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ACRONYMS AND ABBREVIATIONS

ER	Environmental Report
HAER	Historic American Engineering Record
NRC	United States Nuclear Regulatory Commission

6.0 ENVIRONMENTAL INFORMATION

Westinghouse performed an evaluation of potential environmental impacts associated with remaining decommissioning activities and license termination. As detailed further in the Hematite Environmental Report (ER) (Reference 6-1), this evaluation demonstrates that site decommissioning activities and license termination will not have a significant adverse impact on the environment. The ER was written to include the environmental information described in NUREG–1748, (Reference 6-2).

6.1 WETLANDS AND SURFACE WATER

Jurisdictional wetlands and surface water issues will be taken into consideration during decommissioning remediation operations and activities. Additional information regarding wetlands and surface water issues is provided in the ER.

6.1.1 WETLANDS

In preparation for the site remediation investigation, a wetland and surface water assessment was conducted in November 2003, to delineate and classify potential jurisdictional wetlands and surface water bodies at the Hematite Site. The single potential wetland identified on the site is located in a small depression south of the active rail line between the railroad berm and a gravel road that goes from the Central Tract Area south towards Joachim Creek (See Figure 6-1). This potential wetland is a small, isolated forested/scrub area confined to the southeast by the gravel road, and to the north and northwest by the railroad berm. There are no planned activities at or near this wetland; therefore, the wetland will not be disturbed during the course of remedial actions. There are no inputs or outputs to the area, and hydrology appears to be the result of precipitation which collects between the road and railroad.

6.1.2 SURFACE WATER

Five intermittent tributaries (North Lake Tributary, East Lake Tributary, Northeast Site Creek, Site Creek, and Lake Virginia/Site Creek Tributary) and one perennial stream (Joachim Creek) flow across or run adjacent to the site (see Figure 6-2). A lake and a pond (East Lake and Site Pond) are also on the property. These water resources, just as wetlands, are under the jurisdiction of the federal government and the State of Missouri.

Water flow for the Site Creek/Pond and the Northeast Site Creek are planned to be diverted to support decommissioning as discussed in Section 8.6. Remediation of the Site Creek/Pond is addressed in Section 8.5.3.4 and conceptual Class 1 survey units are identified in Figure 4-14. The Northeast Site Creek does not require remediation and is in a conceptual Class 3 survey unit with a portion crossed by a haul road designated as part of a conceptual Class 2 survey unit.

6.2 THREATENED AND ENDANGERED SPECIES

A letter dated December 10, 2004 from the U.S. Fish and Wildlife Service (Reference 6-3) states that "…no federally listed, proposed or candidate species or critical habitat occurs on or near the project site…" Observations made during pedestrian surveys of the central site tract indicate that it contains neither sensitive nor unique ecological resources, nor the types of habitat to support these resources. Additional information relative to threatened and endangered species is provided in the ER.

6.3 CULTURAL RESOURCES MANAGEMENT

Historic and cultural resources (including prehistoric or historic sites, buildings, districts, structures and objects) are protected under the National Historic Preservation Act (Reference 6-4), Executive Order 11593—Protection and Enhancement of the Cultural Environment, the Archaeological and Historic Preservation Act (Reference 6-5), and the Historic Sites Act (Reference 6-6). These regulations require federal agencies take into account the effects of their actions (including permitting and licensing activities) on potential historic or cultural resources, and if necessary, resolve potential impact issues with appropriate state and federal agencies.

The historical significance of the Hematite facility relates to the role the facility filled during the "Cold War" era. From 1956 to 1974 the Hematite facility supplied high-enriched nuclear fuel for the U.S. Navy nuclear submarine program and other reactor programs. The Hematite facility was also the first privately owned and operated uranium fuel production plant in the United States.

Plans for removal of facility buildings are discussed in Section 8.2. The United States Nuclear Regulatory Commission (NRC) approved removal of the above-grade portion of site buildings in 2006. The plan to remove these buildings would result in the permanent loss of these buildings from the historical record. Due to the potential historical significance and the proposed impacts to these buildings, the National Park Service and State Historic Preservation Officer required a Historic American Engineering Record (HAER) be compiled for each of the buildings at the facility.

The HAER process has been completed for the site, including photographic documentation of both the process equipment and buildings (HAER file No. MO-113, Reference 6-7). The National Park Service provided review and approval of the HAER (Reference 6-8). The completion of the HAER adequately documents the historical resources and satisfies the requirements of Section 106 of the National Historic Preservation Act. Additional information relative to cultural resources management is provided in the ER.

6.4 REFERENCES FOR CHAPTER 6.0

6-1 Westinghouse Electric Company Document No. DO-08-009, "Environmental Report."

6-2 U.S. Nuclear Regulatory Commission, NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs," August 2003.

6-3 U. S. Department of the Interior, Fish and Wildlife Services, Letter from C. Scott (U.S. FWS) to A. Kouhestani (NRC), dated December 10, 2004.

6-4 Code of Federal Regulations, Title 16, Parts 470a-470w, "National Historic Preservation Act."

6-5 Code of Federal Regulations, Title 16, Parts 469-469c, "Archeological And Historic Preservation Act."

6-6 Code of Federal Regulations, Title 16, Parts 461-467, "Historic Sites Act."

6-7 Westinghouse Electric Company, HAER No. MO-113, "Historic American Engineering Record, Hematite Facility," 2003.

