *[Details will appear in the Final Status Survey Plan.]* | 9.3<br>9.6.1 | 9-8<br>9-28 |
| ☐ A description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide | Table 9-8<br>9.6.1 | 9-26<br>9-26 |
| ☐ 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 9-8<br>9.6.1 | 9-26<br>9-26 |
| ☐ 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<br>Table 9-5 | 9-26<br>9-12 |
| ☐ A description of how the samples to be analyzed in the laboratory will be collected, controlled, and handled | 9.6.1 | 9-27 |
| ☐ A description of the final status survey investigation levels and how they were determined | Appen G | G-9 |
| ☐ A summary of any significant additional residual radioactivity that was not accounted for during site characterization | 9.6.1 | 9-24 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ 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 | 9-28 |
| ☐ 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 | 9-28 |
| **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 | 9-45 |
| ☐ 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 | 9-45 |
| ☐ A description of the method by which the number of samples was determined for each survey unit | 9.8.3 | 9-46 |
| ☐ A summary of the values used to determine the number of samples and a justification for these values | 9.8.4 | 9-46 |
| ☐ The survey results for each survey unit include: | 9.8.5 | 9-46 |
| — The number of samples taken for the survey unit; | 9.8.5 | 9-46 |
| — 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 | 9-46 |
| — The measured sample concentrations in units that are comparable to the DCGL; | 9.8.5 | 9-46 |
| — The statistical evaluation of the measured concentrations; | 9.8.5 | 9-46 |
| — Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation; | 9.8.5 | 9-46 |
| — 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 | 9-46 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| — 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 | 9-46 |
| ☐ 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 | 9-46 |
| ☐ A description of how ALARA practices were employed to achieve final activity levels | 9.8.5 | 9-46 |
| ☐ 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 | 9-46 |
| ☐ 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 | 9-47 |
| **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 |
| **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:~~<br>— ~~Trust fund;~~<br>— ~~Escrow account;~~<br>— ~~Government fund;~~<br>— ~~Certificate of deposit;~~<br>— ~~Deposit of government securities;~~<br>— ~~Surety bond;~~ | NA | NA |

| CONTENT | SECTION | PAGE |
|---|---|---|
| — ~~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.~~ | | |
| **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 |
| □ ~~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**<br>*Because there will be no facility or property release associated with the Phase 1 of the 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**<br>*DOE will continue to manage the project premises and provide for monitoring and maintenance until the actions required by the WVDP Act have been completed. DOE's site management plan for the post-Phase 1 period will provide de facto institutional control of the site during this period. Accordingly, DOE will briefly describe this plan, addressing the topics identified as applicable* | | |

| CONTENT | SECTION | PAGE |
|---|---|---|
| *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-32 |
| ☐ A description of the design features of the site that support institutional controls | App D | D-32 |
| ☐ A discussion of the durability of the institutional control(s), including the performance of any engineered barriers used | App D | D-8 |
| ☐ ~~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 |
| ☐ ~~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-18 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| ☐ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| □ ~~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 |
| □ ~~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 |

| CONTENT | SECTION | PAGE |
|---|---|---|
| ☐ ~~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 |
| ☐ ~~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 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. (In 2008, the frequency was reduced further to once every five years, and sampling at most locations was discontinued.) 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.)

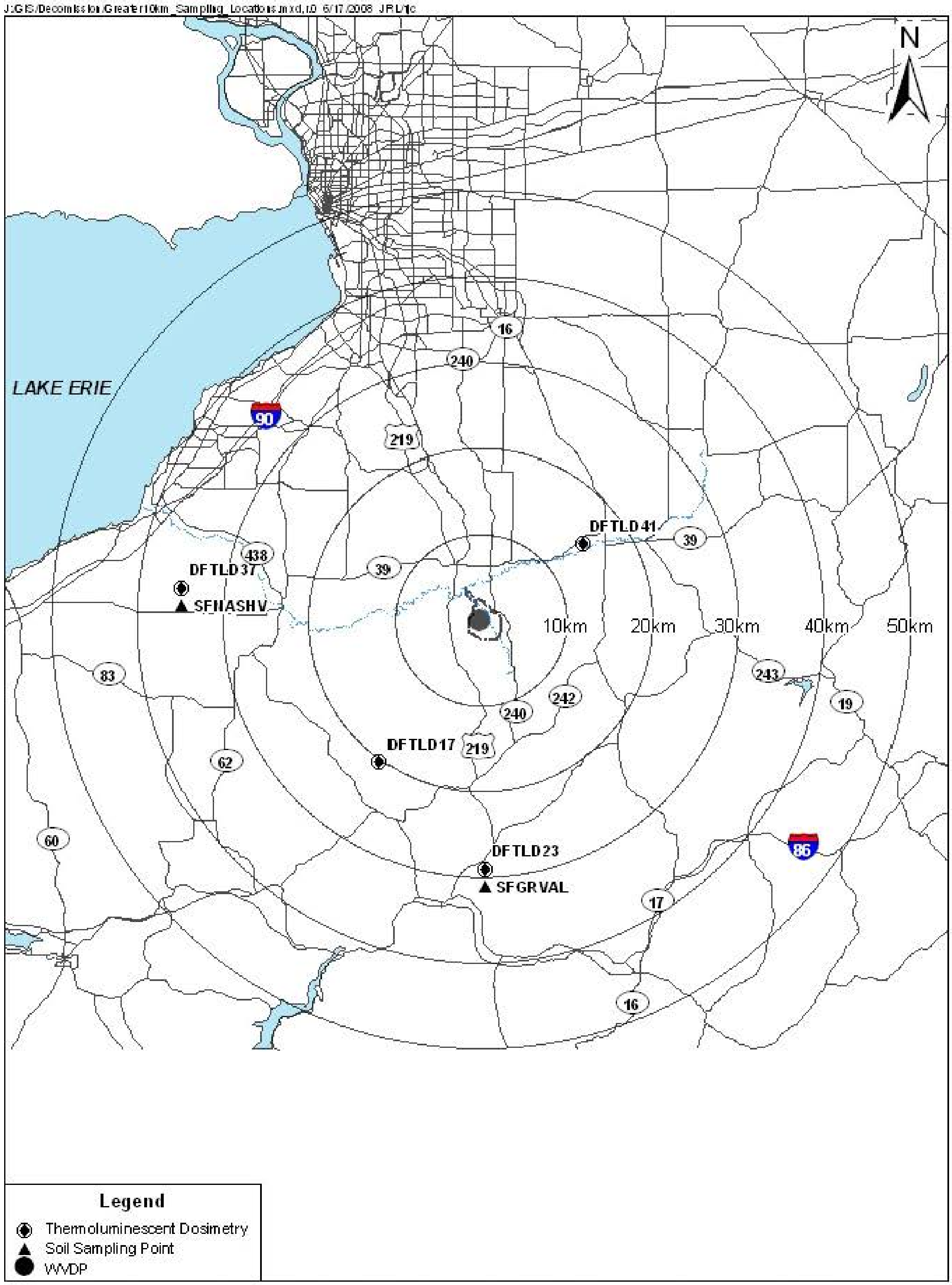


**Figure B-1. Background Sampling Locations More Than 10 Kilometers From the WVDP**

**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 alpha-emitting radionuclides – 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 when Revision 0 to this plan was prepared. 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 initially as reference values because there were too few points. Instead, surface soil background results were used to evaluate the presence of radionuclide concentrations in excess of background in subsurface soil samples.

In 2008, subsurface soil background locations in the sand and gravel and unweathered Lavery till geological units underlying the site were sampled as part of the North Plateau Characterization Program (Michalczak 2007, Klenk 2008). Results from the sand and gravel and unweathered Lavery till samples were statistically indistinguishable, so all were combined, together with the 1993 results, to produce a subsurface soil background for the site.

### 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.)

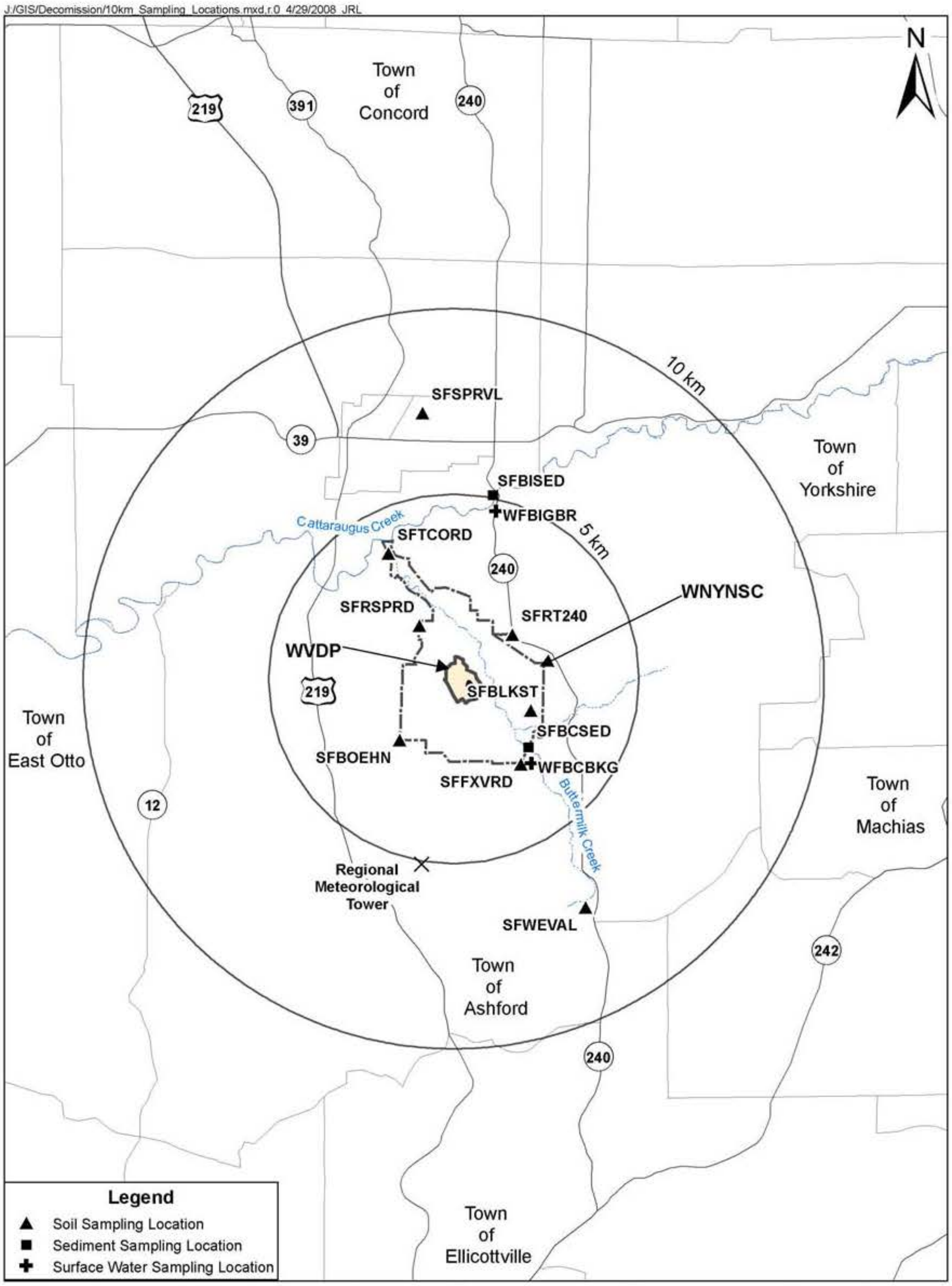


**Figure B-2. Sampling Locations Within 10 Kilometers of the WVDP Used for Background Calculations**

### 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.

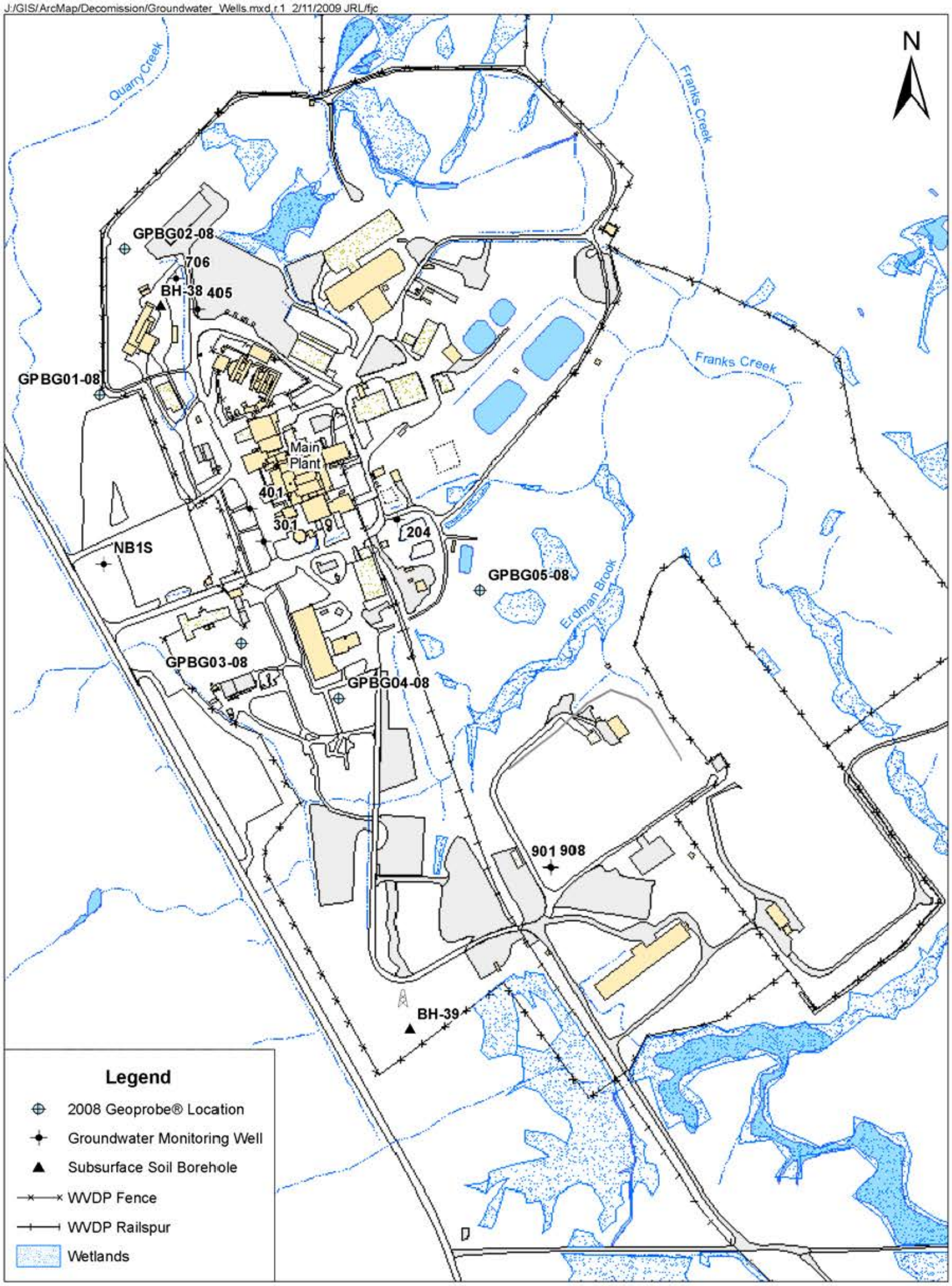


**Figure B-3. Onsite Groundwater and Subsurface Soil Locations Used as Background**

## 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:<br>uncertainty = SQRT((uncertainty$_1^2$ + … + uncertainty$_N^2$) / N)<br>where uncertainty$_1$ = the uncertainty term from sample 1<br>uncertainty$_N$ = the uncertainty term from sample N<br>N = the total number of samples<br>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**

| Constituent | Average concentration (pCi/g) Result | ± | Uncertainty | Median (pCi/g) | Maximum (pCi/g) | N | % NDs | Years | Data Source Locations[1] |
|---|---|---|---|---|---|---|---|---|---|
| Gross alpha | 1.20E+01 | ± | 4.76E+00 | 1.26E+01 | 1.69E+01 | 18 | 0% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| Gross beta | 3.19E+01 | ± | 3.99E+00 | 2.86E+01 | 6.10E+01 | 18 | 0% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| Sr-90 | 1.80E-02 | ± | 2.59E-02 | < 2.30E-02 | 1.24E-01 | 18 | 89% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| Cs-137 | 4.51E-03 | ± | 2.43E-02 | < 2.41E-02 | 1.49E-01 | 18 | 94% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| U-232 | -2.65E-03 | ± | 2.55E-02 | < 2.44E-02 | < 4.19E-02 | 18 | 100% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| U-233/234 | 6.83E-01 | ± | 1.19E-01 | 7.91E-01 | 1.08E+00 | 18 | 0% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| U-235/236 | 5.14E-02 | ± | 3.47E-02 | 4.25E-02 | 1.17E-01 | 18 | 33% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| U-238 | 7.19E-01 | ± | 1.22E-01 | 8.64E-01 | 1.11E+00 | 18 | 0% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| Pu-238 | 4.32E-04 | ± | 1.30E-02 | < 1.15E-02 | < 2.41E-02 | 18 | 100% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| Pu-239/240 | 1.72E-03 | ± | 1.19E-02 | < 1.04E-02 | < 1.87E-02 | 18 | 100% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |
| Am-241 | -1.93E-03 | ± | 1.07E-02 | < 1.09E-02 | < 1.27E-02 | 18 | 100% | 1993, 2008 | BH-38 and 39 (1993); GPBG01-08, GPBG02-08, GPBG03-08, GPBG04-08, and GPBG05-08 (2008) |

LEGEND: N = Number of samples
ND = Nondetect

NOTE: (1) Background locations are shown on Figure B-3. After testing to ensure that subsurface soil results for the sand and gravel unit and the unweathered Lavery till were statistically indistinguishable, values were combined into a single subsurface soil background value for each radionuclide.

**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) | | | Median | Maximum | N | Years | Data Source Locations[1] |
|---|---|---|---|---|---|---|---|---|
| | Result | ± | Uncertainty | | | | | |
| 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; Geoprobe® studies done in 1994, 1997, 1998, and 2008); 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. 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 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

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**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 | 45 | 0.31 | 303 | 63,419 | 5.0E-02 | Inside MPPB (GP7898, 21-23' depth) |
| | Tc-99 | 6 | 0.0027 | 2.3 | 5.6 | 1.1E-01 | Outside MPPB, south of FRS (GP7208, 14-16' depth) |
| | U-232 | 1 | 0.023 | 0.023 | 0.023 | 2.0E+00 | Outside southeast corner of MPPB (GP2908, 14-16' depth) |
| | U-233/234 | 9 | 0.0074 | 0.79 | 12 | 7.2E-02 | Inside MPPB (GP10008, 30-32' depth) |
| | U-235/236 | 5 | 0.013 | 0.063 | 1.1 | 1.4E-01 | Outside eastern wall of MPPB (GP3008, 4-6' depth) |
| | U-238 | 7 | 0.82 | 6.1 | 18 | 7.2E-02 | Outside MPPB, north of FRS (GP10108, 20-22' depth) |
| | Pu-238 | 5 | 0.0025 | 0.019 | 0.18 | 1.5E-01 | Outside MPPB, south of FRS (GP7208, 4-6' depth) |
| | Pu-239/240 | 8 | 0.015 | 0.067 | 0.80 | 5.5E-02 | East of laundry building (BH-18, 14-16' depth) |
| | Am-241 | 16 | 0.025 | 0.19 | 2.7 | 3.6E-02 | Inside MPPB (GP77, 19-23' depth) |
| | Cm-243/244 | 1 | 0.015 | 0.015 | 0.015 | 1.0E+01 | Inside MPPB (GP8008, 25-27' depth) |
| WMA 2 | Sr-90 | 27 | 0.037 | 1.9 | 750 | 4.8E-02 | Northwest of Lagoon 1 (BH-09, 10-12' depth) |
| | U-232 | 11 | 0.0050 | 0.021 | 1.0 | 4.8E-02 | Northwest of Lagoon 1 (BH-09, 10-12' depth) |
| | U-233/234 | 8 | 0.0046 | 1.9 | 7.0 | 2.7E-01 | Solvent dike (BH-11, 10-12' depth) |
| | U-235/236 | 7 | 0.000038 | 0.55 | 1.1 | 2.7E-01 | Solvent dike (BH-11, 10-12' depth) |
| | U-238 | 7 | 0.00052 | 0.052 | 4.4 | 2.7E-01 | Solvent dike (BH-11, 10-12' depth) |
| | Pu-238 | 15 | 0.0049 | 0.023 | 0.089 | 1.9E+00 | Between Interceptors and Lagoon 1 (BH-14, 14-16' depth) |
| | Pu-239/240 | 15 | 0.0046 | 0.031 | 0.11 | 1.6E-01 | Maintenance Shop Leach Field (BH-35, 18-20' depth) |
| | Pu-241 | 7 | 0.030 | 0.11 | 0.21 | 1.6E+01 | East of Test and Storage Building (BH-35, 6-8' depth) |
| | Am-241 | 18 | 0.010 | 0.051 | 0.23 | 2.7E-01 | Solvent dike (BH-11, 10-12' depth) |
| WMA 4 | Sr-90 | 2 | 0.73 | 0.75 | 0.77 | 8.8E-02 | Southeast corner of CDDL (BH-28, 6-8' depth) |
| WMA 5 | Sr-90 | 1 | 6.3 | 6.3 | 6.3 | 4.8E-02 | Between LSA 3 and LSA 4 (BH-30, 10-12' depth) |
| WMA 6 | Sr-90 | 5 | 1.1 | 174 | 1115 | 1.3E-01 | Downgradient of MPPB (GP10208, 16-18' depth) |
| | U-232 | 1 | 0.087 | 0.087 | 0.087 | 1.1E+00 | Downgradient of MPPB (GP10208, 14-16' depth) |

**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 |
|---|---|---|---|---|---|---|---|
| | U-233/234 | 2 | 1.2 | 4.6 | 8.0 | 1.3E-01 | Downgradient of MPPB (GP10208, 16-18' depth) |
| | U-235/236 | 2 | 0.33 | 0.82 | 1.3 | 1.3E-01 | Downgradient of MPPB (GP10208, 16-18' depth) |
| | U-238 | 2 | 1.3 | 5.2 | 9.0 | 1.3E-01 | Downgradient of MPPB (GP10208, 16-18' depth) |
| | Pu-238 | 2 | 0.025 | 0.030 | 0.035 | 4.3E+00 | Southeast of FRS (BH-19A, 12-14' depth) |
| | Pu-239/240 | 3 | 0.040 | 0.047 | 0.047 | 1.1E+00 | Downgradient of MPPB (GP10208, 14-16' depth) |
| | Pu-241 | 1 | 0.35 | 0.35 | 0.35 | 4.3E+00 | Southeast of FRS (BH-19A, 12-14' depth) |
| | Am-241 | 4 | 0.13 | 0.20 | 0.33 | 1.3E-01 | Downgradient of MPPB (GP10208, 16-18' depth) |
| WMA 7 | Sr-90 | 1 | 2.6 | 2.6 | 2.6 | 5.4E-02 | Northern corner of NDA (BH-42, 25-27' depth) |
| WMA 12 | Sr-90 | 1 | 1.5 | 1.5 | 1.5 | 4.4E-02 | Northwest of the NDA (BH-24, 6-8' depth) |

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 the nearest integer.

(2) No subsurface soil data were available for WMAs 3 and 9. No Cs-137 results exceeding background were found in WMA 10.

(3) Cs-137 concentration at the location with the maximum ratio.

LEGEND: N = Number of Samples; MPPB = Main Plant Process Building; FRS = Fuel Receiving and Storage; CDDL = Construction and Demolition Debris Landfill; LSA = Lag Storage Area; NDA = Nuclear Regulatory Commission Licensed Disposal Area

**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 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) Result | ± | Uncertainty | Maximum (pCi/g) | Exceeded Background?(1) |
|---|---|---|---|---|---|---|---|---|---|
| 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) Result | ± | Uncertainty | Maximum (pCi/L) | Exceeded Background?(1) |
|---|---|---|---|---|---|---|---|---|---|
| WMA 2 | WNSP001 | Gross alpha | 232 | 1.75E+01 | 1.92E+01 | ± | 1.32E+01 | 1.01E+02 | Yes |
| | Lagoon 3 | Gross beta | 433 | 2.56E+02 | 3.01E+02 | ± | 2.25E+01 | 8.18E+02 | Yes |
| | Discharge | H-3 | 231 | 2.47E+03 | 2.75E+03 | ± | 1.42E+02 | 7.17E+03 | Yes |
| | Weir | 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 | Gross alpha | 450 | <1.87E+00 | 2.86E-01 | ± | 2.28E+00 | 7.25E+00 | No |
| | Northeast | Gross beta | 451 | 3.01E+03 | 3.24E+03 | ± | 5.33E+01 | 9.98E+03 | Yes |
| | Swamp | H-3 | 451 | 1.13E+02 | 1.13E+02 | ± | 8.21E+01 | 5.20E+02 | Yes |
| | Drainage | 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 | Gross alpha | 450 | <2.17E+00 | 3.88E-02 | ± | 3.09E+00 | 7.89E+00 | No |
| | North | Gross beta | 450 | 1.17E+01 | 1.21E+01 | ± | 4.34E+00 | 4.24E+01 | Yes |
| | Swamp | H-3 | 450 | <8.18E+01 | -2.14E+00 | ± | 8.07E+01 | 2.80E+02 | No |
| | Drainage | 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 | Gross alpha | 324 | <2.62E+00 | 1.37E-01 | ± | 3.32E+00 | 4.80E+00 | No |
| | Sanitary | Gross beta | 324 | 1.45E+01 | 1.53E+01 | ± | 5.02E+00 | 4.05E+01 | Yes |
| | Waste | H-3 | 324 | <8.25E+01 | 2.26E+01 | ± | 8.18E+01 | 1.53E+03 | No |
| | Discharge | 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 | Gross alpha | 73 | <1.91E+00 | 5.65E-01 | ± | 2.03E+00 | 5.81E+00 | No |
| | Cooling | Gross beta | 73 | 6.83E+00 | 9.05E+00 | ± | 3.64E+00 | 3.43E+01 | Yes |
| | Tower Water | 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) Result | ± | Uncertainty | Maximum (pCi/L) | Exceeded Background?(1) |
|---|---|---|---|---|---|---|---|---|---|
| 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) Result | ± | Uncertainty | Maximum (pCi/L) | Exceeded Background?[4] |
|---|---|---|---|---|---|---|---|---|---|
| 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) Result | ± | Uncertainty | Maximum (pCi/L) | Exceeded Background?[4] |
|---|---|---|---|---|---|---|---|---|---|
| WMA 2 | WNW0116 | Gross alpha | 40 | <3.08E+00 | 8.94E-01 | ± | 4.35E+00 | 7.03E+00 | No |
| | S&G | 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 | Gross alpha | 35 | <4.87E+00 | 4.37E-01 | ± | 7.67E+00 | <2.73E+01 | No |
| | S&G | 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 | Gross alpha | 35 | <2.47E+00 | 6.69E-01 | ± | 3.33E+00 | 5.02E+00 | No |
| | LTS | 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 | Gross alpha | 40 | <3.58E+00 | -7.91E+00 | ± | 9.05E+00 | 6.44E+00 | No |
| | S&G | 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) Result | ± | Uncertainty | Maximum (pCi/L) | Exceeded Background?[4] |
|---|---|---|---|---|---|---|---|---|---|
| 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 | Gross alpha | 40 | <4.40E+00 | 7.94E-01 | ± | 8.04E+00 | 1.46E+01 | No |
| | S&G | 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 | Gross alpha | 41 | <5.02E+00 | 3.92E-01 | ± | 7.89E+00 | 9.30E+00 | No |
| | S&G | 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 | Gross alpha | 35 | <4.68E+00 | 1.07E+00 | ± | 7.83E+00 | 9.00E+00 | No |
| | S&G | 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 | Gross alpha | 40 | 9.11E+00 | 8.46E+00 | ± | 7.66E+00 | 2.08E+01 | Yes |
| | S&G | 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 | Gross alpha | 12 | <3.95E-01 | 9.03E-01 | ± | 2.74E+00 | <6.92E+00 | No |
| | S&G | 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 | Gross alpha | 13 | 6.08E+00 | 7.90E+01 | ± | 2.33E+01 | 7.42E+02 | Yes |
| | S&G | 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 | Gross alpha | 40 | <3.10E+00 | -3.75E-01 | ± | 5.55E+00 | 3.84E+00 | No |
| | S&G | 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) Result | ± | Uncertainty | Maximum (pCi/L) | Exceeded Background?[4] |
|---|---|---|---|---|---|---|---|---|---|
| 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) |
| GP2908 | 2008 | 892784.10 | 1129167.91 | 1410.50 | 17-19, 29-31, 35-37 |
| 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) |
| GP3008 | 2008 | 892837.12 | 1129147.27 | 1409.83 | 20-22, 28-30, 35-37 |
| 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 |

**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) |
|---|---|---|---|---|---|
| GP59 | 1994 | 892859.54 | 1129363.33 | 1399.83 | 09-11, 17-19 |
| GP60 | 1994 | 892870.18 | 1129409.83 | 1400.01 | 12-14, 17-19 |
| 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) |
| GP7208 | 2008 | 892871.89 | 1129180.55 | 1410.07 | 20-22, 25-27, 31-33, 38-40 |
| 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 |
| GP7608 | 2008 | 892824.00 | 1129049.00 | 1415.00 | 20-22, 34-36 |
| 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 |
| GP7808 | 2008 | 892843.00 | 1129107.00 | 1410.21 | 20-22, 28-30, 34-36 |
| 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) |
| GP8008 | 2008 | 892812.00 | 1129141.00 | 1415.00 | 25-27, 32-34, 39-41 |
| GP8198 | 1998 | 893048.83 | 1129217.96 | 1403.98 | 15-17, 20-22, 25-27, 30-32, 35-37 |

**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) |
|---|---|---|---|---|---|
| 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 |
| GP8308 | 2008 | 892980.71 | 1129181.86 | 1409.79 | 22-24, 30-32, 38-40 |
| 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 |
| 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 |
| GP10008 | 2008 | 892805.00 | 1129048.00 | 1415.00 | 20-22, 35-37 |
| GP10108 | 2008 | 892924.08 | 1129094.92 | 1410.30 | 21-23, 28-30 |
| GP10208 | 2008 | 892838.12 | 1129224.43 | 1409.11 | 27-29 |
| GP10308 | 2008 | 892977.38 | 1129140.72 | 1410.53 | 21-23, 30-32, 35-37 |
| GP10408 | 2008 | 892953.72 | 1129241.54 | 1405.91 | 21-23 |
| GP10508 | 2008 | 893026.27 | 1129223.72 | 1405.04 | 16-18, 28-30, 34-36 |
| GP10608 | 2008 | 893026.76 | 1129312.67 | 1403.39 | 16-18, 20-22, 28-30 |
| GP10708 | 2008 | 893119.33 | 1129306.52 | 1403.80 | 15-17, 22-24, 30-32 |
| GP10908 | 2008 | 893138.89 | 1129224.21 | 1402.60 | 14-16, 28-30, 34-36 |

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.

# 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 deterministic 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 decommissioning will 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.
- Attachment 2 provides an additional electronic file (a Microsoft Excel spreadsheet) used in the preliminary dose assessments.

## 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 ($K_d$s) 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 and also in the probabilistic uncertainty analysis described in Section 5 and Appendix E.

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. Alternate receptors were considered as discussed in Section 5, including acute dose from subsurface material to a well driller during cistern installation, dose from subsurface material during installation of a natural gas well, and dose from surface and subsurface material to a resident gardener.

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 maximum detection level concentration for non-detected radionuclides. (It should be noted that the minimum detection levels for non-detected radionuclides may range several orders of magnitude. Use of the maximum detection level concentration for non-detected radionuclides results in added conservatism in the reported preliminary dose assessment.) The results were as follow:

WMA 1, a maximum of 1.3 mrem a year

WMA 2, a maximum of 0.04 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-4B in Attachment 2 shows how these doses were estimated.

Note that the probabilistic uncertainty analysis described in Section 5 and Appendix E produced somewhat different results, as did other analyses such as the multi-source analysis for subsurface soil.

**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. Alternative configurations were considered in the sensitivity analysis. |
| | 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 over a 100 $m^2$ area. Alternative configurations were considered in the sensitivity analysis. |
| Length parallel to aquifer flow (m) | 1.00E+02 | 1.65E+02 | SS | Selected to achieve site specific groundwater dilution factor of 0.2, based on DEIS groundwater model correlation. 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 $K_d$s 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. |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| Evapotranspiration coefficient | 5.00E-01 | 7.80E-01 | All | Evapotranspiration and runoff coefficients selected to achieve infiltration rate of 0.26 m/y. |
| 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. |
| | 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 | 4.10E-01 | All | Runoff and evapotranspiration coefficients selected to achieve infiltration rate of 0.26 m/y. |
| 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 $mi^2$ for Buttermilk Creek. |
| 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 and site-specific groundwater dilution factor. 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) |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| 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. |
| 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 | SS, SB | WVNSCO 1993a and WVNSCO 1993c. |
| Unsaturated zone total porosity | 4.00E-01 | 3.60E-01 | SS, SB | WVNSCO 1993c. |
| Unsaturated zone effective porosity | 2.00E-01 | 2.50E-01 | SS, SB | WVNSCO 1993c. |
| Unsaturated zone field capacity | 2.00E-01 | 2.00E-01 | SS, SB | WVNSCO 1993c. |
| Unsaturated zone hydraulic conductivity (m/y) | 1.00E+01 | 1.40E+02 | SS, SB | 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 | SS, SB | 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. |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| 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 | All | Beyeler, et al. 1999. |
| Mass loading for inhalation ($g/m^3$) | 1.00E-04 | 1.48E-05 | All | Beyeler, et al. 1999. Based on relative time fractions and mean dust loadings. Assumes 288 hours of active farming per year. |
| Exposure duration (y) | 3.00E+01 | 1.00E+00 | All | Yearly dose estimates calculated. |
| Filtration factor, inhalation | 4.00E-01 | 1.00E+00 | SS, SB | Beyeler, et. al. 1999. |
| 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 | 1.12E+02 | SS, SB | Beyeler, et al. 1999. |
| Leafy vegetable consumption (kg/y) | 1.40E+01 | 2.10E+01 | SS, SB | Beyeler, et al. 1999. |
| Milk consumption (L/y) | 9.20E+01 | 2.33E+02 | SS, SB | Beyeler, et al. 1999. |
| 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. |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| | 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. |
| 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 | Assumes all ingestion is from the contaminated source. |
| Contamination fraction of meat | -1 | 1.0 | All | Assumes all ingestion is from the contaminated source. |
| Contamination fraction of milk | -1 | 1.0 | SS, SB | Assumes all ingestion is from the contaminated source. |
| 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 | All | 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. |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| 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. |
| 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. |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| 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. |
| 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. |

**Table C-1. RESRAD Input Parameters**

| RESRAD Parameter (Units) | Default | Value | Medium | Comment/Reference |
|---|---|---|---|---|
| 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. |
| 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[(1)]**

| 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 | 6760 (780 – 22,970) | 6760 (780 – 22,970) | 6760 (780 – 22,970) | 6760 (780 – 22,970) | 6760 (780 – 22,970) |
| Cesium | 4600 | 280[(4)] (48 - 4800) | 480[(5)] (48 - 4800) | 480[(5)] (48 - 4800) | 280[(4)] (48 - 4800) | 280[(4)] (48 - 4800) |
| Iodine | 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 | 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) |
| Strontium | 30 | 5[(9)] (1 - 32) | 15[(5)] (1 - 32) | 15[(5)] (1 - 32) | 5[(9)] (1 - 32) | 5[(9)] (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)] (10 - 350) | 10[(9)] (1 - 100) | 10[(9)] (1 - 100) | 35[(4)] (10 - 350) | 35[(4)] (10 - 350) |
| Progeny Elements[(10)] | | | | | | |
| Actinium | 20 | 1740 | 1740 | 1740 | 1740 | 1740 |
| Lead | 100 | 2400 | 2400 | 2400 | 2400 | 2400 |

**Table C-2. Soil/Water Distribution Coefficients[(1)]**

| 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) |
|---|---|---|---|---|---|---|
| Protactinium | 50 | 2040 | 2040 | 2040` | 2040 | 2040 |
| Radium | 70 | 3550 | 3550 | 3550 | 3550 | 3550 |
| Thorium | 60,000 | 5890 | 5890 | 5890 | 5890 | 5890 |

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; Sheppard, et. al. 2006, 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.

(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) Beyeler, et. al. 1999

(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) Site specific data (Dames and Moore 1995a, 1995b). The Sr-90 value of 5 mL/g is consistent with the value used in the Decommissioning EIS.

(10) Progeny $K_d$s were not included in the sensitivity analysis; DEIS 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 Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas[1]**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| BH-17 (WMA 6, 1993)<br>Depth to Lavery till - 27 ft | 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)<br>Depth to Lavery till - 37.5 ft | 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 |
| | 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 |
| GP3098 (WMA 1, 1998)<br>Depth to Lavery till - 37 ft | 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 |
| GP3008 (WMA 1, 2008)<br>Depth to Lavery till - 37 ft | C-14 | < 3.0E-01 | 37-39 |
| | Sr-90 | 1.7E+00 | 37-39 |
| | Tc-99 | < 5.5E-01 | 37-39 |
| | I-129 | < 1.1E-01 | 37-39 |
| | Cs-137 | < 2.0E-02 | 37-39 |

**Table C-4. Radiological Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas(1)**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| | U-232 | < 2.2E-02 | 37-39 |
| | U-233/234 | 9.7E-01 | 37-39 |
| | U-235/236 | 1.3E-01 | 37-39 |
| | U-238 | 1.1E+00 | 37-39 |
| | Np-237 | < 9.8E-03 | 37-39 |
| | Pu-238 | < 1.1E-02 | 37-39 |
| | Pu-239/240 | < 1.2E-02 | 37-39 |
| | Pu-241 | < 4.8E-01 | 37-39 |
| | Am-241 | < 1.2E-02 | 37-39 |
| | Cm-243/244 | < 1.2E-02 | 37-39 |
| GP7398 (WMA 1, 1998) Depth to Lavery till - 39 ft | 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 |
| GP7608 (WMA 1, 2008) Depth to Lavery till - 38 ft | C-14 | 1.1E-01 | 38-40 |
| | Sr-90 | 1.5E+01 | 38-40 |
| | Tc-99 | < 2.7E-01 | 38-40 |
| | I-129 | < 2.9E-01 | 38-40 |
| | Cs-137 | 3.9E+00 | 38-40 |
| | U-232 | < 2.7E-02 | 38-40 |
| | U-233/234 | 2.3E+00 | 38-40 |
| | U-235/236 | 1.0E-01 | 38-40 |
| | U-238 | 8.1E-01 | 38-40 |
| | Np-237 | < 1.6E-02 | 38-40 |
| | Pu-238 | < 2.3E-02 | 38-40 |
| | Pu-239/240 | 6.4E-02 | 38-40 |
| | Pu-241 | < 5.7E-01 | 38-40 |
| | Am-241 | 1.3E-01 | 38-40 |
| | Cm-243/244 | < 2.3E-02 | 38-40 |
| GP7808 (WMA 1, 2008) Depth to Lavery till - 37 ft | C-14 | < 2.9E-01 | 37-39 |
| | Sr-90 | 8.6E+00 | 37-39 |
| | Tc-99 | < 4.4E-01 | 37-39 |
| | I-129 | < 2.3E-01 | 37-39 |
| | Cs-137 | < 2.2E-02 | 37-39 |
| | U-232 | < 1.3E-02 | 37-39 |
| | U-233/234 | 8.2E-01 | 37-39 |
| | U-235/236 | 9.2E-02 | 37-39 |
| | U-238 | 1.1E+00 | 37-39 |
| | Np-237 | < 2.1E-02 | 37-39 |
| | Pu-238 | < 1.1E-02 | 37-39 |

**Table C-4. Radiological Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas(1)**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| | Pu-239/240 | < 1.5E-02 | 37-39 |
| | Pu-241 | < 4.9E-01 | 37-39 |
| | Am-241 | < 1.7E-02 | 37-39 |
| | Cm-243/244 | < 1.6E-02 | 37-39 |
| GP8098 (WMA 1, 1998)<br>Depth to Lavery till - 41 ft | 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 |
| GP8008 (WMA 1, 2008)<br>Depth to Lavery till - 40 ft | C-14 | < 2.8E-01 | 39-41 |
| | C-14 | < 2.8E-01 | 41-43 |
| | Sr-90 | 5.3E+00 | 39-41 |
| | Sr-90 | 1.4E+00 | 41-43 |
| | Tc-99 | < 3.4E-01 | 39-41 |
| | Tc-99 | < 3.7E-01 | 41-43 |
| | I-129 | < 1.2E-01 | 39-41 |
| | I-129 | < 1.2E-01 | 41-43 |
| | Cs-137 | < 2.3E-02 | 39-41 |
| | Cs-137 | < 2.8E-02 | 41-43 |
| | U-232 | < 1.0E-02 | 39-41 |
| | U-232 | < 1.3E-02 | 41-43 |
| | U-233/234 | 5.2E-01 | 39-41 |
| | U-233/234 | 1.1E+00 | 41-43 |
| | U-235/236 | 3.9E-02 | 39-41 |
| | U-235/236 | 1.1E-01 | 41-43 |
| | U-238 | 8.2E-01 | 39-41 |
| | U-238 | 1.4E+00 | 41-43 |
| | Np-237 | < 1.1E-02 | 39-41 |
| | Np-237 | < 1.2E-02 | 41-43 |
| | Pu-238 | < 1.5E-02 | 39-41 |
| | Pu-238 | < 1.5E-02 | 41-43 |
| | Pu-239/240 | < 1.6E-02 | 39-41 |
| | Pu-239/240 | < 1.5E-02 | 41-43 |
| | Pu-241 | < 4.4E-01 | 39-41 |
| | Pu-241 | < 5.2E-01 | 41-43 |
| | Am-241 | < 1.2E-02 | 39-41 |
| | Am-241 | < 1.5E-02 | 41-43 |
| | Cm-243/244 | < 1.3E-02 | 39-41 |
| | Cm-243/244 | < 1.6E-02 | 41-43 |
| GP8308 (WMA 1, 2008) | C-14 | < 3.5E-01 | 40-42 |

**Table C-4. Radiological Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas[1]**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| Depth to Lavery till - 41.5 ft | Sr-90 | 1.5E+00 | 40-42 |
| | Tc-99 | < 3.6E-01 | 40-42 |
| | I-129 | 2.4E-01 | 40-42 |
| | Cs-137 | < 2.7E-02 | 40-42 |
| | U-232 | < 2.4E-02 | 40-42 |
| | U-233/234 | 9.8E-01 | 40-42 |
| | U-235/236 | 2.2E-01 | 40-42 |
| | U-238 | 1.1E+00 | 40-42 |
| | Np-237 | < 1.3E-02 | 40-42 |
| | Pu-238 | < 1.1E-02 | 40-42 |
| | Pu-239/240 | < 1.1E-02 | 40-42 |
| | Pu-241 | < 2.7E-01 | 40-42 |
| | Am-241 | < 1.2E-02 | 40-42 |
| | Cm-243/244 | < 1.8E-02 | 40-42 |
| GP8698 (WMA 1, 1998)<br>Depth to Lavery till - 39 ft | 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 |
| GP10008 (WMA 1, 2008)<br>Depth to Lavery till - 37 ft | C-14 | < 3.0E-01 | 37-39 |
| | Sr-90 | 6.7E+00 | 37-39 |
| | Tc-99 | < 4.0E-01 | 37-39 |
| | I-129 | < 1.4E-01 | 37-39 |
| | Cs-137 | < 2.7E-02 | 37-39 |
| | U-232 | < 1.3E-02 | 37-39 |
| | U-233/234 | 7.6E-01 | 37-39 |
| | U-235/236 | 7.5E-02 | 37-39 |
| | U-238 | 9.5E-01 | 37-39 |
| | Np-237 | < 1.2E-02 | 37-39 |
| | Pu-238 | < 2.2E-02 | 37-39 |
| | Pu-239/240 | < 1.1E-02 | 37-39 |
| | Pu-241 | < 4.3E-01 | 37-39 |
| | Am-241 | < 1.4E-02 | 37-39 |
| | Cm-243/244 | < 2.3E-02 | 37-39 |
| GP10108 (WMA 1, 2008)<br>Depth to Lavery till - 33 ft | C-14 | < 3.1E-01 | 32-34 |
| | Sr-90 | 6.3E-01 | 32-34 |
| | Tc-99 | < 5.4E-01 | 32-34 |
| | I-129 | < 9.1E-02 | 32-34 |

**Table C-4. Radiological Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas(1)**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| | Cs-137 | < 2.6E-02 | 32-34 |
| | U-232 | < 1.6E-01 | 32-34 |
| | U-233/234 | 6.0E-01 | 32-34 |
| | U-235/236 | 5.0E-02 | 32-34 |
| | U-238 | 7.3E-01 | 32-34 |
| | Np-237 | < 1.0E-02 | 32-34 |
| | Pu-238 | < 9.5E-03 | 32-34 |
| | Pu-239/240 | < 8.8E-03 | 32-34 |
| | Pu-241 | < 4.7E-01 | 32-34 |
| | Am-241 | < 1.1E-02 | 32-34 |
| | Cm-243/244 | < 1.1E-02 | 32-34 |
| GP10408 (WMA 1, on border of WMA 2) Depth to Lavery till - 24 ft | C-14 | < 3.6E-01 | 24-26 |
| | Sr-90 | 7.4E+00 | 24-26 |
| | Tc-99 | < 5.1E-01 | 24-26 |
| | I-129 | < 1.1E-01 | 24-26 |
| | Cs-137 | < 5.5E-02 | 24-26 |
| | U-232 | 4.1E-02 | 24-26 |
| | U-233/234 | 8.8E-01 | 24-26 |
| | U-235/236 | 1.4E-01 | 24-26 |
| | U-238 | 7.9E-01 | 24-26 |
| | Np-237 | < 6.9E-03 | 24-26 |
| | Pu-238 | < 1.2E-02 | 24-26 |
| | Pu-239/240 | < 1.2E-02 | 24-26 |
| | Pu-241 | < 3.1E-01 | 24-26 |
| | Am-241 | < 1.3E-02 | 24-26 |
| | Cm-243/244 | < 1.4E-02 | 24-26 |
| BH-05 (WMA 2, 1993), located downgradient of Lagoon 1 Depth to Lavery till - 12 ft | 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 |
| | 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) Depth to Lavery till - 13 ft | Sr-90 | 1.3E-01 | 12-14 |
| | Cs-137 | 7.5E-02 | 12-14 |

**Table C-4. Radiological Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas(1)**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| | 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), located downgradient of Lagoon 1<br>Depth to Lavery till - 11.5 ft | 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)<br>Depth to Lavery till - 15.5 ft | 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 |
| BH-13 (WMA 2, 1993)<br>Depth to Lavery till - 19 ft | 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 |

**Table C-4. Radiological Concentrations from Soil Samples Containing Lavery Till in the WMA 1 and WMA 2 Excavation Areas(1)**

| Location | Nuclide | Result (pCi/g) | Sample Depth Interval (ft) |
|---|---|---|---|
| | 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)<br>Depth to Lavery till - 15 ft | 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 other 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 – Rev1.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
  - Hydraulic gradient
  - Gamma shielding factor
  - Indoor air filtration factor
  - Mass loading for dust inhalation
  - Depth of roots
  - Food 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
- Gamma shielding factor
- Indoor air filtration factor
- Mass loading for dust inhalation
- Depth of roots
- Food transfer factors

• Stream Bank Sediment sources:

- Outdoor time fraction
- Source thickness
- Unsaturated zone thickness
- Soil/water distribution coefficients
- Runoff/Evapotranspiration coefficients/ Infiltration rate
- Mass loading for dust inhalation
- Root depth
- Food 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, the source volume was evaluated for three thickness/area configurations to conserve the total amount of excavated material. The source thickness/area was varied from 0.1m/300m$^2$ to 0.6 m/50 m$^2$, to bound the base case of 0.3 m/100 m$^2$. 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 (assumed to be ~30 m$^3$).

***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. The sediment model assumes that there is no unsaturated zone for the stream bank.

***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 63 m/y (2.0E-04 cm/s) to 220 m/y (7.0E-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 630 to 2200 m/y 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. Values were selected to ensure that the site-specific groundwater conceptual model assumptions (that the well captures the entire width of the plume, but that there is some vertical dilution within the water table) were maintained.

***Runoff/evapotranspiration coefficient*** – varied from 0.41/0.6 to 0.41/0.9 to bound the base case of 0.41/0.78. The base case was selected to achieve infiltration rate of 0.26m/y which corresponds to the calibrated three dimensional groundwater model used in the Decommissioning EIS (DEIS Appendix E). The upper and lower bounds are assumed values for these parameters that maintain the site-specific groundwater dilution assumptions.

***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. The upper and lower bound values for these parameters also maintain the site-specific groundwater dilution assumptions.

***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 165 m. Base value was selected to achieve site-specific groundwater dilution factor of 0.2. Values were selected to ensure that the site-specific groundwater conceptual model assumptions (that the well captures the entire width of the plume, but that there is some vertical dilution within the water table) were maintained.

***Hydraulic gradient*** – applicable to non-dispersion model only (surface soil base case). Varied from 0.02 to 0.04 to bound the base case of 0.03.

***Gamma shielding factor*** – applicable to the surface and subsurface soil models. Varied from 0.17 to 0.51 to bound base case of 0.273, representing a range of possible home construction methods.

***Indoor air filtration factor*** – applicable to the surface and subsurface soil models. Varied from 0.4 to 0.75 to evaluate less conservative assumptions than the base case value of 1.0.

***Mass loading for inhalation*** – applicable to all models. For the soil models, the range of 4.5E-06 to 2.5E-05 bound the base case of 1.5E-05 g/m$^3$. For sediment, the base case of 3.2E-06 is bounded by the range of 1E-06 to 1E-05.

***Root depth*** – applicable to all models. Varied from 0.3 to 3.0 from the base case of 0.9 m to reflect a range of potential crops.

***Food transfer factors*** – 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 $K_d$ value)
- WV Surface - SENS12.RAD (increased $K_d$ 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 hydraulic gradient)
- WV Surface – SENS20.RAD (increased hydraulic gradient)
- WV Surface – SENS21.RAD (decreased gamma shielding factor)
- WV Surface – SENS22.RAD (increased gamma shielding factor)
- WV Surface – SENS23.RAD (decreased indoor air filtration factor)
- WV Surface – SENS24.RAD (increased indoor air filtration factor)
- WV Surface – SENS25.RAD (decreased mass loading factor for inhalation)
- WV Surface – SENS26.RAD (increased mass loading factor for inhalation)
- WV Surface – SENS27.RAD (decreased root depth)
- WV Surface – SENS28.RAD (increased root depth)
- WV Surface - SENS29.RAD (decreased food transfer factors)
- WV Surface – SENS30.RAD (increased food transfer factors)
- WV Surface - SENS31.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 $K_h$ value)
- WV Subsurface - SENS12.RAD (increased $K_h$ value)
- WV Subsurface - SENS13.RAD (decreased runoff/evapotranspiration)
- WV Subsurface - SENS14.RAD (increased runoff/evapotranspiration)
- WV Subsurface – SENS15.RAD (decreased gamma shielding factor)
- WV Subsurface – SENS16.RAD (increased gamma shielding factor)
- WV Subsurface – SENS17.RAD (decreased indoor air filtration factor)
- WV Subsurface – SENS18.RAD (increased indoor air filtration factor)
- WV Subsurface – SENS19.RAD (decreased mass loading factor for inhalation)
- WV Subsurface – SENS20.RAD (increased mass loading factor for inhalation)
- WV Subsurface – SENS21.RAD (decreased root depth)
- WV Subsurface – SENS22.RAD (increased root depth)
- WV Subsurface - SENS23.RAD (decreased food transfer factors)
- WV Subsurface – SENS24.RAD (increased food 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 root depth)

- WV Sediment – SENS12.RAD (increased root depth)
- WV Sediment - SENS13.RAD (decreased food transfer factors)
- WV Sediment – SENS14.RAD (increased food 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 hydraulic gradient)
  - WV Surface - SENS20_sum.TXT (increased hydraulic gradient)
  - WV Surface - SENS21_sum.TXT (decreased gamma shielding factor)
  - WV Surface - SENS22_sum.TXT (increased gamma shielding factor)
  - WV Surface - SENS23_sum.TXT (decreased indoor air filtration factor)
  - WV Surface - SENS24_sum.TXT (increased indoor air filtration factor)
  - WV Surface - SENS25_sum.TXT (decreased mass loading factor for inhalation)
  - WV Surface - SENS26_sum.TXT (increased mass loading factor for inhalation)
  - WV Surface - SENS27_sum.TXT (decreased root depth)
  - WV Surface - SENS28_sum.TXT (increased root depth)
  - WV Surface - SENS29_sum.TXT (decreased food transfer factors)
  - WV Surface - SENS30_sum.TXT (increased food transfer factors)
  - WV Surface - SENS31_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.RAD (decreased gamma shielding factor)
- WV Subsurface – SENS16.RAD (increased gamma shielding factor)
- WV Subsurface – SENS17.RAD (decreased indoor air filtration factor)
- WV Subsurface – SENS18.RAD (increased indoor air filtration factor)
- WV Subsurface – SENS19.RAD (decreased mass loading factor for inhalation)
- WV Subsurface – SENS20.RAD (increased mass loading factor for inhalation)
- WV Subsurface – SENS21.RAD (decreased root depth)
- WV Subsurface – SENS22.RAD (increased root depth)
- WV Subsurface - SENS23_sum.TXT (decreased food transfer factors)
- WV Subsurface – SENS24_sum.TXT (increased food 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 root depth)
- WV Sediment – SENS12_sum.TXT (increased root depth)

- WV Sediment - SENS13_sum.TXT (decreased food transfer factors)
- WV Sediment – SENS14_sum.TXT (increased food 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;

$$DCGL_w = \text{Dose Limit / Peak radionuclide DSR} \qquad \text{(Eq.2)}$$

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_Rev1.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 “WV Surface DCGLs_Rev1.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_Rev1.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_Rev1.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 SENS31 (Tables C-17 to C-47). Also included in the file are a summarization of the calculated DCGLs (Table C-48) and a summary of the percent change from the base case (Table C-49) for each of the sensitivity runs (also presented in Table 5-9). Table C-50 below presents a summary of the surface soil sensitivity results.

**Table C-50 Summary of Surface Soil DCGL Sensitivity Analysis**

| Parameter | Run | Change in Sensitivity Parameter | Minimum | | Maximum | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Indoor/Outdoor Fraction | 1 | -32% | -22% | U-232 | 0% | Cm-244 |
| | 2 | 21% | 0% | C-14 I-129 Np-237 Tc-99 U-234 | 28% | U-232 |
| Source Thickness | 3 | -50% | 9% | U-232 | 231% | C-14 |
| | 4 | 200% | -57% | C-14 | 0% | Am-241 Cm-243 Cm-244 Cs-137 Pu-239 Pu-240 |
| Unsaturated Zone Thickness | 5 | -50% | -10% | Tc-99 | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 |
| | 6 | 150% | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 | 12% | U-235 |
| Irrigation/Pump Rate | 7 | -57% | -1% | U-232 | 65% | I-129 |
| | 8 | 70% | -36% | I-129 | 1% | U-232 |
| Soil/Water Distribution Coefficients ($K_d$) | 9 | lower | -99% | Pu-239 | 2% | C-14 |
| | 10 | higher | -3% | U-232 | 867% | U-234 |

**Table C-50 Summary of Surface Soil DCGL Sensitivity Analysis**

| Parameter | Run | Change in Sensitivity Parameter | Minimum | | Maximum | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Hydraulic Conductivity ($K_h$) | 11 | -55% | -36% | I-129 | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 |
| | 12 | 57% | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 | 40% | I-129 |
| Runoff/Evaporation Coefficient | 13 | -23% | -29% | U-234 | 2% | U-232 |
| | 14 | 15% | -2% | U-232 | 81% | Np-237 |
| Depth of Well Intake | 15 | -40% | -40% | I-129 | 0.0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 |
| | 16 | 100% | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 | 99% | I-129 |
| Length Parallel to Aquifer Flow | 17 | -30% | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 | 30% | I-129 |
| | 18 | 21% | -12% | I-129 | 0.0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Pu-241 Sr-90 U-232 |
| Hydraulic Gradient | 19 | -33% | -23% | I-129 | 0.0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 |

**Table C-50 Summary of Surface Soil DCGL Sensitivity Analysis**

| Parameter | Run | Change in Sensitivity Parameter | Minimum | | Maximum | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| | **20** | 33% | 0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 U-232 | 23.3% | I-129 |
| Gamma Shielding Factor | **21** | -38% | 0% | no change | 0.0% | no change |
| | **22** | 87% | -24% | U-232 | 0.0% | Np-237 |
| Indoor Dust Filtration Factor | **23** | -60% | 0% | C-14 Cs-137 I-129 Np-237 Sr-90 Tc-99 U-234 | 0.6% | Cm-244 |
| | **24** | -25% | 0% | C-14 Cs-137 I-129 Np-237 Sr-90 Tc-99 U-233 U-234 | 0.3% | Pu-241 |
| Dust Loading Factor | **25** | -70% | 0% | C-14 Cs-137 I-129 Np-237 Sr-90 Tc-99 U-234 | 1.0% | Cm-244 |
| | **26** | 67% | -1% | Cm-244 | 0.0% | C-14 Cs-137 I-129 Sr-90 Tc-99 U-235 U-238 |
| Root Depth | **27** | -67% | 0% | no change | 0.0% | no change |
| | **28** | 233% | 0% | I-129 | 199.7% | C-14 |
| Food Transfer Factors | **29** | lower | -38% | U-235 | 875% | Sr-90 |
| | **30** | higher | -97% | Sr-90 | -14% | Np-237 |
| Mass Balance Model | **31** | NA | -67% | U-234 | 0.0% | Am-241 C-14 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 Sr-90 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_Rev1.XLS" in the sheet "Base" (TableC-51), 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_Rev1.XLS". The results for each source area are presented in individual sheets (Tables C-52 to C-55). The sheet "Summary" presents the DCGL$_{EMC}$ values (Table C-56) and subsurface soil area factors (Table C-57) 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_Rev1.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 SENS24 (Tables C-58 to C-81). Also included in the file is a summarization of the calculated DCGLs (Table C-82) and a summary of the percent change from the base case (Table C-83) for each of the sensitivity runs (also presented in Table 5-10). Table C-84 below presents a summary of the subsurface soil sensitivity results.

**Table C-84 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 | 0.5% | Pu-238 |
| | 2 | 21% | 0% | C-14 | 35% | U-232 |

**Table C-84 Summary of Subsurface Soil DCGL Sensitivity Analysis**

| Parameter | Run | Change in Sensitivity Parameter | Minimum | | Maximum | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Source Thickness | 3 | -67% | -65% | U-238 | 204% | Tc-99 |
| | 4 | 233% | -33% | C-14 | 98% | U-234 |
| Unsaturated Zone Thickness | 5 | -50% | -2% | Np-237 | 58% | U-238 |
| | 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 | 2218% | U-234 |
| Irrigation/Pump Rate | 7 | -57% | -39% | I-129 | 57% | U-238 |
| | 8 | 70% | 0% | Am-241 Cm-243 Cm-244 Pu-238 Pu-239 Pu-240 | 20% | I-129 |
| Soil/Water Distribution Coefficients (Kd) | 9 | lower | -99% | Pu-239 | 116% | U-232 |
| | 10 | higher | -20% | U-232 | 2168% | U-234 |
| Hydraulic Conductivity ($K_h$) | 11 | -55% | 0% | No change | 0% | No change |
| | 12 | 57% | 0% | No change | 0% | No change |
| Runoff/Evaporation Coefficient | 13 | -23% | -44% | U-234 | 61% | U-238 |
| | 14 | 15% | -11% | U-232 | 117% | U-234 |
| Indoor Gamma Shielding Factor | 15 | -38% | 0% | U-238 | 19% | U-232 |
| | 16 | 87% | -27% | Cs-137 | 1% | U-238 |
| Indoor Dust Filtration Factor | 17 | -60% | 0% | U-238 | 13% | Cm-244 |
| | 18 | -25% | 0% | C-14 Cs-137 I-129 Np-237 Sr-90 Tc-99 U-233 U-234 U-238 | 5% | Cm-244 |
| Inhalation Dust Loading | 19 | -70% | 0% | U-238 | 22% | Cm-244 |
| | 20 | 67% | -15% | Cm-244 | 0% | C-14 Cs-137 I-129 Np-237 Sr-90 Tc-99 |
| Root Depth | 21 | -67% | -67% | Tc-99 | 1% | U-233 |
| | 22 | 233% | 0% | U-238 | 227% | Tc-99 |

**Table C-84 Summary of Subsurface Soil DCGL Sensitivity Analysis**

| Parameter | Run | Change in Sensitivity Parameter | Minimum | | Maximum | |
|---|---|---|---|---|---|---|
| | | | Change | Nuclide(s) | Change | Nuclide(s) |
| Food Transfer Factors | 23 | lower | -0.1% | U-238 | 582% | Tc-99 |
| | 24 | higher | -93% | Sr-90 | 0% | 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_Rev1.XLS" in the sheet "Base" (Table C-85). 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_Rev1.XLS". The results for each source area are presented in individual sheets (Tables C-86 to C-91). The sheet "Summary" presents the $DCGL_{EMC}$ values (Table C-92) and sediment area factors (Table C-93) 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_Rev1.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-94 to C-107). Also included in the file is a summarization of the calculated DCGLs (Table C-108) and percent change from the base case (Table C-109) for each of the sensitivity runs (also presented in Table 5-11). Table C-110 below presents a summary of the sediment sensitivity analysis.

**Table C-110 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 | 98% | Cm-243 |
| | 2 | 100% | -50% | Cm-243 | 0% | C-14 |
| Source Thickness | 3 | -50% | 0% | Am-241 Cm-243 | 157% | C-14 |
| | 4 | 200% | -52% | C-14 | 0% | Am-241 Cm-243 Cm-244 Pu-238 Pu-239 Pu-240 |
| Soil/Water Distribution Coefficients (Kd) | 5 | lower | -91% | Pu-239 | 26% | U-232 |
| | 6 | higher | -65% | U-233 | 52% | U-234 |
| Runoff/Evaporation Coefficient | 7 | -23% | 0% | Am-241 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 | 4% | U-232 |
| | 8 | 15% | -3% | I-129 | 0% | Am-241 Cm-243 Cm-244 Cs-137 Pu-238 Pu-239 Pu-240 |
| Mass Loading for Inhalation | 9 | -70% | 0% | Np-237 | 1% | Cm-244 |
| | 10 | 67% | -4% | Cm-244 | 0% | C-14 Cs-137 I-129 Sr-90 |
| Root Depth | 11 | -67% | 0% | no change | 0% | no change |
| | 12 | 233% | 0% | Cm-243 U-232 U-235 | 50% | Sr-90 |
| Food Transfer Factors | 13 | lower | 1% | Cm-243 | 852% | Sr-90 |
| | 14 | higher | -98% | Sr-90 | -11% | Cm-243 |

**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_Rev1.xls” (Table C-111) 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_Rev1.xls” presents the calculated $DCGL_W$ and $DCGL_{EMC}$ values when considering the combined doses from surface soil (90% x 25 mrem/y = 22.5 mrem/y) and sediment sources (10% x 25 mrem/y = 2.5 mrem/y), which are summarized in Tables C-112, C-113, and C-114 (also presented in Table 5-13). In the same Excel file, Table C-115 presents the cleanup goals to be used as the criteria for the remediation activities. Values in Table C-115 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 percent 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 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 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-116).

**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 cannot be performed until additional site characterization data is available. In the interim, Table C-117 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 will be refined as additional site data are collected.

**Table C-117 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 |

**Table C-117 Summary of Primary Dose Pathways**

| Nuclide | Primary Pathway for Dose | Key Parameters(1) | Year of Peak Dose |
|---|---|---|---|
| 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 uptake) | source thickness, Kd, runoff/evap coefficients | 1.37E+01 |
| 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 | outdoor fraction | 0.00E+00 |

**Table C-117 Summary of Primary Dose Pathways**

| Nuclide | Primary Pathway for Dose | Key Parameters(1) | Year of Peak Dose |
|---|---|---|---|
| | independent (meat uptake) | | |
| 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 |
| 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.

## 3.0 References

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**Attachments**

1. Electronic Files Described in Section 2 (provided separately)
2. Electronic File Described in Section 1 (provided separately)

# 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 decommissioning and describe the post-remediation monitoring, maintenance, and institutional control program to be implemented for the WVDP premises following Phase 1 Decommissioning.

**INFORMATION IN THIS APPENDIX**

This appendix includes information on engineered barrier conceptual designs and the conceptual 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 decommissioning;
- Section 2 describes the conceptual post-remediation site monitoring and maintenance program that will be implemented for the project premises at the conclusion of Phase 1 decommissioning;
- Section 3 describes the conceptual post-remediation site institutional control program that will be implemented for the project premises at the conclusion of Phase 1 decommissioning.

**RELATIONSHIP TO OTHER PLAN SECTIONS**

Information provided in Section 1 on the project background and Section 7 on decommissioning activities, will 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 decommissioning, supplementing the physical descriptions previously presented in Section 7. Engineered barriers will 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.

The final design of the barrier walls and French drain will be prepared by the site decommissioning contractor after Phase 1 decommissioning activities start in 2011. The final design details of the hydraulic barriers and French drain will be provided to the NRC for technical review before their installation, as indicated in Section 1.6 of this plan.

The development of the WMA 1 and WMA 2 hydraulic barrier walls and French drain designs will be supported by the collection of subsurface soil geotechnical data, the installation of groundwater monitoring wells to provide groundwater elevation monitoring data, and groundwater modeling to evaluate the potential impacts these structures have on groundwater flow patterns in WMA 1 and WMA 2 and in surrounding areas such as WMA 3.

> 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 decommissioning will 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 will require the installation of temporary and permanent subsurface hydraulic barrier walls prior to excavation as described in Section 7. A French drain system will be installed in the backfilled excavation to prevent mounding of groundwater against the permanent barrier wall as described in Section 7. The WMA 1 barrier walls and French drain will be designed to result in minimal changes to groundwater flow patterns and water levels in WMA 3. 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 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 will be removed down into the underlying Lavery till to meet the unrestricted release criteria in 10 CFR 20.1402. Much of the WMA 1 excavation will be within the saturated sand and gravel unit within the north plateau groundwater plume.

Subsurface hydraulic barrier walls will 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 will 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 will be used to locate the temporary interlocking sheet piling which will 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 permanent hydraulic barrier of slurry wall type construction will 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 will:

- 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 will 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 their current levels. The French drain system will:

- 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 will 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 will 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 will be installed in uncontaminated soils along the upgradient and cross-gradient sides of the excavation boundary before excavation begins (Figure D-1). The piles will 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.

[1]The recontamination potential is low since groundwater flows northeast away from WMA 1.

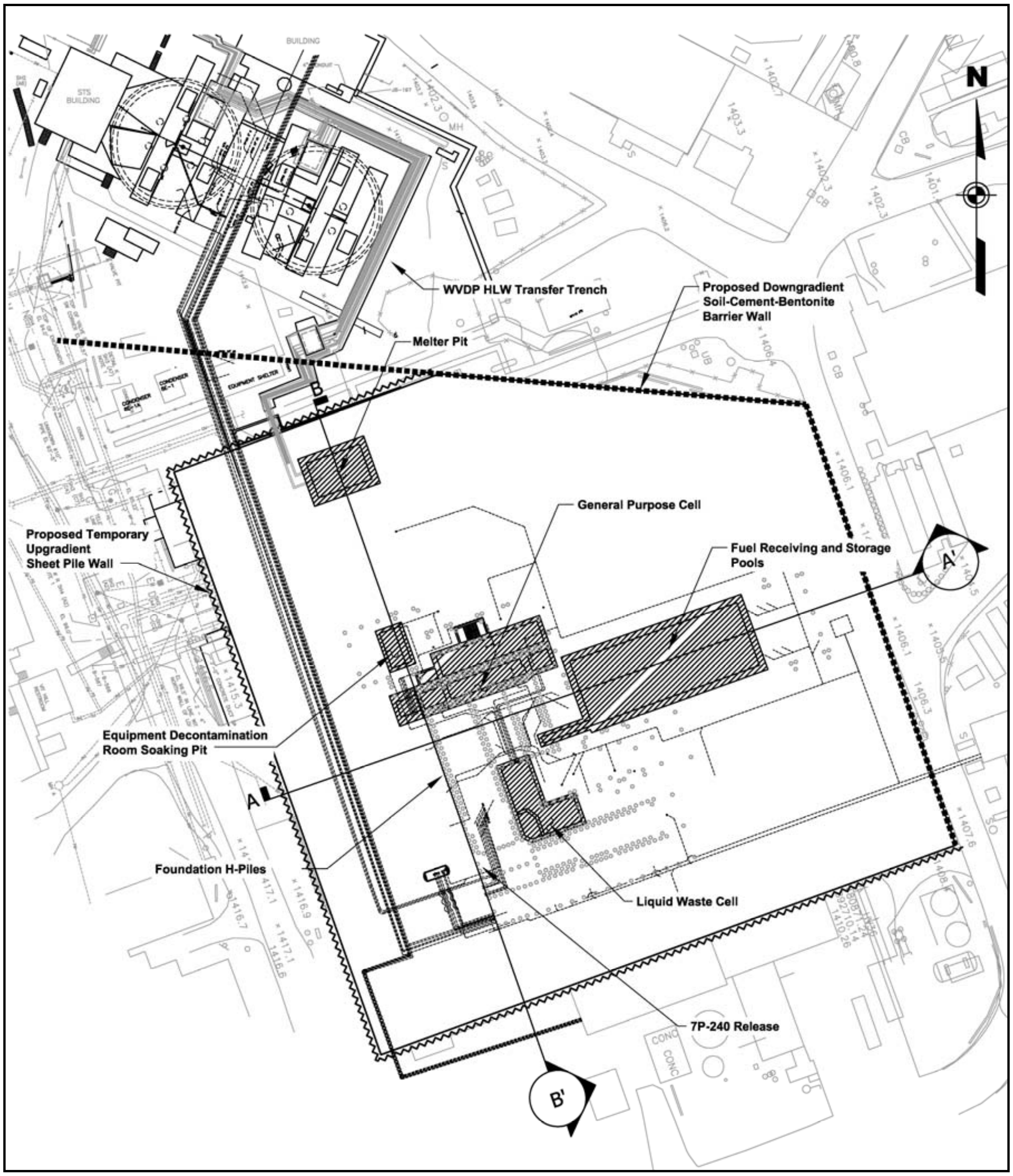


**Figure D-1. Plan View of the WMA 1 Excavation**

Contaminated soil exceeding the subsurface soil cleanup criteria specified in Section 5 will 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 will be surveyed during the Phase 1 final status surveys as specified in Sections 7 and 9 of this plan. The sheet pile barrier wall will be removed as specified in Section 7 once the final status survey, the independent verification survey, and backfilling of the

[2] Figure 7-8 in Section 7 of this plan shows typical excavation slopes.

WMA 1 excavation is completed to allow a return to typical groundwater flow patterns within the sand and gravel unit.

**Permanent Downgradient Hydraulic Barrier Wall**

The permanent hydraulic barrier wall constructed on the downgradient side of the WMA 1 excavation (Figure D-1) will 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 will 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 will be installed under a carefully planned and rigorous quality control-quality assurance program as described in Section 8.

The soil-cement-bentonite barrier wall will be a mixture of 85 percent soil, five percent Portland cement, and 10 percent bentonite. The Portland cement will provide internal stability to the barrier wall and it will have an initial maximum design hydraulic conductivity of 6.0E-06 cm/s.

The soil-cement-bentonite barrier wall will be approximately 750 feet long, two to 13 feet wide, and will be up to 50 feet deep with an average depth of 27 feet. The wall will 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 will be two to three feet thick. The remaining 525 feet of barrier wall within the boundary of the excavation will be at least 13 feet thick to allow the excavation of subsurface soils up to and into the barrier wall. The thickness will 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 will be surveyed during the Phase 1 final status survey.[3]

The upper three feet of the barrier wall will be constructed of clean backfill similar to the surrounding sand and gravel unit. This material will allow vehicular traffic over the barrier wall without damaging the underlying barrier wall.

**French Drain System**

A French drain system will be installed upgradient of the permanent hydraulic barrier wall during the backfilling of the WMA 1 excavation (Figure D-1). The French drain will 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 will 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 drain will meet at a concrete manhole located near the mid-point of the barrier wall. The French

[3] As explained in Section 7 of this plan, any soil found to exceed cleanup goals will 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 will be sloped to the southeast to discharge by gravity flow to a surface water drainage discharging to Erdman Brook.

### 1.2 Waste Management Area 2

The Phase 1 decommissioning activities in WMA 2 will 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-2). The removal of the lagoons, sub-grade structures, and surrounding soils will 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 will 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 will be installed on the northwest and northeast 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 will 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 will 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 and northeast side of the WMA 2 excavation will 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 will 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 will be installed under a carefully planned and rigorous quality control-quality assurance program as described in Section 8. The barrier wall will 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 will 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.

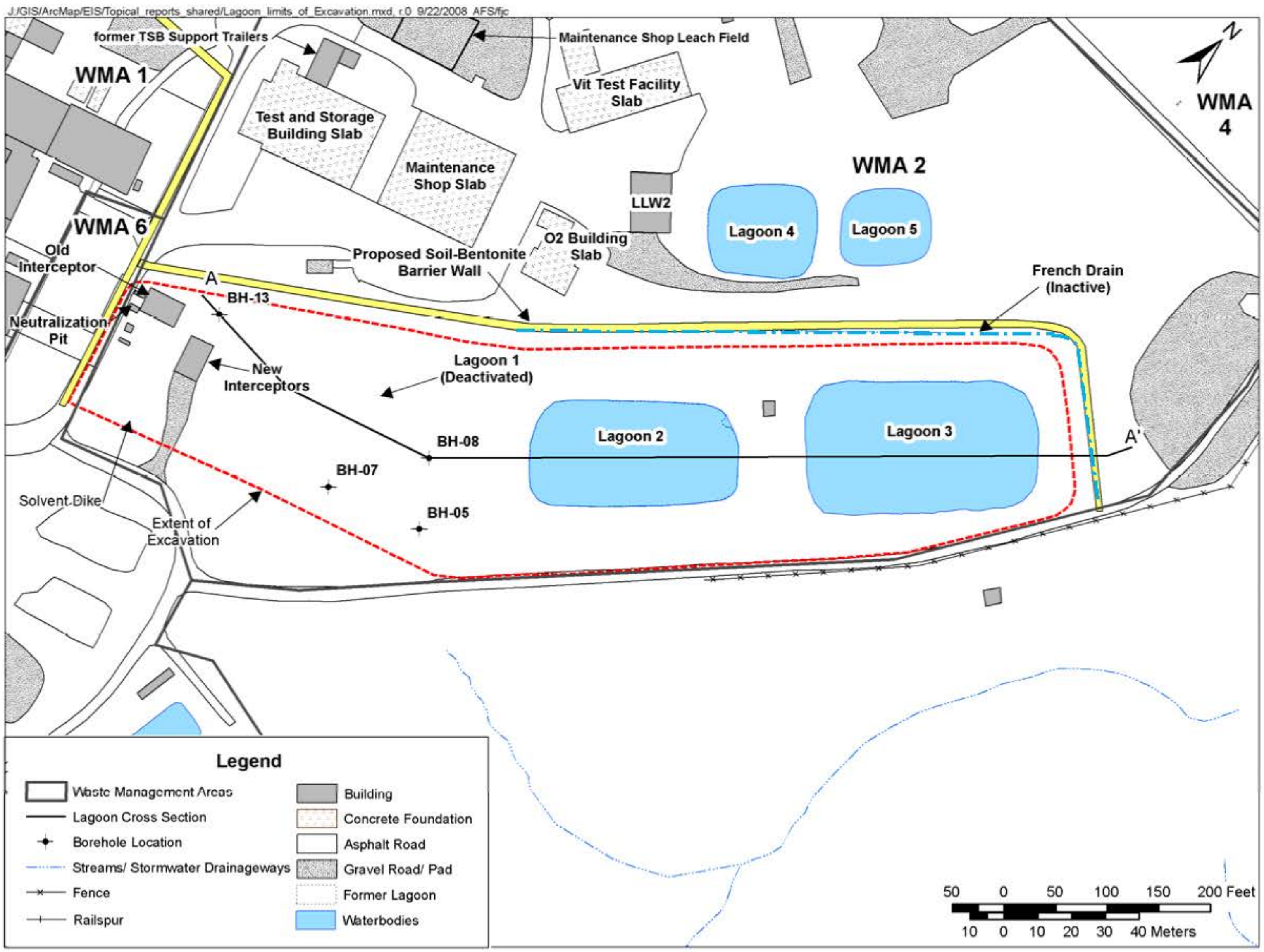


**Figure D-2. Plan View of the WMA 2 Excavation**

The upper three feet of the barrier wall will be constructed of clean backfill similar to the surrounding sand and gravel unit. This material will allow vehicular traffic over the barrier wall without damaging the underlying barrier wall.

### 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 will 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 (Mitchell 1986 and Mitchell 1993). 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 will be a mixture of soil, Portland cement, 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 will 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 will 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-cement-bentonite slurry wall has been evaluated under combined static and seismic loading conditions. The evaluation results indicate that the soil-cement-bentonite slurry wall will 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 will be constructed of natural (stone backfill) and man-made (perforated drain pipe, geotextile) materials. The French drain trench backfill will be designed to minimize silting of the drainpipe. The French drain will be periodically monitored and maintained until the start of Phase 2 decommissioning to ensure it is functioning properly.

### 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 to be installed on the downgradient side of the WMA 1 excavation will be nearly perpendicular to the current groundwater flow path in the sand and gravel unit in the north plateau.

### 1.4.1 Conceptual Model

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]. This model is a revised version of the near-field model described in Appendix E to the Decommissioning EIS. Figure D-3 shows the boundaries of the north plateau near-field model.


**Figure D-3. North Plateau Groundwater Flow Model Boundary**

The north plateau model mimics the shape of the lateral extent of the sand and gravel unit. It is oriented from the southwest to the northeast and extends downward from the ground surface to the top of the Kent Recessional Sequence.

Hydrogeologic units represented in the model are the thick-bedded unit, the slack-water sequence and the unweathered Lavery till. Together, the thick-bedded unit and the slack-water sequence comprise the surficial sand and gravel unit. The thick-bedded unit comprises glaciofluvial gravel and alluvial deposits that range from one to six meters in thickness overlying the unweathered Lavery till. The slack-water sequence is a depositional sequence with layers of gravel, sand and silt filling a southwest-to-northeast trending channel in the upper portion of the unweathered Lavery till. The slack-water sequence varies in thickness from zero to five meters with the thickest portions beneath the Process Building. The unweathered Lavery till is a glacial till with a thickness range of 10 to 17 meters in the model volume.

[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.

The hydrogeologic units incorporated into the north plateau near-field flow model are represented in Figures D-4 through D–8. The slack-water sequence appears in the northeastern portion of the model as shown in Figures D-6 through D–8. The hydraulic conductivities of these units are assumed constant over the model domain with values of 2.5 E-03, 5.3 E-03, and 6.0 E-08 centimeters per second for the thick-bedded unit, slack-water sequence, and unweathered Lavery till, respectively. Two variants of the north plateau near-field model were developed to simulate current north plateau groundwater flow conditions and to evaluate north plateau groundwater flow conditions associated with the hydraulic barriers to be installed during Phase 1.

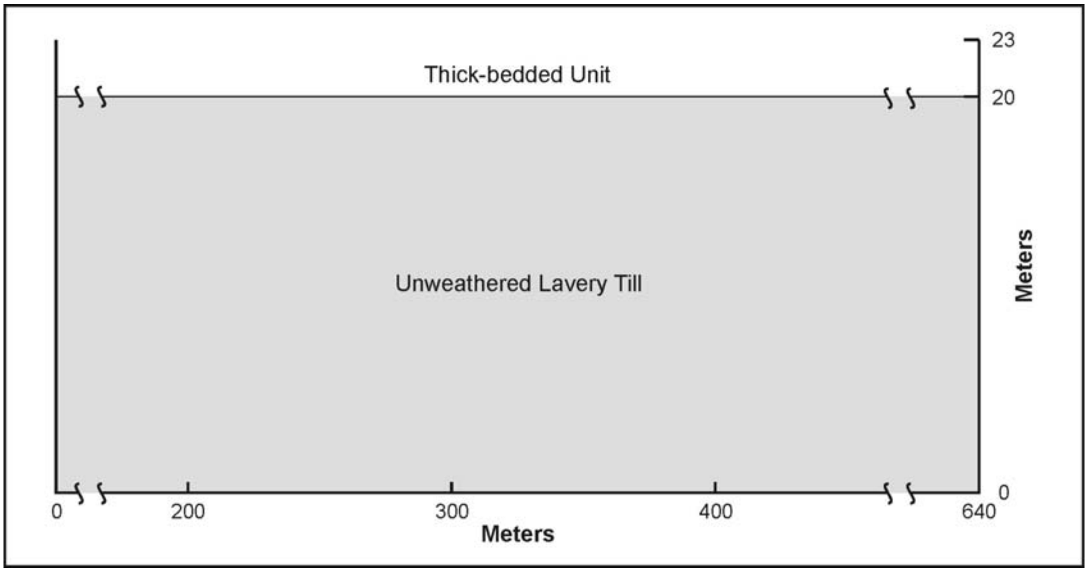


**Figure D-4. Cross Section of North Plateau Near-Field Model – Southwest to Northeast Distance of 0 to 80 Meters**

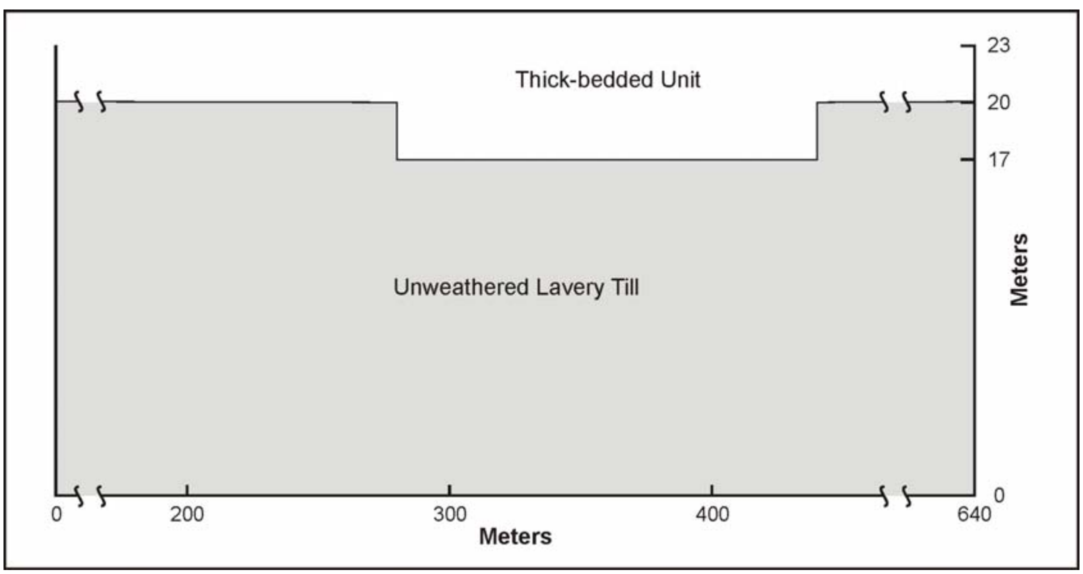


**Figure D-5. Cross Section of North Plateau Near-Field Model – Southwest to Northeast Distance of 80 to 120 Meters**

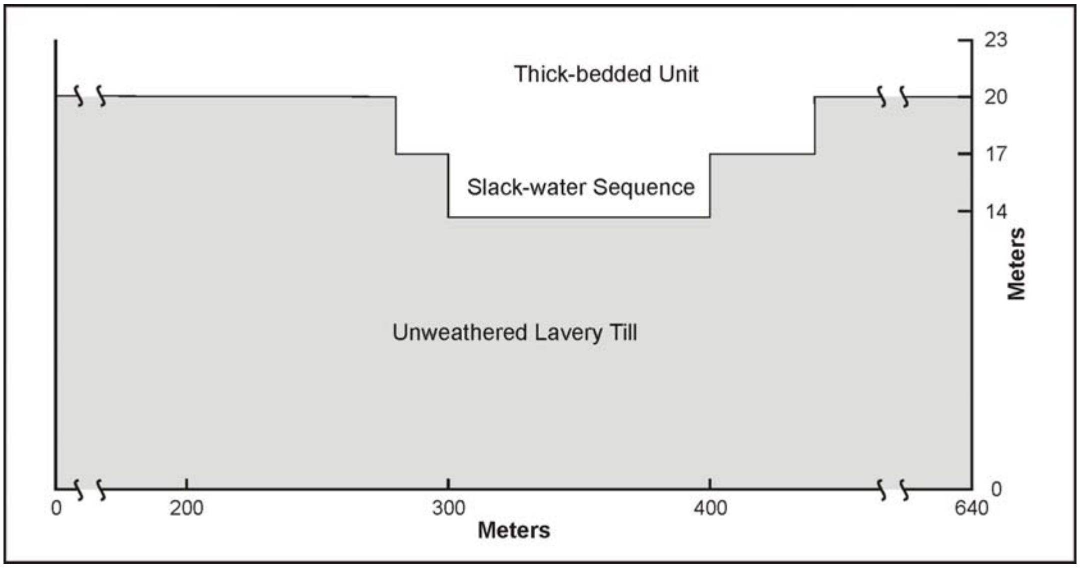


**Figure D-6. Cross Section of North Plateau Near-Field Model – Southwest to Northeast Distance of 120 to 250 Meters**

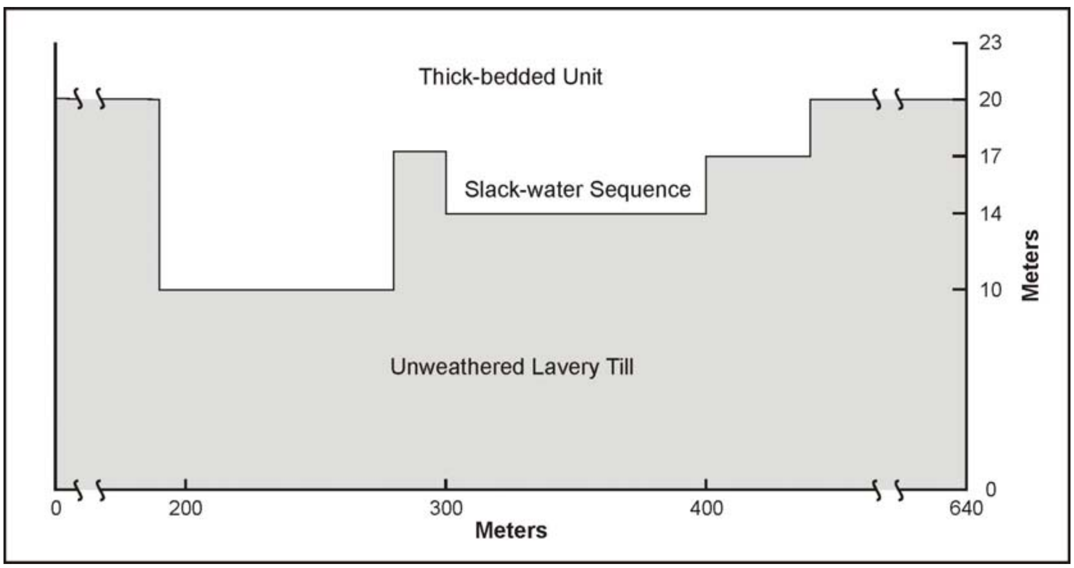


**Figure D-7. Cross Section of North Plateau Near-Field Model – Southwest to Northeast Distance of 250 to 310 Meters**

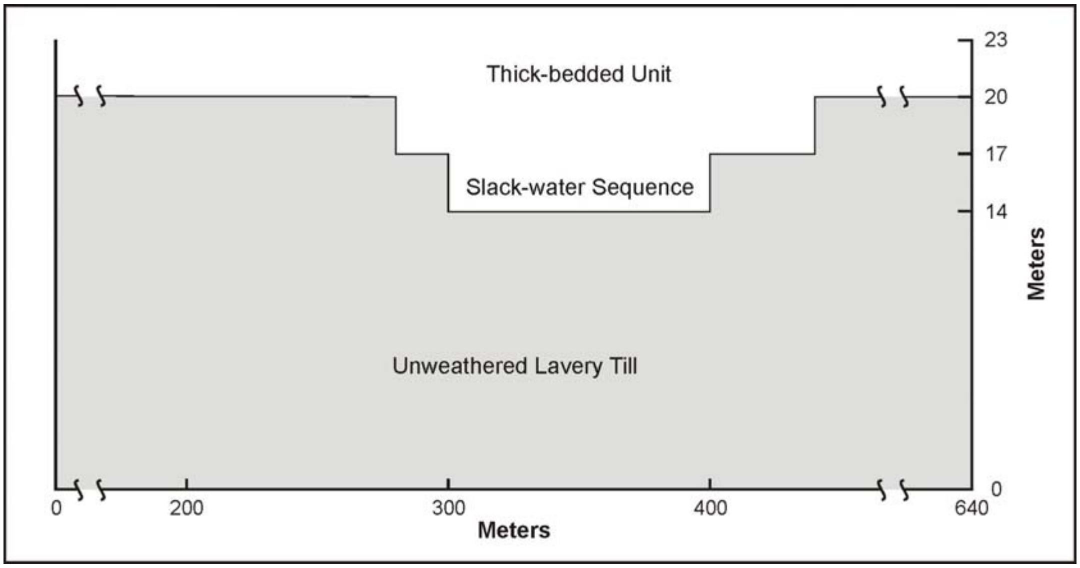


**Figure D-8. Cross Section of North Plateau Near-Field Model – Southwest to Northeast Distance of 310 to 820 Meters**

### 1.4.2 Modeling Current Conditions

To simulate current conditions, the horizontal portion of the near-field groundwater model grid comprised rectangular blocks with 81 blocks in the southwest-to-northwest direction and 64 blocks in the southwest-to-southeast direction. Grid blocks with horizontal dimension as large as 50 meters were used along the west and north boundaries while grid block horizontal dimensions range from 1 to 10 meters over most of the model domain. For the vertical direction, the upper three meters were represented using 15 0.2-meter-thick layers, the next three meters were represented using six 0.5-meter-thick layers, and the bottom 17 meters were represented using 17 1.0-meter-thick layers. With these dimensions, the model utilized approximately 174,000 grid blocks.

Boundary conditions applied for the near-field model are consistent with site observations and with those applied for the site-wide model. At the bottom of the unweathered Lavery till, atmospheric pressure was applied representing the presence of a water table in the Kent Recessional Sequence. On the sides of the model, no flow conditions were applied for the unweathered Lavery till. On the southwest side of the model, lateral recharge into the thick-bedded unit of 20 cubic meters per day was applied. On the northwest, southeast, and northeast sides of the model, atmospheric pressure conditions were applied for the thick-bedded unit and slack-water sequence to represent seepage to Quarry Creek, Erdman Brook, and Franks Creek, respectively.

Evaluation of simulated pressures and measured conditions in target groundwater wells showed that a uniform recharge of 26 centimeters per year produced the closest match to existing conditions. Table D-1 compares measured hydraulic heads in wells screened in the sand and gravel unit from the north plateau with predicted hydraulic heads generated by the near-field model for three different recharge rates. Figure D-9 shows the resulting plot of water table elevation in the thick bedded unit for a recharge of 26 centimeters per year. These water table elevations are consistent with the measured heads and the predictions of the site-wide groundwater model described in Appendix E to the Decommissioning EIS. Table D-2 shows the modeled flow balance.

**Table D-1. North Plateau Near-field Flow Model Calibration for Head(1)**

| Groundwater Well | Measured Head (ft) | Predicted Head (ft) at Specified Recharge | | |
|---|---|---|---|---|
| | | 18 cm/y | 26 cm/y | 34 cm/y |
| 103 | 1391.4 | 1386.8 | 1391.6 | 1394.5 |
| 104 | 1385.5 | 1379.6 | 1383.1 | 1385.7 |
| 116 | 1380.5 | 1372.4 | 1376.8 | 1379.4 |
| 203 | 1394.4 | 1400.2 | 1401.6 | 1404.2 |
| 205 | 1393.1 | 1397.9 | 1399.2 | 1401.2 |
| 301 | 1410.7 | 1401.9 | 1406.8 | 1410.6 |
| 401 | 1410.3 | 1401.5 | 1406.4 | 1409.5 |
| 406/86-08 | 1393.5 | 1394.1 | 1397.4 | 1400.0 |
| 601 | 1377.3 | 1376.9 | 1378.9 | 1380.9 |
| 603 | 1391.9 | 1395.0 | 1397.0 | 1399.6 |
| 604 | 1391.6 | 1389.7 | 1391.9 | 1394.6 |
| 86-09 | 1391.8 | 1391.6 | 1396.5 | 1399.8 |
| 86-12 | 1364.8 | 1343.6 | 1345.2 | 1346.8 |
| 408 | 1391.8 | 1391.0 | 1394.8 | 1398.4 |
| 501 | 1391.3 | 1386.8 | 1391.5 | 1394.5 |
| 403 | 1408.0 | 1401.1 | 1405.8 | 1409.1 |
| 801 | 1376.6 | 1369.3 | 1373.1 | 1375.7 |
| 804 | 1369.9 | 1356.0 | 1359.2 | 1360.4 |
| Sum of Squared Residuals ($ft^2$)(2) | | 1111.4 | 730.1 | 831.4 |

NOTES: (1) This specified recharge is the net inflow at the ground surface that results from the balance of precipitation, evapotranspiration, and run-off.

(2) Sum of squared residuals = (Measured Head – Predicted Head)$^2$ for each location, then summed.

**Table D-2. Summary of Sand and Gravel Unit Flow Balance(1)**

| Inflow | | Outflow | |
|---|---|---|---|
| Location | Rate ($m^3$/y) | Location | Rate ($m^3$/y) |
| Recharge at the Ground Surface | 107,624 | Down Flow to the KRS | 9,060 |
| Recharge from Bedrock from the | 7,304 | Seepage to Quarry Creek | 8,456 |
| | | Seepage to Erdman Brook | 15,238 |

**Table D-2. Summary of Sand and Gravel Unit Flow Balance(1)**

| Inflow | | Outflow | |
|---|---|---|---|
| **Location** | **Rate ($m^3$/y)** | **Location** | **Rate ($m^3$/y)** |
| Southwest | | Seepage to Frank's Creek | 66,713 |
| | | Seepage to North Plateau Ditch | 15,445 |
| Totals | 114,928 | | 114,912 |

NOTE: (1) For a recharge rate of 26 centimeters per year
LEGEND: KRS = Kent Recessional Sequence

The relationship between rate of flow in the slack-water sequence and the thick-bedded unit above the slack-water sequence was investigated through tabulation of groundwater velocities along a flow path extending from the location of the Process Building to the north plateau ditch. Average linear velocities predicted by the near-field model for this path are presented in Table D-3. An effective porosity value of 0.225 was used for the thick-bedded unit and an effective porosity value of 0.35 for the slack-water sequence. For the slack-water sequence and thick-bedded unit above the slack-water sequence, the travel time and average velocity along the flow path are 1.90 years and 161 meters per year and 2.0 years and 157 meters per year, respectively.