6-8 U.S. Department of the Interior, National Park Service, Letter from Rachel Franklin Weekley (Architectural Historian) to Dr. Steve J. Dasovitch (SCI Engineering, Inc.), dated December 17, 2008.

Figure 6-1

Surface Water Features And Flood Boundaries

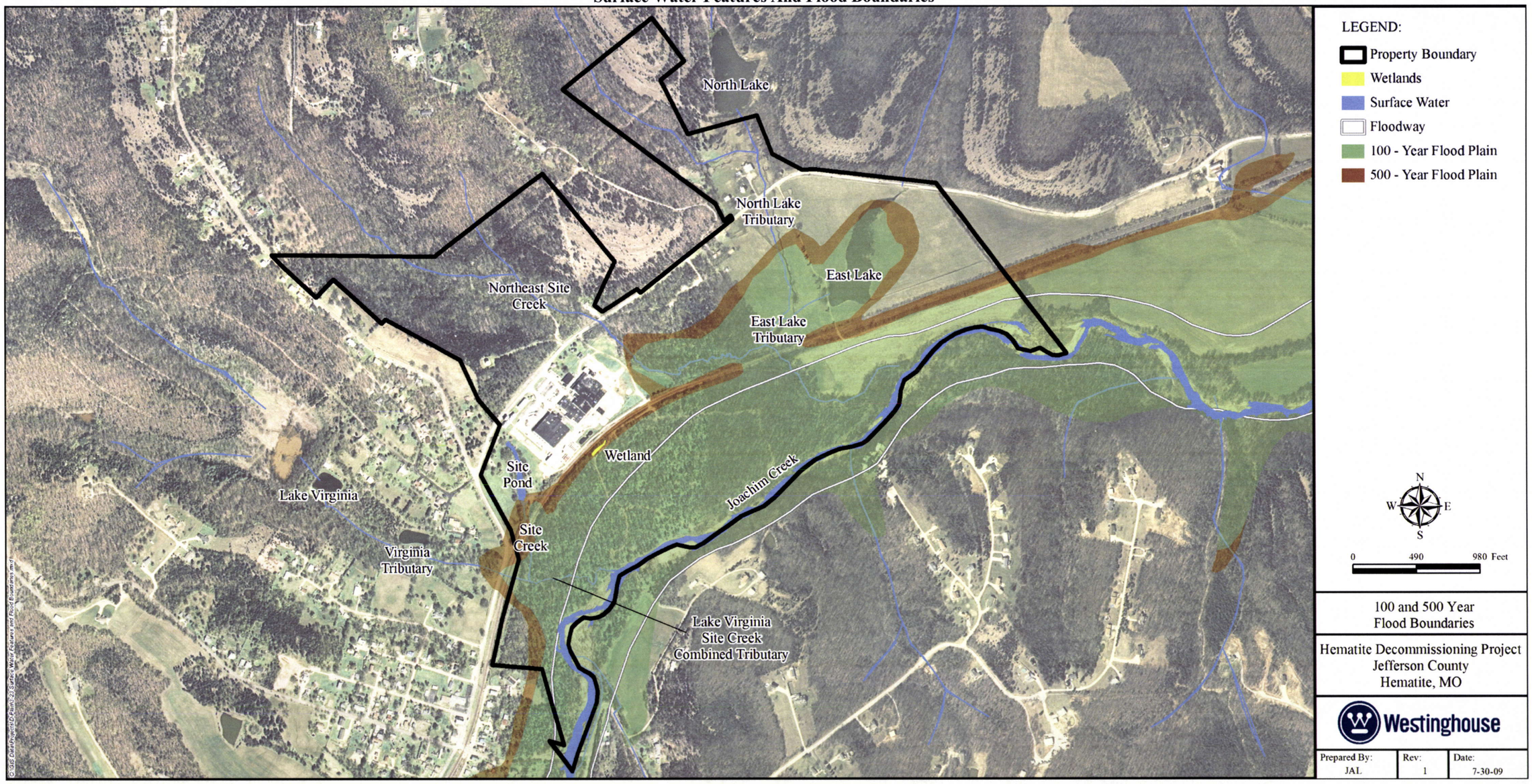
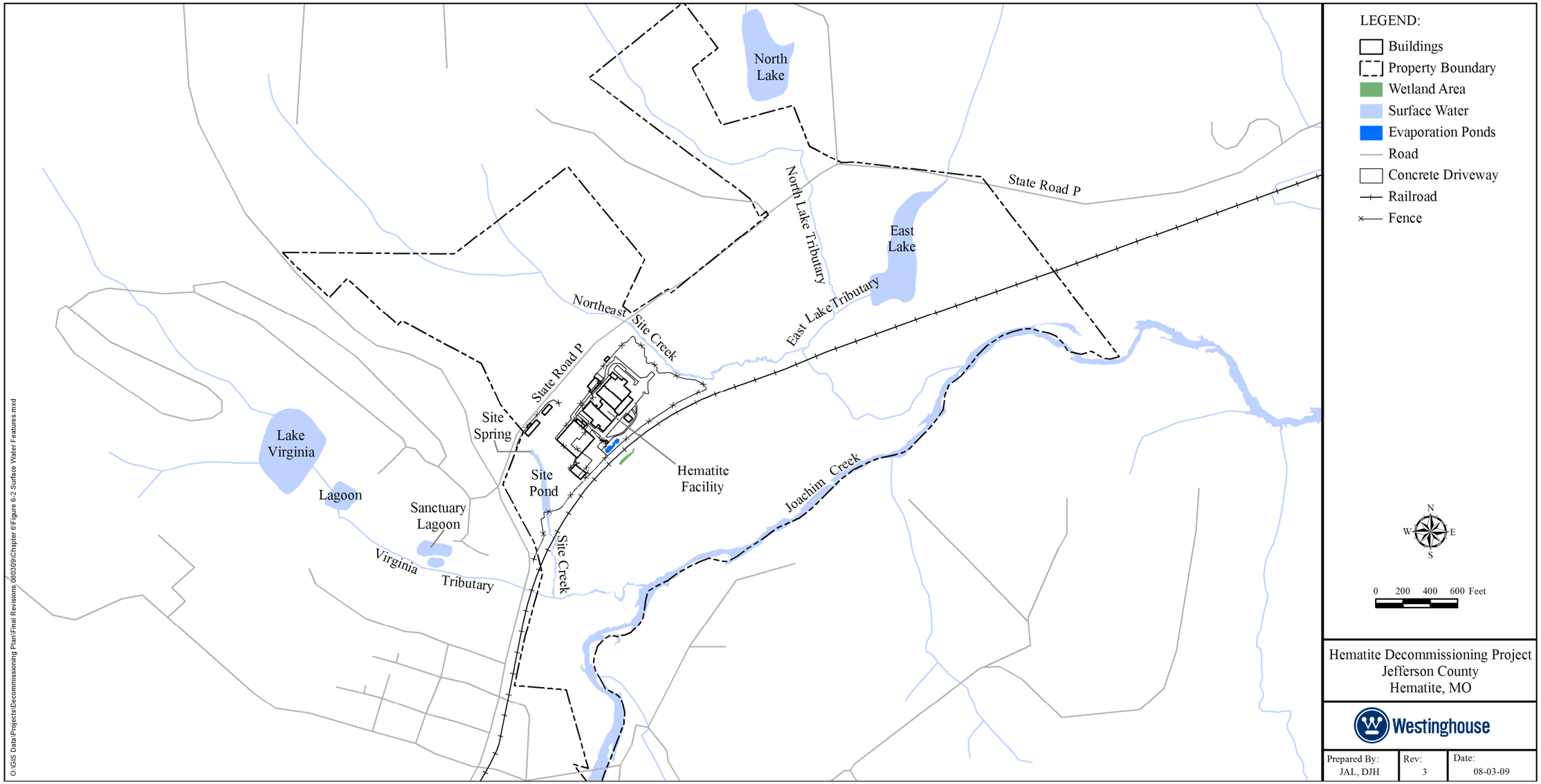


Figure 6-2

Surface Water Features



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ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
CFR	Code Of Federal Regulations
cm^2	square centimeters
DCGL	Derived Concentration Guideline Level
dpm	disintegration per minute
ft	foot
ft^2	square foot
HDP	Hematite Decommissioning Project
hr	hour
km	kilometer
m^2	square meters
m^3	cubic meters
mrem/yr	millirem per year
mSv	milliSieverts (1 mSv = 100 mrem)
NRC	Nuclear Regulatory Commission
pCi/g	picoCuries per gram
rem	Roentgen equivalent man
rem/yr	Roentgen equivalent man per year
TEDE	Total Effective Dose Equivalent
U	Uranium
yr	year

7.0 ALARA ANALYSIS

This chapter provides the methods, results and conclusions of an As Low As Reasonably Achievable (ALARA) analysis for the soils and building surface criteria developed for use at the Hematite Decommissioning Project (HDP).

The ALARA criteria are provided in the Code of Federal Regulations (CFR), 10 CFR 20.1402 (Reference 7-1), which states:

> *"A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) 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."*

In order to demonstrate the Hematite Site meets these requirements for site release, site-specific release criteria or derived concentration guideline levels (DCGLs) were developed using dose modeling, as described in Chapter 5.0. The established DCGLs, as described in Chapter 5.0, represent the radionuclide specific release criteria, that when met will ensure regulatory dose limits are satisfied.

Based on the HDP objective to remediate to unrestricted release criteria, and use appropriate dose modeling as described in Chapter 5.0 to relate concentration to dose,
Section N.1.5, Appendix N of NUREG-1757, Volume 2 (Reference 7-2) is applicable:

> *"In certain circumstances, the results of an ALARA analysis are known on a generic basis and an analysis is not necessary. For residual radioactivity in soil at sites that may have unrestricted release, generic analyses show that shipping soil to a low-level waste disposal facility is unlikely to be cost effective for unrestricted release, largely because of the high costs of waste disposal. Therefore, shipping soil to a low-level waste disposal facility generally does not have to be evaluated for unrestricted release."*

As indicated by this statement, the results of an ALARA analysis for soils are "known on a generic basis and an analysis is not necessary." However, because Westinghouse Electric Company LLC (Westinghouse) is committed to the ALARA philosophy, a simplified analysis has been performed to ensure compliance with ALARA principles.