**Table D-3. Average Linear Velocity for Flow Path Originating at the Process Building**

| Distance Along Flow Path (m) | Average Linear Velocity (m/y) | |
|---|---|---|
| | **Slack-water Sequence** | **Thick-bedded Unit** |
| 0 to 10 | 114 | 105 |
| 10 to 63 | 130 | 132 |
| 63 to 110 | 143 | 147 |
| 110 to 160 | 156 | 161 |
| 160 to 210 | 171 | 174 |
| 210 to 260 | 192 | 180 |
| 260 to 310 | 220 | 176 |

NOTE: To convert meters per year to feet per year, multiply by 3.2803.

### 1.4.3 Modeling Conditions Following Phase 1 of the Decommissioning

The near-field groundwater flow model developed to assess current groundwater flow conditions was used to evaluate groundwater flow following the installation of the Phase 1 hydraulic barriers and WMA 1 French drain. The WMA 1 and WMA 2 slurry walls are modeled as one-meter thick extending downward to the unweathered Lavery till with a hydraulic conductivity of 1.0 E-06 cm/s. The WMA 1 hydraulic barrier wall downgradient of the Process Building is oriented parallel to the groundwater elevation contours and perpendicular to groundwater flow as shown in Figure D-9. 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. The French drain for WMA 1 was modeled as one-meter thick with a depth of three meters and a hydraulic conductivity of 10 cm/s.

The cross-sectional structure of the aquifer is that represented in Figures D-4, D-5, D-6, D-7, and D-8 with the same vertical discretization as the current conditions case.

Figure D-9 shows the distribution of hydraulic heads predicted following completion of Phase 1 of the decommissioning. The results indicate an overall increase in water table elevation of several feet across the large backfilled WMA 1 and WMA 2 excavations formerly occupied by the Process Building and the lagoons, respectively.

The higher groundwater elevations in the backfilled WMA 1 excavation suggest that groundwater would flow through the WMA 1 slurry wall to the northeast, towards the non-source area of the north plateau groundwater plume. However, a significant volume of this flow would be diverted by the French drain and discharged to Erdman Brook (Table D-4). Groundwater elevations coincide on either side of the slurry wall separating the backfilled WMA 1 excavation from the Waste Tank Farm, suggesting little potential for groundwater flow from the backfilled WMA 1 excavation toward the Waste Tank Farm.

Groundwater elevations coincide with the bottom of the French drain near the WMA 1 barrier wall. Groundwater elevations on the downgradient side of the WMA 1 barrier wall are approximately 10 feet lower than on the upgradient side, resulting in a steep hydraulic gradient across the barrier wall and a shallower gradient along the non-source area of the north plateau groundwater plume.

Groundwater levels in the backfilled WMA 2 excavation are several feet higher than modeled in the current conditions scenario and would be below grade across the backfilled WMA 2 excavation. Groundwater elevations are up to 10 feet lower on the north plateau plume side of the WMA 2 barrier wall, suggesting groundwater flow to the northwest and northeast through the WMA 2 slurry wall towards the non-source area of the north plateau groundwater plume and to the southeast towards Erdman Brook.

Table D-4 summarizes the modeled flow balance. Table D-5 shows the average linear velocities predicted by the near-field model for conditions after Phase 1.

**Table D-4. Summary of Sand and Gravel Unit Flow Balance After Phase 1[(1)]**

| Inflow | | Outflow | |
|---|---|---|---|
| **Location** | **Rate ($m^3$/y)** | **Location** | **Rate ($m^3$/y)** |
| Recharge at the Ground Surface | 107,624 | Down Flow to the KRS | 8,909 |
| Recharge from Bedrock from the Southwest | 7,304 | Seepage to Quarry Creek | 8,780 |
| | | Seepage to Erdman Brook (TBU) | 14,915 |
| | | French Drain to Erdman Brook | 21,698 |
| | | Seepage to Frank's Creek | 46,791 |
| | | Seepage to North Plateau Ditch | 13,783 |
| Total | 114,928 | | 114,876 |

NOTE: (1) For a recharge rate of 26 centimeters per year.

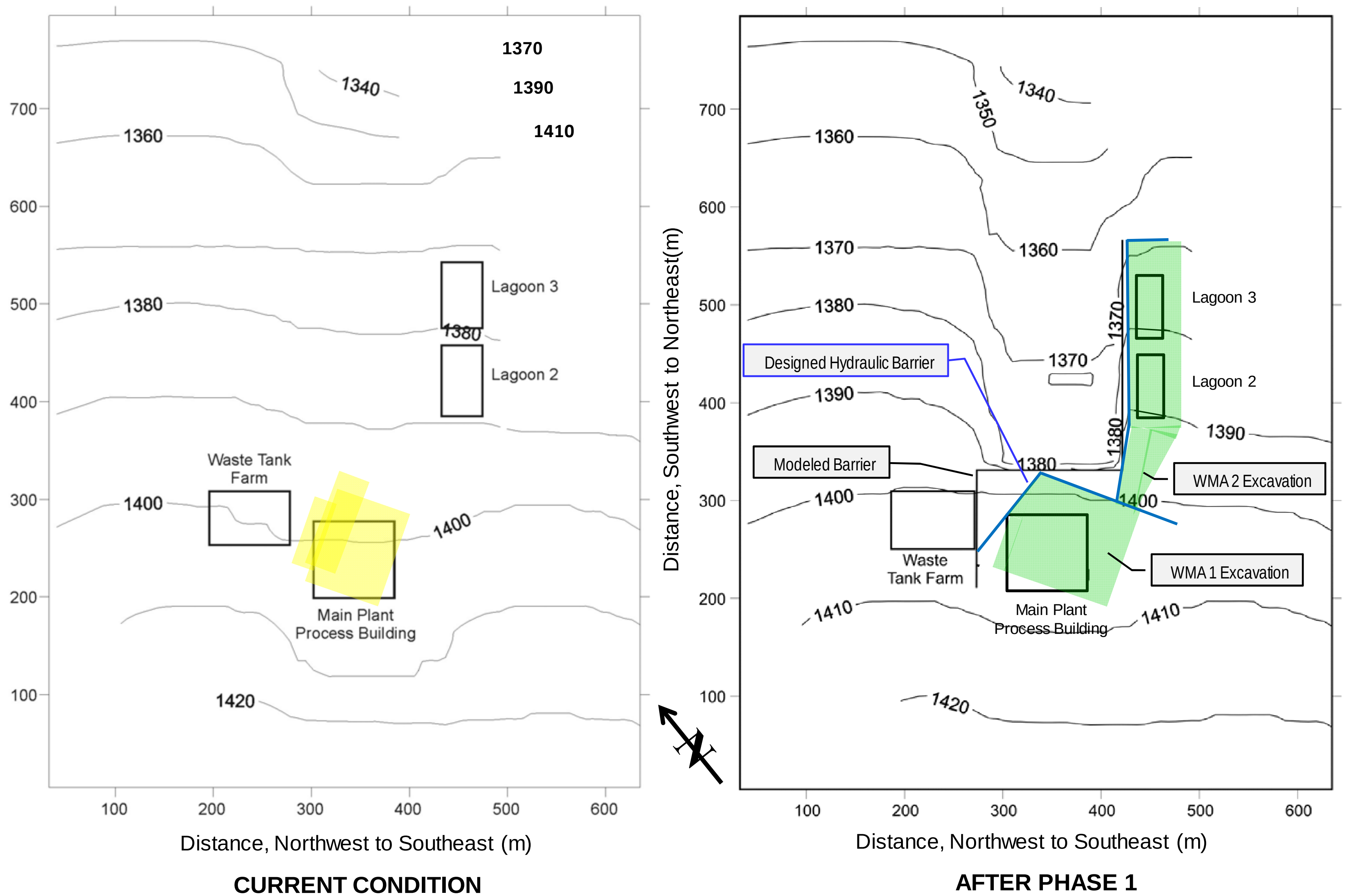


**Figure D-9. Groundwater Flow Associated with the WMA 1 and WMA 2 Engineered Barriers**

**Table D-5. Average Linear Velocity for Flow Path Originating at the Process Building Area After Phase 1**

| Distance Along Flow Path (m) | Average Linear Velocity (m/y) | |
|---|---|---|
| | Slack-water Sequence | Thick-bedded Unit |
| 0 to 40 | 81.0 | 81.2 |
| 40 to 80 | 79.2 | 82.2 |
| 80 to 120 | 22.5 | 1.9 |
| 120 to 160 | 61.2 | 1.8 |
| 160 to 200 | 104.3 | 1.9 |
| 200 to 240 | 95.6 | 6.0 |
| 240 to 280 | 112.6 | 84.7 |
| 280 to 320 | 131.3 | 111.5 |

NOTE: To convert meters per year to feet per year, multiply by 3.2803.

In calculation of linear velocities shown in Table D-5, the value of effective porosity of 0.35 was used for the slack-water sequence while the moisture content of the thick-bedded unit was used to reflect unsaturated conditions that develop along the flow path north of the location of the slurry wall. For the slack-water sequence and thick-bedded unit above the slack-water sequence, the travel time and average velocity along the flow path are 6.37 years and 50 meters per year and 70 years and 4.6 meters per year, respectively.

### 1.4.4 Groundwater Modeling Predictions for Conditions Following Phase 1

The revised near-field groundwater model for the north plateau suggests that the engineered barriers to be installed during Phase 1 decommissioning would have the following effect on groundwater flow in the north plateau:

- Groundwater flow patterns upgradient of the WMA 1 barrier wall and French drain would be similar to current flow patterns in the sand and gravel unit shown in Figure D-9.
- Water table elevations in WMA 1 would be approximately 10 feet higher on the upgradient side of the northeastern segment of the WMA 1 barrier wall compared to water levels immediately downgradient of this wall segment.
- This steep hydraulic gradient suggests that groundwater would preferentially flow from the backfilled WMA 1 excavation to the northeast 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.
- Groundwater elevations coincide on either side of the northwestern segment of the WMA 1 barrier wall separating the backfilled WMA 1 excavation from the Waste Tank Farm, suggesting low potential for groundwater flow across the barrier wall from either the backfilled excavation or Waste Tank Farm.

- Flow contours southeast of the WMA 1 barrier wall suggest that groundwater would flow to the east into the area of the backfilled WMA 2 excavation, as discussed in Section 1.4.3 of this appendix.
- Downgradient of the WMA 1 barrier wall groundwater flow in the sand and gravel unit would continue to the northeast across the north plateau. However, the upgradient diversion of groundwater flow by the barrier wall system would result in an overall reduction in the hydraulic gradient of the non-source area of the north plateau groundwater plume.
- Groundwater elevations in the backfilled WMA 2 excavation are expected to be up to 10 feet higher than present in the non-source area of the north plateau groundwater plume.
- Higher groundwater elevations within the backfilled WMA 2 excavation suggests groundwater would flow across the WMA 2 barrier wall to the northwest and northeast toward the non-source area of the north plateau groundwater plume and also to the southeast toward Erdman Brook.

## 2.0 Conceptual Post-Remediation Site Monitoring and Maintenance

DOE will be responsible for maintaining institutional controls and for monitoring and maintenance of the project premises until the completion of Phase 2 of the WVDP decommissioning.

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 decommissioning. The Phase 1 program will 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 will continue until the start of Phase 2 of the decommissioning, when the program requirements will be re-evaluated. DOE concludes that this program will 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 decommissioning will 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, will also be routinely monitored and maintained as part of the DOE monitoring and maintenance program. Corrective actions will 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 will 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 will include routine inspections of these systems for signs of degradation or loss of performance.

**Hydraulic Barrier Walls**

A series of nested piezometers screened at different depth intervals will be installed at regular intervals upgradient and downgradient of the permanent hydraulic barrier walls installed downgradient of the WMA 1 and northwest of the WMA 2 excavations (Figure D-10) to monitor their performance. These piezometers will 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 will be routinely monitored to identify any changes in water levels that may indicate the development of defects within the barrier walls that require corrective action. Groundwater will be routinely sampled and analyzed for 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. Changes in groundwater concentrations of these radiological indicator parameters may identify defects associated with the barrier walls that require corrective action to limit the potential recontamination of the backfilled WMA 1 and WMA 2 excavations.

If groundwater monitoring suggests repairs to the walls are required, these repairs will 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 will 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, will be determined based on analysis of the groundwater level data, which will 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 will 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 – will 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 will 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 will 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 will be routinely monitored and maintained throughout Phase 1.

**Geomembrane Cover**

The geomembrane cover will be routinely inspected for signs of deterioration or damage to the membrane. The seams connecting the geomembrane panels will be inspected to evaluate their condition. The geomembrane cover will 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 will 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 will be routinely monitored during Phase 1 to evaluate the performance of the barrier wall in limiting groundwater flow into the NDA.

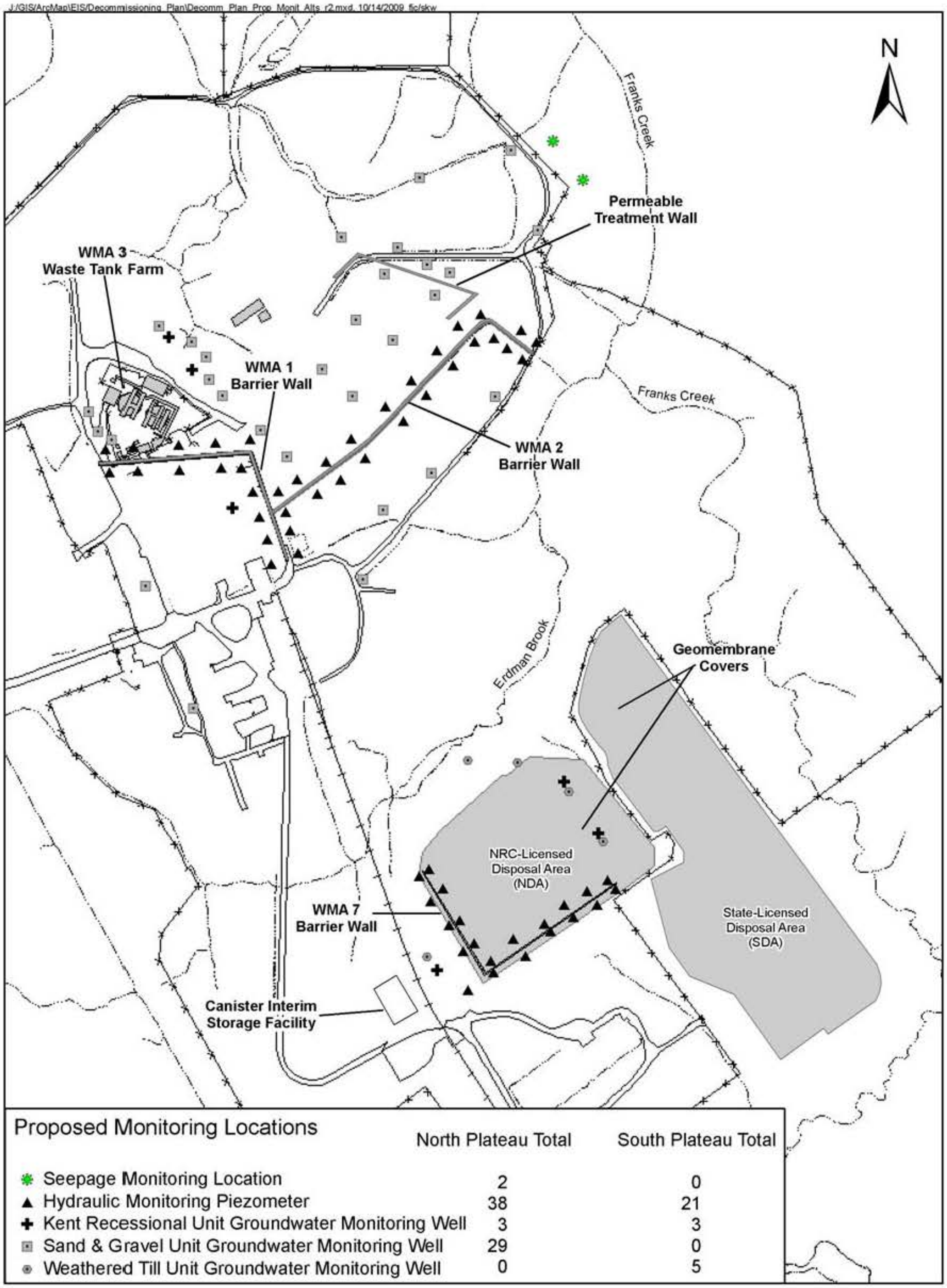


**Figure D-10. Groundwater Monitoring Locations within the Project Premises during the Phase 1 Institutional Control Period**

#### 2.1.5 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 will 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 decommissioning activities will 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 will not be considered during Phase 1 decommissioning but will 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 will 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 will include onsite groundwater, storm water, and air monitoring, and onsite and offsite surface water, sediment, and radiation monitoring as described below. Annual reports will be issued summarizing the monitoring results. These reports will 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 will 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 will 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-10. A total of 40 groundwater wells will 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 will 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 will 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 will 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 decommissioning.

Groundwater from these piezometers and monitoring wells will 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 will 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 will 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 will 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 will be routinely monitored at WMA 3 during the Phase 1 institutional control period. Eight wells will be screened in the sand and gravel unit with three wells upgradient and five wells downgradient of the Waste Tank Farm. Two wells screened in the Kent Recessional Sequence will be installed downgradient of the Waste Tank Farm.

Groundwater from these wells will 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 will 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 will 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 will 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 will 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 will be routinely monitored by five wells screened in the weathered Lavery till and three wells screened in the Kent Recessional Sequence. One well cluster will be located upgradient of the NDA and will 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, will be located downgradient of the burial area. The two remaining wells screened in the weathered Lavery till will be located downgradient of the burial area.

Groundwater at WMA 7 will 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 will 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 will 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 will be routinely monitored both within and outside the project premises during the Phase 1 institutional control period. The 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 will be routinely monitored during the Phase 1 institutional control period (Figure D-11). These locations monitor streams both within (WNDNKEL, WNSP005, WNNDADR, WNFRC67, WNERB53) and leaving the project premises (WNSW74A, WNSWAMP, and WNSP006). Sediment samples will be collected from three locations where surface waters leave the project premises (SNSW74A, SNSWAMP, and SNSP006).

Surface water will be routinely collected and analyzed from three sampling locations outside of the project premises (Figure D-12). These locations will 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 will be collected from all three off-site locations (SFBCSED, SFTCSED, SFCCSED).

Surface water and sediment samples will be collected from these locations semi-annually and will be analyzed for radiological indicator parameters (gross alpha, gross beta, and tritium).

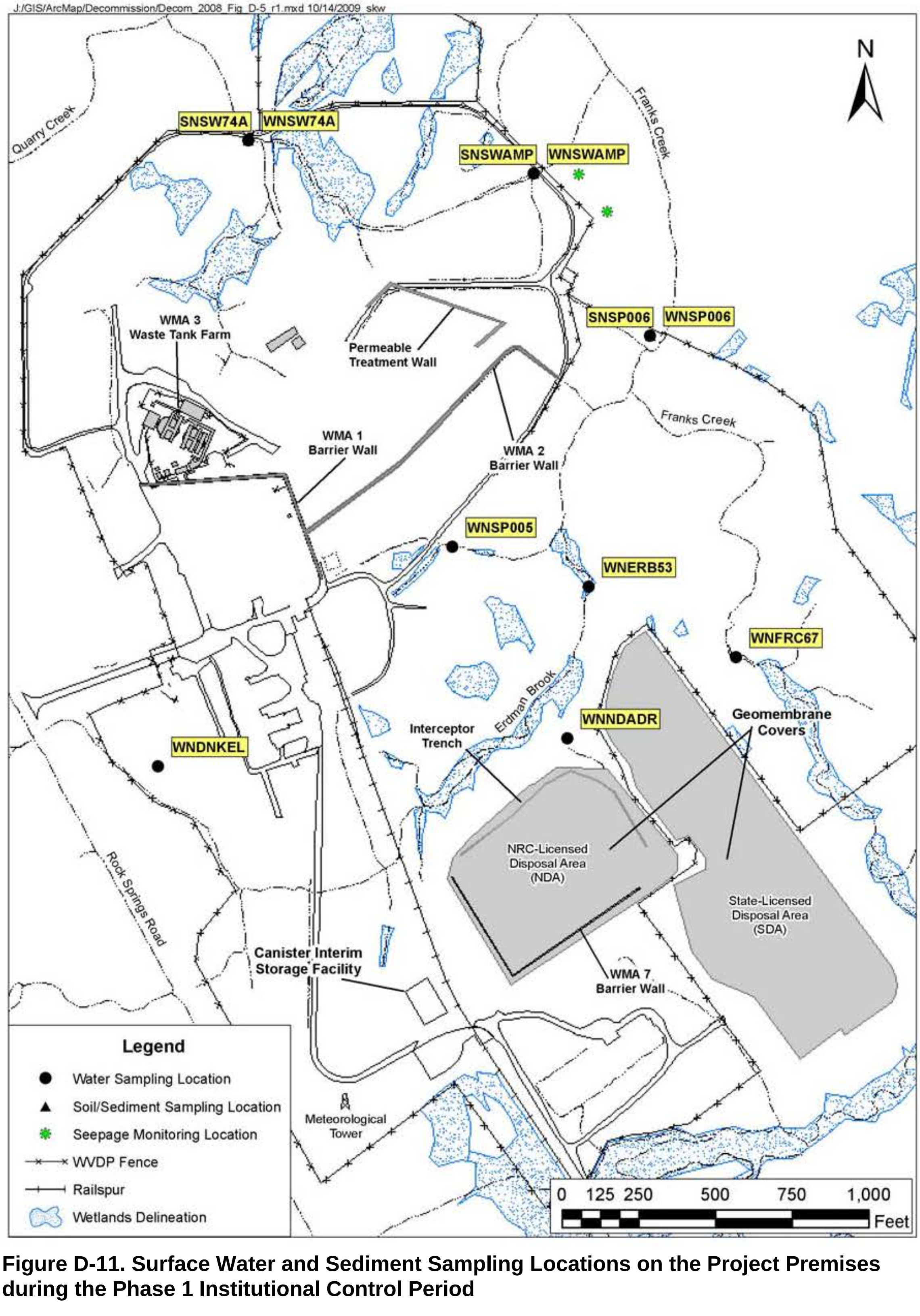


**Figure D-11. Surface Water and Sediment Sampling Locations on the Project Premises during the Phase 1 Institutional Control Period**

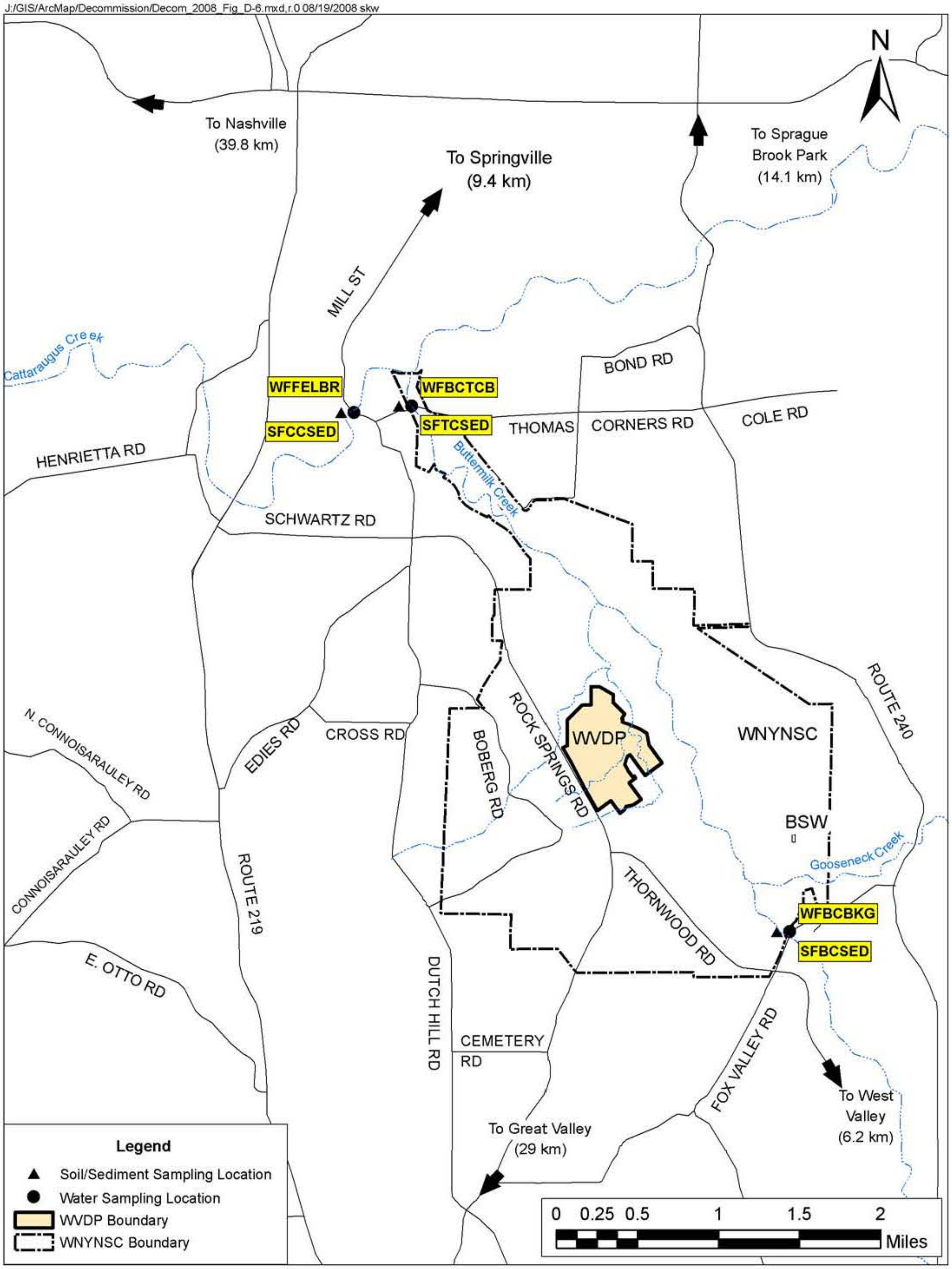


**Figure D-12 – Offsite Surface Water and Sediment Sampling Locations during the Phase 1 Institutional Control Period**

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-13. Sampling will 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 will 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-14).

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 will also be treated in the Permanent Ventilation System Building.

Air discharges from this location will 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 will 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 will be within the project premises (Figure D-15) and eleven stations will be located on the perimeter of the Center (Figure D-16). All locations will be routinely monitored for gamma radiation exposure on a quarterly monitoring schedule.

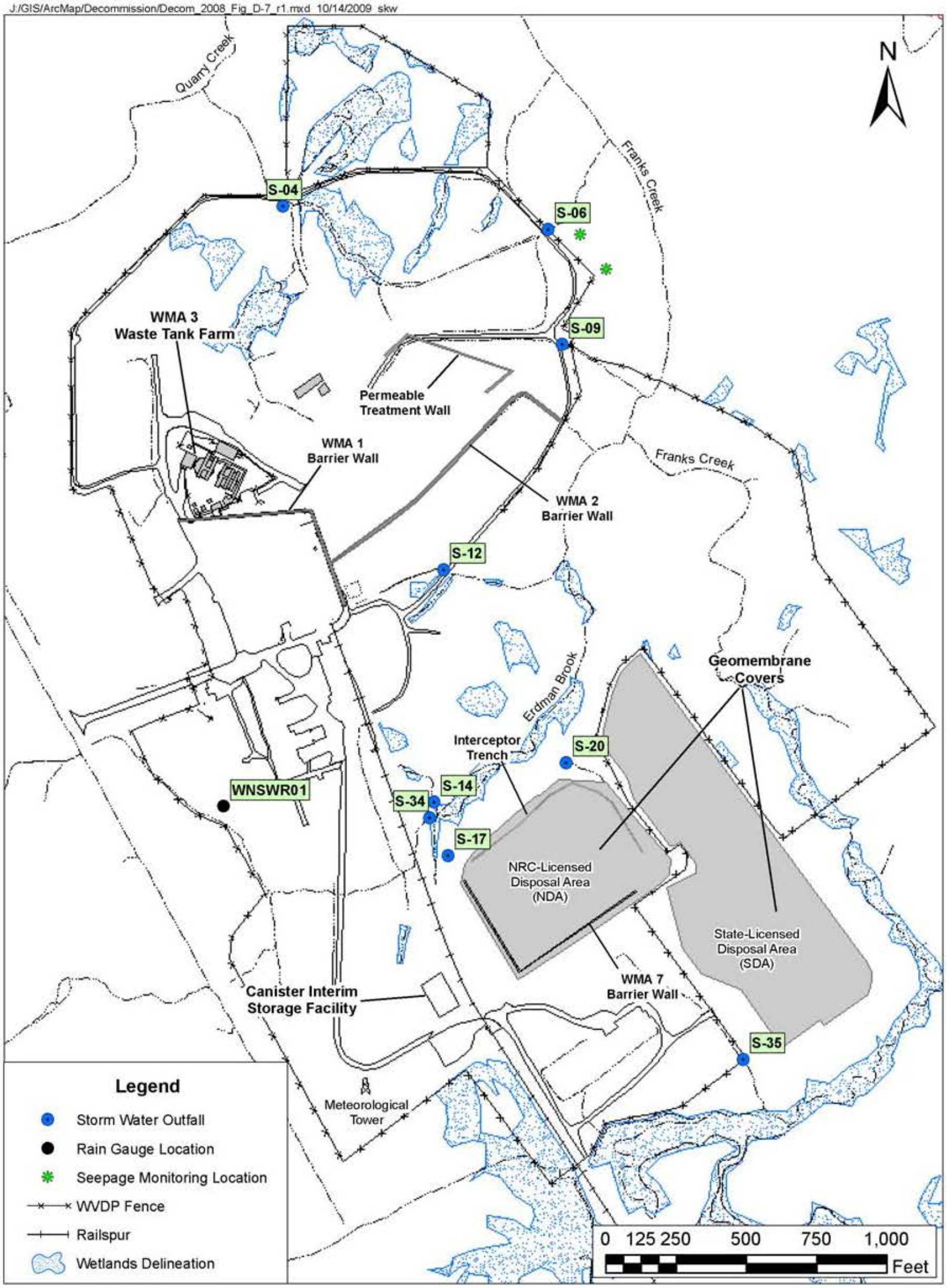


**Figure D-13. Storm Water Sampling Locations on the Project Premises during the Phase 1 Institutional Control Period**

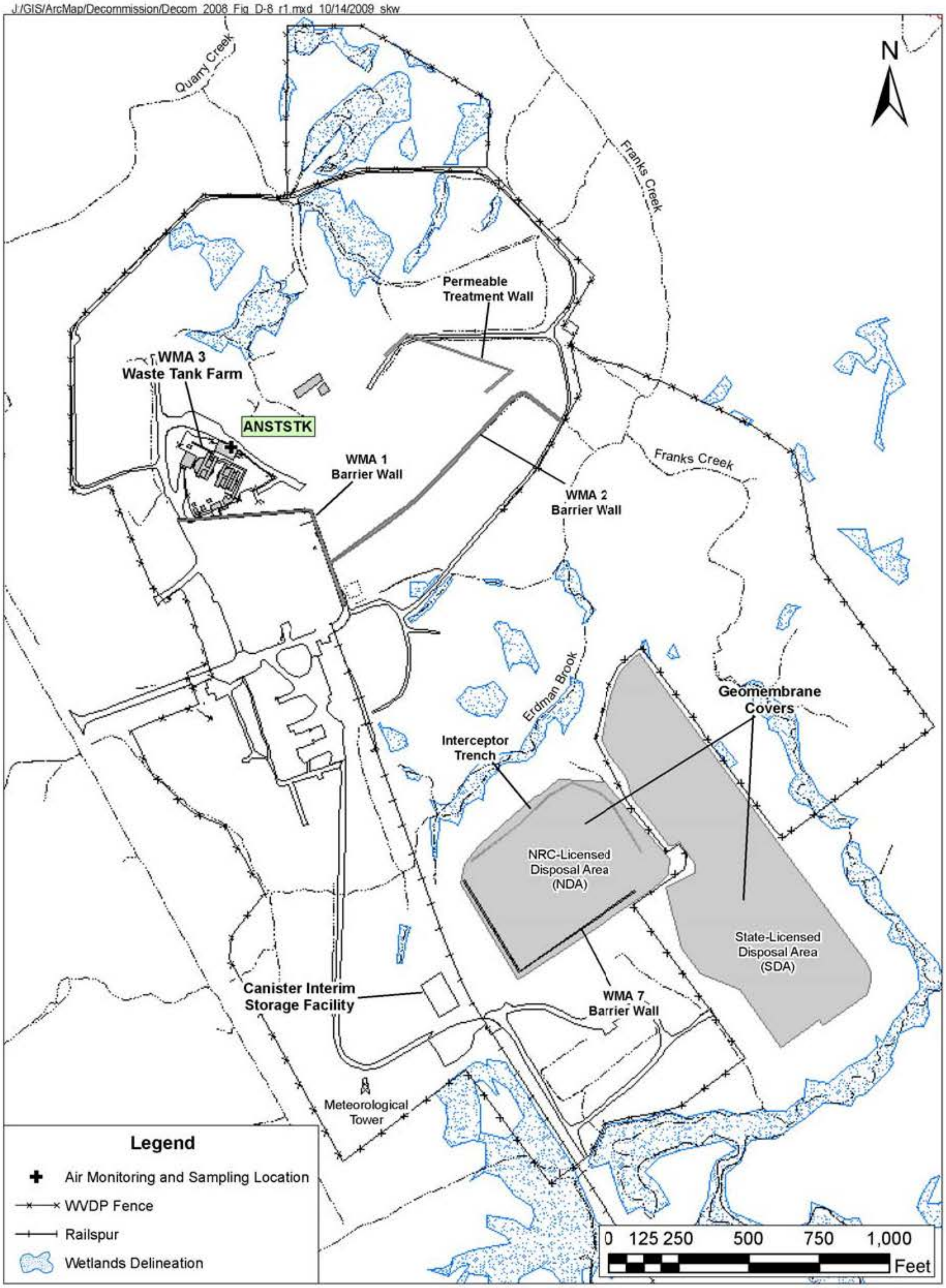


**Figure D-14. Air Monitoring Locations on the Project Premises during the Phase 1 Institutional Control Period**

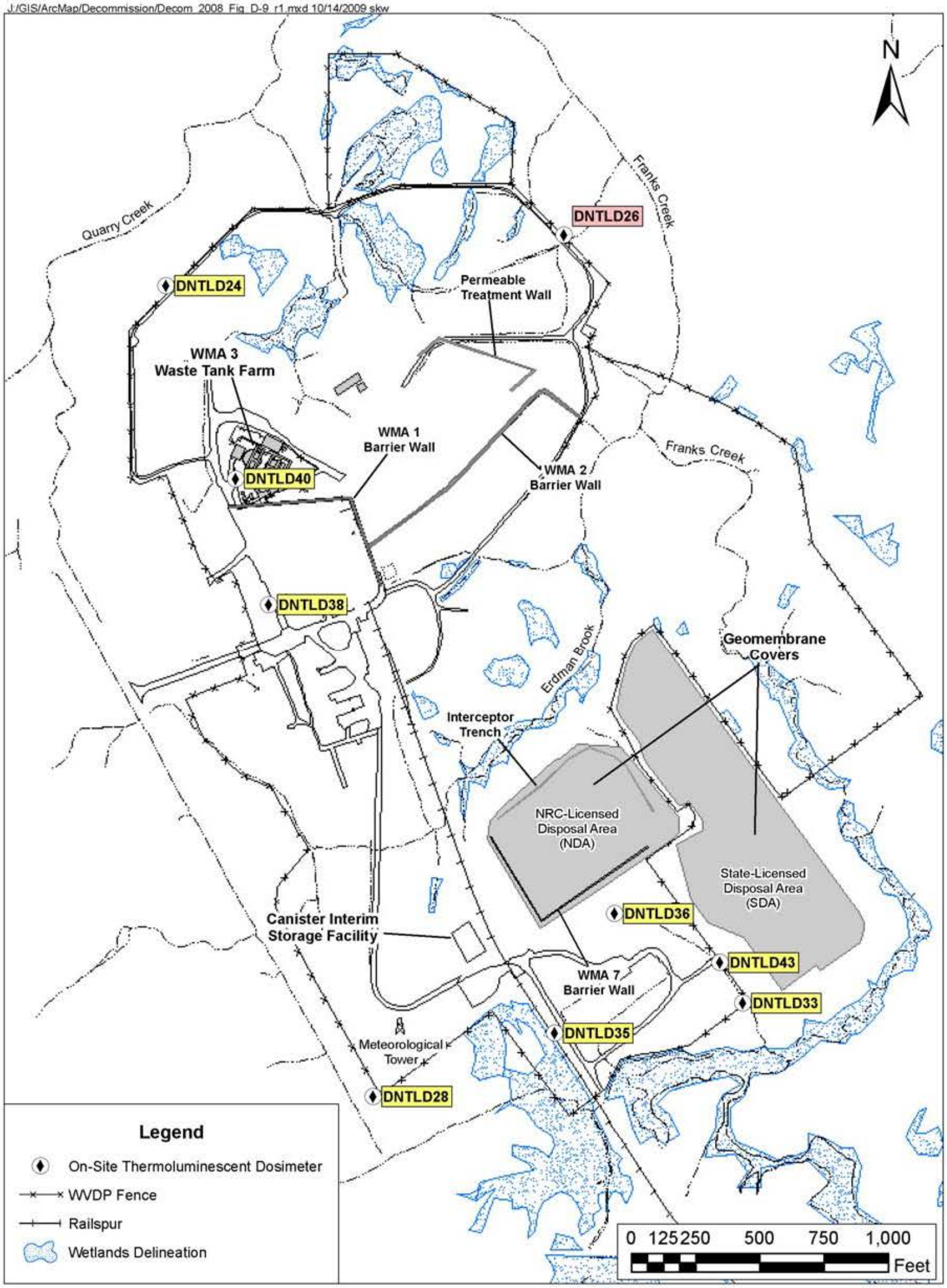


**Figure D-15 – Direct Radiation Monitoring Locations on the Project Premises during the Phase 1 Institutional Control Period**

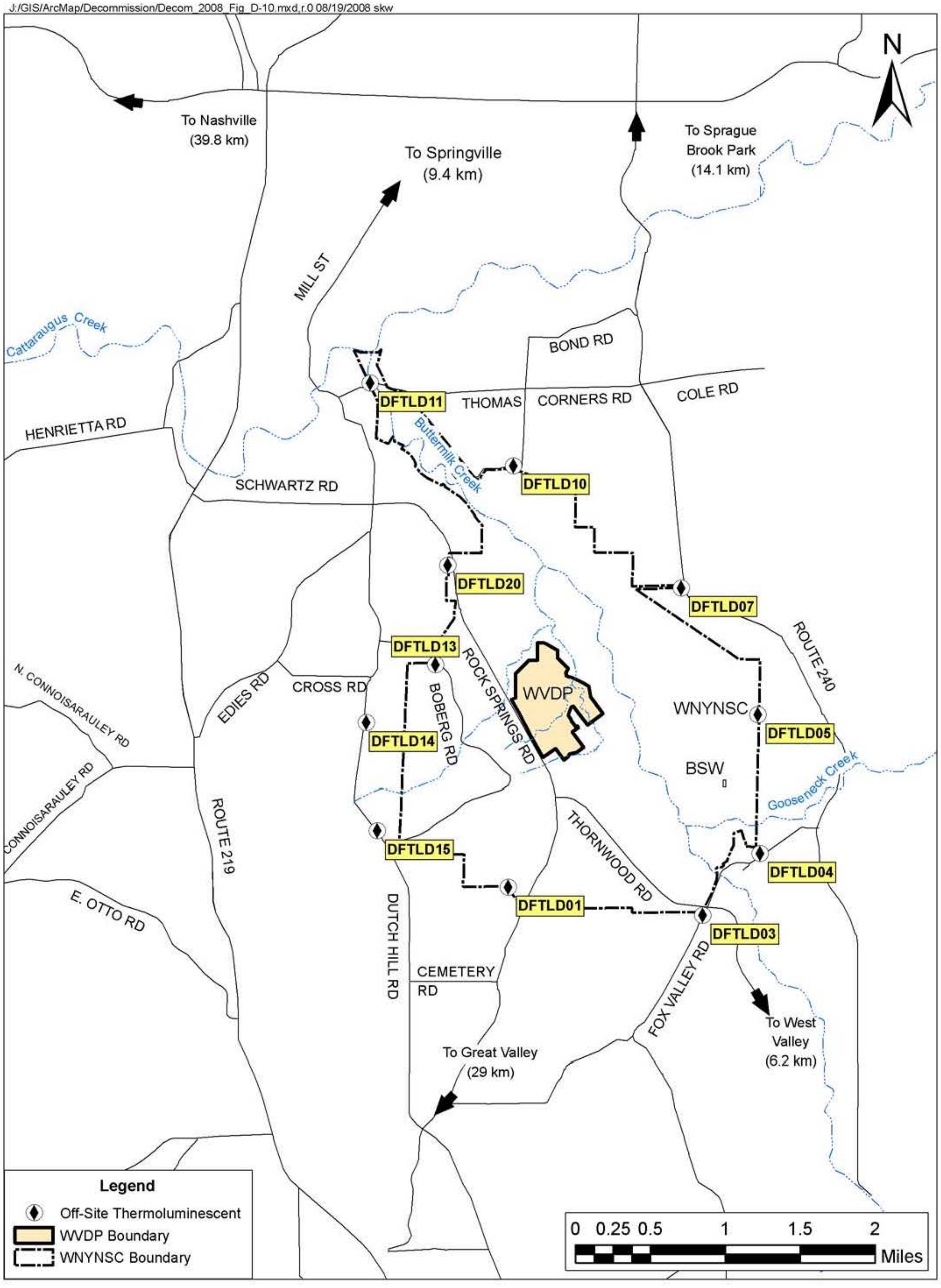


**Figure D-16. Offsite Direct Radiation Monitoring Locations during the Phase 1 Institutional Control Period**

## 3.0 Phase 1 Institutional Control Program

This section describes the institutional control program that will be implemented for the project premises during and 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 will 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 will therefore continue control of the project premises during the implementation of the Phase 1 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 will 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 will 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 will also be involved in a regulatory oversight capacity over the project premises, which will remain under NRC license.

### 3.2 Institutional Control Design Features

The institutional control program for the project premises will prevent its unacceptable use and protect against inadvertent intrusion into the site. DOE in its capacity as the steward of the site will ensure that institutional controls are maintained at the project premises during Phase 1 decommissioning and during the Phase 1 institutional control period. These institutional controls will 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 will 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 will ensure that operations at the site protect members of the public and the environment from radiation risk.

Additional institutional controls will be provided for the new Canister Interim Storage Facility on the south plateau. These will 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

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

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# APPENDIX E
# DOSE MODELING PROBABILISTIC UNCERTAINTY ANALYSES

**PURPOSE OF THIS APPENDIX**

The purpose of this appendix is to describe probabilistic uncertainty analyses performed to evaluate the degree of conservatism in key input parameters for the conceptual models used to develop derived concentration guideline levels (DCGLs) for surface soil, subsurface soil, and streambed sediment, along with the results of these analyses.

**INFORMATION IN THIS APPENDIX**

This appendix provides the following information:

- Section 1 provides introductory information to help place the discussions that follow into context.
- Section 2 defines key terms used in the discussions.
- Section 3 summarizes the probabilistic analysis capabilities of the RESRAD computer code used in the analyses.
- Section 4 describes criteria used for selecting parameters for uncertainty analysis.
- Section 5 describes how parameter distributions were selected.
- Section 6 describes correlation of parameters.
- Section 7 describes the uncertainty analysis results for each of the three conceptual models, including DCGLs expressed as the peak-of-the-mean ($50^{th}$ percentile) and $95^{th}$ percentile.
- Section 8 describes parameter output rank correlations.
- Section 9 provides conclusions and describes actions taken on the analysis results.
- Attachment 1 contains copies of representative probabilistic output plots.
- Attachment 2 contains the electronic files developed in performing the analyses.

**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 Introduction

## 1.1 Purpose

The probabilistic uncertainty analyses discussed in this appendix were performed to evaluate the degree of conservatism in key input parameters for the conceptual models used in developing DCGLs for surface soil, subsurface soil, and streambed sediment that are described in Section 5 of this plan. The DOE letter that forwarded Revision 0 of this plan to NRC for review (DOE 2008) noted that this matter was still under evaluation when Revision 0 was completed.

These probabilistic uncertainty analyses supplement the deterministic sensitivity analyses described in Section 5 of this plan. They compute the total uncertainty in the DCGLs resulting from the uncertainty in or the variability of the input parameters. They also help determine the relative importance of the contributions of different input parameters to the total uncertainty in the DCGLs.

These analyses thereby provide additional perspective on the relationships between conceptual model input parameters and estimated dose, along with sets of DCGLs expressed in probabilistic terms. This information supports a risk-informed approach to establishing cleanup goals for Phase 1 of the decommissioning.

## 1.2 Background

The DCGLs for surface soil, subsurface soil, and streambed sediment were developed using the basic RESRAD deterministic approach in which the analysis is performed by assigning each parameter a single value, as described in Section 5 of this plan. As noted in Section 5, RESRAD was selected as the mathematical model for DCGL development due to its extensive use by DOE and by NRC licensees in developing DCGLs and evaluating doses from residual radioactivity at decommissioned sites.

**General NRC Guidance on Uncertainty and Sensitivity Analyses**

NRC guidance on uncertainty and sensitivity analyses appears in Appendix I to NUREG-1757, Volume 2 (NRC 2006). NRC concludes that while the deterministic modeling approach has the advantage of being simple to implement and easy to communicate to a non-specialist audience, it has significant limitations:

- It does not allow consideration of the effects of unusual combinations of input parameters;
- It does not provide information on uncertainty in the results, which would be helpful to the decision-maker; and
- It often leads to overly conservative evaluations because it has to rely on the use of pessimistic estimates of each parameter of the model to ensure a bounding dose estimate, that is, results that are likely to overestimate the actual peak dose.

The first two limitations apply to the deterministic dose analysis described in Section 5, which did not include evaluation of different parameter combinations or estimates of uncertainty. And while DOE used conservative model input parameters in many cases, it is difficult to demonstrate that the results of the deterministic dose analysis are bounding.

NRC encourages the use of probabilistic techniques to evaluate and quantify the magnitude and effect of uncertainties in dose assessments, and the sensitivity of the calculated risks from individual parameter values and modeling assumptions. Probabilistic uncertainty analysis provides more information to the decision-maker than deterministic analysis, as it characterizes a range of potential doses and the likelihood that a particular dose may be exceeded. (NRC 2006)

Uncertainty analyses in the RESRAD probabilistic modules use Latin hypercube sampling[1], a modified Monte Carlo method, allowing for the generation of representative input parameter values from all segments of the input distributions. Input variables for the models are selected randomly from probability distribution functions for each parameter of interest. Parameter distribution functions may be either independent or correlated to other input variable distributions. The analysis is then performed hundreds of times to obtain a distribution of doses resulting from each set of randomly selected input parameters.

The results of a probabilistic uncertainty analysis provide a distribution of doses illustrating the effects of random combinations of input parameters. It should be recognized that some percentage of the calculated distribution of doses may exceed the regulatory limit, which is expressed as a (deterministic) single value. Compliance can be stated in terms of a metric of the distribution such as the mean falling below the limit, or only a percentage of calculated doses exceeding the limit. (NRC 2006)

NRC indicates that when using probabilistic dose modeling, the “peak-of-the-mean” dose distribution should be used for demonstrating compliance with its License Termination Rule in 10 CFR Part 20, Subpart E (NRC 2006).

**Specific NRC Guidance for Phase 1 of the WVDP Decommissioning**

DOE and NRC held two scoping meeting on DOE’s dose modeling plans. The NRC summary of the second meeting (NRC 2008) included the following statements:

> “NRC indicated that it might not be acceptable to use the mean or most likely value for those parameters that have the largest impact on dose in a deterministic analysis (e.g., for parameters such as $K_d$s that have a large parameter range and uncertainty).”

> “NRC warned of the potential pitfalls of performing a deterministic analysis with a sensitivity analysis in lieu of a probabilistic assessment. Depending on the combination and range of parameter values selected and models employed (e.g., mass balance versus non-dispersion model in RESRAD), key radionuclides and pathways, the results of the sensitivity analysis could be misleading and the full range of uncertainty difficult to determine. Selection of parameter values should be guided by conservative assumptions when uncertainty is large and cannot be reduced. To determine the impact of a particular parameter value on the dose results, DOE must identify key risk drivers and perform a comprehensive sensitivity analysis to ensure that its selection of parameter values in its deterministic analysis errors on the side of conservatism.”

DOE identified key risk (i.e., dose) drivers and included a comprehensive sensitivity analysis in Section 5.2.4 of Revision 1 to the plan. The analyses described in this appendix, complete DOE actions on these matters.

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[1] The Latin hypercube method is a modified Monte Carlo method; see Section 2 below for definitions of terms such as these. NRC supported development of the probabilistic version of RESRAD for use in determining compliance with its License Termination Rule (Yu, et al. 2000). RESRAD probabilistic modeling capabilities are discussed in Section 3 below.

### 1.3 Analyses and Associated Electronic Files

The probabilistic dose analyses discussed herein were performed using the probabilistic modules of RESRAD Version 6.4 (LePoire, et al. 2000; Yu, et al. 2000; Yu, et al. 2001) making use of the stratified sampling of the Latin hypercube method.

For the surface soil model, three groups of results were generated for 1000 sets of input parameters, with calculated statistical parameters (minimum, maximum, mean, percentiles) output by RESRAD for each of the three input parameter datasets. For the subsurface and streambed sediment models, use of the mass balance groundwater option results in long computation times for multiple parameter input sets. Therefore, only a single set of 1000 input values for each parameter was used for the subsurface soil and sediment evaluation where simulation times were extensive.

Included in the electronic files of Attachment 1 are the RESRAD input and output files for surface soil ("RESRAD PROB SURF.zip"), subsurface soil ("RESRAD PROB SUBS.zip"), and sediment ("RESRAD PROB SED.zip"), and a Word file containing output plots of dose over time for each radionuclide in each media ("PROB Dose Plots.doc").

### 1.4 Products of the Probabilistic Uncertainty Analyses

The primary products of these analyses are as follows:

- Sets of peak-of-the-mean $DCGL_W$ values for surface soil, subsurface soil, and streambed sediment, that is, values that have a 50 percent probability that the specified concentration for each radionuclide would correspond to a dose of 25 mrem in the year of peak dose;
- Sets of 95$^{th}$ percentile $DCGL_W$ values for surface soil, subsurface soil, and streambed sediment, that is, values that have a 95 percent probability that the specified concentration for each radionuclide would correspond to a dose of 25 mrem in the year of peak dose;
- Preliminary dose estimates for the remediated Waste Management Area (WMA) 1 excavation expressed as the peak of the mean (50$^{th}$ percentile) and the 95$^{th}$ percentile; and
- Preliminary dose estimates for the remediated WMA 2 excavation expressed as the peak of the mean and the 95$^{th}$ percentile.

> As discussed in Section 9.2 of this appendix, the results of the probabilistic uncertainty analyses indicate that some input parameters used in the deterministic modeling to develop DCGLs may not be sufficiently conservative to ensure bounding results.

## 2.0 Key Terms

Because of the technical nature of the discussions in this appendix, some readers may find the following definitions to be useful. These definitions are tailored to the use of the terms in this appendix.

**Behavioral parameter.** Any conceptual model input parameter whose value would depend on the receptor's behavior within the scenario definition. For the same group of receptors, a behavioral parameter value could change if the scenario changed, e.g., parameters for recreational use could be different from those for residential use. (See also **metabolic parameter** and **physical parameter**.)

**Correlation.** A measure of the strength of the relationship between two variables (e.g., conceptual model input parameters) used to predict the value of one variable given the value of the other.

**Correlation coefficient.** Correlation coefficients (R values) are expressed on a scale from -1.0 to +1.0, with the strongest correlations being at both extremes and providing the best predictions. Negative values reflect inverse relationships. (See also **partial rank correlation coefficient**.)

**Deterministic analysis.** In a deterministic analysis, each input parameter is assumed to be an exactly known single value, as are the analysis results.

**Empirical distribution.** An empirical distribution is a parameter distribution well defined by available data to the extent that additional sampling would not be expected to significantly change the distribution's shape.

**Latin hypercube sampling.** A modified **Monte Carlo method** used to generate random samples of input parameters in the probabilistic version of RESRAD.

**Lognormal distribution.** In a lognormal distribution, the logarithm of the parameter has a **normal distribution**. A lognormal distribution is defined by two parameters, the logarithmic mean and its standard deviation.

**Mean.** The arithmetic mean as used here is the mathematical average of a set of numbers. The mean is calculated by adding a set of values and dividing the total by the number of values in the set.

**Metabolic parameter.** A parameter representing the metabolic characteristics of the potential receptor that is independent of scenario. (Metabolic parameters were not included in the evaluation discussed in this appendix.)

**Monte Carlo method.** A technique which obtains a probabilistic approximation to the solution of a problem by using statistical sampling techniques. Monte Carlo methods rely on repeated random sampling to compute their results, and are often used to simulate complex physical and mathematical systems.

**Normal distribution.** Probability values in a normal distribution follow a bell shaped curve centered about a mean value with the width of the "bell" described by the standard deviation. In a bounded normal distribution, upper and lower limits to the range are specified.

**Overall coefficient of determination.** This coefficient, denoted by $R^2$, provides an indication of the variability in the overall radionuclide dose accounted for by the selected input parameters. It varies between 0 and 1; the higher the value, the greater the influence. A value of 0 indicates the selected parameters do not influence the calculated dose at all.

**Partial rank correlation coefficient.** The partial rank correlation coefficient measures the strength of the relationship between variables after any confounding influences of other variables have been removed. (See also **rank correlation coefficient**.)

**Peak of the mean.** The highest dose value in a plot of the estimated mean dose over time.

**Physical parameter**. Any parameter whose value would not change if a different group of receptors was considered. Physical parameters are site-specific factors determined by the source, its location, and geological or physical characteristics of the site.

**Probabilistic analysis.** In a probabilistic analysis, statistical distributions are defined for input parameters to account for their uncertainty, and the analysis results reflect the resulting uncertainty, e.g., a distribution of values rather than a single value. Such analyses use a random sampling method to select parameter values from a distribution. Results of the calculations appear in the form of a distribution of values.

**Probability density function.** A graphical representation of the probability distribution of a continuously random variable illustrating the range of possible values and the relative frequency (probability) of each value within the range. Uncertainty in a conceptual model input parameter is represented by the probability density function for that parameter. Probability distribution functions provided for in RESRAD include empirical, uniform, triangular, normal, and lognormal.

**Rank correlation coefficient.** A correlation coefficient between two variables that is used for determining the relative importance of input parameters in influencing the resultant dose.

**Regression analysis.** A mathematical method of modeling the relationships among three or more variables used to predict the value of one variable given the values of the others.

**Triangular distribution.** In a triangular distribution of a continuous random variable, the graph of the probability density function forms a triangle, with a range defined by minimum and maximum values and a mode value which is the most frequent (probable) value.

**Uniform distribution.** In a uniform distribution, each value within the range has the same probability of occurrence.

## 3.0 The Probabilistic Version of RESRAD

The probabilistic RESRAD code is an extended and enhanced version of RESRAD. RESRAD Version 6.4, which was used for the dose analyses described in Section 5 of this plan, provides both deterministic and probabilistic analysis capabilities.

The probabilistic version of RESRAD was developed for use in site-specific dose modeling in support of NRC's License Termination Rule compliance process for decontamination and decommissioning of NRC-licensed sites. Probabilistic analysis capabilities were incorporated into RESRAD in external software modules integrated into the code. Three reports describe these probabilistic analyses capabilities and how they are applied:

- NUREG/CR-6676, *Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes* (Kamboj, et al. 2000);

- NUREG/CR-6692, *Probabilistic Modules for the RESRAD and RESRAD-Build Computer Codes, User Guide* (LePoire, et al. 2000); and
- NUREG/CR-6697, *Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes* (Yu, et al. 2000).

Three basic types of input parameters are considered in probabilistic analyses: physical parameters, behavioral parameters, and metabolic parameters[2]. Certain parameters fall into more than one category, e.g., inhalation rate is both a behavioral parameter and a metabolic parameter.

The probabilistic modules in RESRAD Version 6.4 provide default values and distributions for various parameters. Default probability distributions include normal, lognormal, uniform, triangular, and empirical. These default distributions are based primarily on the quantity of relevant data available in reviewed technical literature.[3] For three parameters of interest in this plan – cover depth, precipitation rate, and well pumping rate – a default distribution type is not provided.

In a RESRAD probabilistic analysis, the results from all input samples are analyzed and presented in a statistical format in terms of the average value, standard deviation, minimum value, and maximum value. The cumulative probability distribution of the output is presented in both tabular and graphical forms.

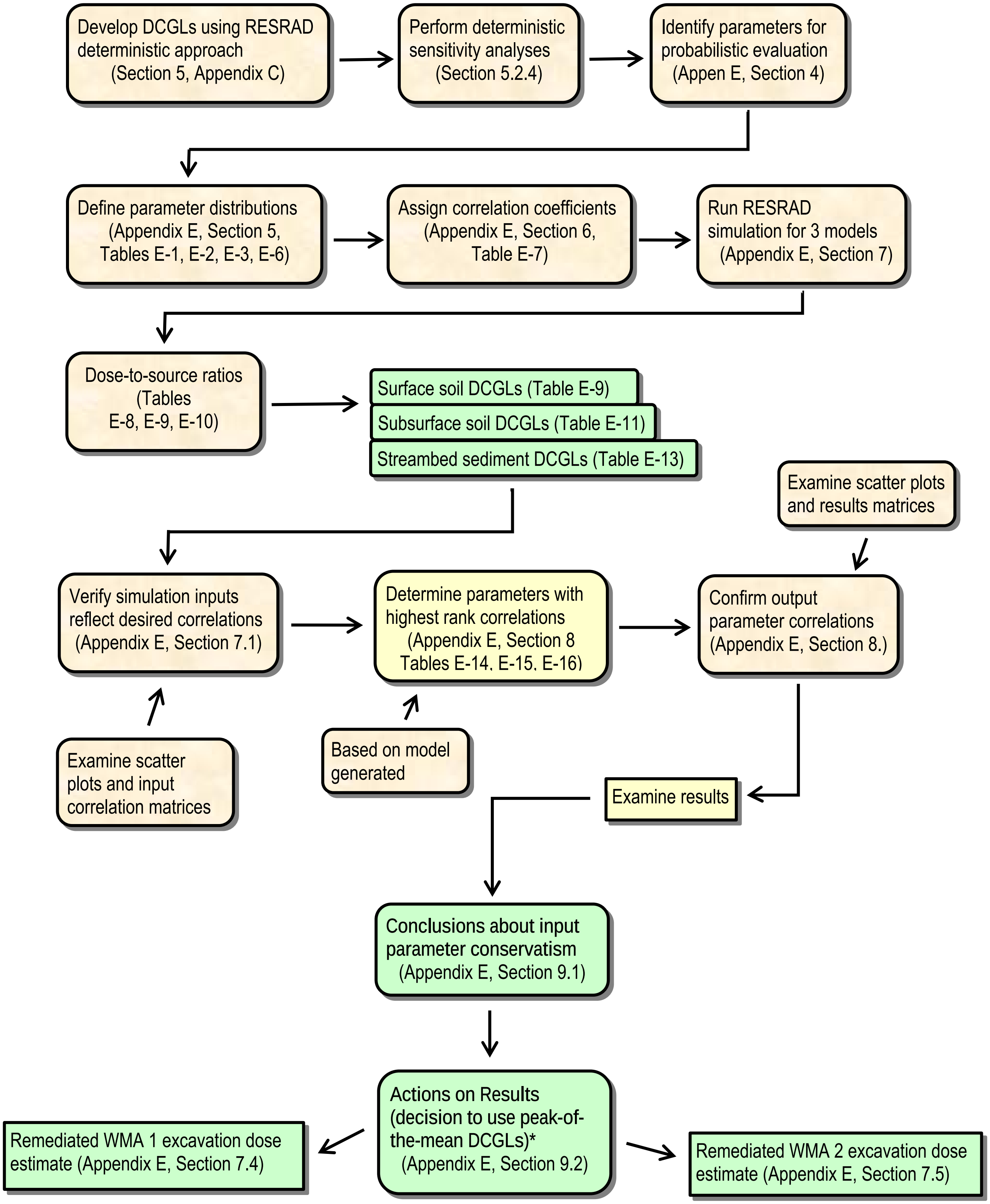
The basic process includes the following steps:

- Identifying parameters for probabilistic evaluation;
- Defining distributions of key parameters;
- Assigning correlations between input parameters, which is done to limit the occurrence of unrealistic physical conditions;
- Verifying that simulation input values reflect the desired correlations by visual inspection of scatter plots of correlated parameters;
- Determining parameters with highest rank correlation coefficients in the results, i.e., those that most influence dose; and
- Confirming output parameter correlations with scatter plots of parameter input values versus calculated dose.

Figure E-1 illustrates the process.

[2] Metabolic parameters were not included in this evaluation because the deterministic values represent means for the generic population, which would be independent of site conditions (Kamboj, et al. 2000).

[3] Parameter distributions developed for use with RESRAD and RESRAD-BUILD and their bases are described in Attachment C to NUREG/CR-6697 (Yu, et al. 2000).




*For surface soil and streambed sediment. See Section 5.2.8 for subsurface soil DCGLs.

**Figure E-1. Probabilistic Uncertainty Analysis Process**

## 4.0 Key Parameter Selection

The main criteria used for identifying key parameters to be evaluated involved the expected parameter influence on dose variability. That is, key parameters are those that have the largest effect on the dose analysis results.

Section 5.2.4 of this plan describes the results of sensitivity analyses for key input parameters for each of the three conceptual models. Tables E-1, E-2, and E-3 identify key parameters for the three conceptual models described in Section 5 of the plan, along with their assigned distributions, which are discussed in the next section.

Section 5.2.4 identifies Sr-90 and Cs-137 as likely to be the primary dose drivers for surface soil, subsurface soil and sediment exposure pathways. However, all eighteen radionuclides of interest were evaluated in the probabilistic analyses for the sake of completeness.

Other factors considered in parameter selection included the availability of site-specific information that could be used to define the distributions and NRC guidance on potentially significant parameters. Preference was also given to including parameters for which input correlations with other input variables could be defined, and where ambiguous input correlations with other input parameters was limited. Additionally, a number of parameters were used to establish a site-specific dilution factor (See Appendix C) corroborated by the detailed three dimensional flow model. These parameters were not varied with the exception of hydraulic conductivity, well pumping rate and length parallel to aquifer flow. For these parameters the probabilistic evaluation included values that would vary the dilution factor within a reasonable site-specific range.

Initial probabilistic simulations included parameters such as soil density, total porosity, and effective porosity for the contaminated, unsaturated, and saturated zones. These parameters consistently had correlation coefficients below 0.25. Because the correlation of these parameters with other more significant input parameters (i.e. hydraulic conductivity) was not clear, these parameters were dropped from subsequent analysis. Additional information regarding parameter input correlation is provided in Section 6.0.

## 5.0 Parameter Distribution Selection

This section first addresses the statistical distributions of model input parameters other than $K_d$ values and then addresses $K_d$ values.

### 5.1 Parameters Other Than Distribution Coefficients

Distributions selected for the input parameters are presented in Tables E-1, E-2, and E-3, and were based on applicable guidance in NUREG/CR-6676 (Kamboj, et al. 2000) and NUREG/CR-6697 (Yu, et al. 2000). Site specific parameters were generally assigned triangular distributions centered on the most likely value (e.g., source thickness, contaminated length parallel to aquifer flow).

Table E-1 identifies parameters of interest and their assigned distributions for the surface soil conceptual model that were varied during the analyses and the distribution used for each parameter, except for distribution coefficients and the plant, meat and milk biotransfer factors. The distribution coefficients for all ten elements associated with the radionuclides of interest were also varied using bounded lognormal distributions.

**Table E-1. Input Parameter Distributions for Surface Soil Model (Other than $K_d$ and Biotransfer Factor Values)[(1)(2)]**

| RESRAD Parameter | Parameter Description | Units | Distribution | Parameters[(3)] | | | |
|---|---|---|---|---|---|---|---|
| THICK0 | Contaminated zone thickness | m | triangular | 0.5 | 1 | 3 | |
| LCZPAQ | Length parallel to aquifer flow | m | triangular | 100 | 165 | 200 | |
| HCSZ | Saturated zone hydraulic conductivity | m/y | triangular | 630 | 1400 | 2200 | |
| UW | Well pumping rate | $m^3/y$ | bounded normal | 5900 | 1270 | 2618 | 7586 |
| RI | Irrigation rate | m/y | bounded normal | 0.47 | 0.12 | 0.14 | 0.64 |
| FIND | Indoor time fraction | none | triangular | 0.45 | 0.66 | 0.8 | |
| FOTD | Outdoor time fraction | none | triangular | 0.1 | 0.25 | 0.45 | |
| HCUZ(1) | Unsaturated zone hydraulic conductivity | m/y | triangular | 63 | 140 | 220 | |
| HCCZ | Contaminated zone hydraulic conductivity | m/y | triangular | 63 | 140 | 220 | |
| DROOT | Root depth | m | triangular | 0.3 | 0.9 | 3 | |
| PRECIP | Precipitation rate | m/y | bounded normal | 1.03 | 0.13 | 0.86 | 1.36 |
| THICK0 | Contaminated zone thickness | m | triangular | 0.5 | 1 | 3 | |
| SHF1 | External gamma shielding factor | none | triangular | (4) | (4) | (4) | |

NOTES: (1) Values in RESRAD file "SUMMARY.REP".

(2) Radionuclide specific $K_d$ values were varied (see Table E-6) and plant, meat, milk transfer factors were assigned the RESRAD default distribution.

(3) Parameters for the distributions are: TRIANGULAR - minimum, mode, maximum and BOUNDED NORMAL - mean, standard deviation, minimum, maximum.

(4) Radionuclide specific distribution. Dose drivers Cs-137 and U-232 were evaluated.

In general, site-specific physical parameters in Table E-1 were described with triangular distributions across the range of values associated with the site, including hydraulic conductivity, and indoor/outdoor time fraction, etc. Depth of roots was assigned a triangular distribution ranging from 0.3 meter (onions, lettuce) to three meters (alfalfa), centered on 0.9 m (corn).

Precipitation was based on a normal distribution described by statistical parameters (mean = 1.03 meter, standard deviation = 0.13 meter) that were calculated from meteorological data collected over the last 30 years in Buffalo, New York (http://www.weatherexplained.com/Vol-4/2001-Buffalo-New-York-BUF.html). The precipitation data was then used to assign a distribution for the irrigation rate, assuming that a total of 1.5 m/y of applied water was needed, and the well pumping rate was assigned a distribution based on the irrigation volume needed. These parameters were also correlated to ensure this relationship in the input values.

The total onsite fraction of 0.91 equates to a total of 33 days each year, or 15 hours each week, away from the site inclusive of time spent taking livestock/crops to market, assisting on neighboring farms, or other travel off-site (vacation, family occasions, religious services, etc.).

The plant-soil, meat-soil, and milk-soil bioaccumulation factors were simulated using the RESRAD default lognormal-N distributions, and were correlated (R = -0.87) with the $K_d$ as described in Section 6.0.

Table E-2 identifies parameters of interest and their assigned distributions for the subsurface soil conceptual model, except for distribution coefficients and the plant, meat and milk biotransfer factors, that were varied during the analyses and the distribution used for each parameter. The distribution coefficients for all ten elements associated with the radionuclides of interest were also varied using bounded lognormal distributions.

**Table E-2. Input Parameter Distributions for Subsurface Soil Model (Other than $K_d$ and Biotransfer Factor Values)(1)(2)**

| RESRAD Parameter | Parameter Description | Units | Distribution | Parameters(3) | | | |
|---|---|---|---|---|---|---|---|
| UW | Well pumping rate | $m^3/y$ | bounded normal | 5900 | 1270 | 2618 | 7586 |
| RI | Irrigation rate | m/y | bounded normal | 0.47 | 0.12 | 0.14 | 0.64 |
| FIND | Indoor time fraction | none | triangular | 0.45 | 0.66 | 0.8 | |
| FOTD | Outdoor time fraction | none | triangular | 0.1 | 0.25 | 0.45 | |
| DROOT | Root depth | m | triangular | 0.3 | 0.9 | 3 | |
| PRECIP | Precipitation rate | m/y | bounded normal | 1.03 | 0.13 | 0.86 | 1.36 |
| SHF1 | External gamma shielding factor | none | triangular | (4) | (4) | (4) | |

NOTES: (1) Values in RESRAD file "SUMMARY.REP".

(2) Radionuclide specific $K_d$ values were varied (see Table E-6) and plant, meat, milk transfer factors were assigned the RESRAD default distribution.

(3) Parameters for the distributions are: TRIANGULAR - minimum, mode, maximum and BOUNDED NORMAL - mean, standard deviation, minimum, maximum.

(4) Radionuclide specific distribution. Dose drivers Cs-137 and U-232 were evaluated

Because the subsurface soil model is based on the well drilling scenario, only a limited amount of material is available from the excavation ( approximately 30 $m^3$). The parameter ranges and correlation described below were selected assuming deterministic values for the contaminated zone area and depth. The sensitivity of the models to specific area and thickness combinations was evaluated in Section 5 of the body of this plan. Note that the subsurface soil evaluation is based on the mass balance groundwater model.

The plant-soil, meat-soil, and milk-soil bioaccumulation factors were simulated using the RESRAD default lognormal-N distributions, and were correlated (R = -0.87) with the $K_d$ as described in Section 6.0.

Table E-3 identifies parameters of interest and their assigned distributions for the streambed sediment conceptual model, except for distribution coefficients and the plant and meat biotransfer factors, that were varied during the analyses and the distribution used for each parameter. The distribution coefficients for all ten elements associated with the radionuclides of interest were also varied using bounded lognormal distributions

**Table E-3. Input Parameter Distributions for Streambed Sediment Model (Other than $K_d$ and Biotransfer Factor Values)(1)(2)**

| RESRAD Parameter | Parameter Description | Units | Distribution | Parameters(3) | | | |
|---|---|---|---|---|---|---|---|
| HCCZ | Contaminated zone hydraulic conductivity | m/y | triangular | 63 | 140 | 220 | |
| PRECIP | Precipitation rate | m/y | bounded normal | 1.03 | 0.13 | 0.86 | 1.36 |
| FOTD | Outdoor time fraction | none | triangular | 0.006 | 0.012 | 0.024 | |

NOTES: (1) Values in RESRAD file "SUMMARY.REP"..

(2) Radionuclide specific $K_d$ values were varied (see Table E-6) and plant, meat, fish transfer factors were assigned the RESRAD default distribution.

(3) Parameters for the distributions are: TRIANGULAR - minimum, mode, maximum and BOUNDED NORMAL - mean, standard deviation, minimum, maximum.

Soil parameters were varied over the same ranges used for the soil models. Parameter values for the fraction of time outdoors were taken from the deterministic sensitivity analysis described in Section 5 of the plan for likely recreational exposures.

The plant-soil and meat-soil bioaccumulation factors were simulated using the RESRAD default lognormal-N distributions, and were correlated (R = -0.87) with the $K_d$ as described previously. Fish transfer factors were also simulated using the RESRAD default lognormal-N distributions, however no correlations were included.

### 5.2 Distribution Coefficients

Table C-2 of this plan identifies the distribution coefficients ($K_d$ values) used in the dose analyses described in Section 5 of the body of this plan. Section 3.7.8 and Table 3-20 of this plan provide information on measurements of the distribution coefficients in soils at the site. However, these data are not sufficient to establish a site-specific distribution of the $K_d$ parameter for each of the 10 chemical elements represented in the 18 radionuclides of interest in dose modeling.

Sheppard and Thibault (Sheppard and Thibault 1990) and NUREG/CR-6697 (Yu, et al. 2000) recommend that the $K_d$ parameter be described as a lognormal distribution. Table E-4 summarizes data on $K_d$ values from two key sources compared to the values used in the dose modeling described in Section 5 of this plan. Table E-5 provides a summary of the parameters describing the lognormal distributions as given in these reports.

Consideration of the data in Table E-5 from the two sources led to the distribution parameters in Table E-6, which were used in the uncertainty analyses. The distributions were bounded based on the values presented in Table E-6 to constrain unreasonably large or small values, which is consistent with the approach suggested in NUREG-6697 (Attachment C). The values in the table were established as follows:

- When Sheppard and Thibault sand values were used for $K_d$ in the basic RESRAD analysis, then the Sheppard and Thibault sand distribution was used in the uncertainty analysis; and
- For cases when WVDP site-specific values are available, a distribution was selected so that the distribution mean [exp(μ)] provides a closer approximation to the $K_d$ used in the basic RESRAD analyses.

**Table E-4. Summary of Data on $K_d$ Parameter (mL/g) for the 10 Elements of Interest**

| Element | RESRAD Default | Geometric Mean and Range [Sheppard and Thibault 1990] | | | | Range [EPA 1999] [EPA 2004] | Values Used in Section 5 Modeling | |
|---|---|---|---|---|---|---|---|---|
| | | Sand | Loam | Clay | Organic | | Surface Soil, Unsaturated Zone, Saturated Zone | Subsurface Soil and Sediment in Contaminated Zone |
| Am | 20 | 1,900<br>8.2 – 300,000 | 9,600<br>400 – 48,309 | 8,400<br>25 – 400,000 | 112,000<br>6,398 – 450,000 | 8.2 - 2,270,000 | 1900 (1)<br>(420 - 111,000) | 4000 (2)<br>(420 - 111,000) |
| C | 0 | 5 | 20 | 1 | 7 | not addressed | 5 (1)<br>(0.7 - 12) | 7 (2)<br>(0.7 - 12) |
| Cm | calculated | 4,000<br>780 – 22,970 | 18,000<br>7,666 – 44,260 | 6,000<br>ND | 6,000<br>0 | 93 – 51,900 | calculated | calculated |
| Cs | 460 | 280<br>0.2 – 10,000 | 4,600<br>560 – 61,287 | 1,900<br>37 – 31,500 | 270<br>0.4 – 145,000 | 10 – 66,700 | 280 (1)<br>(48 - 4800) | 480 (2)<br>(48 - 4800) |
| I | calculated | 1<br>0.04 - 81 | 5<br>0.1 - 43 | 1<br>0.2 - 29 | 25<br>1.4 - 368 | 0.05 – 10,200 | 1 (1)<br>(0.4 - 3.4) | 2 (3)<br>(0.4 - 3.4) |
| Np | calculated | 5<br>0.5-390 | 25<br>1.3-79 | 55<br>0.4-2,575 | 1200<br>857-1,900 | 0.36 – 50,000 | 2.3 (4)<br>(0.5 - 5.2) | 3 (2)<br>(0.5 - 5.2) |
| Pu | 2,000 | 550<br>27-36,000 | 1200<br>100-5,933 | 5100<br>316-190,000 | 1900<br>60-62,000 | 5 – 2,550 | 2600 (4)<br>(5 - 27,900) | 3000 (2)<br>(5 - 27,900) |
| Sr | 30 | 15<br>0.05-190 | 20<br>0.01-300 | 110<br>3.6-32,000 | 150<br>8-4800 | 1 -1,700 | 5 (5)<br>(1 - 32) | 15 (2)<br>(1 - 32) |
| Tc | 0 | 0.1<br>0.01-16 | 0.1<br>0.01-0.4 | 1<br>1.16-1.32 | 1<br>0.02-340 | 0.01 – 340 | 0.1 (1)<br>(0.01 - 4.1) | 4.1 (3)<br>(1 - 10) |
| U | 50 | 35<br>0.03-2,200 | 15<br>0.2-4,500 | 1600<br>46-395,100 | 410<br>33-7,350 | 0.4 – 1,000,000 | 35 (1)<br>(15 - 350) | 10 (3)<br>(1 - 100) |

NOTES: (1) From Sheppard and Thibault 1990, for sand.
(2) Site specific value for the unweathered Lavery till (see Section 3.7.8, Table 3-20).
(3) Site specific value for the Lavery till (see Section 3.7.8, Table 3-20).
(4) Site specific value for the sand and gravel unit (see Section 3.7.8, Table 3-20).
(5) Dames and Moore (1995a, 1995b).

**Table E-5. Lognormal Distribution Parameters for $K_d$ Values from Literature**

| Element | Sand Soil(1) | | | | Clay Soil(2) | | | | RESRAD Default(3) | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | No. of Obs. | μ(4) | σ(5) | exp(μ)(6) | No. of Obs. | μ(4) | σ(5) | exp(μ)(6) | No. of Obs. | μ(4) | σ(5) | exp(μ)(6) |
| Am | 29 | 7.6 | 2.6 | 1,998 | 11 | 9.0 | 2.6 | 8,100 | 219 | 7.28 | 3.15 | 1,451 |
| C | 3 | 1.1 | 0.8 | 3 | 0(7) | 0.8 | | 2.2 | NA | 2.40 | 3.22(8) | 11 |
| Cm | 2 | 8.4 | 2.4 | 4,447 | 0(7) | 8.7 | | 6,000 | 23 | 8.82 | 1.82 | 6,761 |
| Cs | 81 | 5.6 | 2.5 | 270 | 28 | 7.5 | 1.6 | 1,810 | 564 | 6.10 | 2.33 | 446 |
| I | 22 | 0.04 | 2.2 | 1.0 | 8 | 0.5 | 1.5 | 1.7 | 109 | 1.52 | 2.19 | 4.6 |
| Np | 16 | 1.4 | 1.7 | 4.1 | 4 | 4.0 | 3.8 | 55 | 77 | 2.84 | 2.25 | 17 |
| Pu | 39 | 6.3 | 1.7 | 545 | 18 | 8.5 | 2.1 | 4,920 | 205 | 6.86 | 1.89 | 953 |
| Sr | 81 | 2.6 | 1.6 | 13.5 | 24 | 4.7 | 2.0 | 110 | 539 | 3.45 | 2.12 | 32 |
| Tc | 19 | -2.0 | 1.8 | 0.1 | 4 | 0.2 | 0.06 | 1.2 | 59 | -0.67 | 3.16 | 0.51 |
| U | 24 | 3.5 | 3.2 | 33 | 7 | 7.3 | 2.9 | 1,480 | 60 | 4.84 | 3.13 | 126 |

NOTES: (1) From Sheppard and Thibault 1990, Table A-1.
(2) From Sheppard and Thibault 1990, Table A-3.
(3) From Yu, et al. 2000, Table 3.9-1.
(4) The mean of the underlying normal distribution after taking natural logarithm of the $K_d$ values.
(5) The standard deviation of the underlying normal distribution after taking natural logarithm of the $K_d$ values.
(6) Exponential of the mean value [mL/g] or the geometric mean $K_d$.
(7) Default values for μ and exp(μ) have been predicted using soil-to-plant concentration ratios for nuclides with 0 observations.
(8) Standard deviation for data obtained from using the RESRAD default root uptake transfer factor and the correlation between $K_d$ and the concentration ratio for loamy soil was set to 3.22 to consider a potential wide range of distribution.

LEGEND: NA = not available

**Table E-6. Lognormal Distribution Parameters Used for $K_d$ Uncertainty Analyses**

| Element | Surface Soil, Unsaturated Zone Saturated Zone | | | | | Subsurface Soil and Sediment in Contaminated Zone | | | | | Bounding Range |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Source(1) | μ(2) | σ(3) | exp(μ)(4) | DP $K_d$ | Source(1) | μ(2) | σ(3) | exp(μ)(4) | DP $K_d$ | |
| Am | S&T Sand | 7.6 | 2.6 | 1,900 | 1,900 | S&T Sand | 7.6 | 2.6 | 1,900 | 4,000 | 0.5 - 390 |
| C | S&T Sand | 1.1 | 0.8 | 5 | 5 | S&T Sand | 1.1 | 0.8 | 5 | 7 | 0.7 - 12 |
| Cm | RESRAD | 8.82 | 1.82 | 6,761 | 6760 | RESRAD | 8.82 | 1.82 | 6,761 | 6760 | 780 - 22970 |
| Cs | S&T Sand | 5.6 | 2.5 | 280 | 280 | RESRAD | 6.10 | 2.33 | 446 | 480 | 10 - 10000 |
| I | S&T Sand | 0.04 | 2.2 | 1.0 | 1 | S&T Clay | 0.5 | 1.5 | 1 | 2 | 0.4 - 81 |
| Np | S&T Sand | 1.4 | 1.7 | 5 | 2.3 | S&T Sand | 1.4 | 1.7 | 5 | 3 | 0.5 - 390 |
| Pu | RESRAD | 6.86 | 1.89 | 953 | 2,600 | S&T Clay | 8.5 | 2.1 | 5,100 | 3,000 | 27 - 2550 |
| Sr | S&T Sand | 2.6 | 1.6 | 15 | 5 | D&M | 2.6 | 1.6 | 15 | 15 | 1 - 190 |
| Tc | S&T Sand | -2.0 | 1.8 | 0.1 | 0.1 | RESRAD | -0.67 | 3.16 | 0.51 | 4.1 | 0.01 - 16 |
| U | S&T Sand | 3.5 | 3.2 | 35 | 35 | S&T Sand | 3.5 | 3.2 | 35 | 10 | 0.4 - 2200 |

NOTES: (1) Sources**:** S&T Sand is Table A-1, Sheppard and Thibault 1990; S&T Clay is Table A-3, Sheppard and Thibault 1990; D&M from Dames and Moore, 1995a, 1995b, and RESRAD is Table 3.9-1, Attachment C, NUREG/CR-6697 (Yu, et al. 2000)

(2) The mean of the underlying normal distribution after taking natural logarithm of the $K_d$ values.

(3) The standard deviation of the underlying normal distribution after taking natural logarithm of the $K_d$ values.

(4) Exponential of the mean value [mL/g] or the geometric mean.

### 6.0 Parameter Correlation

The RESRAD code allows correlation of input parameters to limit the occurrence of unrealistic physical conditions (e.g., high outdoor and also high indoor time fractions). Parameters were correlated in pairs based on the user specified rank correlation coefficient as presented in Table E-7. The basis for the correlation coefficients for each conceptual model is discussed following the table.

**Table E-7. Input Correlations for Probabilistic Evaluation**(1)

| Parameter 1 | Parameter 2 | Correlation Coefficient | Basis | Surface Soil Model | Subsurface Model | Sediment Model |
|---|---|---|---|---|---|---|
| Indoor time fraction | Outdoor time fraction | -0.95 | Continuity of onsite time | ● | ● | |
| Contaminated zone hydraulic conductivity | Unsaturated zone hydraulic conductivity | 0.95 | Homogeneity in soil column | ● | | |
| Contaminated zone hydraulic conductivity | Saturated zone hydraulic conductivity | 0.95 | Homogeneity in soil column | ● | | |
| Unsaturated zone hydraulic conductivity | Saturated zone hydraulic conductivity | 0.95 | Homogeneity in soil column | ● | | |
| Precipitation rate | Rate of irrigation | -0.95 | Less irrigation when rainy | ● | ● | |
| Precipitation rate | Well pumping rate | -0.95 | Less pumping for irrigation when rainy | ● | ● | |
| Rate of irrigation | Well pumping rate | 0.95 | Pumping volume due mainly to irrigation | ● | ● | |
| Contaminated zone $K_d$ | Unsaturated zone $K_d$ | 0.95 | Homogeneity in soil column | ● | | |
| Unsaturated zone $K_d$ | Saturated zone $K_d$ | 0.95 | Homogeneity in soil column | ● | | |
| Contaminated zone $K_d$ | Saturated zone $K_d$ | 0.95 | Homogeneity in soil column | ● | | |
| Contaminated zone $K_d$ | Plant transfer factor | -0.87 | Baes, et. al. 1984 | ● | ● | ● |
| Contaminated zone $K_d$ | Meat transfer factor | -0.87 | Plant correlation used for meat | ● | ● | ● |
| Contaminated zone $K_d$ | Milk transfer factor | -0.87 | Plant correlation used for milk | ● | ● | |
| Unsaturated zone $K_d$ | Plant transfer factor | -0.87 | Baes, et. al. 1984 | ● | | |
| Unsaturated zone $K_d$ | Meat transfer factor | -0.87 | Plant correlation used for meat | ● | | |
| Unsaturated zone $K_d$ | Milk transfer factor | -0.87 | Plant correlation used for milk | ● | | |
| Saturated zone $K_d$ | Plant transfer factor | -0.87 | Baes, et. al. 1984 | ● | | |
| Saturated zone $K_d$ | Meat transfer factor | -0.87 | Plant correlation used for meat | ● | | |
| Saturated zone $K_d$ | Milk transfer factor | -0.87 | Plant correlation used for milk | ● | | |

NOTES: (1) Presented in the RESRAD probabilistic output files “LHS.REP” for each media.

### 6.1 Surface Soil Model

This section discusses the parameters correlated in the surface soil model, including distribution coefficients, plant transfer factors, hydraulic conductivities, as well as irrigation, precipitation, and well pumping rates.

The strongly negative correlation (R = -0.87) of $K_d$ with plant transfer factors is based on regression results obtained from computer simulation for a range of elements (Baes, et. al. 1984). This Oak Ridge National Laboratory investigation included all areas of the country and therefore represents average results, which are used in lieu of site-specific correlations. Similarly, the meat and milk transfer coefficients were strongly correlated with the contaminated zone $K_d$ for the principal radionuclides. Transfer factors for principal radionuclide daughter products were not correlated. As each additional parameter requires cross correlating with transfer factors for each soil layer, reducing the number of required correlations allows for reasonable code execution times.

The rate of irrigation and the well pumping rate were strongly correlated (R = 0.95) since the majority of water pumped by the well is used for irrigation. The precipitation rate was strongly negatively correlated (R = -0.95) with the irrigation and well pumping rate, assuming less groundwater will be needed to adequately water crops during wet years.

To ensure that the soils reflect relative homogeneity, the hydraulic conductivity in the three zones (contaminated, unsaturated and saturated) were correlated (R = 0.95).

### 6.2 Subsurface Soil Model

The subsurface soil model is based on a cistern excavation scenario, and is therefore based on a limited volume of source material brought to the surface. The potential configurations of contaminated zone area and thickness were evaluated in the deterministic sensitivity analysis presented in Section 5. Alternate parameters were selected for probabilistic evaluation.

### 6.3 Streambed Sediment Model

Parameters correlated in the streambed sediment model included:

- Contaminated zone and saturated zone hydraulic conductivity (0.95), and
- Contaminated zone $K_d$ and plant/meat transfer factors (-0.87).

To ensure that intended correlations were reflected in the RESRAD model input vectors, values were viewed graphically to verify the parameter relationships for each media and radionuclide.

## 7.0 RESRAD Output

### 7.1 Basic Approach

The results of the probabilistic evaluation are output from RESRAD in numerous summary data files and graphic displays. As suggested in NUREG/CR-6676 (Kamboj, et al. 2000), the input values generated by the specified distributions and correlations were graphically viewed to verify parameter associations. RESRAD output was tabulated and probabilistic-based DCGLs were calculated as described below.

Additionally, the tabulated output parameter correlation ranks were used to identify the parameters most significantly associated with the modeled dose, as described in subsequent sections. Plots of the modeled dose over time are included in Attachment 1 for each radionuclide and media model. DCGLs were calculated from the RESRAD DSRs in the same manner as described in Appendix C to this plan.

### 7.2 Surface Soil

Key results of the surface soil evaluation are presented in Table E-8. Table E-9 compares the resulting probabilistic DCGLs with the DCGLs developed using the deterministic method.

As can be seen in Table E-9, key dose drivers Cs-137, Sr-90, I-129 and U-232 had probabilistic peak-of-the-mean DCGLs below the deterministic values, as did all radionuclides except Np-237. Radionuclides were identified as key dose drivers based on preliminary characterization data in WMA1 and WMA2 (See Attachment 1, Tables Att-1 and Att-2). Cs-137, Sr-90, I-129 and U-232 are discussed below (See also Table E-14).

- The Cs-137 dose is due primarily to external exposure in the initial years of exposure. However the depth of source thickness and exposure time fractions were the probabilistic parameters that are directly related to the external pathway, and were not highly correlated with resulting dose.
- The Sr-90 dose is due primarily to plant uptake in the initial years of exposure. Plant uptake factors and depth of roots were highly correlated with the resulting dose.
- I-129 dose is primarily due to ingestion of water and milk in the initial decades of exposure. Length parallel to groundwater flow and contaminated zone thickness were the most highly correlated parameters with the resulting dose.
- U-232 dose is primarily due to external exposure during the initial years of the simulation. The gamma shielding factor, and indoor/outdoor time fractions were most highly correlated with the resulting dose.

Attachment 1 presents plots of the probabilistic (peak-of-the-mean and 95$^{th}$ percentile) and deterministic dose-source ratios (DSRs) for comparison, for the radionuclides listed above. Also presented are plots of deterministic results compared with the cumulative probability derived from the probabilistic modeling. For all radionuclides (with the exception of Np-237) the peak-of-the-mean DCGLs were smaller than the deterministic DCGLs.

**Table E-8. Key Output Dose Statistics (DSRs) – Surface Soil Model (mrem/y per pCi/g)(1)**

| Radionuclide | Year of Peak Dose | Minimum | Maximum | Mean | 95$^{th}$ Percentile |
|---|---|---|---|---|---|
| Am-241 | 2.01E+02 | 4.04E-02 | 3.49E+01 | 8.68E-01 | 1.32E+00 |
| C-14 | 0.00E+00 | 2.12E-01 | 2.83E+00 | 1.53E+00 | 2.56E+00 |
| Cm-243 | 0.00E+00 | 2.70E-01 | 4.69E+00 | 7.21E-01 | 1.60E+00 |
| Cm-244 | 0.00E+00 | 4.94E-02 | 7.38E+00 | 3.85E-01 | 1.04E+00 |
| Cs-137 | 0.0E+00 | 1.8E+00 | 2.2E+01 | 3.3E+00 | 6.3E+00 |
| I-129 | 3.43E+00 | 3.31E-01 | 1.86E+03 | 7.68E+01 | 4.68E+02 |
| Np-237 | 1.18E+01 | 9.16E-01 | 1.02E+03 | 9.59E+01 | 5.17E+02 |
| Pu-238 | 0.00E+00 | 8.51E-02 | 8.10E+00 | 6.26E-01 | 1.78E+00 |

**Table E-8. Key Output Dose Statistics (DSRs) – Surface Soil Model (mrem/y per pCi/g)(1)**

| Radionuclide | Year of Peak Dose | Minimum | Maximum | Mean | 95th Percentile |
|---|---|---|---|---|---|
| Pu-239 | 8.84E+02 | 2.73E-02 | 1.48E+01 | 9.86E-01 | 5.83E+00 |
| Pu-240 | 7.81E+02 | 5.28E-02 | 1.32E+01 | 9.48E-01 | 5.84E+00 |
| Pu-241 | 5.18E+01 | 3.34E-03 | 2.47E-01 | 2.15E-02 | 6.00E-02 |
| Sr-90 | 0.00E+00 | 2.12E-01 | 2.11E+02 | 1.22E+01 | 4.17E+01 |
| Tc-99 | 0.00E+00 | 2.30E-02 | 1.39E+01 | 1.19E+00 | 3.64E+00 |
| U-232 | 1.2E+01 | 1.5E+00 | 5.6E+02 | 1.7E+01 | 1.1E+02 |
| U-233 | 1.51E+01 | 2.07E-02 | 8.61E+01 | 3.02E+00 | 2.96E+01 |
| U-234 | 1.33E+01 | 1.41E-02 | 1.35E+02 | 2.96E+00 | 2.60E+01 |
| U-235 | 6.63E+01 | 7.77E-01 | 2.20E+01 | 7.20E+00 | 1.60E+01 |
| U-238 | 1.33E+01 | 3.34E-02 | 6.82E+01 | 2.54E+00 | 2.27E+01 |

NOTE: (1) From RESRAD probabilistic output file “MCSUMMARY.REP”.

**Table E-9. Surface Soil $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g**

| Nuclide | Deterministic(1) | Probabilistic(2) | | Percent Difference Deterministic and Peak of the Mean |
|---|---|---|---|---|
| | | Peak-of-the-Mean | 95th Percentile | |
| Am-241 | 4.31E+01 | 2.88E+01 | 1.89E+01 | -33% |
| C-14 | 2.00E+01 | 1.63E+01 | 9.77E+00 | -18% |
| Cm-243 | 4.06E+01 | 3.47E+01 | 1.56E+01 | -15% |
| Cm-244 | 8.22E+01 | 6.49E+01 | 2.40E+01 | -21% |
| **Cs-137(3)(4)** | **2.43E+01** | **1.52E+01** | **7.95E+00** | -37% |
| **I-129(4)** | 3.47E-01 | 3.26E-01 | 5.34E-02 | -6% |
| Np-237 | 9.42E-02 | 2.61E-01 | 4.84E-02 | 177% |
| Pu-238 | 5.03E+01 | 3.99E+01 | 1.40E+01 | -21% |
| Pu-239 | 4.53E+01 | 2.54E+01 | 4.29E+00 | -44% |
| Pu-240 | 4.53E+01 | 2.64E+01 | 4.28E+00 | -42% |
| Pu-241 | 1.42E+03 | 1.16E+03 | 4.17E+02 | -18% |
| **Sr-90(3)(4)** | **6.25E+00** | **4.10E+00** | **1.20E+00** | -34% |
| Tc-99 | 2.37E+01 | 2.10E+01 | 6.87E+00 | -11% |
| **U-232(4)** | 5.84E+00 | 1.51E+00 | 2.23E-01 | -74% |
| **U-233(4)** | 1.90E+01 | 8.28E+00 | 8.45E-01 | -56% |
| **U-234(4)** | 1.97E+01 | 8.45E+00 | 9.62E-01 | -57% |
| **U-235(4)** | 1.87E+01 | 3.47E+00 | 1.79E+00 | -81% |
| **U-238(4)** | 2.06E+01 | 9.84E+00 | 1.10E+00 | -52% |

NOTES: (1) From Table 5-8 of Section 5.
(2) From RESRAD probabilistic output file “MCSUMMARY.REP”.
(3) DCGLs for these radionuclides are multiplied by a factor of two to account for decay during 30 year institutional control period.
(4) Dose driver radionuclide (see Section 5.2.4 of the plan).

### 7.3 Subsurface Soil

Key results of the subsurface soil evaluation are presented in Table E-10. Table E-11 compares the resulting probabilistic DCGLs with the DCGLs developed using the deterministic method. Note that the deterministic DCGLs used in this table for comparison purposes are the DCGLs from Table 5-8, which are based on the original base-case conceptual model. The DCGLs from the multi-source analysis that takes into account continuing releases from the bottom of the deep excavations are not directly comparable with the peak-of-the-mean DCGLs because the model used in development of the latter does not account for this secondary source. Table 5-11c in Section 5 of this plan compares all of the different subsurface soil DCGLs.

Note also that the DCGLs presented in Table E-11 reflect a 10 fold dilution of the source term (i.e. using 1/10$^{th}$ the DSRs presented in Table E-10) as described in Section 5 of the DPlan.

As can be seen in Table E-11, only Sr-90, Tc-99, and U-232 had probabilistic peak-of-the-mean DCGLs at least 10 percent below the deterministic values. These radionuclides are discussed below (See also Table E-15).

- The Sr-90 dose is due primarily to plant uptake in the initial years of exposure. Depth of roots and plant uptake factors were highly correlated with the resulting dose.
- The Tc-99 dose is due primarily to plant uptake in the initial years of exposure. Depth of roots and plant uptake factors were highly correlated with the resulting dose.
- The U-232 dose is due primarily to external exposure in the initial years of the simulation. The contaminated zone $K_d$ and gamma shielding factors were most highly correlated with the resulting dose.