NUREG-1757, Volume 2, Appendix N provides information outlining a simplified method to estimate when a proposed remediation guideline is cost effective. Possible benefits, as well as possible costs, are identified and compared. If the benefits from the remedial action are greater than the cost of the action, the remediation action is considered cost-effective and should be performed. Conversely, if the benefits are less than the cost, the level of residual radioactivity is considered to be ALARA, without taking additional remediation action. A list of possible benefits and costs to be considered in the analysis is provided in Table 7-1. The simplified ALARA analysis was conducted to demonstrate that the dose from residual radioactivity in soil is ALARA, satisfying the established dose criteria in 10 CFR 20, Subpart E (Reference 7-3).

A simplified ALARA analysis was also conducted for building surfaces. Based on the analysis results and Westinghouse's commitment to ALARA, Westinghouse will use typical good-practice construction efforts such as job-site cleanliness as part of the decommissioning process.

This analysis has been performed in accordance with Appendix N of NUREG-1757. However, at the time of preparation of this ALARA analysis the Nuclear Regulatory Commission (NRC) had withdrawn some sections of Appendix N, without providing interim guidance for replacement of the items removed. The withdrawal of portions of Appendix N was published in the "Federal Register Volume 72, No. 158, Page 46102" (Reference 7-4) on August 16, 2007. Specifically, the values for 'r', "monetary discount rate for dose averted" were withdrawn. Assumptions made to complete this analysis are noted in sections that follow.

The calculations shown in the following sections are based on conservative, low-end estimates of waste volume and cost per unit volume of waste. The results of calculations based on estimates of soil volume and typical unit volume costs, are summarized in Table 7-2.

7.1 DETERMINATION OF BENEFITS

7.1.1 COLLECTIVE DOSE AVERTED

As indicated above, remediation of site soils to levels that meet the unrestricted release criteria in 10 CFR 20, Subpart E, demonstrates those levels are ALARA. Therefore, calculation of collective dose averted is not required. However, collective dose costs are considered in the ALARA calculations below (Sections 7.3 and 7.4).

7.1.2 REGULATORY COST AVOIDED

Based on the Hematite Site objective to remediate to unrestricted release, costs which may be associated with restricted release (e.g., licensing fees, financial assurance costs, costs associated with public meetings or the community review committee, future liability costs, etc.) are avoided. Therefore, such costs are not taken into account in this analysis.

7.1.3 CHANGES IN LAND VALUE

Because of the relatively low land values in the Hematite vicinity and the small portion of the site assigned to industrial uses, the potential land value benefit is assumed to be small and was not considered.

7.1.4 AESTHETICS

Following removal of contaminated soil during remediation, the excavation will be refilled and contoured to the surroundings, including restoration of vegetation for erosion control. However, if a decision was made to remediate below the DCGL value, an increasing quantity of previously undisturbed land might be disrupted and removed. This additional remedial action would increase the overall environmental disturbance of the land, resulting in either no benefit, or an adverse aesthetics benefit.

7.2 DESCRIPTION OF HOW COSTS WERE ESTIMATED

Information regarding disposal costs, waste transportation and guidance in NUREG-1757, Volume 2, Appendix N was used to estimate total costs. The determination of costs does not include environmental impacts and loss of economic use of the site/facility, which need not be considered for the Hematite Site objective of remediation to unrestricted release criteria found in Code of Federal Regulations Title 10, Part 20. This level of remediation ensures the site will be available for any future proposed activity, thus preventing the loss of economic use. Additionally, land contours and vegetation will be restored following site remediation, for the purpose of erosion control. Because these costs are incurred regardless of the remediation method, they were excluded.

The calculations in the following sections are based on conservative, low-end estimates of waste volume and cost per unit volume of waste. Additional calculations for soil waste were performed using middle and high-end estimates of waste volume and a typical cost per unit volume, and are summarized in Table 7-2.

The total cost of a given decommissioning alternative is calculated using Equation N-3 of NUREG-1757, Volume 2, Appendix N:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{Acc} + Cost_{TF} + Cost_{Wdose} + Cost_{Pdose} + Cost_{Other}$$

Where:

$Cost_T$ = Total cost;

$Cost_R$ = Monetary cost of the remediation action;

$Cost_{WD}$ = Monetary cost for transport and disposal of the waste generated;

$Cost_{Acc}$ = Monetary cost of worker accidents during the remediation action;

$Cost_{TF}$ = Monetary cost of traffic fatalities during transport of the waste;

$Cost_{Wdose}$ = Monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility;

$Cost_{Pdose}$ = Monetary cost of the dose to the public from excavation, transport and disposal of the waste; and,

$Cost_{Other}$ = Other costs as appropriate for the particular situation.

The remediation cost ($Cost_R$) is the cost of the remediation action, including costs such as mobilization, demobilization, etc.

The costs for waste disposal were evaluated using the following formula, Equation N-4 from NUREG-1757, Volume 2, Appendix N:

$$Cost_{WD} = V_A * Cost_V$$

Where:

V_A = volume of waste produced in m^3; and

$Cost_V$ = cost of waste disposal per unit volume, including transportation, in dollars ($)/$m^3$.

The cost of non-radiological workplace accidents were evaluated using the following formula, Equation N-5 from NUREG-1757, Volume 2, Appendix N:

$$Cost_{Acc} = \$3{,}000{,}000 * F_W * T_A$$

Where:

F_W = workplace fatality rate in fatalities/hr;

= 4.2 E-8 /hr (NUREG 1757, Volume 2, Table N.2 (originally obtained from "NUREG 1496, Volume 2, Appendix B, Table A.1" [Reference 7-5]); and

T_A = worker-hours.