Attachment 1 presents the plots of the probabilistic (peak-of-the-mean and 95$^{th}$ percentile) and deterministic DSRs for comparison, for the key dose drivers Sr-90, Cs-137, and U-232. Also presented are plots of deterministic results compared with the cumulative probability derived from the probabilistic modeling. For seven other radionuclides, the peak-of-the-mean DCGLs were greater than or equal to the deterministic.

**Table E-10. Key Output Dose Statistics (DSRs) – Subsurface Soil Model (mrem/y per pCi/g)[(1)]**

| Radionuclide | Year of Peak Dose | Minimum | Maximum | Mean | 95$^{th}$ Percentile |
|---|---|---|---|---|---|
| Am-241 | 0.0E+00 | 2.4E-02 | 2.4E-01 | 3.7E-02 | 5.8E-02 |
| C-14 | 0.0E+00 | 1.4E-04 | 1.2E-03 | 3.5E-04 | 6.9E-04 |
| Cm-243 | 0.0E+00 | 1.6E-01 | 3.8E-01 | 2.2E-01 | 2.7E-01 |
| Cm-244 | 0.0E+00 | 6.0E-03 | 7.3E-02 | 1.1E-02 | 2.3E-02 |
| Cs-137 | 0.0E+00 | 1.4E+00 | 2.4E+00 | 1.7E+00 | 1.8E+00 |
| I-129 | 1.2E+01 | 2.1E-03 | 1.7E+00 | 3.7E-01 | 9.6E-01 |

**Table E-10. Key Output Dose Statistics (DSRs) – Subsurface Soil Model (mrem/y per pCi/g)[(1)]**

| Radionuclide | Year of Peak Dose | Minimum | Maximum | Mean | 95th Percentile |
|---|---|---|---|---|---|
| Np-237 | 2.5E+01 | 6.5E-08 | 2.3E+01 | 2.7E+00 | 8.5E+00 |
| Pu-238 | 0.0E+00 | 9.7E-03 | 1.6E-01 | 1.8E-02 | 3.7E-02 |
| Pu-239 | 0.0E+00 | 1.1E-02 | 1.9E-01 | 2.0E-02 | 4.1E-02 |
| Pu-240 | 0.0E+00 | 1.1E-02 | 4.7E-01 | 2.1E-02 | 3.9E-02 |
| Pu-241 | 5.2E+01 | 2.0E-04 | 7.7E-03 | 1.0E-03 | 1.6E-03 |
| Sr-90 | 0.0E+00 | 1.3E-02 | 5.0E+00 | 1.5E-01 | 4.8E-01 |
| Tc-99 | 0.0E+00 | 5.5E-04 | 5.2E-01 | 1.7E-02 | 5.7E-02 |
| U-232 | 6.4E+00 | 5.4E-03 | 5.1E+00 | 3.4E+00 | 4.6E+00 |
| U-233 | 3.7E+02 | 2.3E-14 | 6.3E-01 | 2.5E-02 | 7.4E-02 |
| U-234 | 3.7E+02 | 4.5E-07 | 1.3E+00 | 2.0E-02 | 6.7E-02 |
| U-235 | 0.0E+00 | 1.5E-01 | 3.6E-01 | 2.7E-01 | 3.3E-01 |
| U-238 | 0.0E+00 | 3.3E-02 | 1.1E-01 | 5.4E-02 | 6.6E-02 |

NOTE: (1) From RESRAD probabilistic output file “MCSUMMARY.REP”.

**Table E-11. Subsurface Soil $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g**

| Nuclide | Deterministic[(1)] | Probabilistic[(2)] | | Percent Difference Deterministic and Peak-of-the-Mean |
|---|---|---|---|---|
| | | Peak-of-the-Mean | 95th Percentile | |
| Am-241 | 7.16E+03 | 6.81E+03 | 4.30E+03 | -5% |
| C-14 | 5.59E+05 | 7.18E+05 | 3.64E+05 | 28% |
| Cm-243 | 1.15E+03 | 1.12E+03 | 9.33E+02 | -3% |
| Cm-244 | 2.37E+04 | 2.21E+04 | 1.08E+04 | -7% |
| **Cs-137**[(3)(4)] | **4.36E+02** | **3.01E+02** | **2.72E+02** | -31% |
| **I-129**[(4)] | 6.46E+02 | 6.70E+02 | 2.60E+02 | 4% |
| Np-237 | 5.77E+01 | 9.33E+01 | 2.95E+01 | 62% |
| Pu-238 | 1.47E+04 | 1.37E+04 | 6.83E+03 | -7% |
| Pu-239 | 1.33E+04 | 1.23E+04 | 6.11E+03 | -7% |
| Pu-240 | 1.33E+04 | 1.21E+04 | 6.44E+03 | -9% |
| Pu-241 | 2.41E+05 | 2.50E+05 | 1.59E+05 | 4% |
| **Sr-90**[(3)(4)] | **4.36E+03** | **3.42E+03** | **1.03E+03** | -21% |
| Tc-99 | 1.59E+04 | 1.44E+04 | 4.36E+03 | -10% |
| **U-232**[(4)] | 1.06E+02 | 7.40E+01 | 5.43E+01 | -30% |
| **U-233**[(4)] | 2.72E+03 | 9.92E+03 | 3.39E+03 | 264% |

**Table E-11. Subsurface Soil $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g**

| Nuclide | Deterministic(1) | Probabilistic(2) | | Percent Difference Deterministic and Peak-of-the-Mean |
|---|---|---|---|---|
| | | Peak-of-the-Mean | 95th Percentile | |
| **U-234**(4) | 2.81E+03 | 1.26E+04 | 3.75E+03 | 349% |
| **U-235**(4) | 9.41E+02 | 9.33E+02 | 7.60E+02 | -1% |
| **U-238**(4) | 2.94E+03 | 4.60E+03 | 3.79E+03 | 57% |

NOTES: (1) From Table 5-8 of Section 5. More limiting deterministic values for the resident gardener are available as an alternative comparison for some radionuclides. Refer to Section 5.2.8 for a comparison between the probabilistic DCGLs and all other sets of subsurface soil DCGLs.

(2) From RESRAD probabilistic output file "MCSUMMARY.REP" for the resident farmer with a contamination zone of 100 $m^2$.

(3) DCGLs for these radionuclides are multiplied by a factor of two to account for decay during 30 year institutional control period.

(4) Dose driver radionuclide (see Section 5.2.4 of the plan).

### 7.3 Streambed Sediment

Key results of the streambed sediment evaluation are presented in Table E-12. Table E-13 compares the resulting probabilistic DCGLs with the DCGLs developed using the deterministic method.

As can be seen in Table E-13, all radionuclides had probabilistic peak-of-the-mean DCGLs at least 10 percent below the deterministic values. Key dose drivers for sediment are Sr-90 and Cs-137. These radionuclides are discussed below (See also Table E-16).

- Sr-90 dose is due primarily to ingestion of venison in the initial years of exposure. The resulting dose is highly correlated to the contaminated zone $K_d$ value; however, the plant and fish biotransfer factors were more closely correlated than the meat biotransfer factors.
- Cs-137 dose is primarily due to external exposure in the initial years of exposure. As expected, the outdoor time fraction was highly correlated with dose.

Attachment 1 presents the plots of the probabilistic (peak-of-the-mean and 95th percentile) and deterministic DSRs for comparison. Also presented are plots of deterministic results compared with the cumulative probability derived from the probabilistic modeling.

**Table E-12. Key Output Dose Statistics (DSRs) – Streambed Sediment Model (mrem/y per pCi/g)(1)**

| Radionuclide | Year of Peak Dose | Minimum | Maximum | Mean | 95th Percentile |
|---|---|---|---|---|---|
| Am-241 | 1.0E+00 | 9.1E-04 | 5.7E-02 | 2.5E-03 | 4.8E-03 |
| C-14 | 0.0E+00 | 5.8E-03 | 4.5E-01 | 1.4E-02 | 3.4E-02 |
| Cm-243 | 0.0E+00 | 3.7E-03 | 1.4E-02 | 8.2E-03 | 1.2E-02 |
| Cm-244 | 0.0E+00 | 2.6E-04 | 2.4E-03 | 6.5E-04 | 9.9E-04 |
| Cs-137 | 0.0E+00 | 2.3E-02 | 8.8E-02 | 4.8E-02 | 6.9E-02 |
| I-129 | 0.0E+00 | 6.1E-03 | 6.6E-01 | 3.2E-02 | 7.2E-02 |

**Table E-12. Key Output Dose Statistics (DSRs) – Streambed Sediment Model (mrem/y per pCi/g)[1]**

| Radionuclide | Year of Peak Dose | Minimum | Maximum | Mean | 95th Percentile |
|---|---|---|---|---|---|
| Np-237 | 0.0E+00 | 1.0E-02 | 2.2E+00 | 7.7E-02 | 2.3E-01 |
| Pu-238 | 1.0E+00 | 6.9E-04 | 1.4E-01 | 2.0E-03 | 3.6E-03 |
| Pu-239 | 1.0E+00 | 8.8E-04 | 2.3E-02 | 2.1E-03 | 4.1E-03 |
| Pu-240 | 1.0E+00 | 9.0E-04 | 1.6E-02 | 2.1E-03 | 4.2E-03 |
| Pu-241 | 5.2E+01 | 2.8E-05 | 1.9E-03 | 7.3E-05 | 1.3E-04 |
| Sr-90 | 0.0E+00 | 1.4E-03 | 1.5E-01 | 1.1E-02 | 3.0E-02 |
| Tc-99 | 0.0E+00 | 3.4E-06 | 1.1E-03 | 3.8E-05 | 1.1E-04 |
| U-232 | 7.2E+00 | 4.6E-02 | 9.3E-01 | 1.1E-01 | 1.7E-01 |
| U-233 | 0.0E+00 | 1.1E-04 | 5.2E-02 | 1.2E-03 | 3.9E-03 |
| U-234 | 0.0E+00 | 1.2E-04 | 2.9E-02 | 1.2E-03 | 4.2E-03 |
| U-235 | 0.0E+00 | 4.9E-03 | 4.0E-02 | 1.1E-02 | 1.6E-02 |
| U-238 | 0.0E+00 | 1.1E-03 | 9.0E-02 | 3.1E-03 | 5.5E-03 |

NOTE: (1) From RESRAD probabilistic output file "MCSUMMARY.REP".

**Table E-13. Streambed Sediment DCGL$_W$ Values for 25 mrem in Peak Year in pCi/g**

| Nuclide | Deterministic[1] | Probabilistic[2] | | Percent Difference Deterministic and Peak-of-the-Mean |
|---|---|---|---|---|
| | | Peak-of-the-Mean | 95th Percentile | |
| Am-241 | 1.55E+04 | 1.02E+04 | 5.19E+03 | -34% |
| C-14 | 3.44E+03 | 1.84E+03 | 7.42E+02 | -46% |
| Cm-243 | 3.59E+03 | 3.06E+03 | 2.08E+03 | -15% |
| Cm-244 | 4.84E+04 | 3.83E+04 | 2.52E+04 | -21% |
| **Cs-137**[3][4] | **1.29E+03** | **1.04E+03** | **7.24E+02** | -19% |
| I-129 | 3.69E+03 | 7.91E+02 | 3.49E+02 | -79% |
| Np-237 | 5.19E+02 | 3.25E+02 | 1.11E+02 | -37% |
| Pu-238 | 1.99E+04 | 1.24E+04 | 7.02E+03 | -38% |
| Pu-239 | 1.79E+04 | 1.19E+04 | 6.08E+03 | -33% |
| Pu-240 | 1.79E+04 | 1.20E+04 | 5.98E+03 | -33% |
| Pu-241 | 5.11E+05 | 3.44E+05 | 1.92E+05 | -33% |
| **Sr-90**[3][4] | **9.49E+03** | **4.72E+03** | **1.67E+03** | -50% |
| Tc-99 | 2.17E+06 | 6.61E+05 | 2.38E+05 | -70% |
| U-232 | 2.61E+02 | 2.23E+02 | 1.49E+02 | -15% |
| U-233 | 5.75E+04 | 2.16E+04 | 6.38E+03 | -62% |
| U-234 | 6.04E+04 | 2.16E+04 | 5.94E+03 | -64% |

**Table E-13. Streambed Sediment $DCGL_W$ Values for 25 mrem in Peak Year in pCi/g**

| Nuclide | Deterministic(1) | Probabilistic(2) | | Percent Difference Deterministic and Peak-of-the-Mean |
|---|---|---|---|---|
| | | Peak-of-the-Mean | 95th Percentile | |
| U-235 | 2.89E+03 | 2.34E+03 | 1.58E+03 | -19% |
| U-238 | 1.25E+04 | 8.17E+03 | 4.55E+03 | -34% |

NOTES: (1) From Table 5-8 of Section 5.
(2) From RESRAD probabilistic output file "MCSUMMARY.REP".
(3) DCGLs for these radionuclides are multiplied by a factor of two to account for decay during 30 year institutional control period.
(4) Dose driver radionuclide (see Section 5.2.4 of the plan).

### 7.4 Preliminary Dose Assessment for Remediated WMA 1 Excavation

As indicated in Section 5.4.4 of this plan, the preliminary dose assessment for the remediated WMA 1 excavated area estimated by using information from the multi-source deterministic analysis was a maximum of approximately 8 mrem per year. Using the probabilistic modeling results, the estimates are as follows:

- A peak-of-the-mean estimate of 1.9 mrem per year
- A $95^{th}$ percentile value of 2.8 mrem per year

Table Att-1 of Attachment 1 shows the calculations of these values. The probabilistic results were not used because they were lower than the 8 mrem per year estimate produced using information from the multi-source deterministic analysis.

### 7.5 Preliminary Dose Assessment for Remediated WMA 2 Excavation

As indicated in Section 5.4.4 of this plan, the preliminary dose assessment for the remediated WMA 2 excavated area estimated by using information from the multi-source deterministic analysis was a maximum of approximately 0.2 mrem per year. Using the probabilistic modeling results, the estimates are as follows:

- A peak-of-the-mean estimate of 0.11 mrem per year
- A $95^{th}$ percentile value of 0.13 mrem per year

Table Att-2 of Attachment 1 shows the calculations of these values. The probabilistic results were not used because they were lower than the 0.2 mrem per year estimate produced using information from the multi-source deterministic analysis.

## 8.0 Parameter Output Rank Correlations

The RESRAD results include several correlations of input parameters with the output modeled dose. Several correlations are available based on actual numerical calculated values and relative rankings.

Guidance for RESRAD probabilistic modeling in NUREG/CR-6676 (Kamboj, et al. 2000) indicates that correlation coefficients based on relative rankings are preferable where nonlinear relationships, widely disparate scales, or long tails are present in the input and outputs. Therefore, determinations of parameter significance presented in this section are based on the partial rank correlation coefficient (PRCC). Where strong correlations between an input parameter and the dose were indicated in the output ranking, scatter plots were inspected to confirm the conclusion.

RESRAD also calculates the overall coefficients of determination ($R^2$) for each model, which provides an indication of the variability in the overall radionuclide dose accounted for by the selected input parameters.

As described previously, numerous parameters were selected for probabilistic evaluation for each radionuclide. The tables presented and discussed below focus on the three highest ranked parameter correlations for all included parameters for each radionuclide in each media.

To ensure sufficient model iterations were being used to allow for convergence of the results, three sets of 1,000 iterations were selected. This was considered to be appropriate as the peak-of-the-mean doses for the three datasets were within approximately +/-10 percent. The run with the largest peak-of-the-mean dose was selected as the basis for the information in the summary tables.

### 8.1 Surface Soil Model

Table E-14 presents a summary of the parameters which correlate most closely with the overall dose for each radionuclide. In general, $K_d$, plant transfer factors, and root zone depth were most strongly correlated with dose. The plant transfer factors have the higher correlations (mostly >0.7) when compared with $K_d$ (<0.7).

The $R^2$ values ranged from 0.71 (U-232) to 0.99 (I-129). Where the overall correlation is low, identification of additional probabilistic parameters for these radionuclides may better describe the variability in the model output.

**Table E-14. Summary of Parameter Rankings – Surface Soil Model**[(1)]

| Nuclide | Parameter Ranking | | | Simulation No. ($R^2$) |
|---|---|---|---|---|
| | 1 | 2 | 3 | |
| Am-241 | Plant transfer factor for Am (0.78) | Contaminated zone Thickness (0.54) | Depth of roots (-0.49) | 3 (0.93) |
| C-14 | Contaminated zone thickness (0.98) | Depth of roots (-0.79) | Plant transfer factor for C (0.08) | 3 (0.96) |
| Cm-243 | Plant transfer factor for Cm (0.86) | Contaminated zone Thickness (0.65) | Depth of roots (-0.64) | 2 (0.96) |
| Cm-244 | Plant transfer factor for Cm (0.87) | Contaminated zone Thickness (0.68) | Depth of roots (-0.67) | 3 (0.96) |
| Cs-137 | Plant transfer factor for Cs (0.71) | Depth of roots (-0.56) | Contaminated zone Thickness (0.52) | 3 (0.95) |
| I-129 | Length parallel to groundwater flow (0.64) | Contaminated zone Thickness (0.62) | Irrigation rate (0.34) | 2 (0.99) |
| Np-237 | Length parallel to groundwater flow (0.73) | Contaminated zone Thickness (0.60) | Saturated zone hydraulic conductivity (-0.45) | 2 (0.99) |

**Table E-14. Summary of Parameter Rankings – Surface Soil Model(1)**

| Nuclide | Parameter Ranking | | | Simulation No. ($R^2$) |
|---|---|---|---|---|
| | 1 | 2 | 3 | |
| Pu-238 | Plant transfer factor for Pu (0.86) | Depth of roots (-0.67) | Contaminated zone Thickness (0.66) | 3 (0.96) |
| Pu-239 | Plant transfer factor for Pu (0.72) | Depth of roots (-0.44) | Contaminated zone Thickness (0.43) | 1 (0.91) |
| Pu-240 | Plant transfer factor for Pu (0.74) | Depth of roots (-0.44) | Contaminated zone Thickness (0.43) | 1 (0.91) |
| Pu-241 | Plant transfer factor for Am (0.81) | Contaminated zone Thickness (0.39) | Depth of roots (-0.37) | 1 (0.75) |
| Sr-90 | Plant transfer factor for Sr (0.84) | Depth of roots (-0.62) | Contaminated zone thickness (0.60) | 3 (0.96) |
| Tc-99 | Contaminated zone Thickness (0.67) | Plant transfer factor for Tc (0.55) | Depth of roots (-0.33) | 3 (0.92) |
| U-232 | Gamma shileding factor (0.38) | Outdoor time fraction (0.34) | Indoor time fraction (0.21) | 1 (0.67) |
| U-233 | Contaminated zone Thickness (0.23) | Meat transfer factor for U (-0.19) | Plant transfer factor for Th (0.18) | 3 (0.92) |
| U-234 | Contaminated zone Thickness (0.32) | Meat transfer factor for U (-0.15) | Depth of roots (-0.13) | 3 (0.95) |
| U-235 | Length parallel to groundwater flow (0.78) | Contaminated zone Thickness (0.77) | Saturated zone Kd (-0.46) | 3 (0.93) |
| U-238 | Contaminated zone Thickness (0.23) | Length parallel to groundwater flow (0.16) | Depth of roots (-0.16) | 1 (0.96) |

NOTE: (1) From RESRAD probabilistic output file "MCSUMMARY.REP". Simulation (out of three) with largest peak-of-the-mean dose was used to determine the parameter ranking, based on the PRCCs with statistic (either R or R2) in parentheses.

### 8.2 Subsurface Soil Model

As shown in Table E-15, the most highly correlated parameters for the subsurface model, like with the surface soil model, are the $K_d$, plant transfer coefficients, and root depth. The highest correlations (-0.99) were calculated for the depth of roots; however the $K_d$ correlations were generally lower than those for the plant transfer factors. The $R^2$ values ranged from 0.17 (U-233) to 1.00 (Np-237).

**Table E-15. Summary of Parameter Rankings - Subsurface Soil Model(1)**

| Nuclide | Parameter Ranking | | | Simulation No. ($R^2$) |
|---|---|---|---|---|
| | 1 | 2 | 3 | |
| Am-241 | Depth of roots (-0.82) | Plant transfer factor for Am (0.76) | Outdoor time fraction (0.58) | 1 (0.93) |
| C-14 | Depth of roots (-0.99) | Meat transfer factor for C (0.18) | Plant transfer factor for C (0.17) | 2 (0.98) |
| Cm-243 | Outdoor time fraction (0.91) | Indoor time fraction (0.53) | Plant transfer factor for Cm (-0.44) | 1 (0.96) |

**Table E-15. Summary of Parameter Rankings - Subsurface Soil Model(1)**

| Nuclide | Parameter Ranking | | | Simulation No. ($R^2$) |
|---|---|---|---|---|
| | 1 | 2 | 3 | |
| Cm-244 | Depth of roots (-0.93) | Plant transfer factor for Cm (0.89) | Indoor time fraction (0.40) | 1 (0.97) |
| Cs-137 | Outdoor time fraction (0.93) | Gamma shielding factor (0.92) | Indoor time fraction (0.81) | 3 (0.96) |
| I-129 | Contaminated zone $K_d$ for I (-0.94) | Well pumping rate (-0.56) | Irrigation rate (0.27) | 1 (0.99) |
| Np-237 | Contaminated zone $K_d$ for Np (-0.95) | Well pumping rate (-0.55) | Irrigation rate (0.29) | 3 (1.00) |
| Pu-238 | Depth of roots (-0.93) | Plant transfer factors for Pu (0.32) | Outdoor time fraction (0.32) | 1 (0.97) |
| Pu-239 | Depth of roots (-0.93) | Plant transfer factor for Pu (0.89) | Outdoor time fraction (0.29) | 2 (0.97) |
| Pu-240 | Depth of roots (-0.93) | Plant transfer factor for Pu (0.90) | Indoor time fraction (0.33) | 1 (0.97) |
| Pu-241 | Plant transfer factor for Am (0.81) | Depth of roots (-0.62) | Contaminated zone $K_d$ for Am (0.52) | 1 (0.77) |
| Sr-90 | Depth of roots (-0.94) | Plant transfer factor for Sr (0.91) | Contaminated zone $K_d$ for Cs (-0.10) | 1 (0.98) |
| Tc-99 | Depth of roots (-0.93) | Plant transfer factor for Tc (0.90) | Well pumping rate (-0.10) | 1 (0.97) |
| U-232 | Contaminated zone $K_d$ for U (0.49) | Gamma shielding factor (0.48) | Outdoor time fraction (0.41) | 3 (0.87) |
| U-233 | Contaminated zone $K_d$ for U (-0.34) | Milk transfer factor for U (-0.31) | Plant transfer factor for U (-0.29) | 3 (0.17) |
| U-234 | Contaminated zone $K_d$ for U (-0.31) | Milk transfer factor for U (-0.24) | Meat transfer factor for U (-0.22) | 3 (0.25) |
| U-235 | Outdoor time fraction (0.71) | Indoor time fraction (0.28) | Meat transfer factor for U (-0.15) | 2 (0.85) |
| U-238 | Outdoor time fraction (0.48) | Milk transfer factor for U (-0.22) | Meat transfer factor for U (-0.21) | 1 (0.62) |

NOTE: (1) From RESRAD probabilistic output file “MCSUMMARY.REP”. Simulation (out of three) with largest peak-of-the-mean dose was used to determine the parameter ranking, based on the Partial Rank Correlation Coefficients (PRCC) with statistic (either R or R2) in parentheses.

### 8.3 Streambed Sediment Model

Table E-16 shows the correlation coefficients and highest ranked sediment parameters for streambed sediment. Fourteen radionuclides have a correlation coefficient greater than or equal to 0.85 and one radionuclide has a coefficient below 0.5. The $R^2$ values ranged from 0.23 (U-233) to 0.99 (Cm-243). The outdoor time fraction accounted for the majority of the highest correlations.

**Table E-16. Summary of Parameter Rankings – Streambed Sediment Model**[1]

| Nuclide | Parameter Ranking | | | Simulation No. ($R^2$) |
|---|---|---|---|---|
| | 1 | 2 | 3 | |
| Am-241 | Outdoor time fraction (0.86) | Fish transfer factor for Am (0.43) | Meat transfer factor for Am (0.13) | 1 (0.81) |
| C-14 | Fish transfer factor for C (0.98) | Contaminated zone $K_d$ for C (-0.43) | Meat transfer factor for C (0..07) | 1 (0.97) |
| Cm-243 | Outdoor time fraction (1.00) | Contaminated zone $K_d$ for Cm (-0.14) | Fish transfer factor for Cm (0.11) | 1 (0.99) |
| Cm-244 | Outdoor time fraction (0.92) | Fish transfer factor for Cm (0.29) | Meat transfer factor for Cm (0.26) | 1 (0.89) |
| Cs-137 | Outdoor time fraction (0.99) | Meat transfer factor for Cs (0.33) | Plant transfer factor for Cs (0.18) | 1 (0.98) |
| I-129 | Fish transfer factor for I (0.81) | Contaminated zone $K_d$ for I (-0.48) | Meat transfer factor for I (0.44) | 1 (0.95) |
| Np-237 | Fish transfer factor for Np (0.89) | Outdoor time fraction (0.52) | Contaminated zone $K_d$ for Np (-0.47) | 1 (0.93) |
| Pu-238 | Outdoor time fraction (0.82) | Fish transfer factor for Pu (0.74) | Contaminated zone $K_d$ for Pu (-0.23) | 1 (0.87) |
| Pu-239 | Outdoor time fraction (0.81) | Fish transfer factor for Pu (0.74) | Contaminated zone $K_d$ for Pu (-0.27) | 1 (0.86) |
| Pu-240 | Outdoor time fraction (0.81) | Fish transfer factor for Pu (0.74) | Contaminated zone $K_d$ for Pu (-0.30) | 1 (0.96) |
| Pu-241[2] | Outdoor time fraction (0.79) | Contaminated zone $K_d$ for Am (-0.58) | Fish transfer factor for Am (0.38) | 1 (0.72) |
| Sr-90 | Contaminated zone $K_d$ for Sr (-0.73) | Fish transfer factor for Sr (0.59) | Plant transfer factor for Sr (0.30) | 1 (0.97) |
| Tc-99 | Fish transfer factor for Tc (0.91) | Plant transfer factor for Tc (0.17) | Meat transfer factor for Tc (0.13) | 1 (0.86) |
| U-232 | Outdoor time fraction (0.96) | Fish transfer factor for U (0.27) | Plant transfer factor for U (-0.14) | 1 (0.93) |
| U-233 | Contaminated zone $K_d$ for Th (-0.21) | Outdoor time fraction (0.26) | Meat transfer factor for Tc (0.20) | 1 (0.23) |
| U-234 | Fish transfer factor for U (0.45) | Outdoor time fraction (0.28) | Contaminated zone $K_d$ for U (-0.26) | 3 (0.78) |
| U-235 | Outdoor time fraction (0.94) | Fish transfer factor for U (0.35) | Meat transfer factor for U (0.20) | 1 (0.90) |
| U-238 | Outdoor time fraction (0.85) | Fish transfer factor for U (0.41) | Contaminated zone $K_d$ for U (-0.23) | 1 (0.85) |

NOTES: (1) From RESRAD probabilistic output file "MCSUMMARY.REP". Simulation (out of three) with largest peak-of-the-mean dose was used to determine the parameter ranking, based on the Partial Rank Correlation Coefficients (PRCC) with statistic (either R or R2) in parentheses.

(2) This analog was assumed give the decay of Pu-241 to Am-241.

## 9.0 Conclusions from the Uncertainty Analyses and Related Actions

### 9.1 Conclusions

The following conclusions can be drawn from the results of the probabilistic modeling described above.

**Surface Soil DCGLs**

Table E-9 shows that deterministic DCGLs for 17 of the 18 radionuclides of interest are not bounding because they are greater than the peak-of-the mean probabilistic DCGLs. Parameters highly correlated with the output are plant transfer factors, depth of roots, and length parallel to aquifer flow.

The length parallel to aquifer flow is a parameter selected to vary the dilution factor in groundwater.

These input parameters therefore lack sufficient conservatism insofar as the 17 radionuclides are concerned. This group of radionuclides includes three that have been identified as dose drivers: Sr-90, Cs-137, and U-235.

The lack of conservatism in these surface soil criteria can be quantified in another manner by considering the average soil concentrations at the deterministic DCGLs. If the average residual concentration of Sr-90, for example, were to be 6.25 pCi/g (the deterministic DCGL for surface soil), then the probabilistic modeling would indicate that the probability that the resulting dose would not exceed 25 mrem in the peak year would be approximately 55 percent (see Figure Att-2 in Attachment 1).

The primary conclusion for the surface soil model is that some input parameters used in the deterministic modeling are not sufficiently conservative and, consequently, the deterministic DCGLs for 17 radionuclides are not bounding.

**Subsurface Soil DCGLs**

Table E-11 shows that 10 of the deterministic DCGLs are not bounding because they exceed the peak-of-the mean probabilistic DCGLs, however only three radionuclides were below the deterministic DCGL by more than 10 percent. The comparisons above are based on the deterministic values for the resident farmer scenario, however more limiting values are available for the resident gardener scenario for comparison. The most limiting of all deterministic and probabilistic scenarios will be used to establish the cleanup levels (See Section 5). Parameters highly correlated with the output are depth of roots, contaminated zone $K_d$, and outdoor time fraction. The outdoor time fraction is based on assumptions of anticipated activity and may be refined with additional site-specific considerations. Refer to Section 5.2.8 for comparisons between the probabilistic DCGLs and other sets of subsurface soil DCGLs.

**Streambed Sediment DCGLs**

Table E-13 indicates that none of the deterministic DCGLs are bounding because they all exceed the peak-of-the-means DCGLs. For the key sediment dose drivers Sr-90 and Cs-137, the probabilistic values less than the deterministic by 50 percent and 19 percent respectively. The outdoor time fraction is most highly correlated with the dose for Cs-137, and Sr-90 was most highly correlated with the contaminated zone $K_d$. The outdoor time fraction is based on assumptions of anticipated activity and may be refined with additional site-specific considerations.

**Preliminary Dose Assessments**

The probabilistic dose estimates for the WMA 1 excavation area show that doses are likely to be less than 1.9 mrem/y, due primarily to Sr-90. The probabilistic dose estimates for the WMA 2 excavation area show that the doses are likely to be less than 0.11 mrem/y, due primarily to Cs-137.

Based on these results, it is anticipated that a small number of radionuclides will account for the majority of the dose.

**Input Parameters and Dose Variability**

The determination of which input parameters account for the majority of variability in the output was accomplished by inspection of the output correlation coefficients, which indicated the following:

- For surface soil, output dose results were well described by the input parameters, as only two radionuclides (Pu-241 and U-232) had coefficients of determination <+/-0.9. The highest parameter correlations (>+/-0.7) were for plant transfer factors and contaminated zone thickness.
- For subsurface soil, the variability in the calculated dose was moderately well described by the input parameters (six radionuclides with $R^2$ <+/-0.9). The highest correlations for individual parameters (>+/-0.9) were the depth of roots, contaminated zone $K_d$, and outdoor time fraction
- Sediment dose variability was well described by the input parameters (nine radionuclides with $R^2$ <+/-0.9), with the highest correlations (>+/-0.9) observed for the outdoor time fraction and fish transfer factor.

The probabilistic evaluation has identified parameters that are well correlated with the calculated dose. Based on these results, the input parameters that account for the majority of variability in the output are plant transfer factors, contaminated zone thickness, depth of roots, contaminated zone $K_d$ , outdoor time fraction, and fish transfer factors.

## 9.2 Actions

The conclusions on the probabilistic uncertainty analysis results just described led to the decision to make use of the probabilistic peak-of-the-mean DCGLs in place of the deterministic DCGLs provided in Revision 0 to this plan for surface soil and streambed sediment. The probabilistic peak-of-the-mean DCGLs were used for subsurface soil for three radionuclides as discussed in Section 5.2.8. Changes in Section 5 made as part of Revision 2, including changes to the cleanup goals, reflect these decisions.

# 10.0 References

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## 11.0 ATTACHMENTS

(1) Plots of Probabilistic and Deterministic Results

(2) Electronic Files Described in Section 1.3 (provided separately)

# ATTACHMENT 1

## Plots of Probabilistic and Deterministic Results

Note that the deterministic results used in this attachment are the deterministic results based on the original base-case conceptual model. The multi-source analysis results were not used because they are not directly comparable with the probabilistic results.

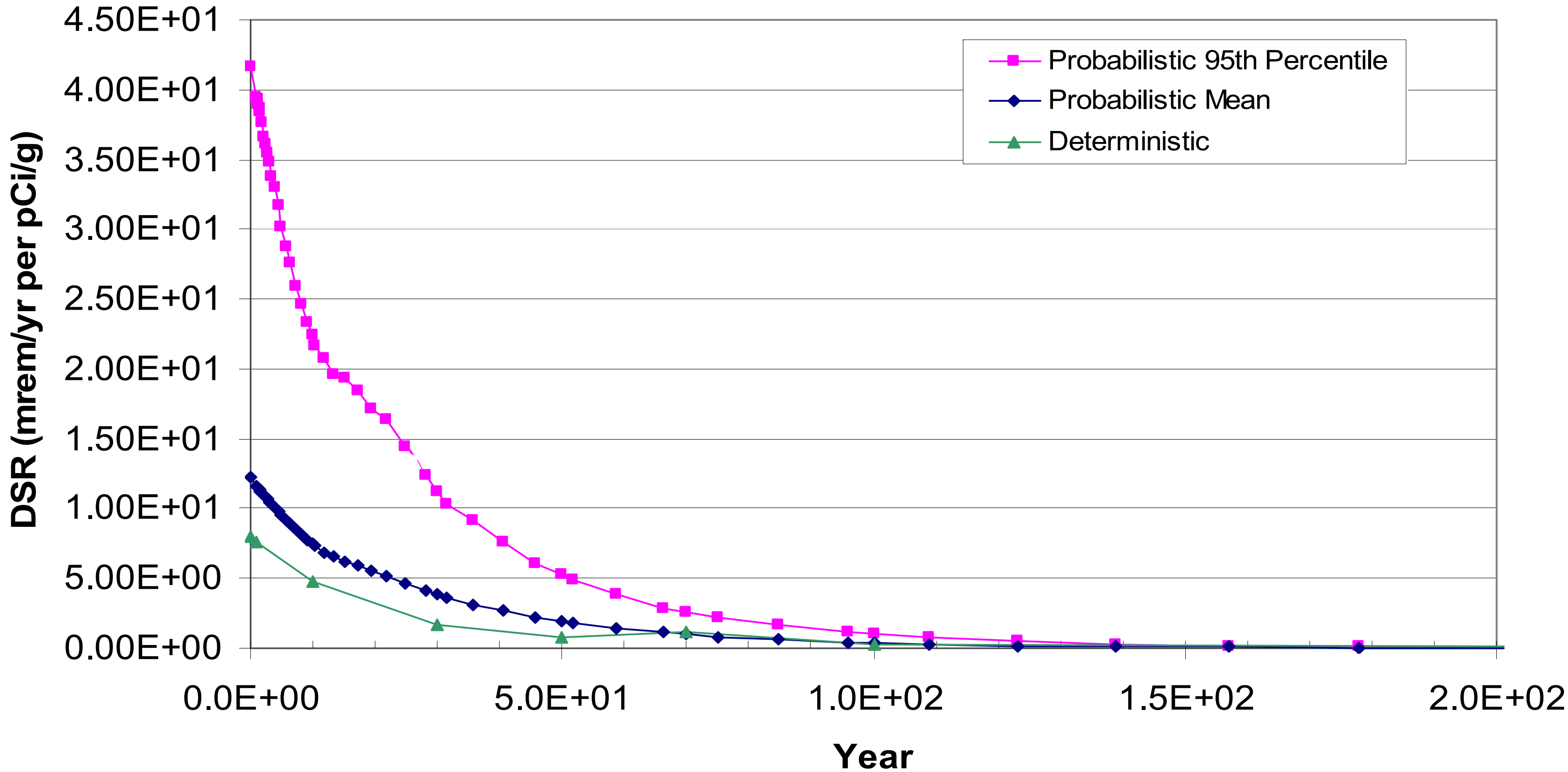



**Figure Att-1. Probabilistic and Deterministic Dose-Source Ratio vs. Time, Sr-90 – Surface Soil**

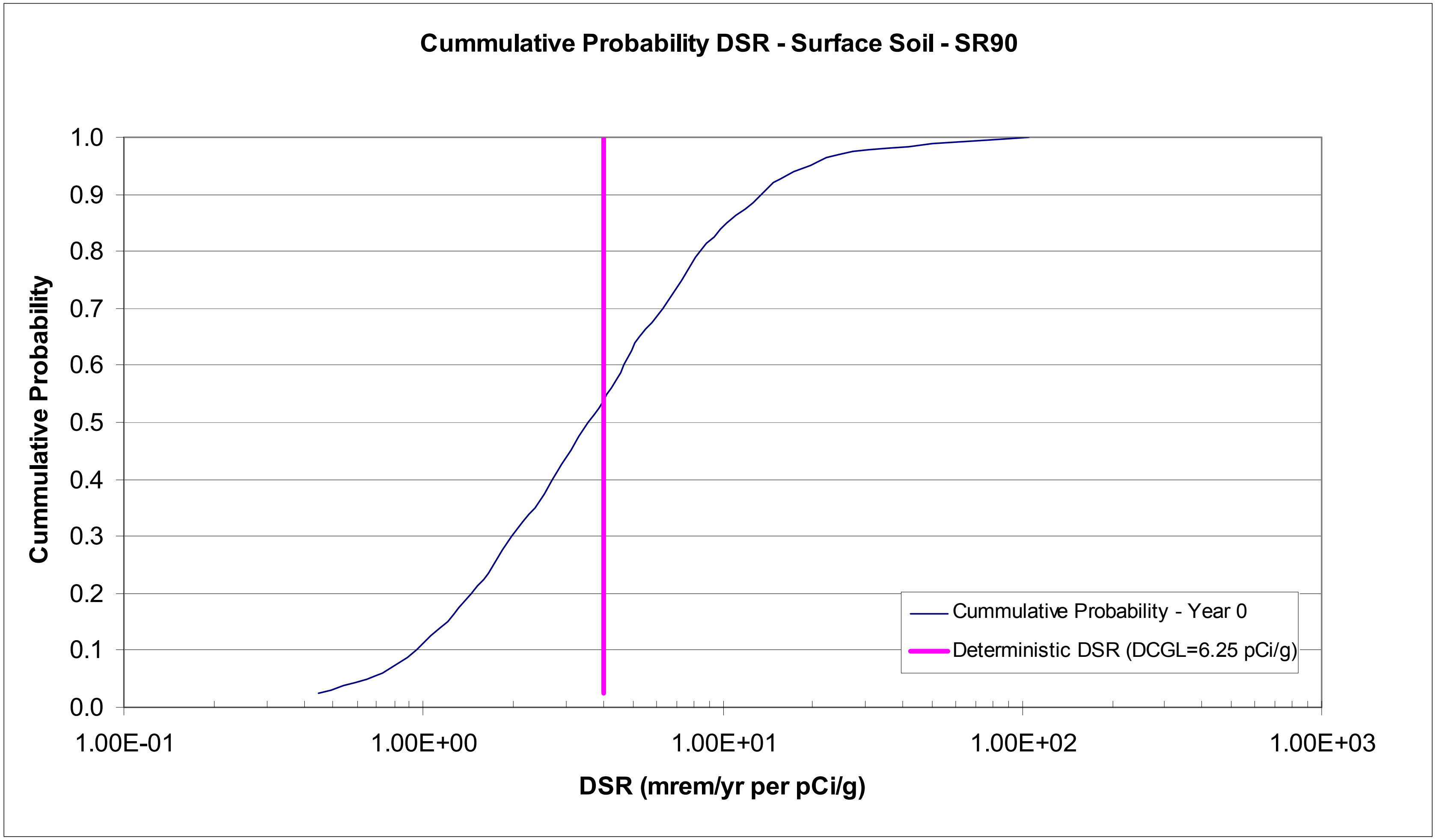



**Figure Att-2. Cumulative Probability Dose-Source Ratio, Sr-90 – Surface Soil**

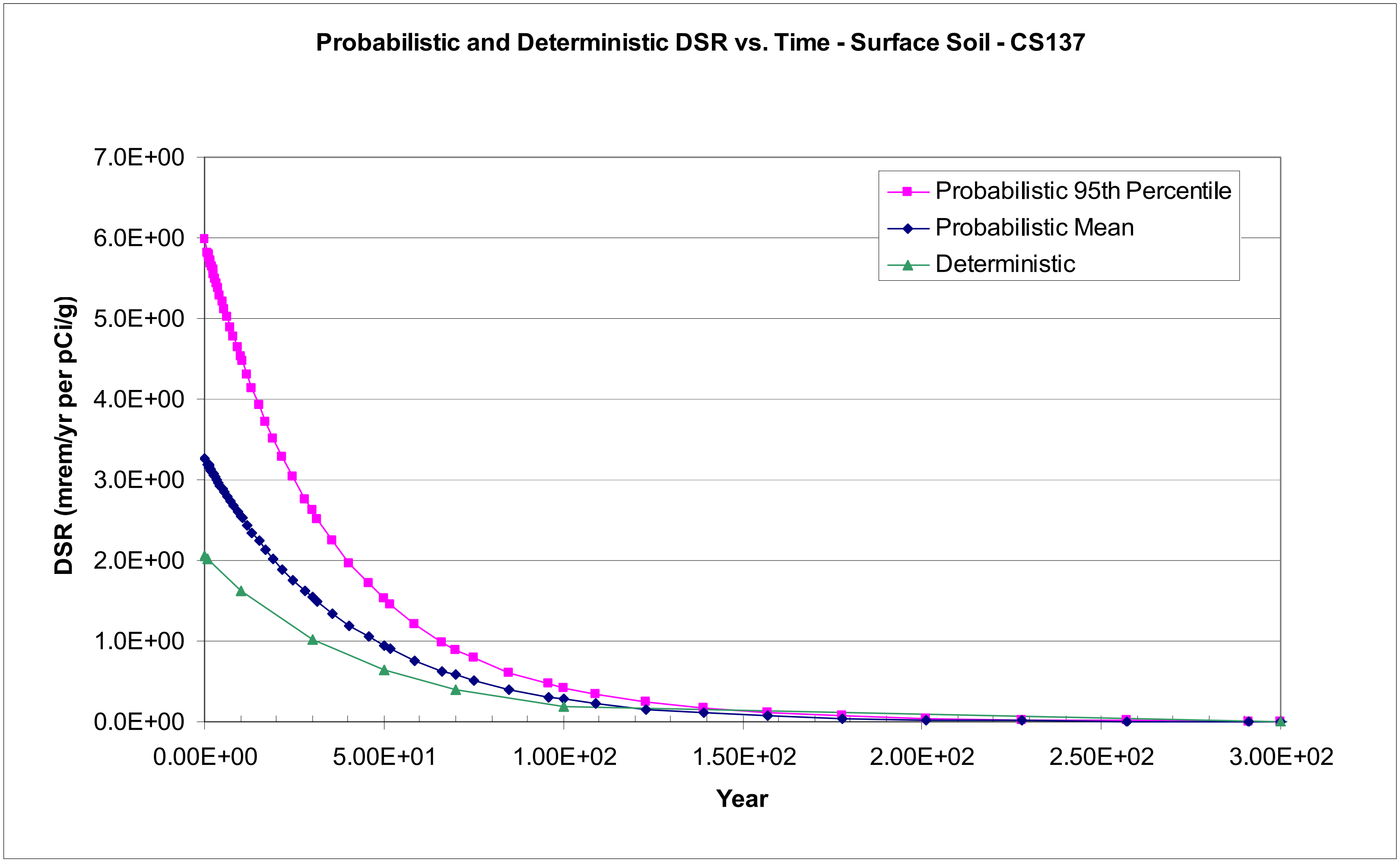



**Figure Att-3. Probabilistic and Deterministic Dose-Source Ratio, Cs-137 – Surface Soil**

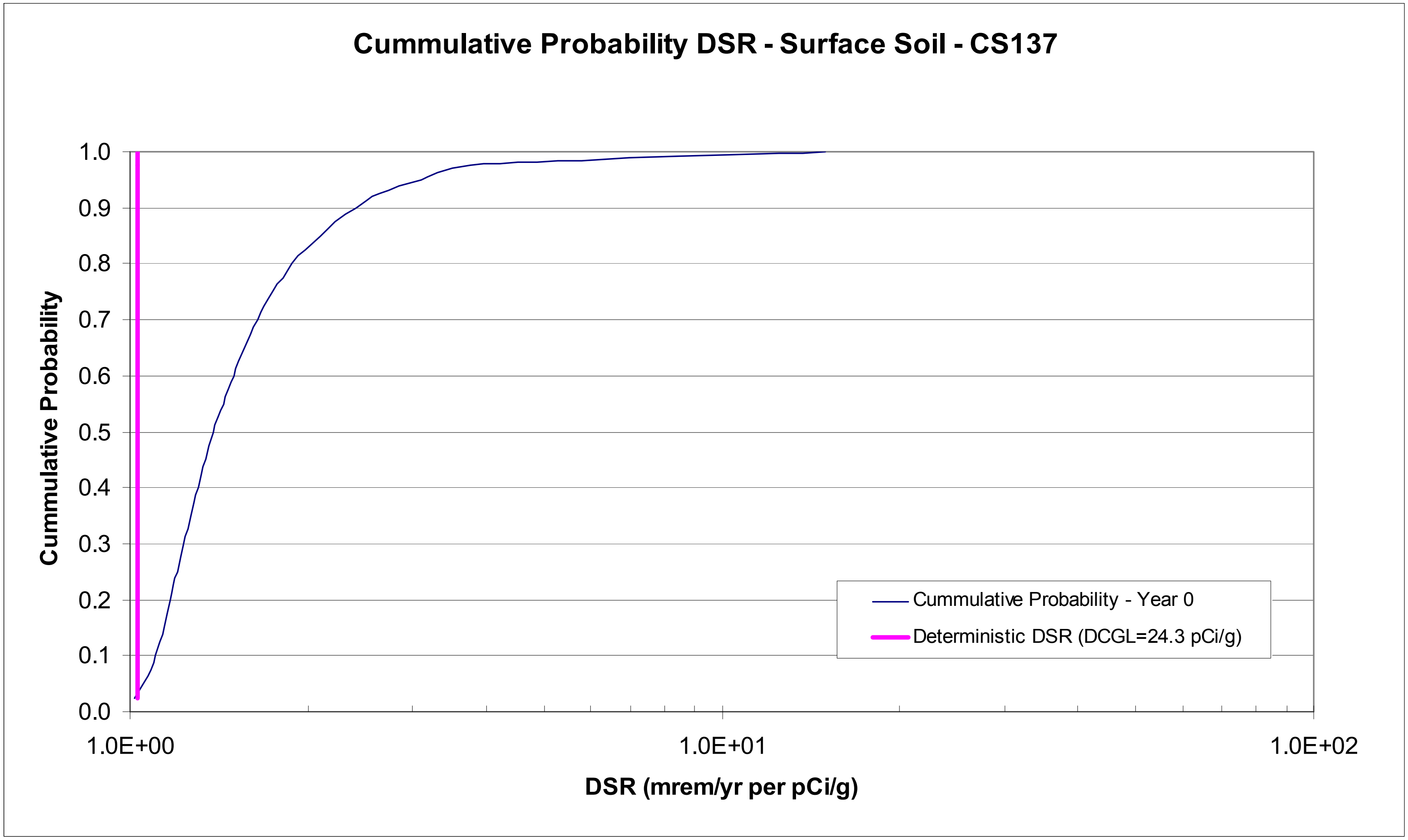



**Figure Att-4. Cumulative Probability Dose-Source Ratio, Cs-137 – Surface Soil**

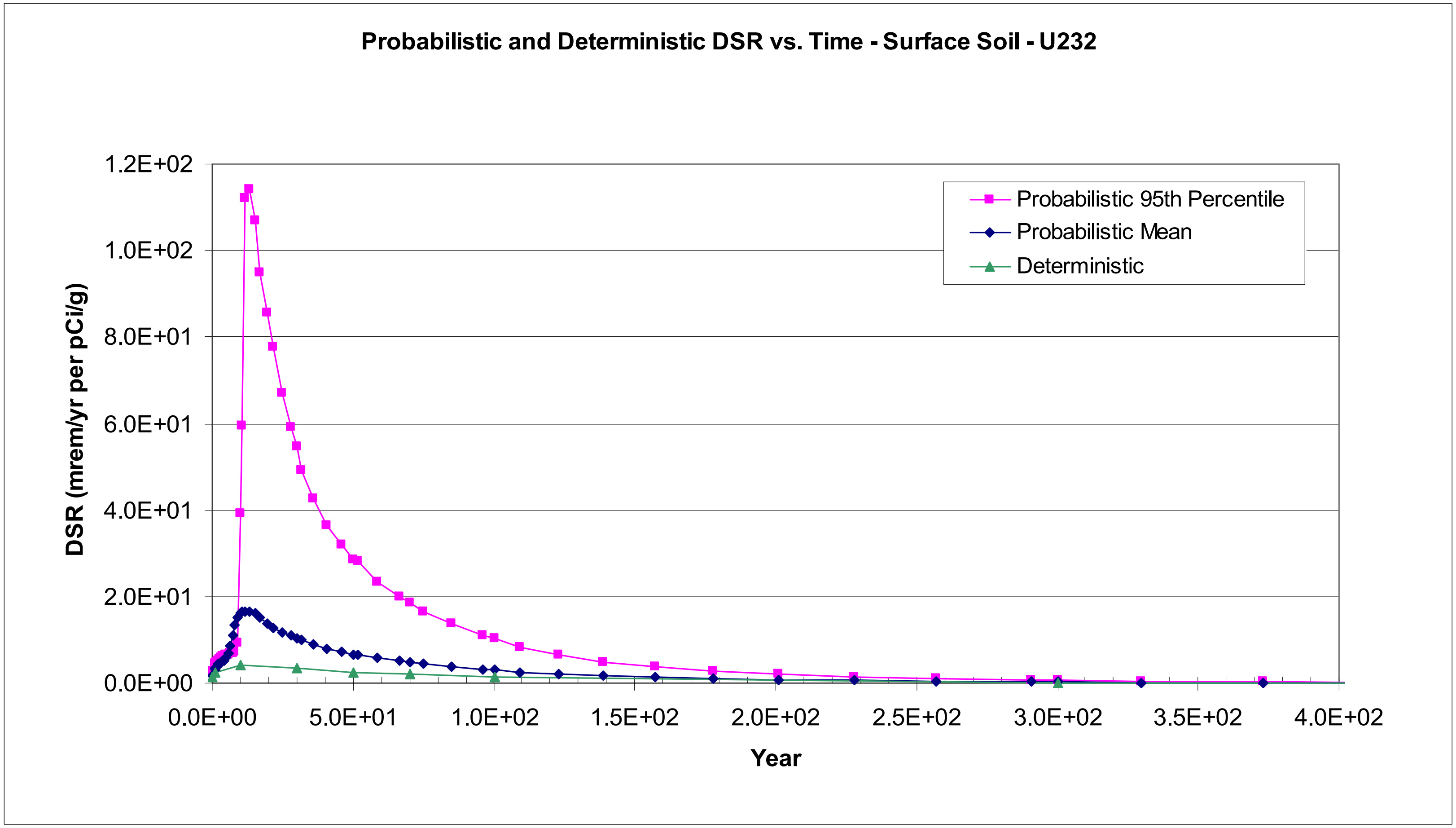



**Figure Att-5. Probabilistic and Deterministic Dose-Source Ratio vs. Time, U-232 – Surface Soil**

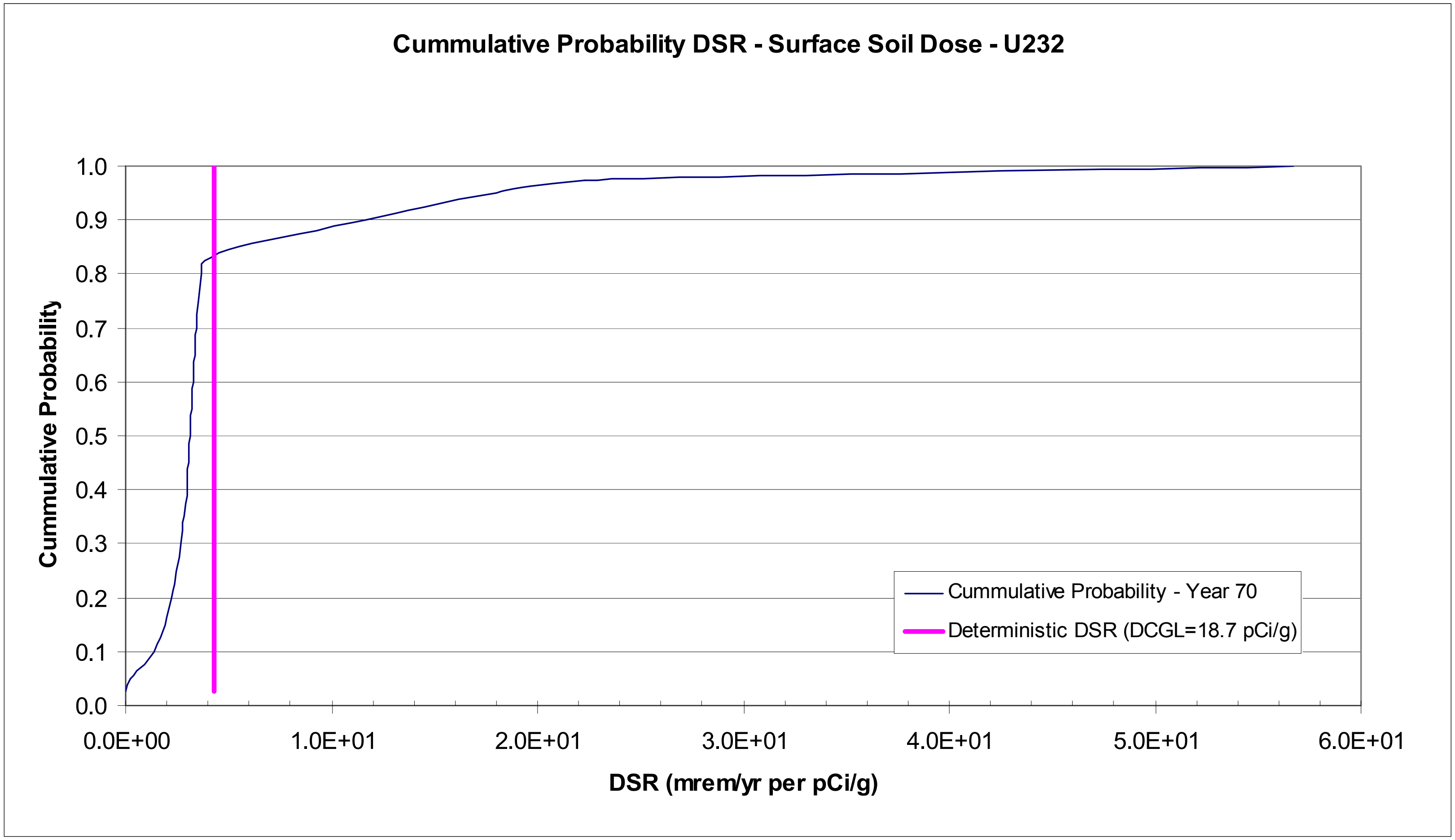



**Figure Att-6. Cumulative Probability Dose-Source Ratio, U-232 – Surface Soil**

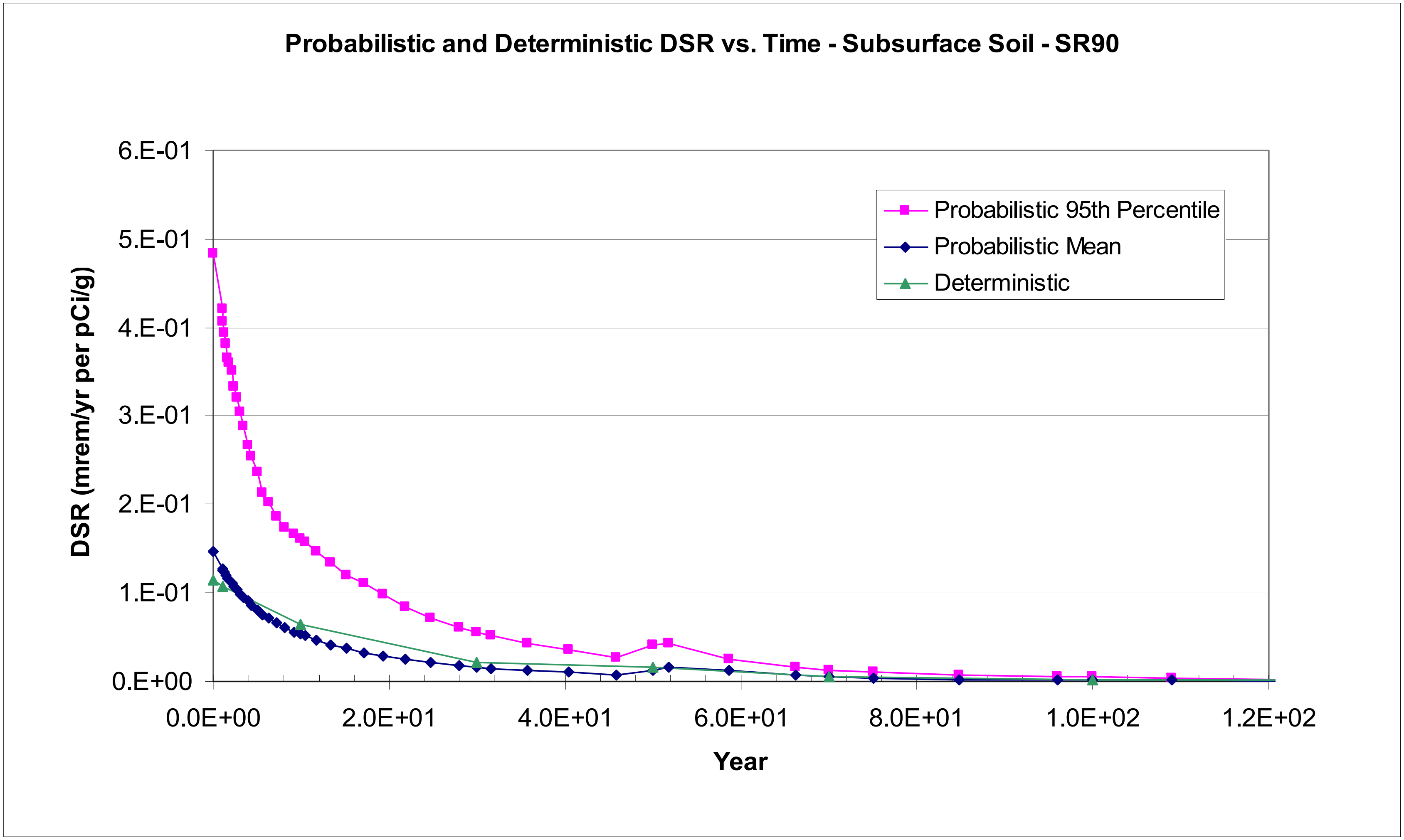



**Figure Att-7. Probabilistic and Deterministic Dose-Source Ratio vs. Time, Sr-90 – Subsurface Soil**

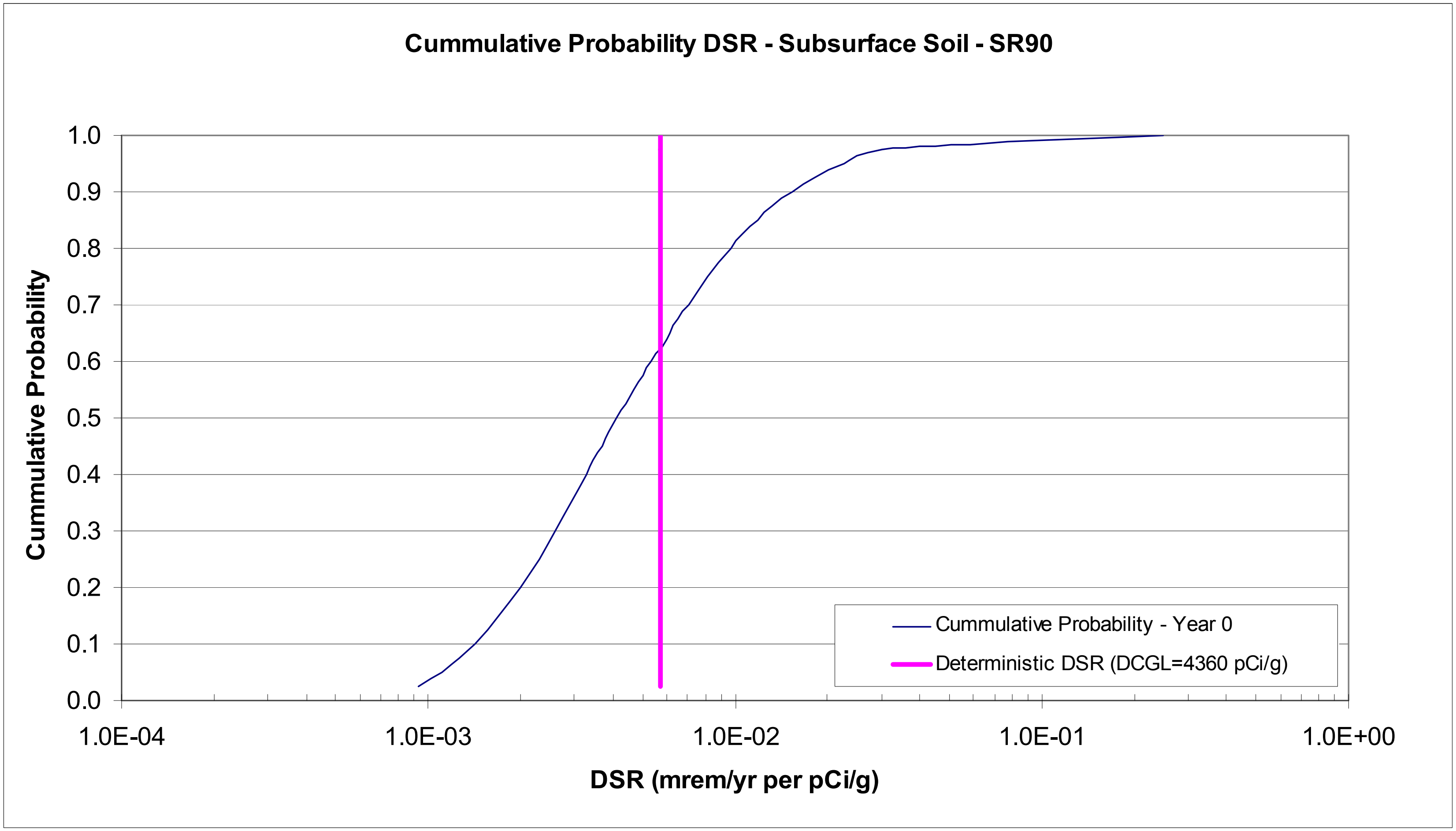



**Figure Att-8. Cumulative Probability Dose-Source Ratio, Sr-90 – Subsurface Soil**

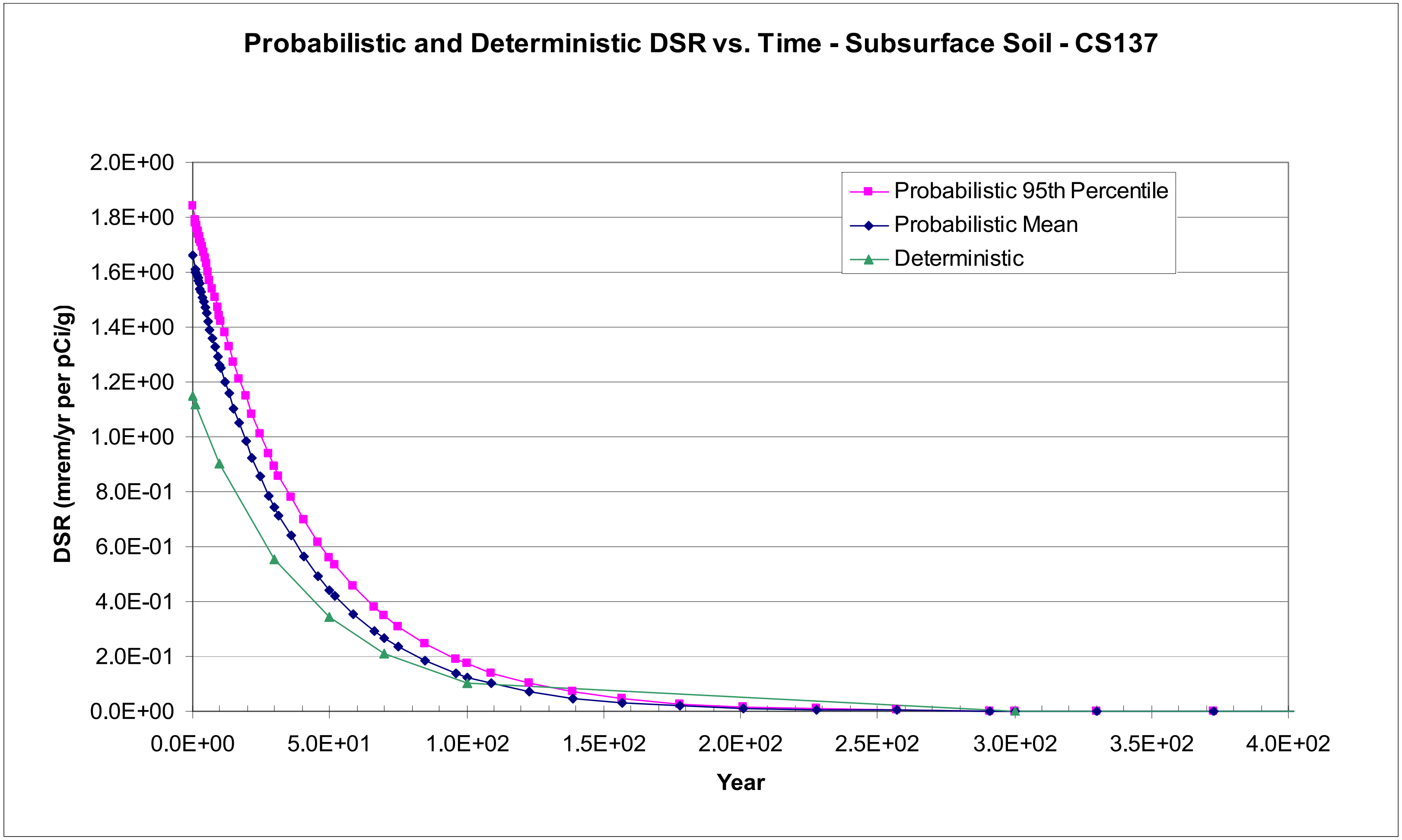



**Figure Att-9. Probabilistic and Deterministic Dose-Source Ratio vs. Time, Cs-137 – Subsurface Soil**

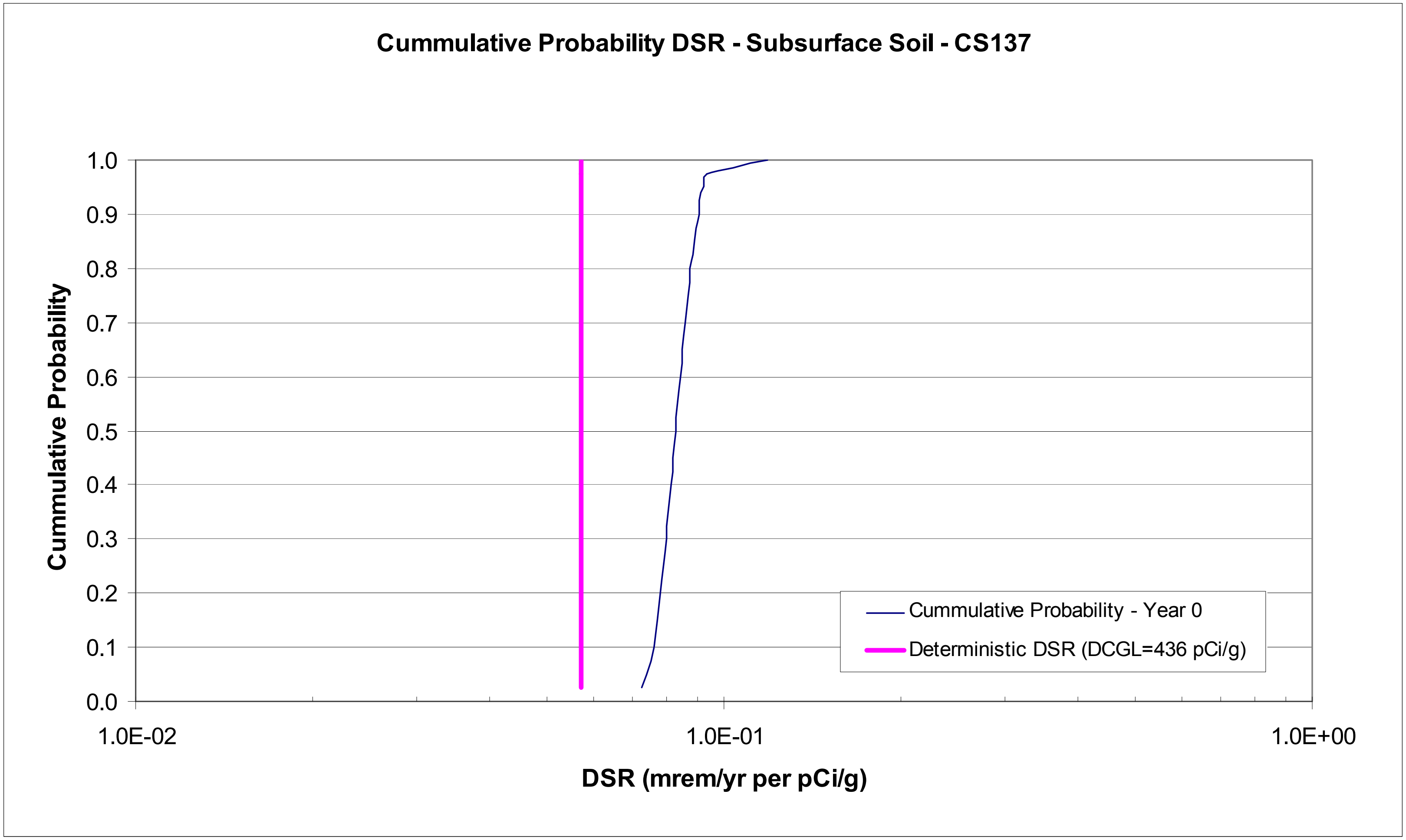



**Figure Att-10. Cumulative Probability Dose-Source Ratio, Cs-137 – Subsurface Soil**

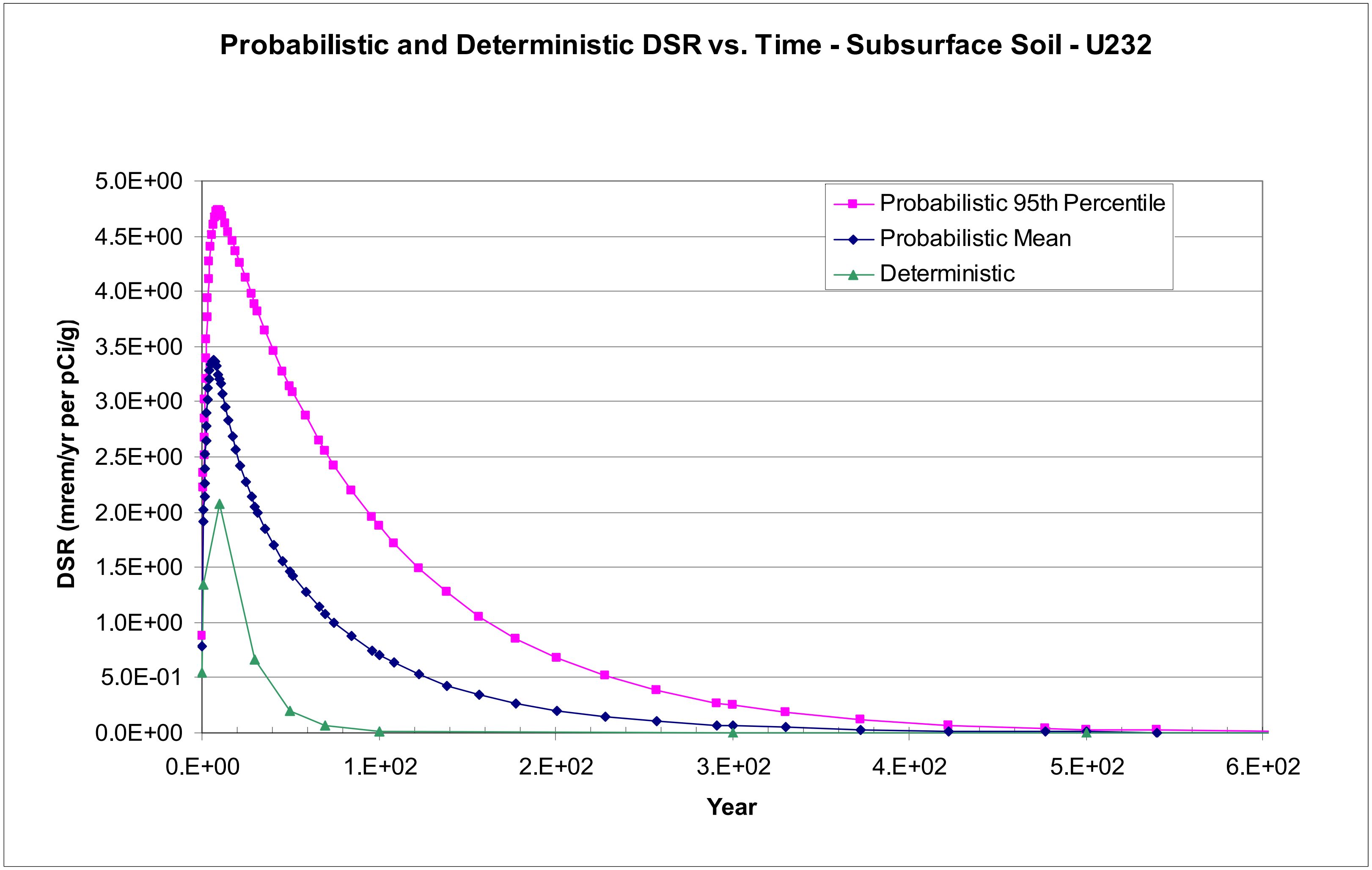



**Figure Att-11. Probabilistic and Deterministic Dose-Source Ratio vs. Time, U-232 – Subsurface Soil**

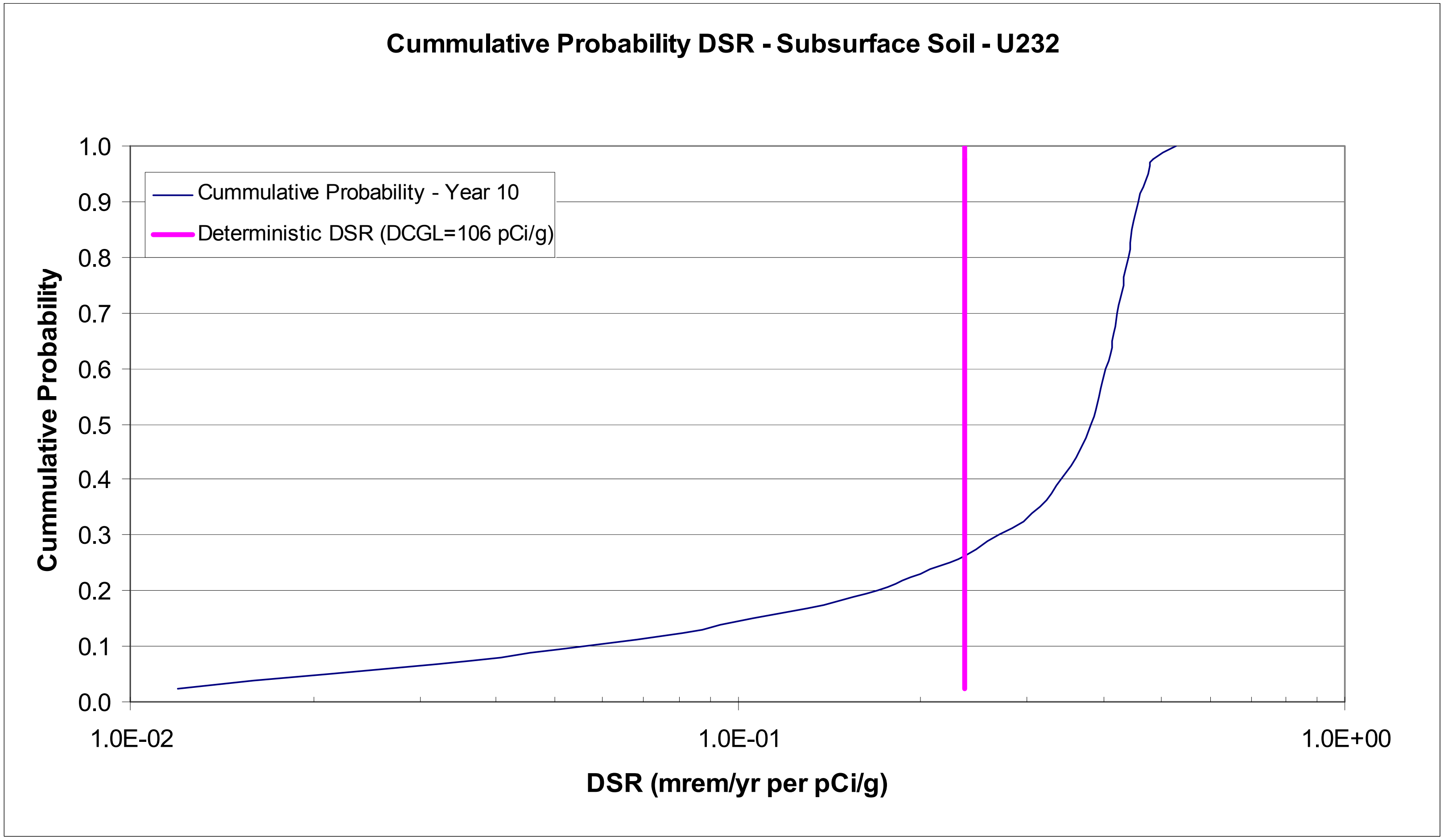



**Figure Att-12. Cumulative Probability Dose-Source Ratio, U-232, Subsurface Soil**

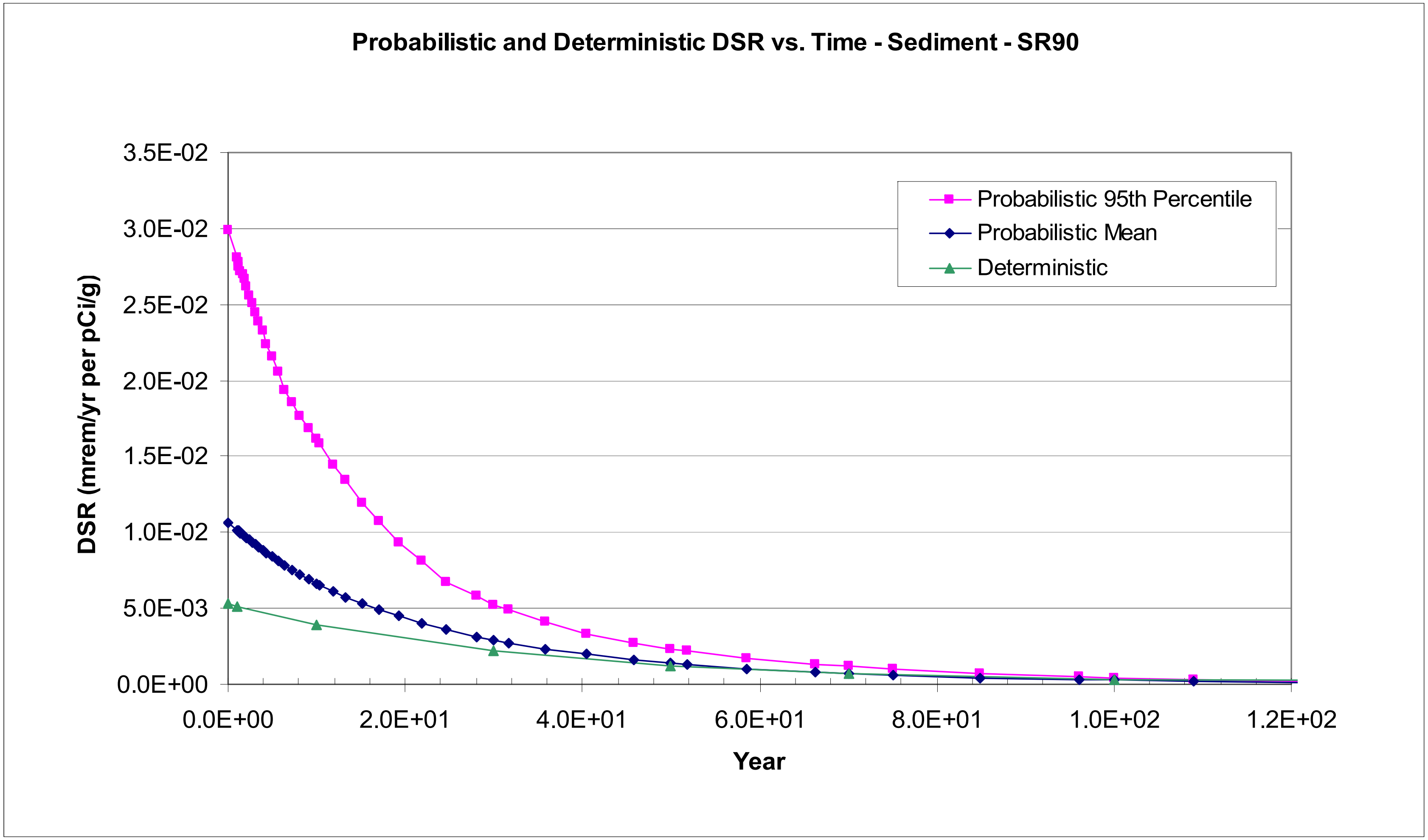



**Figure Att-13. Probabilistic and Deterministic Dose-Source Ratio vs. Time, Sr-90 – Streambed Sediment**

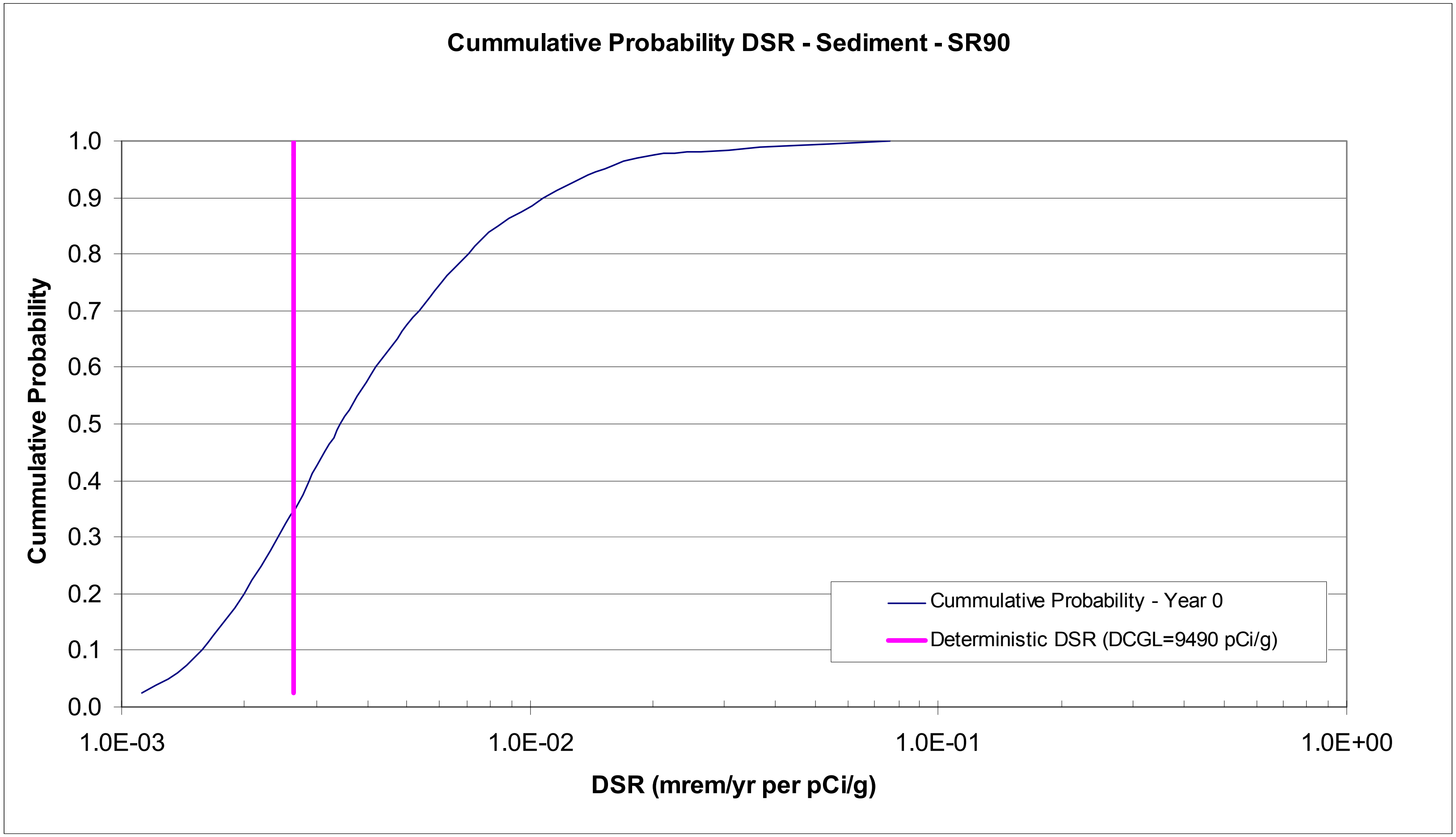



**Figure Att-14. Cumulative Probability Dose-Source Ratio, Sr-90 – Streambed Sediment**

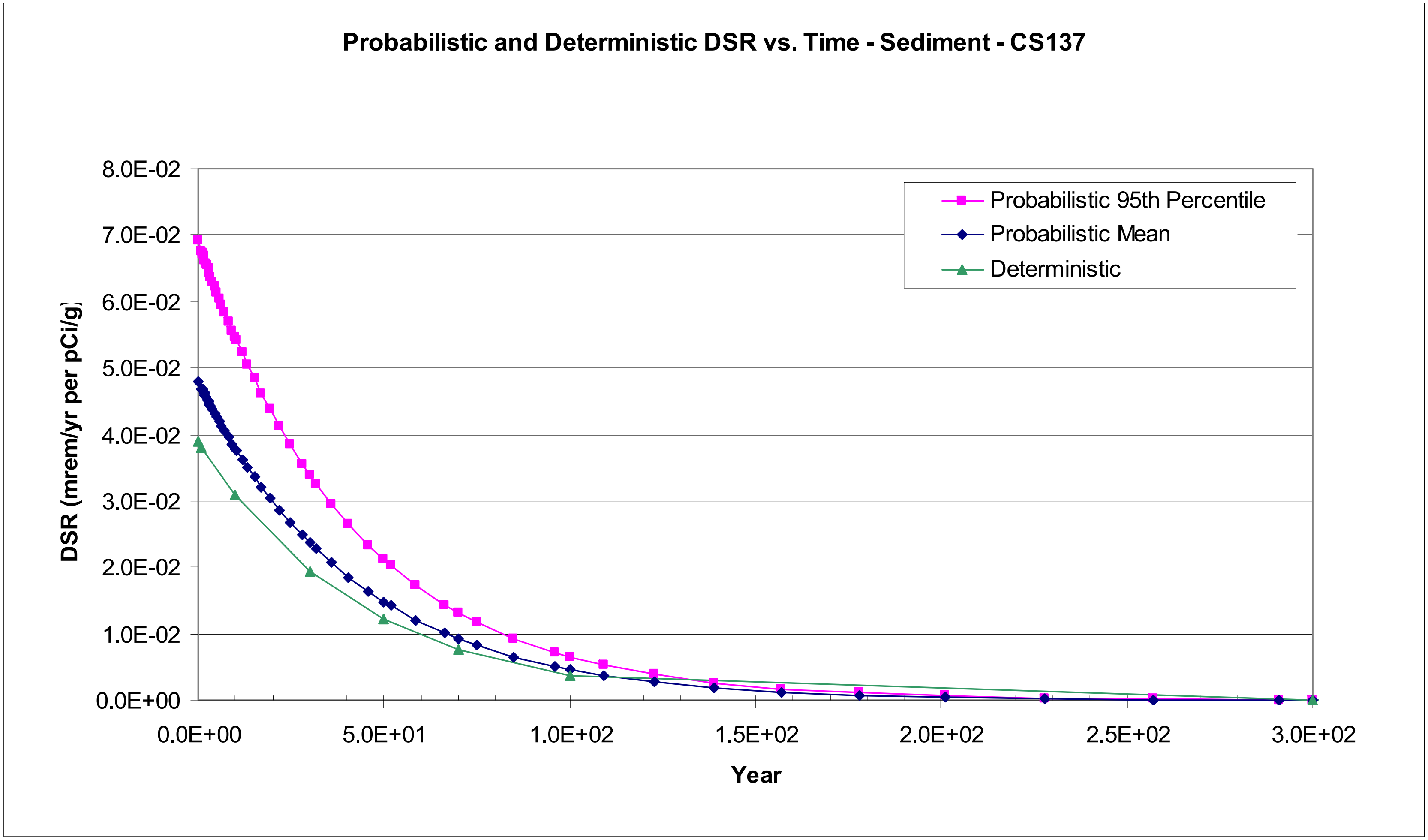



**Figure Att-15. Probabilistic and Deterministic Dose-Source Ratio vs. Time, Cs-137 – Streambed Sediment**

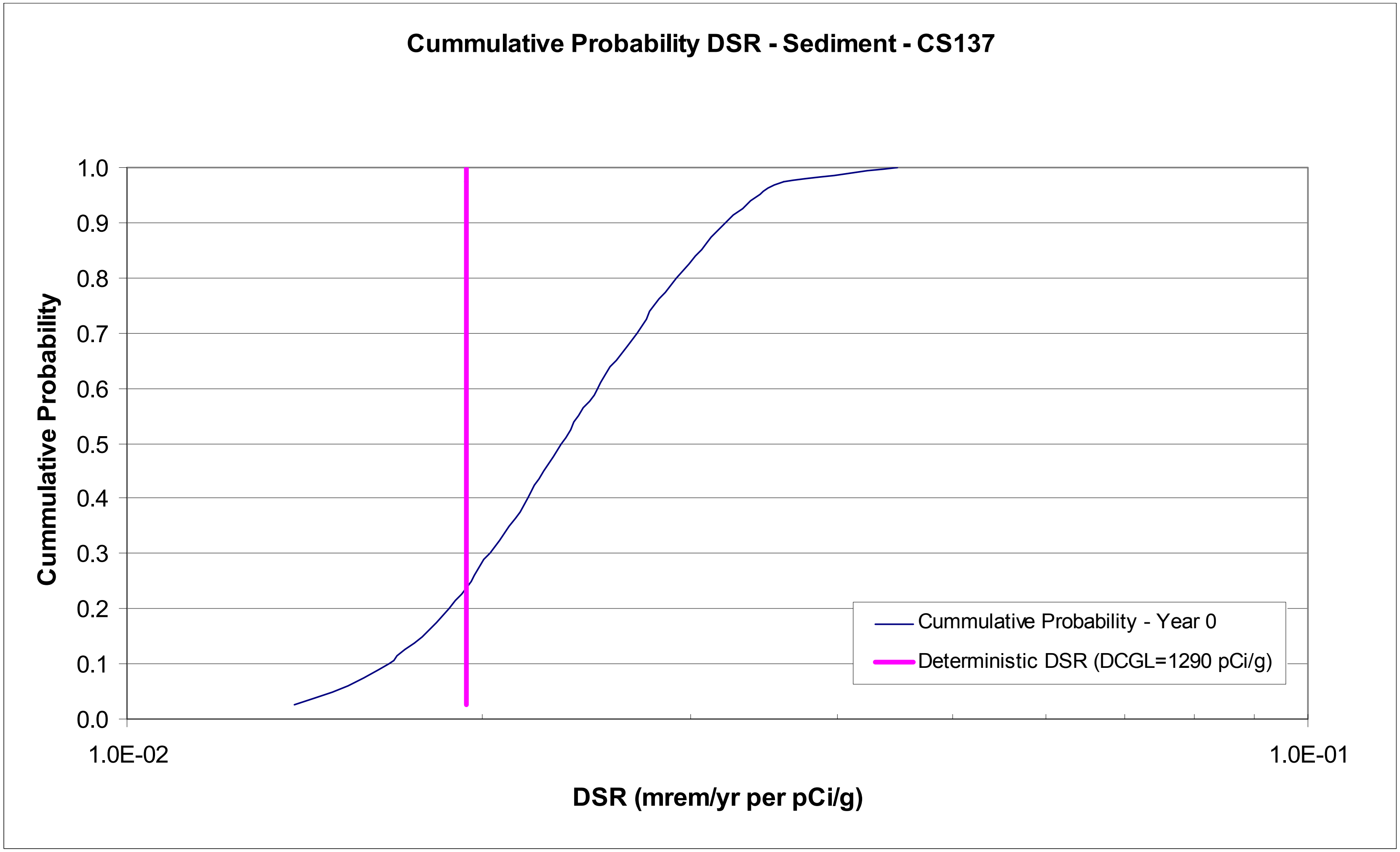



**Figure Att-16. Cumulative Probability Dose-Source Ratio, Cs-137 – Streambed Sediment**

**Table Att-1. Estimated WMA 1 Doses from Observed Maximum Radionuclide Concentrations in the Lavery Till**

| Radionuclide | Maximum Detection (pCi/g)(1) | Depth (ft) | Peak-of-the-Mean Subsurface Soil $DCGL_w$ (pCi/g)(2) | 95th Percentile Subsurface Soil $DCGL_w$ (pCi/g) | Peak-of-the-Mean Estimated Dose (mrem/y)(3) | 95th Percentile Estimated Dose (mrem/y)(3) |
|---|---|---|---|---|---|---|
| Am-241 | 1.3E-01 | 38-40 | 6.8E+03 | 4.3E+03 | 4.8E-04 | 7.6E-04 |
| C-14 | 1.1E-01 | 38-40 | 3.7E+05 | 3.6E+05 | 7.3E-06 | 7.5E-06 |
| Cs-137 | 3.9E+00 | 38-40 | 3.0E+02 | 2.7E+02 | 3.6E-01 | 3.6E-01 |
| Cm-243 | 2.3E-02 | 38-40 | 1.1E+03 | 9.3E+02 | 6.2E-04 | 6.2E-04 |
| Cm-244 | 2.3E-02 | 38-40 | 2.2E+04 | 1.1E+04 | 5.3E-05 | 5.3E-05 |
| I-129 | 2.9E-01 | 38-40 | 5.2E+01 | 5.2E+01 | 1.4E-01 | 1.4E-01 |
| Np-237 | 2.1E-02 | 37-39 | 4.3E+00 | 4.3E+00 | 1.2E-01 | 1.2E-01 |
| Pu-238 | 2.3E-02 | 38-40 | 1.4E+04 | 6.8E+03 | 4.2E-05 | 8.4E-05 |
| Pu-239 | 6.4E-02 | 38-40 | 1.2E+04 | 6.1E+03 | 1.3E-04 | 2.6E-04 |
| Pu-240 | 6.4E-02 | 38-40 | 1.2E+04 | 6.4E+03 | 1.3E-04 | 2.5E-04 |
| Pu-241 | 5.7E-01 | 38-40 | 2.4E+05 | 1.6E+05 | 5.9E-05 | 8.9E-05 |
| Sr-90 | 5.9E+01 | 38.5-39 | 3.2E+03 | 1.0E+03 | 4.6E-01 | 1.4E+00 |
| Tc-99 | 5.5E-01 | 37-39 | 1.1E+04 | 4.4E+03 | 1.2E-03 | 3.2E-03 |
| U-232 | 4.1E-02 | 24-26 | 7.4E+01 | 5.4E+01 | 1.4E-02 | 1.9E-02 |
| U-233 | 2.3E+00 | 38-40 | 1.9E+02 | 1.9E+02 | 3.0E-01 | 3.0E-01 |
| U-234 | 2.3E+00 | 38-40 | 2.0E+02 | 2.0E+02 | 2.9E-01 | 2.9E-01 |
| U-235 | 1.4E-01 | 24-26 | 2.1E+02 | 2.1E+02 | 1.7E-02 | 1.7E-02 |
| U-238 | 1.4E+00 | 41-43 | 2.1E+02 | 2.1E+02 | 1.7E-01 | 1.7E-01 |
| | | | | **Total Estimated Dose** | 1.9E+00 | 2.8E+00 |

NOTES: (1) Maximum detections from Table 5-1. Radionuclides with maximum detections below the detection limit were evaluated at the detection limit.