The cost of transportation risks from fatalities incurred during transportation were evaluated using Equation N-6 from NUREG-1757, Appendix N:

$$Cost_{TF} = \$3{,}000{,}000 * \left(\frac{V_A}{V_{Ship}}\right) * F_T * D_T$$

Where:

V_A = volume of waste produced in m^3;

V_{Ship} = volume of a rail shipment in m^3,

= 65 m^3 per rail car x 10 rail cars/shipment,

= 650 m^3 (Reference 7-6);

F_T = fatality rate per rail-kilometer traveled (fatalities/rail-km),

= 9.96 E-9 (8 yr average rail accident fatality rate, Reference 7-7); and,

D_T = distance traveled in km.

The cost of contaminated soil removal for unrestricted release does not include land restoration costs. The cost of remediation action ($Cost_R$), and the cost for transport and disposal of the waste generated by the action ($Cost_{WD}$), are combined into one value ($Cost_{R+WD}$) for this assessment. The $Cost_T$ is largely dependent on the volume of waste and disposal costs. Example calculations are provided below:

$Cost_{R+WD}$ = $Cost_R + Cost_{WD}$ = V_A x $Cost_{V(remediation + waste\ disposal)}$

= (Volume of waste produced) x
(cost of remediation, disposal and transportation per unit volume),

= 5,000 m^3 x \$1,100/m^3,

= \$5,500,000;

$Cost_{Acc}$ = Monetary value of a fatality equivalent to ($3,000,000/fatality) x (workplace fatality rate in fatalities/hour worked) x (worker time required for remediation in units of worker-hours),

= ($3,000,000/fatality) x (4.2E-8 fatalities/person-hr) x (5,000 m^3 x 1.62 person-hr/m^3),

= $1,021;

$Cost_{TF}$ = Monetary value of a fatality equivalent to ($3,000,000/person-rem) x (volume of waste produced/volume of a shipment) x (fatality rate per kilometer traveled) x (distance traveled),

= ($3,000,000/fatality) x (5,000m3/650m^3/shipment) x (9.96E-9 fatalities/km) x (4,571 km),

= $1,050;

$Cost_{Wdose}$ = This cost is not applicable. Based on dose modeling, the dose to an average construction worker is estimated to be 8 mrem/yr, at a soil concentration of 1,500 pCi/g Total Uranium. Therefore, $2,000 per person-rem x 0.008 rem/yr = $16/yr per construction worker. This dollar value is insignificant relative to the total cost of remediation, and need not be evaluated for the different alternatives;

$Cost_{Pdose}$ = This cost is not applicable. Dose to the public from excavation, transport and disposal of the waste is negligible; thus, the monetary cost of dose to the public is also negligible relative to the total cost of remediation;

$Cost_{Other}$ = This cost is not applicable. Land restoration costs are not included in this analysis; and,

$Cost_T$ = $5,502,072.

7.3 ALARA RESIDUAL RADIOACTIVITY LEVELS – SOIL

The purpose of this ALARA analysis is to demonstrate that the DCGLs established for soil remediation action in Chapter 5.0 are ALARA. The intent of the calculation below is to determine if additional soil remediation should be performed to further reduce dose below the 25 mrem/yr dose basis of the DCGLs. Therefore, only the cost associated with the additional remediation is used as input for these calculations.

The ALARA concentration of residual radioactivity (Conc) is that for which the benefit associated with additional remediation equals the cost of that remediation effort. Equation N-8 from NUREG-1757, Volume 2, Appendix N, can be applied as a ratio of soil concentrations to the $DCGL_W$.

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

Where:

Conc	=	Average concentration of residual radioactivity in the area being evaluated;
$DCGL_W$	=	Derived concentration guideline level equivalent to the average concentration of residual radioactivity per unit volume;
$Cost_T$	=	Total cost (see Table 7-2);
r	=	Monetary discount rate (0/yr for soil), Note: This variable was previously established at 0.03/yr for soil, but was removed from the equation through Federal Register Notice (72FR46102, August 16, 2007); therefore, for this analysis, the value has been conservatively set to 0 (i.e., no discounting for soils)
λ	=	Radiological decay constant for the radionuclide (1.55E-10/yr), assumed to be predominantly Uranium 238 (U-238). Note: The result is not impacted by the decay constant. When the decay constant for Uranium 234 (U-234) is used (2.82E-6/yr), the same result is obtained;
P_D	=	Population density (4E-4 person/m^2 for land);

F = Fraction of the residual radioactivity removed by the remediation action (0.8 - assuming 80 percent of the source term removed during remediation activities);

A = Area being evaluated (92,539 m^2 for impacted area);

N = Number of years over which the collective dose will be evaluated (1,000 yr for soil).

This calculation provides an estimate of the concentration at which a remediation action will be cost effective, prior to starting remediation. The ratio is in effect a cost-to-benefit ratio; therefore, a ratio greater than one (1) indicates the costs associated with additional remediation exceed the benefit to be realized by that remediation.

$$\frac{Conc}{DCGL_W} = \frac{\$5{,}502{,}072}{\$2{,}000 \times (4E-4) \times 0.025 \times 0.8 \times 92{,}539} \times \frac{0 + 1.55E-10}{1 - e^{-(0+1.55E-10)1000}}$$

$$\frac{Conc}{DCGL_W} = 3.72$$

Because this value is greater than one (1), the conclusion is that the soil DCGL is ALARA. Therefore, additional remediation action to achieve residual radioactivity concentrations in soil less then the DCGL values is not warranted.