(2) Subsurface DCGLs are presented in Appendix E and account for 10 to 1 dilution of contaminated till with clean overlying soil during excavation. Subsurface DCGL are the lower of the deterministic values for the resident gardener and farmer or the probabilistic value for the farmer.

(3) Estimated dose (mrem/y) = 25 (mrem/y) x (maximum detection / $DCGL_W$)

**Table Att-2. Estimated WMA 2 Doses from Observed Maximum Radionuclide Concentrations in the Lavery Till**

| Radionuclide | Maximum Detection (pCi/g)(1) | Depth (ft) | Peak-of-the-Mean Subsurface Soil $DCGL_w$ (pCi/g)(2) | 95th Percentile Subsurface Soil $DCGL_w$ (pCi/g) | Peak-of-the-Mean Estimated Dose (mrem/y)(3) | 95th Percentile Estimated Dose (mrem/y)(3) |
|---|---|---|---|---|---|---|
| Am-241 | 3.0E-02 | 12-14 | 6.8E+03 | 4.3E+03 | 1.1E-04 | 1.7E-04 |
| C-14 | None | None | 3.7E+05 | 3.6E+05 | NA | NA |
| Cm-243 | None | None | 1.1E+03 | 9.3E+02 | NA | NA |
| Cm-244 | None | None | 2.2E+04 | 1.1E+04 | NA | NA |
| Cs-137 | 4.5E-01 | 12-14 | 3.0E+02 | 2.7E+02 | 4.1E-02 | 4.1E-02 |
| Np-237 | None | None | 4.3E+00 | 4.3E+00 | NA | NA |
| I-129 | None | None | 5.2E+01 | 5.2E+01 | NA | NA |
| Pu-238 | 1.0E-02 | 12-14 | 1.4E+04 | 6.8E+03 | 1.8E-05 | 3.7E-05 |
| Pu-239 | 5.9E-03 | 12-14 | 1.2E+04 | 6.1E+03 | 1.2E-05 | 2.4E-05 |
| PU-240 | 5.9E-03 | 12-14 | 1.2E+04 | 6.4E+03 | 1.2E-05 | 2.3E-05 |
| Pu-241 | 1.3E+00 | 12-14 | 2.4E+05 | 1.6E+05 | 1.4E-04 | 2.0E-04 |
| Sr-90 | 8.5E-01 | 12-14 | 3.2E+03 | 1.0E+03 | 6.7E-03 | 2.1E-02 |
| Tc-99 | None | None | 1.1E+04 | 4.4E+03 | NA | NA |
| U-232 | 1.2E-02 | 12-14 | 7.4E+01 | 5.4E+01 | 4.1E-03 | 5.5E-03 |
| U-233 | 1.8E-01 | 12-14 | 1.9E+02 | 1.9E+02 | 2.3E-02 | 2.3E-02 |
| U-234 | 1.8E-01 | 12-14 | 2.0E+02 | 2.0E+02 | 2.3E-02 | 2.3E-02 |
| U-235 | 5.9E-03 | 12-14 | 2.1E+02 | 2.1E+02 | 7.1E-04 | 7.1E-04 |
| U-238 | 1.1E-01 | 12-14 | 2.1E+02 | 2.1E+02 | 1.3E-02 | 1.3E-02 |
| | | | | **Total Estimated Dose** | **1.1E-01** | **1.3E-01** |

NOTES: (1) Maximum detections from Table 5.1. Radionuclides with maximum detections below the detection limit were evaluated at the detection limit.

(2) Subsurface DCGLs are presented in Appendix E and account for 10 to 1 dilution of contaminated till with clean overlying soil during excavation. Subsurface DCGL are the lower of the deterministic values for the resident gardener and farmer or the probabilistic value for the farmer.

(3) Estimated dose (mrem/y) = 25 (mrem/y) x (maximum detection / $DCGL_W$)

LEGEND: NA = not available

# APPENDIX F

# ESTIMATED RADIOACTIVITY IN SUBSURFACE PIPING

**PURPOSE OF THIS APPENDIX**

The purpose of this appendix is to provide conservative estimates of residual radioactivity in underground piping to supplement information on the radiological status of facilties discussed in Section 4.1.

**INFORMATION IN THIS APPENDIX**

Information in this appendix was drawn from a radioisotope inventory report completed in July 2004. Included are a list of all buried pipelines and estimates for residual activity in pipelines in three areas: (1) beneath the Process Building, (2) west of the Process Building, and (3) east of the Process Building. An estimate is also included for residual radioactivity in the Leachate Transfer Line that runs from the NRC-Licensed Disposal Area (NDA) to Lagoon 2.

**RELATIONSHIP TO OTHER PARTS OF THE PLAN**

The information in this appendix supplements the information provided in Section 4 and supports the decommissioning activities described in Section 7.

## 1.0 Introduction

Various underground lines in WMA 1 and WMA 2 carried radioactive liquid during NFS and WVDP operations. All were evaluated and conservative estimates of residual radioactivity were made as described in the radioisotope inventory report (Luckett, et al. 2004). During this evaluation, the sources were divided into categories, including:

- Lines beneath the footprint of the Process Building,
- High-activity lines primarily west of the Process Building,
- Low-activity lines primarily east of the Process Building, and
- The leachate transfer line from the NDA to Lagoon 2.

The evaluation process included the following steps:

- Collection and review of available information and data on pipe design and location;
- Consideration of process history to determine which lines had actually carried radioactive liquid;
- Review of radiological data and inventories generated by the Facility Characterization Project;
- Preparation of activity estimates for indicator radionuclides based on (1) data on fluids carried by the pipes and an empirical relationship between the activity of the HLW fluid

and the resulting residual contamination on the pipe interior or (2) the results of surveys of rooms and systems where the pipe contents originated;

- Application of conservative radionuclide distribution scaling factors from the point of origin of the contamination to produce a conservative estimate of the activity in each line; and
- Combining individual line estimates into conservative curie estimates, that were corrected for decay and ingrowth to 2011, for groups of related lines appropriate to dose modeling.

A listing of the underground lines identified in the evaluation is provided in Table F-1. The column "Radionuclide Distribution Surrogate" refers to the distribution of radionuclide ratios assigned to each line, based on process history, the origin and terminus of the line, and the geographic location category. Note that acronyms used in the table are defined in the legend at the end of the table. Residual activity estimated to remain inside the lines is summarized below in Section 2 through 4 of this appendix. Details of the calculations, a discussion of the basis for the assignment of the surrogate radionuclide distribution, and the surface contamination ($\mu Ci/m^2$) for each radionuclide in each of the distributions are provided in Luckett, et al. 2004.

**Table F-1. List of Buried Pipelines**

| Line Number | Pipe Dia. (in) | From | To | Length (feet) | | | Radionuclide Distribution Surrogate |
|---|---|---|---|---|---|---|---|
| | | | | Below Process Bldg | West of Process Bldg | East of Process Bldg | |
| 1P64-1 | 1 | FRS | MSM Valve Pit | 25 | 0 | 400 | CD Pit |
| 7P19-1 | 1 | Miniature Cell | Tank 7D-14 | 70.6 | 0 | 0 | Not Used |
| 7P331a-3 | 0.25 | Tank 7D-13 | capped | 0 | 30 | 0 | Tank 7D-13 |
| 7P331b-3 | 0.25 | Tank 7D-13 | 7D-13 Sample station southwest stairwell | 0 | 30 | 0 | Tank 7D-13 |
| 7P331c-2 | 0.50 | Tank 7D-13 | 7D-13 Sample station southwest stairwell | 0 | 30 | 0 | Tank 7D-13 |
| 7P63-1 | 1 | Tank 7D-8 | Miniature Cell | 76.6 | 0 | 0 | Not Used |
| 7P71-3 | 3 | CPC Floor | 59 ft Outside Bldg Capped | 70 | 59 | 0 | Not Used |
| 7P74-3 | 3 | CPC Floor | 59 ft Outside Bldg Capped | 70 | 59 | 0 | Not Used |
| 7P90-3 | 3 | CPC Floor | 59 ft Outside Bldg Capped | 70 | 59 | 0 | Not Used |
| 7P112-3 | 3 | CPC Floor | Tank 8D-1 | 65.8 | 462 | 0 | Not Used |
| 7P113-3 | 3 | Tank 7D-10/ CPC Floor | Tank 8D-2 | 64.3 | 462 | 0 | 7P113 |
| 7P114-3 | 3 | CPC Floor | 59 ft Outside Bldg Capped | 67.5 | 59 | 0 | Not Used |
| 7P115-3 | 3 | CPC Floor | 59 ft Outside Bldg Capped | 67.6 | 59 | 0 | Not Used |
| 7P116-3 | 3 | CPC Floor | 59 ft Outside Bldg Capped | 67.7 | 59 | 0 | Not Used |
| 7P120-3 | 3 | Tank 7D-4/ CPC Floor | THOREX to 8D-4 | 58.7 | 462 | 0 | 7P120 |
| 7P151-3 | 3 | Tank 7D-10 | Future HLW Storage Capped 59 ft Outside Bldg | 68.2 | 59 | 0 | Not Used |

**Table F-1. List of Buried Pipelines**

| Line Number | Pipe Dia. (in) | From | To | Length (feet) | | | Radionuclide Distribution Surrogate |
|---|---|---|---|---|---|---|---|
| | | | | Below Process Bldg | West of Process Bldg | East of Process Bldg | |
| 7P156-2 | 2 | Tank 7D-13 Vent | OGC | 35.6 | 20 | 0 | Tank 7D-13 |
| 7P159-2 | 2 | Tank 7D-13 Jet | GP Catch Tank 7C-5 | 0 | 60 | 0 | Tank 7D-13 |
| 7P170-2 | 2 | 7C-5 Jet | Tank 8D-1 | 0 | 482 | 0 | Tank 8D-1 |
| 7P177-12 | 1.5 | 7 E-13 GP Evap. | 7D-13 | 0 | 60 | 0 | Tank 7D-13 |
| 7P180-12 | 1.5 | 7 E-13 via 7P177 | 15WW568 | 0 | 10 | 0 | WW |
| 7P271-2 | 2 | 7D-6 Weak Acid Catch Tank Pump 7G-1 | Interceptor | 0 | 10 | 0 | WW |
| 8P11-2 | 2 | Tank 8D-1 8G-4 | Lagoon | 0 | 0 | 825 | Vault Drip Pan |
| 8P12-3 | 3 | Waste Tank Off Gas Knockout Drum 8D-6 | Tank 8D-1 | 0 | 41 | 0 | Tank 8D-1 |
| 8P27-3 | 3 | Waste Tank Off Gas Knockout Drum 8D-6 | Tank 8D-2 | 0 | 52 | 0 | Tank 8D-2 |
| 8P29-16 | 16 | Tanks 8D-1 via 8P13; and 8D-2 via 8P28; and PVS | Waste Tank Off Gas Condensers and Relief Knock Out Drum 8D-7 | 0 | 52 | 0 | 8P29-16 |
| 8P34-2 | 2 | Waste Tank O/H Condensate Pump 8G-1 | 7C-5 | 0 | 425 | 0 | Tank 8D-2 |
| 8P35-2 | 2 | Waste Tank Cond. Pump 8G-1 via 8P34 | 8D-2 via 7P170 | 0 | 5 | 0 | Tank 8D-2 |
| 8P38-2 | 2 | Waste Tank Blowers 8K-1/ 8K-1A VIA 8P-46 | Tank 8D-2 via 8P-27 | 0 | 5 | 0 | Tank 8D-2 |
| 8P46-6 (old) | 6 | Waste Tank Blowers 8K-1/8K-1A | Stack 15F-1 | 0 | 435 | 0 | 8P46-6 |
| 8P46-6 (new) | 6 | Waste Tank Blowers 8K-1/8K-1A | To line 6P95-8 | 0 | 415 | 0 | 8P46-6 |
| 8P68-2 | 2 | Equipment shelter Manifold | Lagoon | 0 | 52 | 0 | Vault Drip Pan |
| 8P95-3 | 3 | Con Ed Tank 8C-1 Caustic Scrubber | Tank 8D-6 Off-Gas Knockout Drum | 0 | 52 | 0 | Tank 8D-4 |
| 8P120-3 | 3 | | Tank 8D-1 | 0 | 52 | 0 | Tank 8D-1 |
| 4P92-12 | 1.5 | Tank 4D-2 Jet 4H-60 | 59 ft Outside Bldg Capped | 61.8 | 59 | 0 | Not Used |
| 15CH739-3 | 3 | PMC Floor Drain | GPC Sump via 15CH760-3 | 13.2 | 0 | 0 | PMCR |
| 15CH750-3 | 3 | CCR Drain | Tank 35104 via 12CH240-6 | 40.2 | 0 | 0 | CCR |
| 15CH752-3 | 3 | Equipment Decon Room | Tank 35104 via 12CH240-6 | 65.8 | 0 | 0 | EDR |

**Table F-1. List of Buried Pipelines**

| Line Number | Pipe Dia. (in) | From | To | Length (feet) | | | Radionuclide Distribution Surrogate |
|---|---|---|---|---|---|---|---|
| | | | | Below Process Bldg | West of Process Bldg | East of Process Bldg | |
| 15CH753-2 | 2 | GPC Sump Jet and Tank 35104 Eductor | 1st U Cycle Tank 4D-10 | 66.8 | 0 | 0 | GCR |
| 15CH754-12 | 1.5 | From GCR Sump Jet | Tank 7D-2 | 77 | 0 | 0 | GCR |
| 15CH758-3 | 3 | Mechanical Crane Room | Tank 35104 via 12CH240-6 | 65.5 | 0 | 0 | PMCR |
| 15CH760-3 | 3 | PMC Floor Drain | GPC Sump | 47.6 | 0 | 0 | PMCR |
| 15CH763-3 | 3 | Scrap Removal | Tank 35104 via 12CH240-6 | 57.9 | 0 | 0 | SRR |
| 15CH773-3 | 3 | Tank 35104 Eductor 15H-1 | Tank 7D-2 | 98.2 | 0 | 0 | Tank 35104 |
| 15CH774-3 | 3 | CPC/EDR Door Slot Drain | Tank 35104 via 12CH240-6 | 6.6 | 0 | 0 | CPC |
| 1WW48-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 20 | 0 | 0 | CD Pit |
| 1WW49-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 20 | 0 | 0 | CD Pit |
| 1WW50-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 1WW51-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 1WW52-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 1WW53-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 1WW54-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 1WW55-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 1WW56-4 | 4 | FRS Cask Decon Drain | Interceptor via 15WW571-6 | 6.5 | 0 | 0 | CD Pit |
| 02WW359-3 | 3 | Lagoon 1 | Lagoon 2 | 0 | 0 | 540 | WW |
| 02WW360-6 | 6 | LLWTF underslab piping drains | LLWTF Sump | 0 | 0 | 80 | WW |
| 02WW362-6 | 6 | LLWTF underslab piping drains | LLWTF Sump | 0 | 0 | 40 | WW |
| 02WW363-8 | 8 | Sump Manhole, LLWTF | Lagoon 1 | 0 | 0 | 167 | WW |
| 02WW364-3 | 3 | LLWTF underslab piping drains | Lagoon 2 | 0 | 0 | 150 | WW |
| 15WW533-6 | 6 | Neutralization Pit | Interceptor | 0 | 0 | 10 | WW |
| 15WW534-6 | 6 | Neutralization Pit | New Interceptor thru West Valve Pit | 0 | 0 | 120 | WW |
| 15WW536-2 | 2 | West Valve Pit | New Interceptor A | 0 | 0 | 30 | WW |
| 15WW538-4 | 4 | Interceptor B thru E Valve Pit | Lagoon 2 thru new 15WW549-4 | 0 | 0 | 35 | WW |
| 15WW539-4 | 4 | New Interceptor A | E Valve Pit | 0 | 0 | 10 | WW |
| 15WW549-4 | 4 | East of Interceptor | Lagoon 1 | 0 | 0 | 200 | WW |

**Table F-1. List of Buried Pipelines**

| Line Number | Pipe Dia. (in) | From | To | Length (feet) | | | Radionuclide Distribution Surrogate |
|---|---|---|---|---|---|---|---|
| | | | | Below Process Bldg | West of Process Bldg | East of Process Bldg | |
| 15WW567-2 | 2 | Tank 7D-13 | Interceptor thru 15WW568-2 | 80 | 0 | 0 | WW |
| 15WW568-2 | 2 | Tank 7D-13 | Interceptor thru 15WW569-6 | 50 | 0 | 0 | WW |
| 15WW569-6 | 6 | Trunk Line S side Process Bldg | Interceptor thru 15WW533-6 | 100 | 0 | 110 | WW |
| 15WW570-4 | 4 | N side Process Bldg / FRS | Interceptor thru 15WW571-6 | 0 | 0 | 200 | WW |
| 15WW571-6 | 6 | FRS Cask Decon Drains | Interceptor thru 15WW843-6 | 60 | 0 | 13 | CD Pit |
| 15WW841-4 | 4 | N Side of MSM Repair | Interceptor thru 15WW852-3 | 12 | 0 | 25 | WW |
| 15WW842-3 | 3 | E Side of MSM Repair | Interceptor thru 15WW570-4 | 19 | 0 | 15 | WW |
| 15WW843-6 | 6 | Trunk Line East of Process Bldg | Interceptor thru 15WW569-6 | 72 | 0 | 120 | WW |
| 15WW846-3 | 3 | Under Lower Warm Aisle | Interceptor thru 15WW569-6 | 5 | 0 | 0 | WW |
| 15WW847-3 | 3 | Under Lower Warm Aisle | Interceptor thru 15WW569-6 | 5 | 0 | 0 | WW |
| 15WW848-3 | 3 | Trunk line, upper floors South side Process Bldg | Interceptor thru 15WW569-6 | 5 | 0 | 0 | WW |
| 15WW850-4 | 4 | Under Floor RAM Equipment Room | Interceptor thru 15WW843-6 | 16 | 0 | 0 | WW |
| 15WW851-3 | 3 | Under Floor CPC | Interceptor thru 15WW895-4 | 80 | 0 | 0 | WW |
| 15WW852-3 | 3 | Equipment Decon Room | Interceptor thru 15WW570-4 | 13.3 | 0 | 55 | WW |
| 15WW857-3 | 3 | Under Floor PMC | Interceptor thru 15WW851-3 | 45 | 0 | 0 | WW |
| 15WW858-3 | 3 | Under Floor RAM Equipment Room | Interceptor thru 15WW895-4 | 6 | 0 | 0 | WW |
| 15WW859-3 | 3 | Under Floor RAM Equipment Room | Interceptor thru 15WW895-4 | 20 | 0 | 0 | WW |
| 15WW860-3 | 3 | Under Floor Cell Access Aisle | Interceptor thru 15WW851-3 | 16 | 0 | 0 | WW |
| 15WW861-3 | 3 | Under Floor W Main Op Aisle | Interceptor thru 15WW895-4 | 25 | 0 | 0 | WW |
| 15WW863-3 | 3 | Under Floor W Main Op Aisle | Interceptor thru 15WW895-4 | 6 | 0 | 0 | WW |
| 15WW885-2 | 2 | Sink Drains | Tank 7D-13 | 120 | 0 | 0 | WW |
| 15WW887-2 | 2 | Sink Drains | Tank 7D-13 via 15WW885-2 | 25 | 0 | 0 | WW |
| 15WW892-3 | 3 | Scrap Removal Room | Interceptor thru 15WW852-3 | 10 | 0 | 10 | WW |

**Table F-1. List of Buried Pipelines**

| Line Number | Pipe Dia. (in) | From | To | Length (feet) Below Process Bldg | West of Process Bldg | East of Process Bldg | Radionuclide Distribution Surrogate |
|---|---|---|---|---|---|---|---|
| 15WW895-4 | 4 | Under Floor RAM Equipment Room | Interceptor thru 15WW843-6 | 25 | 0 | 0 | WW |
| 15WW896-3 | 3 | GOA Sump ejector | Interceptor thru 15WW841-4 | 3 | 0 | 0 | WW |
| 15WW899-3 | 3 | Floor PPS | Interceptor thru 15WW843-6 | 3 | 0 | 0 | WW |
| 15WW900-3 | 3 | Floor UPC | Interceptor thru 15WW843-6 | 15 | 0 | 0 | WW |
| 15WW916-6 | 6 | FRS Resin Wash Pit | Interceptor thru 15WW843-6 | 5 | 0 | 20 | WW |
| 15WW917-4 | 4 | Tank 14D-1 and Tank 14D-2 | Interceptor thru 15WW920-4 | 0 | 0 | 15 | WW |
| 15WW918-4 | 4 | Tank 14D-1 and Tank 14D-2 | Interceptor thru 15WW920-4 | 0 | 0 | 15 | WW |
| 15WW919-4 | 4 | Tank 14D-1 and Tank 14D-2 | Interceptor thru 15WW920-4 | 0 | 0 | 15 | WW |
| 15WW920-4 | 4 | Tank 14D-1 and Tank 14D-2 | Interceptor thru 15WW569-6 | 0 | 0 | 125 | WW |
| 15WW923-6 | 6 | Utility Room Floor Drain | Interceptor thru 15WW569-6 | 30 | 0 | 0 | WW |
| 15WW924-4 | 4 | Utility Room Floor Drain | Interceptor thru 15WW569-6 | 30 | 0 | 0 | WW |
| 15WW925-6 | 6 | Utility Room Floor Drain | Interceptor thru 15WW569-6 | 30 | 0 | 0 | WW |
| 15WW926-2 | 2 | Utility Room Floor Drain | Interceptor thru 15WW569-6 | 30 | 0 | 0 | WW |
| 15WW927-4 | 4 | Utility Room Floor Drain | Interceptor thru 15WW569-6 | 30 | 0 | 0 | WW |
| 15WW929-3 | 3 | Tank 15D-6 | New Interceptor East Valve Pit | 0 | 0 | 660 | WW |
| 15WW1231-3 | 3 | Floor Drain PPS | Interceptor via 15WW569-6 | 15 | 0 | 0 | WW |
| 15WW1232-3 | 3 | Floor Drain Acid Rec Pump Room | Interceptor via 15WW569-6 | 15 | 0 | 0 | WW |
| 15WW1744-2 | 3 | Laundry Sump | New Interceptor A | 0 | 0 | 175 | WW |
| 6-71-6-001 | 6 | 6-50-2-015, 6-71-2-019, 6-71-2-675, 6-50-2-015 | Tank 35104 | 0 | 0 | 15 | WW |
| 6-71-2-003 | 2 | 12CH241 | Tank 35104 Pump Suction | 0 | 0 | 15 | WW |
| 6-71-1-006 | 1 | Tank 35104 Pump Discharge | LWTS Evaporator | 0 | 0 | 40 | WW |
| 6-71-3-016 | 3 | Floor Drain in 35104 pump niche | General crane Room extension | 0 | 0 | 30 | WW |
| 6-71-2-019 | 2 | Truck Fill | Tank 35104 via 6-71-6-001 | 0 | 0 | 4 | WW |
| 6-71-2-020 | 2 | Tank 7D-13 Eductor 7H-19 via 7P159 | PPC manifold via 01/14 & Pipe Chase | 0 | 0 | 45 | WW |

**Table F-1. List of Buried Pipelines**

| Line Number | Pipe Dia. (in) | From | To | Length (feet) | | | Radionuclide Distribution Surrogate |
|---|---|---|---|---|---|---|---|
| | | | | Below Process Bldg | West of Process Bldg | East of Process Bldg | |
| 6-71-2-021 | 2 | Tank 7D-13 Eductor 7H-19 via 7P159 | Interceptor via 15WW848 | 0 | 0 | 25 | WW |
| 6-71-4-022 | 4 | CSS Drain Header | Tank 7D-13 | 0 | 0 | 70 | WW |
| 6-71-2-023 | 2 | Tank 35104 Pump Discharge | 6-50-2-153, return to STS | 0 | 0 | 10 | WW |
| 6-71-2-031 | 2 | Drain from 7D-13 valve pit | Tank 7D-13 via 6-71-4-022 | 0 | 0 | 15 | WW |
| 6-71-2-032 | 0.5 | Tank 35104 Pump Discharge | 35104 Sample Station GPC-CR Lower Air lock | 0 | 0 | 50 | WW |
| 6-71-2-675 | 0.5 | 35104 Sample Station GPC-CR Lower Air lock | 35104 Waste Catch tank via 6-71-6-001 | 0 | 0 | 50 | WW |
| 12CH240-6 | 6 | Drains | Tank 35104 | 0 | 0 | 30 | WW |
| 12CH241-3 | 3 | Tank 35104 Eductor | Tank 7D-2 LWC or Tank 35104 Pump Suction | 0 | 0 | 20 | WW |
| 12CH365-1/8 | 0.125 | 35104 Pit | Cut and Capped 18"below grade | 0 | 0 | 10 | WW |
| 12CH366-2 | 0.5 | 35104 Pit | Cut and Capped 18"below grade | 0 | 0 | 10 | WW |
| 12CH367-1 | 1 | 35104 Pit | Cut and Capped 18"below grade | 0 | 0 | 10 | WW |
| undesignated | 2 | Tank 15D-6 | MSM Valve Pit | 0 | 0 | 150 | Tank 5D-6 |
| undesignated | 2 | MSM Shop 2 Floor Drains | Tank 15D-6 | 50 | 0 | 50 | Tank 15D-6 |
| Leachate Line | 2 | NDA Hardstand | LLWTF Lagoon 2 | 0 | 0 | 2,000 | n/a |

LEGEND: Tanks referred to are located within the Process Building, except 15D-6 that is an underground tank located northeast of the Process Building. CCR is the Chemical Process Cell Crane Room. CD Pit is the Cask Decon Pit. CPC is the Chemical Process Cell. CSS is the Cement Solidification System. EDR is the Equipment Decontamination Room. FRS is Fuel Receiving and Storage. GOA is General Purpose Cell Operating Aisle. GP is General Purpose. GPC is General Purpose Cell. GPC-CR is the General Purpose Cell Crane Room. LWC is the Liquid Waste Cell. LWTS is the Liquid Waste Treatment System. MSM is Master-Slave Manipulator. OGC is the Off-Gas Cell. PMCR is the Process Mechanical Cell Crane Room. PPC is the Product Purification Cell. SRR is the Scrap Removal Room. STS is the Supernatant Treatment System. WW is wastewater.

## 2.0 Lines Beneath the Process Building

Review of drawings and process history established that 57 pipelines or portions of pipelines located beneath the Process Building, Utility Room, or Utility Room Expansion carried radioactive liquid. These include:

- Eleven process drains,
- Two waste transfer lines,

- Eleven Fuel Receiving and Storage Area cask decon lines,
- Thirty-three wastewater drains.

There were 11 lines under the Process Building that were designed to carry radioactive fluids, but were spares that were never used as designed. Their inventory is considered negligible (zero).

Figure F-1 shows the lines that were estimated to contribute more than 98 percent of the total activity in the lines beneath the Process Building. The lines in each category and the estimated source terms are described below.

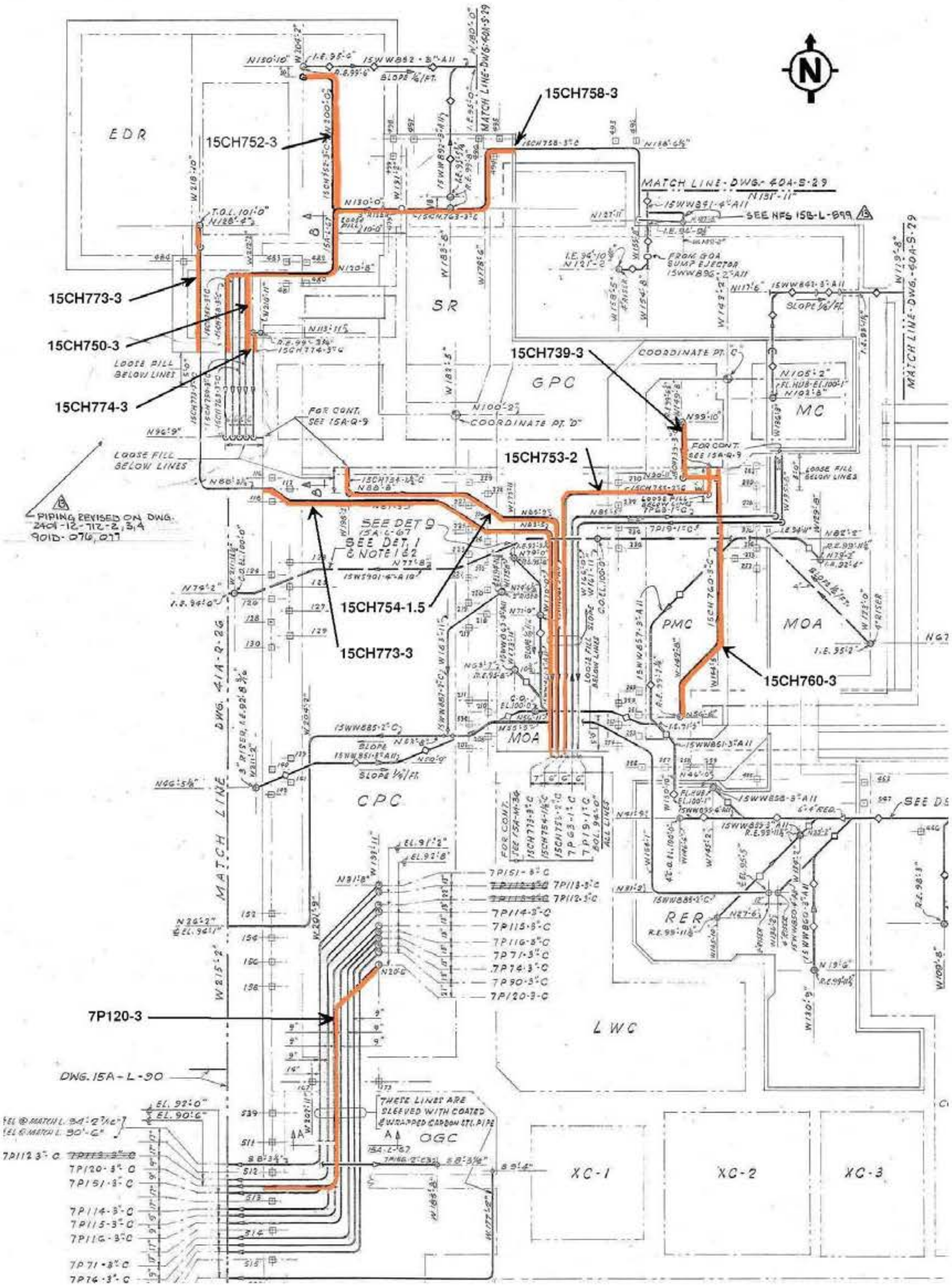



**Figure F-1. Location of Pipelines Beneath the Process Building.** (Marked lines are estimated to contain more than 98 percent of the activity in piping under the building.)

### 2.1 Process Drain Lines

All 11 lines are stainless steel pipe designated for chemical service. Eight are three-inch, two are two-inch, and the other is 1.5-inch in diameter. Each line is encased in an outer carbon steel pipe providing double containment. They are located in side-by-side runs within earth fill beneath the Process Building's reinforced concrete floor slabs.

The lines run typically about 10 feet below grade (reference elevation approximately 90 feet) and are sloped downward in the direction of flow, typically about 0.25 inch per foot. Table F-2 shows conservative estimates of the total activity within all 11 lines.

**Table F-2. Estimated Process Drain Line Activity in Curies** (as of 2011)

| Nuclide | Activity | Nuclide | Activity | Nuclide | Activity |
|---|---|---|---|---|---|
| Am-241 | 7.5E-02 | Np-237 | 3.7E-05 | Tc-99 | 3.9E-04 |
| C-14 | 1.3E-04 | Pu-238 | 1.8E-02 | U-232 | 4.4E-05 |
| Cm-243 | 7.8E-05 | Pu-239 | 1.7E-02 | U-233 | 4.2E-05 |
| Cm-244 | 1.8E-03 | Pu-240 | 1.1E-02 | U-234 | 1.6E-05 |
| Cs-137 | 8.0E-01 | Pu-241 | 2.6E-01 | U-235 | 6.8E-05 |
| I-129 | 2.0E-06 | Sr-90 | 4.6E-01 | U-238 | 2.0E-05 |

### 2.2 Waste Transfer Lines

Both lines are three-inch stainless steel pipe; each is encased within an outer six-inch carbon steel pipe. These lines run approximately 10 feet below grade within a concrete pipe trench. The lines are sloped downward in the direction of flow, about 0.25 inch per foot. Estimated activity in the lines is shown in Table D-3 below.

Line 7P120-3 contains much more radioactivity than the other line, 7P113-3. Line 7P120-3, which runs from the Chemical Process Cell to HLW Tank 8D-4, was used by NFS to transfer THOREX process waste during one fuel reprocessing campaign. Line 7P113-3 was used by NFS to transfer PUREX process wastes to Tank 8D-2; this line was flushed with decontamination solutions and with lower level waste solutions after reprocessing operations ended. Table F-3 shows conservative estimates of the total activity within both lines.

**Table F-3. Estimated Waste Transfer Line Activity in Curies** (as of 2011)

| Nuclide/Line | 7P113-3 | 7P120-3 | Nuclide/Line | 7P113-3 | 7P120-3 |
|---|---|---|---|---|---|
| Am-241 | 1.1E-05 | 1.0E-02 | Pu-240 | 1.3E-06 | 3.3E-04 |
| C-14 | 1.9E-07 | 5.4E-06 | Pu-241 | 1.7E-05 | 1.1E-02 |
| Cm-243 | 3.8E-08 | 5.3E-06 | Sr-90 | 2.9E-04 | 1.0E+01 |
| Cm-244 | 8.9E-07 | 2.2E-04 | Tc-99 | 2.2E-07 | 4.3E-03 |
| Cs-137 | 3.6E-03 | 1.1E+01 | U-232 | 3.6E-08 | 8.9E-05 |

**Table F-3. Estimated Waste Transfer Line Activity in Curies** (as of 2011)

| Nuclide/Line | 7P113-3 | 7P120-3 | Nuclide/Line | 7P113-3 | 7P120-3 |
|---|---|---|---|---|---|
| I-129 | 1.6E-07 | 7.4E-06 | U-233 | 1.6E-08 | 8.7E-05 |
| Np-237 | 9.9E-09 | 1.3E-05 | U-234 | 7.9E-09 | 9.1E-05 |
| Pu-238 | 2.4E-06 | 1.6E-02 | U-235 | 6.3E-11 | 2.1E-07 |
| Pu-239 | 1.7E-06 | 6.4E-04 | U-238 | 8.0E-10 | 2.9E-09 |

### 2.3 Cask Decon Lines

Nine lines are four inches in diameter and are associated with floor drains for the Fuel Receiving and Storage Building; these lines connect to the six-inch trunk line (15WW571-6). Line 1P64-1, a one-inch discharge line running toward the Low-Level Waste Treatment Facility (LLWTF) Interceptor, is also grouped with the cask decon lines.

The estimated activity in these lines, based on the assumption that their average interior surface contamination is similar to that remaining on the floor of the Cask Decon Pit, is shown in Table F-4.

**Table F-4. Estimated Cask Decon Line Activity in Curies** (as of 2011)

| Nuclide | Activity | Nuclide | Activity | Nuclide | Activity |
|---|---|---|---|---|---|
| Am-241 | 1.9E-02 | Np-237 | 2.3E-06 | Tc-99 | 5.2E-05 |
| C-14 | 2.5E-05 | Pu-238 | 2.8E-03 | U-232 | 2.9E-06 |
| Cm-243 | 7.4E-06 | Pu-239 | 5.4E-03 | U-233 | 6.9E-06 |
| Cm-244 | 1.5E-04 | Pu-240 | 2.8E-03 | U-234 | 5.9E-07 |
| Cs-137 | 1.3E-01 | Pu-241 | 7.6E-02 | U-235 | 8.4E-07 |
| I-129 | 1.2E-07 | Sr-90 | 1.2E-01 | U-238 | 7.1E-06 |

### 2.4 Wastewater Drain Lines

These lines deliver low-level or uncontaminated wash water and spills from various drains in the Process Building to the LLWTF Interceptor. This piping is made of Duriron, a high silicone cast iron, in diameters ranging from two-inch to six-inch. Beneath the Process Building, the runs are encased within concrete of 12-inch-square cross section. They are located eight to 12 feet below grade, sloping about 0.25 inch per foot.

The estimated activity in these lines was based on an empirical relationship between the residual contamination and the radioactivity in the fluid carried by the lines observed in HLW lines. (This relationship is based on WVDP experience with residual contamination measured in other piping where the activity of the liquid that passed through the piping was known.) The LLWTF Interceptor operating limit (0.005 µCi/mL) was used in the calculations for conservatism; many discharges though the lines likely had radioactivity concentrations well below this value. The use of the bounding spent nuclear fuel distribution as the surrogate for the waste water also provides a level of conservatism by assigning the maximum radionuclide ratio observed in any spent fuel batch to the residual in the waste water pipes. The total estimated activity in all the lines is shown in Table F-5.

**Table F-5. Estimated Wastewater Drain Line Activity in Curies** (as of 2011)

| Nuclide | Activity | Nuclide | Activity | Nuclide | Activity |
|---|---|---|---|---|---|
| Am-241 | 2.1E-06 | Np-237 | 1.3E-09 | Tc-99 | 5.6E-09 |
| C-14 | 3.2E-11 | Pu-238 | 2.3E-07 | U-232 | 5.8E-10 |
| Cm-243 | 1.2E-08 | Pu-239 | 7.2E-08 | U-233 | 2.4E-10 |
| Cm-244 | 2.6E-07 | Pu-240 | 5.2E-08 | U-234 | 9.7E-11 |
| Cs-137 | 1.4E-04 | Pu-241 | 1.1E-06 | U-235 | 2.5E-12 |
| I-129 | 2.6E-14 | Sr-90 | 1.3E-04 | U-238 | 2.3E-11 |

### 2.5 Total Estimated Inventory in Lines Beneath the Process Building Footprint

As shown in Table F-6 the total estimated residual inventory for all the combined lines beneath the Process Building footprint is approximately 23 Ci, predominantly Sr-90 and Cs-137 activity. The table indicates that Line 7P120-3 and the process drain lines have over 95 percent of the Cs-137 and Sr-90 activity under the Process Building, as well as 71-98 percent of the Pu and U isotopes.

**Table F-6. Estimated Total Residual Inventory in Lines Under the Process Building** (as of 2011)

| Nuclide | Residual Inventory (Ci) | | | Contribution to Total | |
|---|---|---|---|---|---|
| | Total All Lines | Process Drains | Line 7P120-3 | Line 7P120-3 | Line 7P120-3 and Process Drains |
| Am-241 | 1.0E-01 | 7.5E-02 | 1.0E-02 | 10.0% | 85.0% |
| C-14 | 1.6E-04 | 1.3E-04 | 5.4E-06 | 3.4% | 84.6% |
| Cm-243 | 9.1E-05 | 7.8E-05 | 5.3E-06 | 5.8% | 91.5% |
| Cm-244 | 2.2E-03 | 1.8E-03 | 2.2E-04 | 10.0% | 91.8% |
| Cs-137 | 1.2E+01 | 8.0E-01 | 1.1E+01 | 91.7% | 98.3% |
| I-129 | 9.7E-06 | 2.0E-06 | 7.4E-06 | 76.3% | 96.9% |
| Np-237 | 5.2E-05 | 3.7E-05 | 1.3E-05 | 25.0% | 96.2% |
| Pu-238 | 3.7E-02 | 1.8E-02 | 1.6E-02 | 43.2% | 91.9% |
| Pu-239 | 2.3E-02 | 1.7E-02 | 6.4E-04 | 2.8% | 76.7% |
| Pu-240 | 1.4E-02 | 1.1E-02 | 3.3E-04 | 2.4% | 80.9% |

**Table F-6. Estimated Total Residual Inventory in Lines Under the Process Building** (as of 2011)

| Nuclide | Residual Inventory (Ci) | | | Contribution to Total | |
|---|---|---|---|---|---|
| | Total All Lines | Process Drains | Line 7P120-3 | Line 7P120-3 | Line 7P120-3 and Process Drains |
| Pu-241 | 3.5E-01 | 2.6E-01 | 1.1E-02 | 3.1% | 77.4% |
| Sr-90 | 1.1E+01 | 4.6E-01 | 1.0E+01 | 90.9% | 95.1% |
| Tc-99 | 4.7E-03 | 3.9E-04 | 4.3E-03 | 91.5% | 99.8% |
| U-232 | 1.4E-04 | 4.4E-05 | 8.9E-05 | 63.6% | 95.0% |
| U-233 | 1.4E-04 | 4.2E-05 | 8.7E-05 | 62.1% | 92.1% |
| U-234 | 1.1E-04 | 1.6E-05 | 9.1E-05 | 82.7% | 97.3% |
| U-235 | 6.9E-05 | 6.8E-05 | 2.1E-07 | 0.3% | 98.9% |
| U-238 | 2.8E-05 | 2.0E-05 | 2.9E-09 | 0.0% | 71.4% |

## 3.0 Lines West of the Process Building

The lines west of the Process Building identified in Table F-1 include:

- Four ventilation lines;
- Three waste transfer lines, two of which were used; and
- Twenty-four other lines that carried wastewater or ventilation condensate.

### 3.1 Lines of Interest

**Ventilation Lines**

The ventilation lines are:

- 8P29-16, a 16-inch header line that runs from the Permanent Ventilation System to the Equipment Shelter
- 8P34-2, an abandoned and capped two-inch ventilation condensate line from Tank 8D-2,
- 7P170-2, an abandoned and capped two-inch ventilation condensate line from Tank 8D-1, and
- 8P46-6 (old and new), two six-inch lines that connect the Equipment Shelter to the Main Plant Stack.

**Waste Transfer Lines**

The two waste transfer lines of interest are the downstream ends of those discussed in Section 2.2, 7P120-3 and 7P113-3.

**Other Lines West of the Process Building**

The other 24 lines of interest shown in Table F-1 carried process drain fluids, wastewater, and ventilation condensate.

### 3.2 Estimated Inventory in Lines West of the Process Building

The estimated total inventory of the 31 underground lines west of the Process Building is shown in Table F-7. The total length of all of these lines together is approximately 4,176 feet. The total interior surface area is approximately 3.47E+06 $cm^2$.

**Table F-7. Estimated Total Residual Inventory of Lines West of the Process Building in Curies** (as of 2011)

| Nuclide | Activity | Nuclide | Activity | Nuclide | Activity |
|---|---|---|---|---|---|
| Am-241 | 8.3E-02 | Np-237 | 1.0E-04 | Tc-99 | 3.4E-02 |
| C-14 | 4.6E-05 | Pu-238 | 1.3E-01 | U-232 | 7.1E-04 |
| Cm-243 | 4.4E-05 | Pu-239 | 5.2E-03 | U-233 | 6.9E-04 |
| Cm-244 | 1.8E-03 | Pu-240 | 2.7E-03 | U-234 | 7.2E-04 |
| Cs-137 | 8.5E+01 | Pu-241 | 8.6E-02 | U-235 | 1.8E-06 |
| I-129 | 6.0E-05 | Sr-90 | 8.1E+01 | U-238 | 1.0E-06 |

## 4.0 Lines East of the Process Building

### 4.1 Lines of Interest

Table F-1 identifies 47 lines east of the Process Building. Most deliver low-level radioactive or uncontaminated wastewater, wash water, or liquid from spills from various drains throughout the Process Building to the Interceptor in WMA 2. From the Interceptor, the water can be sampled, diverted to storage tanks, sent to the LLWTF for treatment, or released to the lagoon system through other lines identified in the table. Other lines in WMA 2 connect various tanks with the LLWTF and the LLWTF to the lagoons. From the lagoons, waters can be discharged to surface streams on the Center.

Various underground lines were realigned from Lagoon 1 to Lagoon 2 and from Lagoon 2 to Lagoon 3 in 1984 when Lagoon 1 was removed from service. At that time, Lagoon 2 became the initial receiving lagoon for the LLWTF. Originally, water treatment was performed in the O2 Building, but it was replaced by the LLWTF. The New Interceptors (A and B) were installed in 1967 to replace the single Old Interceptor.

### 4.2 Estimated Inventory in Lines East of the Process Building

The estimated total inventory of the 47 underground lines east of the Process Building is shown in Table F-8. The total length of all of these lines together is approximately 4,559 feet. The total interior surface area is approximately 3.40 E+06 $cm^2$.

**Table F-8. Estimated Total Residual Inventory of Lines East of the Process Building in Curies** (as of 2011)

| Nuclide | Activity | Nuclide | Activity | Nuclide | Activity |
|---|---|---|---|---|---|
| Am-241 | 1.3E-02 | Np-237 | 1.5E-06 | Tc-99 | 3.4E-05 |
| C-14 | 1.6E-05 | Pu-238 | 1.9E-03 | U-232 | 1.9E-06 |
| Cm-243 | 4.9E-06 | Pu-239 | 3.6E-03 | U-233 | 4.6E-06 |
| Cm-244 | 9.9E-05 | Pu-240 | 1.9E-03 | U-234 | 3.9E-07 |
| Cs-137 | 8.5E-02 | Pu-241 | 5.0E-02 | U-235 | 5.6E-07 |
| I-129 | 7.9E-08 | Sr-90 | 7.9E-02 | U-238 | 4.7E-06 |

## 5.0 Leachate Transfer Line

### 5.1 Description

The Leachate Transfer Line is a buried two-inch polyvinylchloride pipe that originates on the south plateau at the NDA and continues northward across WMA 6 to Lagoon 2 in WMA 2. The line was laid within a five-inch sand layer at the base of a 36-inch wide trench located five feet below the surface.

The line was originally used to transfer fluids originating from the SDA Lagoons to Lagoon 1 in the LLWTF via a pumphouse adjacent to the NDA hardstand. More recently, it has been used to transfer groundwater from the NDA interceptor trench to Lagoon 2. The total length of the line is approximately 2,000 feet. The location of the Leachate Transfer Line is shown on Drawing 40C-S-1057, on which Figure F-2 is based.