The above calculation is considered to be a conservative estimate, based on use of conservative (low-range) estimates of waste volume (5,000 m^3) and cost per unit volume of waste (\$1,100/m^3). Table 7-2 provides the results of additional calculations using mid-range and high-range estimates for waste volume, and a typical cost per unit volume of waste.

Westinghouse also assumes that overburden and other excavated soils that are less than the soil DCGLs, will also meet the ALARA criterion since the cost to ship and dispose of them in a low-level waste disposal facility is not justified based on the above calculation. The waste disposal cost accounts for more than 99 percent of the total cost.

7.4 ALARA RESIDUAL RADIOACTIVITY LEVELS – BUILDING SURFACES

DCGLs were developed for buildings which will remain (Building 110, Building 230 and Building 231) following Hematite Site decommissioning and license termination, and are described in Chapter 5.0. These DCGLs were based on a Total Effective Dose Equivalent (TEDE) of 25 mrem/year to a member of the critical group, due solely to residual contamination on building and structural surfaces (building dimensions are shown in Table 7-3).

The purpose of this ALARA analysis is to demonstrate that the DCGLs established for building and structural remediation action are ALARA. The intent of the calculation below is to determine if additional building or structural remediation should be performed, to further reduce dose below the 25 mrem/yr dose basis of the DCGLs. Therefore, only the cost associated with the additional remediation is used as input for these calculations.

Chapter 5.0 describes the development of building and structure DCGLs based on three different scenarios and two different building geometries. The most conservative DCGLs were selected for use at the Hematite Site, and are summarized in Chapter 5.0. The pathway analysis associated with development of the DCGLs demonstrates most of the dose is from inhalation. The value for the contamination re-suspension rate used in development of the DCGLs was the median value from NUREG/CR 6755 (Reference 7-8).

As with the ALARA analysis for soils, the monetary cost (see Section 7.2) of dose to the public ($Cost_{Pdose}$) and workers ($Cost_{Wdose}$) during the site remediation, as well as any other costs ($Cost_{Other}$), were assumed to be zero (0) for conservatism; there is minimal dose associated with the radionuclides of concern.

The ALARA concentration of residual radioactivity ('Conc'- average concentration of residual radioactivity in the area being evaluated) is that for which the benefit associated with additional remediation equals the cost of that remediation effort. Equation N-8 from NUREG 1757, Volume 2, Appendix N, can be applied as a ratio of building surface concentrations to the $DCGL_W$.

$$\frac{Conc}{DCGL_W} = \frac{Cost_T}{\$2{,}000 * P_D * 0.025 * F * A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)*N}}$$

As indicated in Section 7.3, this calculation provides an estimate of the concentration at which a remediation action will be cost effective, prior to starting remediation. Additionally, the calculation result can be applied as a correction factor, or DCGL adjustment, for ALARA purposes. The terms for this equation are further defined below, for each analysis performed.

Consistent with the examples in NUREG 1757, the two remedial actions evaluated for this ALARA analysis were washing building surfaces and scabbling building surfaces. Washing surfaces is assumed to remove all "removable activity"; 10 percent of building surface activity is assumed to be "removable activity" (see Chapter 5.0). Floor scabbling is assumed to remove 80 percent of the residual radioactivity.

Equation N-8 of NUREG 1757, Volume 2, Appendix N was used to calculate the benefits of additional remediation assuming a survey unit size of 100 m^2, and using the parameters in Table 7-4. The results of the ALARA analysis for each of the two remedial actions further described below.

7.4.1 ALARA ANALYSIS - WASHING BUILDING SURFACES

Washing building surfaces is assumed to be effective in removing any "removable contamination". The development basis for building surface DCGLs in Chapter 5.0 assumed "removable contamination" is approximately 10 percent of the total surface contamination (i.e., F = 0.1).

The values used for the remaining equation parameters are shown below:

Conc	=	Average concentration of residual radioactivity in the area being evaluated;
$DCGL_W$	=	Derived concentration guideline level equivalent to the average concentration of residual radioactivity per unit volume;
$Cost_T$	=	Total cost (see Table 7-5);
r	=	Monetary discount rate = 0.07/yr for buildings (Reference 7-9);
λ	=	Radiological decay constant for the radionuclide (1.55E-10/yr), assumed to be predominantly U-238. Note: The result is not impacted by the decay constant. When the decay constant for Uranium 234 (U-234) is used (2.82E-6/yr), the same result is obtained;
P_D	=	Population density (0.09 person/m^2 for buildings);
F	=	Fraction of the residual radioactivity removed by the remediation action (0.1 - assuming 10 percent of the source term removed during remediation activities);

A = Area being evaluated (100 m^2 survey unit); and,

N = Number of years over which the collective dose will be evaluated (70 yr for buildings).

Assuming a reasonable waste volume of 0.227 m^3 and disposal cost of \$1,100/m^3, the calculation yield a value which is approximately 1.02 times the established DCGLs (see Table 7-5). NUREG/BR-0058, Revision 4, 'Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission' indicates that the 7 percent rate approximates the marginal pretax real rate of return on an average investment in the private sector, and is applicable to the use of capital in the private sector. When a discount rate of 0.07 is used, the results indicate that the residual contamination levels and associated DCGLs are ALARA.