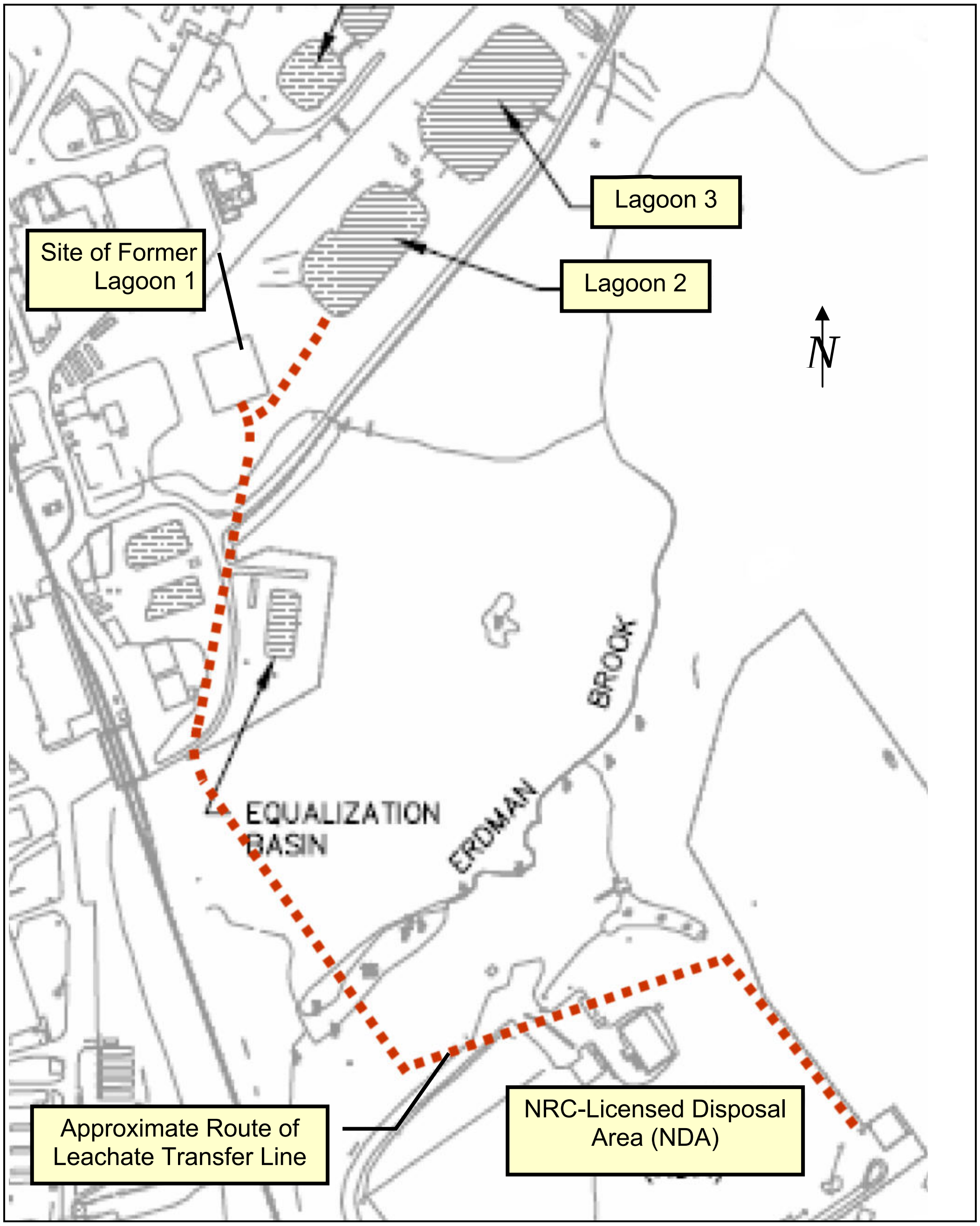



**Figure F-2. Leachate Transfer Line Routing From NDA to Lagoon 1** (based on drawing 40C-S-1057)

### 5.2 Fluids Conveyed by the Line

The use of the Leachate Transfer Line to convey burial trench leachate is described in the RCRA Facility Investigation Report for the NYSERDA-maintained portions of the Center (NYSERDA 1994).

In March 1975 leachate levels in Trenches 4 and 5 of the SDA[1] reached the ground surface and seeped through the earthen covers. NFS began a permitted operation to pump, treat and dispose of leachate[2] from the burial trenches. From 1975 through 1981 NFS pumped over 2,850,000 gals of fluid through the Leachate Transfer Line to Lagoon 1 in WMA 2 for treatment in the LLWTF and eventual discharge to Erdman Brook. Typically, concentrations of radionuclides were in the range of 1 E-03 to 1 E-06 µCi/mL, although in the case of tritium (H-3), concentrations up to ~4 µCi/mL were observed. Before transfer to Lagoon 1 the leachate was chlorinated to destroy biological matter and then treated to reduce water hardness and to precipitate some of the radionuclides. A list of SDA trench-pumping events and volumes is provided in Luckett, et al. 2004. Activity concentrations of radionuclides detected in the leachate are also provided in Luckett, et al. 2004.

The NDA interceptor trench was installed in 1991 on the northeast and northwest boundaries of the NDA to intercept and collect potentially contaminated groundwater migrating from the NDA. The base of the trench extends to a minimum of one foot below the interface of the weathered till with the unweathered till. The trench is drained by a drainpipe that directs accumulated water to a collection sump.

Liquid that collects in the sump is routinely sampled, analyzed, and transferred through the Leachate Transfer Line to Lagoon 2 in WMA 2 for treatment and release. Since its installation, over 3,000,000 gallons of intercepted groundwater have been pumped through the Leachate Transfer Line. Details of fluid volumes pumped through the Leachate Transfer Line from the interceptor trench during the period 1991-2003 are provided in Luckett, et al. 2004.

The NDA interceptor trench is sampled as part of the WVDP environmental monitoring program. Radionuclides detected in samples of the fluid are typically in the range of 1 E-07 to 1 E-10 µCi/mL with two exceptions: Tritium (H-3) is observed in the range of 1 E-05 µCi/mL and uranium, attributed to naturally occurring materials, is observed in the range of 3E-03 µg/mL. A summary of radionuclides detected and their concentrations in the samples of the fluid during the period 1993-2003 are provided in Luckett, et al. 2004

### 5.3 Estimate of Activity Inventory in Leachate Transfer Line

Based on the design, operating history, and radioactivity analyses of fluids conveyed by the line, residual activity remaining in the line is insignificant to the performance assessment. Among the factors which led to this conclusion:

- The line is made of plastic designed to be non-reactive with water-based fluids.

[1] The term “leachate” is used here as a general term for water that has accumulated in a disposal trench and leached constituents from the materials disposed of in the trench. The use of the term does not imply that the water and the associated leached constituents constitute a regulated “leachate” as defined under RCRA or other regulatory regimes.

- The leachates were dilute fluids, which had been treated with a precipitant; there would have been little material in solution to plate out or deposit in the pipe.
- The leachate had been chlorinated; there would have been little opportunity for flora or scum to grow in the pipe and filter or trap radioactive materials conveyed in the fluids.
- The major activity in the leachate was tritium which passed through the pipe with the fluid.
- Since the leachate was conveyed in the pipe, the pipe has been flushed with over 2,600,000 gallons of groundwater that is essentially free of radionuclides.
- Measured radionuclide concentrations are detectable only with the most sensitive analysis and are well below the regulatory limits for the LLWTF inflow waters of 5.0E-03 μCi/ml.
- The total uranium observed is typical of uranium occurring naturally in groundwater, and is well below the EPA drinking water standard of 30 μg/L (or 3.0 E-02 μg/mL) for uranium, as specified in Title 10 CFR 40, Part 141.55.

## 6.0 References

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, Revision 0. Luckett, L., J. Fazio, and S. Marschke, Washington Safety Management Solutions, Aiken, South Carolina, July 6, 2004.

NYSERDA 1994, *RCRA Facility Investigation for NYSERDA-Maintained Portions of the Western New York Nuclear Services Center*, NYSERDA, West Valley, New York, December 1994.

# APPENDIX G

# PHASE 1 FINAL STATUS SURVEY CONCEPTUAL FRAMEWORK

**PURPOSE OF THIS APPENDIX**

The purpose of this appendix is to describe the conceptual basis for the Phase 1 Final Status Survey Plan.

**INFORMATION IN THIS APPENDIX**

This appendix describes the design basis for the Phase 1 Final Status Survey Plan, including the key assumptions, and then outlines the final status survey approach. It closes with a discussion of documentation requirements. Logic diagrams are provided to illustrate the processes involved.

**RELATIONSHIP TO OTHER PARTS OF THE PLAN**

The information in this appendix supplements the requirements for the Phase 1 Final Status Survey Plan described in Section 9.

## 1.0 Introduction

The purpose of this conceptual framework is to describe the design basis and general approach for the WVDP Phase 1 Final Status Survey Plan, thus augmenting the requirements outlined in Section 9 of this plan.

Section 7.2.2 of this plan provides for Phase 1 final status surveys in three types of areas:

(1) The major areas to be made inaccessible during Phase 1 decommissioning activities, that is, the bottom and sides of excavations for removal of key WVDP facilities and contaminated subsurface soil (i.e., the WMA 1 and WMA 2 large excavations);

(2) Excavated soil laydown areas after the soil and ground covering are removed; and

(3) Potentially impacted areas with no subsurface soil contamination that meet the unrestricted release criteria during Phase 1 of the decommissioning.

The primary objective of these surveys is to confirm that cleanup goals specified in Section 5 of this plan have been achieved. However, if an excavated soil laydown area is known to have subsurface contamination, then the objective of the survey of that area will be to determine the radiological status of the surface soil.

Note that the Characterization Sample and Analysis Plan, rather than the Phase 1 Final Status Survey Plan, will provide for radiological status surveys of:

(1) Soil in the footprints of structures, concrete slabs, asphalt pavement, and gravel pads outside of the WMA 1 and WMA 2 large excavations to be removed during Phase 1 decommissioning activities; and

(2) The interior of the HLW transfer trench following removal of piping and equipment in the trench and the associated pump pits and diversion pit.

If DOE chooses to demonstrate that soil in the footprints of selected structures, concrete slabs, asphalt pavement, or gravel pads outside of the WMA 1 and WMA 2 large excavations removed during Phase 1 decommissioning activities meets the unrestricted release criteria, then Phase 1 final status surveys will also be performed in those areas if the characterization data are not sufficient for final status survey purposes.

## 2.0 Final Status Survey Design Basis

As required by Section 9 of this plan, the Phase 1 Final Status Survey Plan will be consistent, to the extent possible, with the MARSSIM (NRC 2000). There are aspects of the WVDP project premises (e.g., buried subsurface soil contamination, etc.) that are beyond MARSSIM’s scope. In those instances, the protocols will be consistent with the intent of MARSSIM.

### 2.1 Project Premises and Phase I Activities

As explained in Section 3 of this plan, the project premises comprise 156.4 acres. The major features of the project premises include existing facilities and associated above-ground and buried infrastructure, disposal areas, wastewater lagoons, roads, hardstands, paved parking lots, a railway spur, streams that drain the parcel, and open land. The project premises were used for spent fuel reprocessing in the 1960s and early 1970s. Reprocessing activities resulted in environmental releases of radionuclides to surrounding soils, surface water, and groundwater as discussed in Section 2 of this plan.

To address known historical releases whose residual environmental contamination pose significant dose concerns, Phase 1 activities include the following planned environmental remediation activities:

(1) A deep (30 – 45 feet), extensive (approximately three acre) excavation of contaminated soils adjacent to and beneath the Main Plant Process Building (WMA 1);

(2) A deep (up to 14 feet), extensive (approximately four acre) excavation of contaminated soils adjacent to and beneath facilities and lagoons associated with the Low-Level Waste Treatment Facility (WMA 2); and

(3) Excavation of contaminated and uncontaminated near-surface soils (approximately two feet below grade) associated with selected building and infrastructure removal in WMA 1, WMA 3, WMA 5, WMA 6, WMA 7, WMA 9, and WMA 10.

In addition to these planned excavations, DOE may also choose to remove additional contaminated soils and/or sediments as part of Phase 1 decommissioning work. Any residual contamination within the project premises that still poses a dose concern will be addressed by Phase 2 decommissioning activities.

### 2.2 Cleanup Criteria

As indicated in Section 5 of this plan, there are 18 radionuclides of interest for the project premises. The DCGL values for each radionuclide are based on a 25 mrem/y dose requirement (incremental to background) assuming a goal of unrestricted release.

The DCGL requirements include a $DCGL_W$ value to be applied as an area-averaged goal to final status survey units and a $DCGL_{EMC}$ value applicable to 1-square meter ($m^2$) areas. Different DCGL values are provided for surface soils (defined as soils to a depth of 1 m), for subsurface soils (defined as soils at significant depth that will be temporarily exposed by Phase 1 excavation activities in WMA 1 and WMA 2), and for streambed sediments. These DCGL values were further refined to reflect cumulative dose concerns, resulting in a final set of cleanup goals reflected in Table 5-14 of this plan[1].

### 2.3 Key Assumptions

This conceptual framework includes several key assumptions:

- ***Decommissioning Plan Changes.*** This conceptual framework is based on DCGLs in Revision 2 to the plan. Any changes in DCGL values or definitions may require changes to this framework.
- ***DCGL Definitions.*** The surface soil DCGLs apply to a vertical interval (contamination zone thickness) of one meter. The planned characterization work

[1] Section 5 of this plan explains the difference between the DCGLs developed to correspond to 25 mrem per year for individual areas and the cleanup goals to be used in remediation activities. As in Section 9 of this plan, the term *DCGL* as used in this appendix from this point on is understood to mean *cleanup goal*.

may identify project premises characteristics that are inconsistent with the conceptual site model used for DCGL derivation (e.g., surface contamination restricted to the top few inches of soil surface, subsurface contamination covered by a few inches of clean soil, or contaminated soils extending to a depth greater than one meter). To address this potential issue:

(1) Surface soil DCGL standards will only be applied when contamination impacts are less than one meter in depth;

(2) Surface soil DCGL standards will be applied separately to the top 15 cm (six inches) of soil and to the top one meter soil interval as part of the final status survey process; and

(3) The presence of thin, highly elevated zones overlain by clean surface soils will be evaluated by Characterization Sampling and Analysis Plan data collection. If near surface contaminated layers are encountered during this data collection effort that result in potential dose concerns but that would not have been identified by the Phase 1 Final Status Survey Plan data collection approach, the Final Status Survey Plan process will be modified to meet the specific needs of those areas.

- ***LBGR*****.** MARSSIM's Lower Bound on the Grey Region (LBGR) corresponds to the average residual activity concentration that will be present when final status survey data collection activities begin. For areas that do not require remediation, the LBGR is the existing average level of contamination present. For areas requiring remediation, the LBGR is the cleanup level targeted by the remediation program. In combination with the Type II error rate and expected sample variability, the LBGR is an important determinant of the number of systematic samples required to demonstrate compliance with the $DCGL_W$ values.

- ***Data Gaps*****.** There are key data gaps that will be addressed as part of the pre-design characterization work discussed in Section 9 of this plan. One example of these is the presence and spatial prevalence of the 18 radionuclides of interest. A second example is the presence and importance of radionuclides other than the 18 identified in this plan. While unlikely, the Final Status Survey Plan framework may need to be revisited if Phase 1 conditions encountered during characterization work are determined to be significantly different from the assumptions and conceptual site model in this plan.

- ***Chemical Contamination*****.** Chemical contamination may exist for portions of the facility. Chemical contamination concerns will be addressed in compliance with RCRA requirements, and are not directly within the scope of the Final Status Survey Plan. Samples collected as part of the Final Status Survey Plan process may also be analyzed for chemical constituents as necessary for waste stream characterization needs, and/or to fulfill RCRA requirements.

- ***Scope of Phase 1 Final Status Survey Plan Data Collection*****.** As part of Phase 1 decommissioning activities, data will be collected to demonstrate that the floors and the sides (at depths greater than three feet) of the WMA 1 and 2 excavations meet the appropriate DCGL requirements. In addition, DOE may also choose to collect data to demonstrate that surface soils for other portions of the WVDP project premises also meet the Phase 1 cleanup goals for those situations where contamination is not present at depths greater than one meter. Examples of these areas include: (1) soils exposed by hardstand, pad, or foundation removal that are believed to be below DCGL requirements; (2) soils with surface contamination above DCGL goals that DOE chooses to remediate; and/or (3) other soils where there is no evidence of contamination above DCGL requirements. The Final Status Survey Plan framework as described applies to soils and does not apply to sediments, surface water or groundwater.

- ***Sign Test Applicability*.** Because all 18 radionuclides identified in the decommissioning plan are either not naturally occurring or have $DCGL_W$ requirements an order of magnitude or more above background levels, the Sign test is considered appropriate for demonstrating compliance with wide-area DCGL ($DCGL_W$) requirements. In the event that DCGL values are lowered it may be necessary to establish a background reference area and use the Wilcoxon Rank Sum (WRS) test instead to demonstrate compliance with the $DCGL_W$ requirements.
- ***$DCGL_{EMC}$ Applicability*.** The $DCGL_{EMC}$ is radionuclide-specific and applies to 1-$m^2$ areas. Gross gamma surveys will be used for demonstrating compliance with the $DCGL_{EMC}$ criteria where appropriate. In addition, appropriate $DCGL_{EMC}$ values will be calculated that correspond to the area represented by systematic samples collected to demonstrate $DCGL_W$ compliance using area factors provided in Tables 9-1 and 9-2 of Section 9 of this plan. The latter approach is intended to address the radionuclides of interest that are not detectable by gamma scans and that may exist in isolation for specific portions of the project premises (e.g., the floor of the WMA 1 dig where Sr-90 may be the principal radionuclide of interest).
- ***Radionuclides of Interest List.*** Because processes and contaminant release scenarios vary from location to location across the project premises, not all 18 radionuclides of interest may be pertinent to specific areas. The assumption is that Characterization Sample and Analysis Plan data collection may be used to determine which of the 18 radionuclides of interest are pertinent to specific areas and that final status survey sampling for those areas may be limited to the smaller set of the pertinent radionuclides of interest.
- ***Use of Sum-of-Ratios Calculations***. Because of the many radionuclides of interest, all final status survey determinations will be based on sample sum-of-ratios calculations. The sum-of-ratios calculation for any particular sample will be based on the radionuclides pertinent to the final status survey unit that was the source of the sample.
- ***Subsurface Soil Contamination*.** The Phase 1 Final Status Survey Plan is not applicable to areas outside the WMA 1 and 2 excavations where subsurface contamination exists at depths greater than one meter.
- ***Null Hypothesis and Acceptable Error Rates*.** For the Sign test, the null hypothesis will be that final status survey units are contaminated above $DCGL_W$ levels based on sample sum-of-ratios values. In this context, the acceptable Type I

error rate (i.e., rejecting the null hypothesis when it should have been accepted) will be 0.05. The Type II error rate (i.e., accepting the null hypothesis when it should have been rejected) will be set based on an engineering cost analysis that weighs the potential for false contaminated conclusions with the costs of final status survey data collection. The Type I error rate establishes the minimum number of systematic samples required for Sign test implementation. In the case of an error rate of 0.05, the minimum number is five samples per survey unit; final status survey units, however, will likely require more systematic samples than this minimum number to meet Type II error rate needs.

- ***Role of Composite Sampling*.** While not discussed in MARSSIM, the use of composite samples is one means for attaining desired Type II error rates while controlling analytical costs when performing $DCGL_W$ evaluations. Composite sampling can also significantly increase the likelihood that $DCGL_{EMC}$ exceedances are identified for radionuclides that are not detectable by gross activity scans. Composite sampling combines soil increments systematically distributed across a portion of a final status survey unit into homogenized composite samples before analysis. The minimum number of composites per survey unit is determined by the desired Type I error rate. The minimum number of soil increments contributing to each composite sample is a function of the desired Type II error rate, the degree of heterogeneity expected within survey units, and the expected average residual activity concentration. Composite sampling will be used when appropriate during the final status survey process to improve overall decision-making performance. Sufficient composite samples are collected from each survey unit to satisfy Sign or WRS test requirements. The type of compositing proposed, and its advantages are well documented, have been used effectively within the RCRA and Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) cleanup programs, and have regulatory support (see EPA 1995, EPA 2002a and EPA 2002b).

NOTE

There currently is insufficient soil characterization information available within the project premises to determine whether the use of composite soil sampling for FSS purposes is appropriate. A decision on whether the use of composite soil sampling for final status survey purposes is appropriate will be made once the soil sampling data collection and interpretation associated with the Characterization Sample and Analysis Plan is completed.

- ***Analytical Methods*.** Some of the radionuclides of interest have relatively low $DCGL_W$ values. The 18 radionuclides span a range of required analytical techniques, including gamma spectroscopy, alpha spectroscopy, liquid scintillation, and gas proportional counting. The Final Status Survey Plan will specify the analytical performance requirements expected for each radionuclide (Table 9-5 of this plan identifies target detection limits). In some cases (e.g., gamma spectroscopy and liquid scintillation), a field-based laboratory may prove advantageous, particularly for those radionuclides that will likely be the primary decision drivers (e.g., Cs-137 and Sr-90). Whether data from field deployable techniques can be used for final status survey compliance demonstration purposes will depend on whether data quality standards can be achieved and documented. There may be cases where a particular field-deployable technique may not have sufficient data quality for final status survey purposes, but where the technique still serves an important and useful role as a screening tool for elevated area concerns, or as part of pre-final status survey/remedial support data collection to determine that an area is ready for final status survey data collection.

- ***Use of Pre-Design Investigation Data for Final Status Survey Purposes*.** The final status survey logic and Final Status Survey Plan were developed in tandem with the Characterization Sample and Analysis Plan for pre-design data collection. The intent is that pre-design data, if collected consistent with Final Status Survey Plan protocols and data quality standards, can potentially be used for final status survey purposes if contamination levels requiring remediation are not identified.

### 2.4 Role of Pre-Design Data Collection

The Characterization Sample and Analysis Plan will address key data gaps pertinent to decommissioning work. Some of those data gaps are also important from the perspective of designing and implementing the final status survey process for the project premises. These include:

- ***Determining whether the list of the 18 radionuclides of interest as identified by the DP is complete***. An additional 12 radionuclides have been identified as possibly (but unlikely to be) present at the site. In addition, the presence of progeny not in equilibrium with the 18 radionuclides of interest has also been identified as a possible concern. Both issues have the potential for requiring changes to the radionuclides of interest list. The Characterization Sample and Analysis Plan will determine whether this is necessary.
- ***Addressing the prevalence, spatial distribution, and potential collocation of the 18 radionuclides of interest***. There are several potential outcomes from this data collection. If particular radionuclides of interest are either not present to any significant degree or are always dominated from a sum-of-ratios perspective by other radionuclides, the analytical list for systematic samples may be reduced to those that are pertinent. The list of “pertinent” radionuclides of interest might vary with location. Alternatively, if a few readily measurable radionuclides of interest (e.g., Cs-137) are ubiquitous and at relatively stable ratios to other radionuclides of interest, a surrogate approach might be adopted for DCGL analysis.
- ***Determining the presence/absence and prevalence of near-surface subsurface soils (e.g., soils that are at depths just below one meter) that exceed DCGL standards***. The Phase 1 surface soil DCGL requirements are only applicable to areas where contamination is not present below a depth of one meter. The Characterization Sample and Analysis Plan will delineate where near-surface subsurface soil contamination is a concern.

- ***Identifying whether thin layers of buried contamination exist within the top one meter of soils that might pose dose concerns if exposed but would be missed by the Final Status Survey Plan sampling logic***. The Characterization Sample and Analysis Plan will determine if this is the case, and if so, identify the areas where this will be a concern. If such areas exist, then the Final Status Survey Plan logic will be adjusted to address those concerns.
- ***Supporting layout of final status survey unit areas for the site***. The MARSSIM defines three different classifications of final status survey units that may potentially be applied to one or more areas of a site. The selection of the appropriate final status survey unit classification for a particular area depends on its expected contamination status relative to the DCGLs. The Characterization Sample and Analysis Plan will provide the data necessary for the correct classification and delineation of MARSSIM final status survey units.
- ***Estimating likely residual radionuclide activity concentrations to be encountered after Phase 1 activities are complete***. Expected average residual activity concentrations, in conjunction with expected heterogeneity and Type II error requirements, will affect final status survey sample numbers.

## 3.0 Final Status Survey Approach

Final status survey data collection will take place for soils within the project premises. In the case of soils, if the final status survey data collection conclusions are that DCGL standards have not been attained, DOE may remediate the area and collect additional final status survey data to demonstrate compliance with DCGL requirements.

For the deep excavated surfaces within WMA 1 and WMA 2, additional remediation will take place if subsurface DCGL requirements are not met. For areas outside the WMA 1 and WMA 2 deep excavations, if a final status unit fails the final status survey process, DOE may choose to remediate the affected area until DCGL requirements are met or to postpone remediation until Phase 2.

If DOE chooses to remediate soils exceeding DCGL standards and the original unit was a Class 1 unit, final status survey data collection will be repeated after additional remediation is complete. If the original unit was an unexcavated Class 2 or Class 3 unit, the affected area will be remediated, reclassified as one or more Class 1 units, and final status survey data collection repeated. DOE may defer remediating areas that are not currently identified as requiring excavation by the DP until Phase 2.

### 3.1 Surface Soils

A complete logged gamma walkover survey of accessible areas within the project premises using an appropriate detector (e.g., Field Instrument for Detecting Low Energy Radiation (FIDLER)) will be performed as part of Characterization Sample and Analysis Plan data collection activities. This walkover survey, in conjunction with biased surface soil sampling and intrusive GeoProbe® data collection, will be used to identify areas likely requiring remediation or impacted at levels approaching soil DCGL levels but not planned for remediation (Class 1 areas), areas impacted but with no evidence of soil DCGL exceedances (Class 2 areas), and areas within the WVDP project premises' boundary that either show no evidence of impacts, or are minimally impacted at very low levels compared to soil DCGL standards (Class 3 areas). Based on data available to date, it is expected that the majority of the project premises will be classified as either Class 1 or Class 2 final status survey units.

As part of Characterization Sample and Analysis Plan data collection, a background reference area will be identified that can be used to assess the background response of the detector used and that can serve as a source of background samples if a WRS test is required to demonstrate $DCGL_W$ compliance. One outcome of reference area gross gamma data collection will be the identification of appropriate field investigation levels to be applied to gross gamma data during routine use of detectors for pre-design characterization, remediation support, and final status survey data collection.

An example of a field investigation level will be a detector response that is not statistically consistent with background readings (e.g., above the 95 percent upper tolerance limit for background data sets). Biased sampling, in conjunction with gamma walkover survey data and associated field investigation levels, will be used during pre-design data collection work in contaminated areas to develop additional field investigation levels that could potentially be used to reliably identify gross activity responses that might be indicative of soil DCGL exceedance concerns.

For areas that are excavated, the final exposed dig face (walls and floors) will be scanned using one or more logged detectors to evaluate the potential presence of either general contamination above soil $DCGL_W$ standards, or very localized contamination potentially associated with soil $DCGL_{EMC}$ concerns. Biased sampling will be used to further evaluate evidence of contamination potentially above soil DCGL standards if encountered by the detector. Detector data will be collected with the goal of complete spatial coverage at a density of one logged measurement per square meter, on average.

Prior to the initiation of final status survey sample collection, the layout of final status survey units will be finalized for surface soils that are considered ready for final status survey data collection. Areas that are candidates for Phase 1 final status survey data collection are areas where there is no evidence or concern about contamination deeper than one meter, and where Characterization Sample and Analysis Plan data indicate that residual contamination levels likely meet surface soil DCGL requirements. Soil Class 1 survey units will not exceed 2,000 $m^2$ in size. Soil Class 2 survey units will not exceed 10,000 $m^2$ in size. There is no size constraint for Class 3 survey units.

For each survey unit the pertinent radionuclides of interest subset will be defined based on historical information, Characterization Sample and Analysis Plan sampling results for that area, and remedial support data in the case of excavated area Class 1 units.

In all cases of sample collection and analysis (systematic and biased), the sum-of-ratios values calculated for samples will be used to test compliance with DCGL standards. Sum-of-ratios values will be calculated based on soil $DCGL_{EMC}$ requirements and based on soil $DCGL_W$ requirements. As part of the sum-of-ratios calculation, background will not be subtracted for those radionuclides that occur naturally. The radionuclides of interest subset used for sum-of-ratios calculation purposes may vary from survey unit to survey unit, depending on which radionuclides of interest have been determined to be pertinent to the area of interest.

The primary determinant of soil $DCGL_{EMC}$ compliance for each survey unit will be scanning results combined with associated biased sampling for radionuclides of interest that lend themselves to scanning, and systematic soil samples for radionuclides of interest that are not detectable via scans. All survey units (Class 1, Class 2, and Class 3) will have complete scanning coverage. Scanning data sets will be logged to allow for post-data collection mapping, analysis, presentation, and data preservation. Biased samples collected in response to scan results, or for any other reason, will be compared to 1-$m^2$ soil $DCGL_{EMC}$ requirements.

If biased soil samples are collected, two samples will be collected and analyzed for each biased sampling location: one that is representative of the top 15 cm of exposed soils, and one that is representative of a 1 m soil depth. Sample results (biased or systematic) that exceed soil $DCGL_{EMC}$ requirements indicate soil conditions requiring further remediation. In addition, appropriate $DCGL_{EMC}$ values will be calculated based on the areas represented by systematic samples collected for $DCGL_W$ purposes using area factors provided by the DP; systematic sample results will also be compared to these additional $DCGL_{EMC}$ values.

The primary determinant of soil $DCGL_W$ compliance will be systematic sample results. Systematic samples will be collected on a random start triangular grid. Systematic samples will be composite samples formed from soil increments distributed across the immediate area the systematic sample represents. Two composite samples will be formed from each grid node, one representative of soils to a depth of 15 cm and one representative of soils to a depth of one meter. The minimum number of systematic soil sample grid locations per survey unit will be five (consistent with achieving a Type I error rate of 0.05). In the case of each composite, sufficient soil mass will be collected to allow analysis for all 18 radionuclides of interest, if necessary.

Figure G-1 contains a decision logic flow diagram for surface soil final status survey units. Sum-of-ratios values for systematic sample results will first be calculated based on soil $DCGL_{EMC}$ requirements. There are two applicable $DCGL_{EMC}$ values of interest. The first is the 1-$m^2$ $DCGL_{EMC}$ value explicitly defined in this plan. This standard will be applied to biased soil sample results. The second is a $DCGL_{EMC}$ value determined from the $DCGL_W$ using area factors (provided in Section 9 of the plan) that are appropriate for the area the systematic sample represents. This approach will be applied to systematic soil sample results.

If there are no soil $DCGL_{EMC}$ concerns, sum-of-ratios values corresponding to soil $DCGL_W$ requirements will be calculated. Samples results representing depths of 15 cm will be evaluated separately from sample results representing a depth of one meter. In each case, if the average of the results is less than unity, the Sign test will be applied assuming a Type I error rate of 0.05. If the null hypothesis is rejected for both depth intervals, the unit will be considered compliant with all relevant soil DCGL standards.

### 3.2 Subsurface Soils

In the case of the final exposed soil surface for the WMA 1 and 2 deep excavations, the general final status survey process will mirror what has already been described in Section 3.1 utilizing the appropriate subsurface DCGL standards. (One exception is that the sample interval for subsurface soil will be 0-1 m; no 0-15 cm samples are required for subsurface soil.)

The primary differences in the case of WMA 1 are the foundation pilings that will remain in place after excavation is complete. There are some 476 pilings and there are concerns that they may have provided vertical preferential flow pathways for contaminated groundwater into the Lavery Till, resulting in soil contamination at levels of potential concern within the till. This issue will be addressed both by remedial support data collection described in the Characterization Sample and Analysis Plan, and by data collection as part of the final status survey process for final status survey units that include foundation pilings.

If foundation piles did serve as preferential pathways for contamination entry into the Lavery Till, the following conditions would be expected:

- Contamination would have occurred between the piling and surrounding soil,
- Contamination that penetrated into the till would have left evidence at the till/sand and gravel unit interface (i.e., soil contamination at that interface), and
- The possibility for till contamination to occur would have been greatest where groundwater contamination was the greatest – beneath the original release point and immediately down gradient.

Based on these assumptions, the final status survey process for demonstrating that there is no significant till contamination concerns associated with pilings would have the following components:

- Excavation work will identify the exact locations of pilings and remedial action support surveys will determine where contaminated soil at levels of concern existed immediately above the Lavery Till.
- Pilings will be considered in two groups: pilings that fell within the greater-than-DCGL footprint of contaminated soils immediately above the Lavery till, and pilings that did not – final status survey data collection will target those pilings falling within the greater-than-DCGL footprint.
- In this set of pilings, sampling will be a combination of biased and systematic data collection:
  - Ten piling locations will be selected for biased sampling to look for $DCGL_{EMC}$ exceedances. This selection will target those pilings most likely to exhibit till contamination, if it existed. The selection will be based on a combination of factors, including proximity to the original release event, level of soil contamination as identified by remedial support sampling immediately above the till, visual evidence of “spaces” between the till and pilings that might have provided preferential flow pathways, etc.
  - A minimum of eight of the pilings in the footprint will be selected for each final status survey unit, at random, for $DCGL_W$ sampling. In the event that this random

selection process identifies a piling already selected for biased sampling, the sample collected from that piling will be used for both $DCGL_{EMC}$ and $DCGL_W$ compliance demonstration purposes.

For those pilings selected for sampling (either biased or systematic) sampling focus on obtaining a soil sample from immediately along the piling at a depth of one meter below the excavation surface.

If any individual soil sample identifies contamination above $DCGL_{EMC}$ requirements, additional excavation will occur to identify the extent of contamination and remove it. Additional samples will be collected from the final exposed dig face to demonstrate that no further $DCGL_{EMC}$ exceedances exist.

For each final status survey unit that includes pilings falling within the greater-than-DCGL overburden footprint, the systematic sample results from pilings will be evaluated using the Sign test. If the pilings satisfy the Sign test and there are no biased piling samples with $DCGL_{EMC}$ exceedances, till contamination associated with pilings will not be considered an issue. If fewer than five systematic piling samples are available, rather than the Sign test all systematic piling samples will be compared to the $DCGL_W$ requirement. If none are above the $DCGL_W$ values, then till contamination associated with pilings will not be considered an issue.

Figure G-2 shows the decision flow logic for final status survey data collection from the deep excavations in WMA 1 and WMA 2 floors.

### 3.2 Sediments

NOTE

The initial issue of the Phase 1 Final Status Survey Plan will not provide for Phase 1 final status surveys of Erdman Brook and Franks Creek. If it is later determined that such surveys will be performed during Phase 1 of the decommissioning, the Phase 1 Final Status Survey Plan will be revised to address these surveys following the protocols described below.

For the purposes of this conceptual framework, sediments are defined as soil or sediment-like materials associated with the bed and banks of Erdman Brook and Franks Creek within the project premises.

Historical data have demonstrated that stream sediments in Erdman Brook and Franks Creek contained within the WVDP fence line are impacted by Phase 1 radionuclides. The Characterization Sample and Analysis Plan pre-design data collection will include stream sediment and stream bank sampling to determine if remediation may be required for portions of the stream within the WVDP fence line. Currently there is no remediation planned for sediments as part of the Phase 1 decommissioning activities. Because of the integrating nature of project premises drainage features, final status survey data collection for stream features will likely be one of the final activities to avoid the possibility of re-contamination occurring post-final status survey data collection due to soil erosion and deposition within drainage features.

However, to support overall final status survey planning, the delineation of final status survey unit areas for stream and drainage features within the WVDP fence line will occur as part of Phase 1 activities. All stream features will be classified as Class 1 areas. Consistent with the sediment DCGL derivation contained in the decommissioning plan, the definition of a stream final status survey unit includes sediments within the streambed itself and three m of bank on either side of the streambed. Each unit will be at most 333 m long, comprising an area of at most 2,000 $m^2$. Subsurface contamination deeper than the 1-m definition of sediments is not considered a plausible scenario for a stream setting; consequently final status survey data collection will focus on surface sediments and adjacent bank soils. This assumption will be tested by Characterization Sample and Analysis Plan data collection.

The decision logic for sediment survey units is identical to surface soils (Figure G-1). As with surface soils across the site, a complete gamma walkover of exposed sediments and associated banks will be performed using an appropriate detector. Biased samples will be collected to clarify scan results that might be indicative of DCGL excedances. For locations where biased samples are collected, two samples will be collected, one representative of a depth of 15 cm, and one representative of a depth of 1 m.

Biased samples collected in response to scan results or for any other reason from within sediment final status survey units will be compared to sediment 1-$m^2$ $DCGL_{EMC}$ requirements. In addition, appropriate $DCGL_{EMC}$ values will be calculated based on the areas represented by systematic samples collected for $DCGL_W$ purposes using area factors provided in Section 9 of this plan; systematic sample results will also be compared to these additional $DCGL_{EMC}$ values. Sample results (biased or systematic) that exceed sediment $DCGL_{EMC}$ requirements indicate conditions requiring remediation.

Sediment $DCGL_W$ compliance will be demonstrated through the use of systematic sediment samples. A minimum of five systematic composite samples will be collected and submitted for laboratory analysis. For each location where a composite sample is obtained, two samples will be formed, one representative of a depth of 15 cm and one representative of a depth of 1 m. The radionuclides of interest subset for the analyses will be determined based on historical data and Characterization Sample and Analysis Plan data collection results.

The systematic sediment sample locations will conform to a linear grid down the length of the survey unit with a fixed grid node separation distance but random start. At each grid node, the sample collected will be formed from three increments, one from the stream centerline, and two collected from randomly selected distances up the bank from the bank's edge. In the case of each composite, sufficient soil/sediment mass will be collected to allow analysis for all 18 radionuclides of interest, if necessary.

Systematic sediment samples will be submitted for analysis based on the radionuclides of interest subset pertinent to that final status survey unit. Sum-of-ratios values for systematic sample results will first be calculated based on sediment $DCGL_W$ requirements corrected by appropriate area factors contained in Section 9 of this plan and evaluated for $DCGL_{EMC}$ exceedances. If there are no sediment $DCGL_{EMC}$ exceedances, sum-of-ratios values corresponding to sediment $DCGL_W$ requirements will be calculated. If the average of these is less than unity, the Sign test will be applied assuming a Type I error rate of 0.05.

This will be done for both depth intervals. If the null hypothesis is rejected in both cases, the unit will be considered compliant with all relevant soil DCGL standards.

In the event that the radionuclides of interest subset does not include all 18 radionuclides, one composite sample per survey unit will be formed by sub-sampling all individual systematic composite samples (after homogenization) representative of a depth of one meter from a survey unit and submitted for a complete analysis of all 18 radionuclides. If the resulting sediment $DCGL_W$ sum-of-ratios value exceeds unity, then the unit will require additional remediation. If the sum-of-ratios value is significantly influenced by radionuclides that were originally not considered pertinent to that final status survey unit, the remaining composite soil mass for each radionuclide will be analyzed for the balance of the 18 radionuclides not already analyzed, $DCGL_W$ sum-of-ratios values recalculated, and compliance with $DCGL_W$ standards re-evaluated.

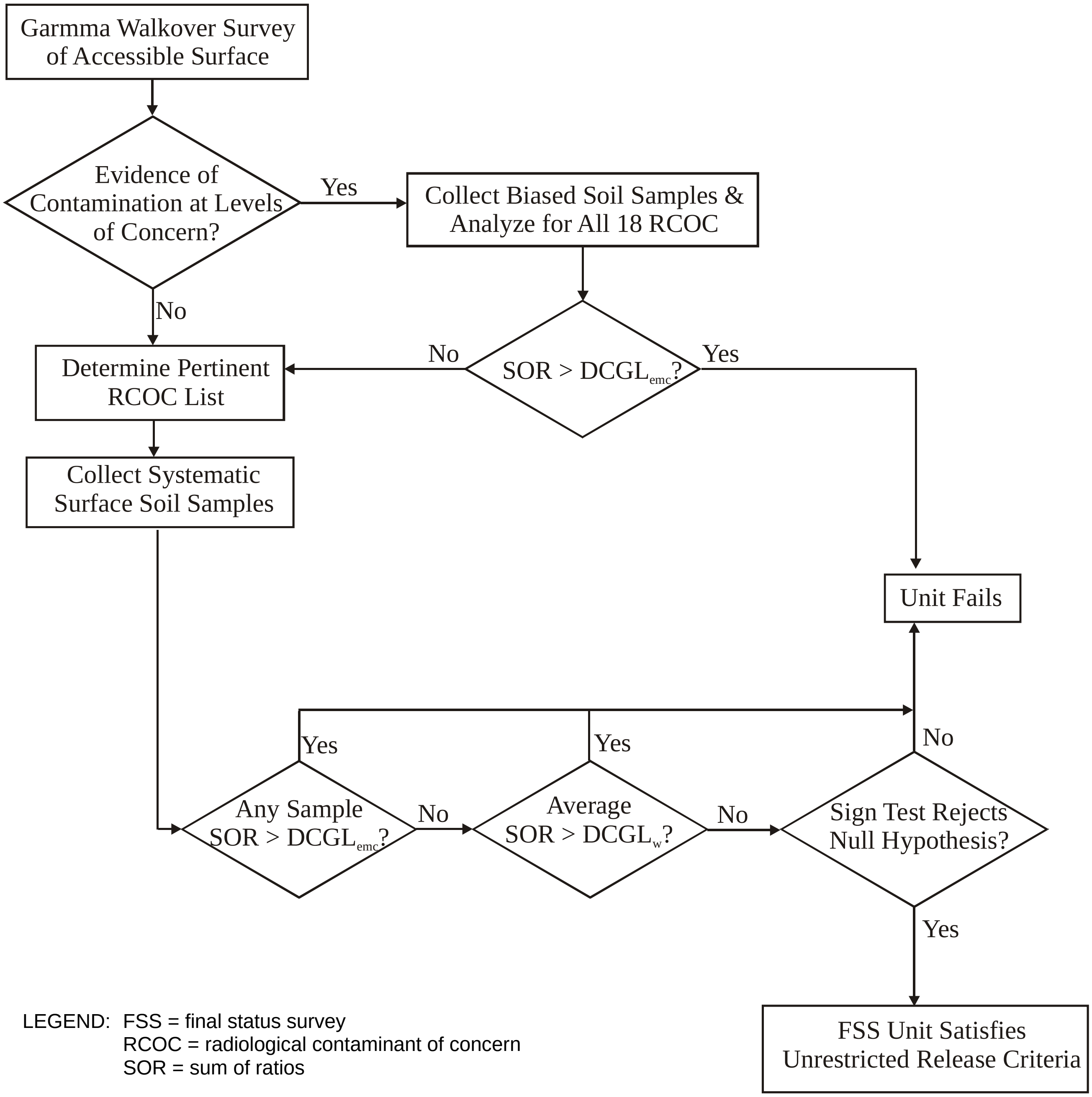
## 4.0 Documentation Requirements

Due to the complexity and time span of the Phase 1 decommissioning activities, it is expected that multiple Final Status Survey Reports will be prepared in accordance with Section 9.8 of this plan. Such reports, for example, may address a group of related survey units, such as those associated with the WMA 1 excavation, or a particular excavated soil laydown area. The use of multiple Final Status Survey Reports will facilitate independent confirmatory surveys and support periodic progress reports to interested stakeholders as the Phase 1 decommissioning activities take place.

Technical data packages will be prepared for individual survey units. Each Final Status Survey Report together with the related technical data packages will contain the information specified in Section 9.8 of this plan, including:

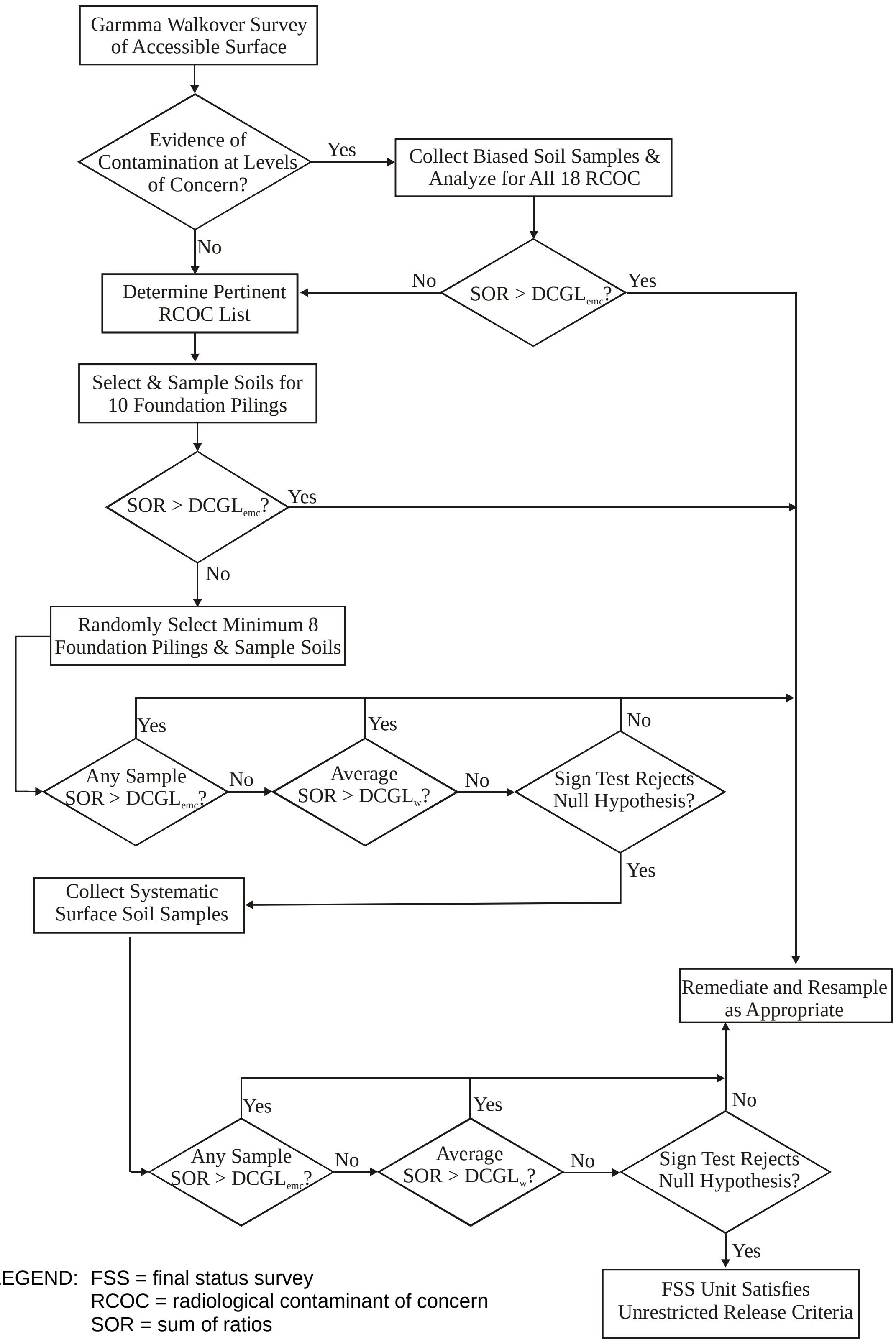
- An overview of the final status survey results;
- A description of the final status survey units comprising the area being evaluated, including any changes from what had been originally planned;
- A summary of the pertinent radionuclides of interest subset and the appropriate $DCGL_W$ and $DCGL_{EMC}$ standards;
- A description of the basis for sample numbers and the analyses used to support sample number determinations for each survey unit;
- A presentation of the gamma scan data for each survey unit, including a map showing the extent of coverage and discussion of the scan data;
- A presentation of the data collected for each survey unit, including a map or drawing of the survey units illustrating the random start systematic sample locations and the location of other samples (i.e., judgmental, biased, and miscellaneous sample data sets which will be reported separately from those samples collected for performing the statistical evaluation);
- A review of quality control parameters associated with data sets;
- A statistical analysis of the data sets with respect to the $DCGL_W$ values in the context of MARSSIM final status survey guidance;

- An evaluation of survey and sampling data to address $DCGL_{EMC}$ standards;
- A conclusion about whether $DCGL_{W}$ and $DCGL_{EMC}$ requirements have been met;
- A description of how ALARA practices were employed to achieve final activity levels; and
- If a unit fails to meet DCGL requirements, the reason for the failure, the implications for other final status survey units, the actions taken to correct the failure, and/or the implications for Phase II activities




LEGEND: FSS = final status survey
RCOC = radiological contaminant of concern
SOR = sum of ratios

**Figure G-1. Decision Logic for Surface Soil and Sediment Survey Units**




LEGEND: FSS = final status survey
RCOC = radiological contaminant of concern
SOR = sum of ratios

**Figure G-2. Decision Logic for WMA 1 and WMA 2 Subsurface Soils**