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\$650}{\$2{,}000 \times 0.09 \times 0.025 \times 0.10 \times 100} \times \frac{0.07 + 1.55\text{E}-10}{1 - e^{-(0.07+1.55\text{E}-10)70}} = 1.02$$

Calculations were also performed using a 0.03 monetary discount rate. The results below show that applying a 0.03 discount rate for building surfaces as a conservative assumption yields a value which is approximately 49 percent of the established DCGLs (see Table 7-5):

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\$650}{\$2{,}000 \times 0.09 \times 0.025 \times 0.10 \times 100} \times \frac{0.03 + 1.55\text{E}-10}{1 - e^{-(0.03+1.55\text{E}-10)70}} = 0.49$$

The guidance in NUREG/BR-0058 is directly applicable to this analysis, as it involves decision-making relative to 'the use of capital in the private sector' and involves the relatively short lifespan of 70 years for buildings when considering intergenerational factors. As the results above indicate, application of a discount rate of 0.07 indicates the DCGLs are reasonable and ALARA. Subsequent to Revision 0.0 of this DP, Westinghouse performed decontamination operations on the accessible building surfaces as necessary to reduce the maximum individual measurement of removable surface contamination to less than 200 dpm/100 cm2, which is approximately 11 percent of the DCGL, based on 10 percent of the total surface contamination being in a removable form. Since the current level of removable surface contamination is less than 11 percent of the DCGL, the incremental costs associated with additional decontamination by surface washing and incremental dose avoidance were not evaluated.

Table 7-5 provides the results of additional calculations using alternative estimates for waste volume and a typical cost per unit volume of waste.

7.4.2 ALARA ANALYSIS - SCABBLING BUILDING SURFACES

The second and most common decontamination practice for structural decontamination, for which an ALARA analysis was performed, is surface scabbling or the mechanical removal of the

structural surface. Scabbling is assumed to remove 80 percent of the residual surface contamination (F = 0.80).

The values used for the remaining equation parameters are shown below:

Conc	=	Average concentration of residual radioactivity in the area being evaluated;
$DCGL_W$	=	Derived concentration guideline level equivalent to the average concentration of residual radioactivity per unit volume;
$Cost_T$	=	Total cost (see Table 7-5);
r	=	Monetary discount rate = 0.07/yr for buildings (Reference 7-9);
λ	=	Radiological decay constant for the radionuclide (1.55E-10/yr), assumed to be predominantly U-238. Note: The result is not impacted by the decay constant. When the decay constant for Uranium 234 (U-234) is used (2.82E-6/yr), the same result is obtained;
P_D	=	Population density (0.09 person/m^2 for buildings);
F	=	Fraction of the residual radioactivity removed by the remediation action (0.80 - assuming 80 percent of the source term removed during remediation activities);
A	=	Area being evaluated (100 m^2 survey unit); and,
N	=	Number of years over which the collective dose will be evaluated (70 yr for buildings).

Using these values and the input parameters from Table 7-4, the estimated costs of remediation by surface scabbling are provided in Table 7-5, using both 0.03 and 0.07 monetary rates as discussed in Section 7.4.1. NUREG/BR-0058, Revision 4, 'Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission' indicates that the 7 percent rate approximates the marginal pretax real rate of return on an average investment in the private sector, and is applicable to the use of capital in the private sector. When a discount rate of 0.07 is used, the results indicate that the residual contamination levels and associated DCGLs are ALARA.

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\$5{,}350}{\$2{,}000 \times 0.09 \times 0.025 \times 0.80 \times 100} \times \frac{0.03 + 1.55\text{E}-10}{1 - e^{-(0.03+1.55\text{E}-10)70}} = 0.51$$

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\$5{,}350}{\$2{,}000 \times 0.09 \times 0.025 \times 0.80 \times 100} \times \frac{0.07 + 1.55\text{E}-10}{1 - e^{-(0.07+1.55\text{E}-10)70}} = 1.05$$

Characterization results for Building 110, Building 230 and Building 231 are provided in the 'Hematite Radiological Characterization Report' (Reference 7-10), and indicate most surface activity in these buildings is well below the most conservative DCGL (Uranium 234 (U-234), See Chapter 5.0 for DCGLs). Subsequent to Revision 0.0 of this DP, Westinghouse performed decontamination operations on elevated radioactivity on accessible building surfaces. As a result, the average total surface contamination levels do not exceed 20 percent of the DCGL. Since the current level of total surface contamination is less than 20 percent of the DCGL, the incremental costs associated with additional decontamination by scabbling and incremental dose avoidance were not evaluated.

In considering remediation of buried drain lines, the estimated cost is $75,000 for excavation and disposal of radioactive waste of a drain line length of approximately 100 meters with a 1 meter excavation width. When this is factored into the ALARA equation, the results show that the currently established DCGLs continue to be ALARA, even with a 3 percent discount rate applied.

$$\frac{\text{Conc}}{\text{DCGL}_W} = \frac{\$75{,}000}{\$2{,}000 \times 0.09 \times 0.025 \times 1 \times 100} \times \frac{0.03 + 1.55\text{E}-10}{1 - e^{-(0.03+1.55\text{E}-10)70}} = 5.7$$

7.5 CONCLUSION

This ALARA analysis demonstrates that the dose from residual radioactivity, at the $DCGL_W$ values established for soil and building surfaces in Chapter 5.0 is ALARA. The Hematite Site approach (Section 7.2) of remediation to unrestricted release values provides net positive benefits with no consideration of costs associated with achieving those benefits.

An ALARA analysis was performed (Section 7.3) for soil remedial action, considering a range of waste volumes and disposal costs per unit volume, and the results indicate the DCGL values established for Hematite Site soils are ALARA; and, further remediation to levels less than those DCGL values is not justified.

An ALARA analysis was performed (Section 7.4) for building remedial action, considering a range of waste volumes and disposal costs per unit volume. The results indicate the DCGL values established for Hematite Site buildings which will remain following decommissioning (Building 110, Building 230 and Building 231) are ALARA, and that further remediation to levels less than those DCGL values is not justified.

7.6 REFERENCES FOR CHAPTER 7.0

7-1 Code of Federal Regulations, Title 10, Part 20.1402, "Standards for Protection Against Radiation – Radiological Criteria For Unrestricted Use," August 28, 2007.

7-2 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria." Volume 2, Revision1, September 2006

7-3 Code of Federal Regulations, Title 10, Part 20, Subpart E, "Standards for Protection Against Radiation – Radiological Criteria for License Termination."

7-4 Federal Register Volume 72, No. 158, Page 46102, "Consolidated Decommissioning Guidance; Notice of Revision to, Withdrawal of Portions of, and Process for Updating," August 16, 2007.

7-5 U.S. Nuclear Regulatory Commission, NUREG-1496, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities," Volume 2, July 1997.

7-6 Westinghouse Electric Company Document DO-05-001, "Environmental Report for Hematite Site Decommissioning," Revision 1, Section 4.2.2.a, "Normal Conditions of Transportation."

7-7 U. S. Department of Transportation, Federal Railroad Administration, Office of Safety Analysis, "Railroad Safety Statistics Annual Report," 1998 through 2005. <*safetydata.fra.dot.gov/OfficeofSafety/publicsite/Publications.aspx*>.

7-8 U.S. Nuclear Regulatory Commission, NUREG/CR 6755, "Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code," 2002.

7-9 U.S. Nuclear Regulatory Commission, NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4, September 2004.

7-10 Westinghouse Electric Company Document No. DO-08-003, "Hematite Radiological Characterization Report."

Table 7-1

Possible Benefits And Costs Related To Decommissioning

Possible Benefits	Possible Costs
• Collective Dose Averted • Regulatory Costs Avoided • Changes In Land Values • Aesthetics • Reduction In Public Opposition	• Remediation Costs • Additional Occupational/Public Dose • Occupational Non-radiological Risks • Transportation Direct Costs And Implied Risks • Environmental Impacts • Loss Of Economic Use Of Site/Facility

Table 7-2

ALARA Calculations – Soil

Estimated Waste Volume (m^3)	Estimated Waste Cost Per Unit Volume ($Cost_V$\$/m^3)	$Cost_{R+WD}$	$Cost_{Acc}$	$Cost_{TF}$	$Cost_T$	$\frac{Conc}{DCGL_W}$
5,000	$1,100	$5,500,000	$1,021	$1,051	$5,502,072	3.72
	$2,800	$14,000,000			$14,002,072	9.46
15,000	$1,100	$16,500,000	$3,062	$3,153	$16,506,215	11.15
	$2,800	$42,000,000			$42,006,215	28.37
30,000	$1,100	$33,000,000	$6,124	$6,307	$33,012,430	22.30
	$2,800	$84,000,000			$84,012,430	56.74

Note: The current soil volume estimate for the project is 801,500 cubic feet, or 22,695 cubic meters which is within the evaluated range.

Table 7-3

Remaining Building Geometry
(Building 110, Building 230 And Building 231)

Building	Length (ft)	Width (ft)	Height of Walls (ft)	Height at Peak (ft)	Floor Area (ft^2)	Floor Area (m^2)
110	60	50	12	NA	3,000	279
230	200	175	24	26	35,000	3,252
231	100	60	20	21	6,000	557
Total Floor Area					**44,000**	**4,088**

Note: Building dimensions obtained from Chapter 5.0

Table 7-4

ALARA Cost Analysis Parameters For Building Surfaces

Parameter	NUREG 1757 Volume 2 Equation	Description	Hematite Site Values
Dollars	N-1	Value in dollars per person-rem averted	* $2,000 /person-rem
P_D	N-2	Population density, people/m^2	* 0.09 person/m^2 for buildings
A	N-2	Area, m^2	100 m^2
F		Fraction of the residual activity removed by remediation	F = 0.1 for floor washing F = 0.8 for floor scabbling
Conc-Buildings	N-2	Average dpm/100cm^2 for each building survey unit (for each radionuclide)	Calculated value in ratio with $DCGL_w$
V_A	N-4	Volume of waste produced, m^3	13 m^3 (scabbling depth = 1/8 inch over 4,088 m^2) and 0.003m^3/m^2 of area scabbled (see Table 7-3 for area) or 0.318 m^3 per 100 m^2 area

Table 7-4 (continued)

ALARA Cost Analysis Parameters For Building Surfaces

Parameter	NUREG 1757 Volume 2 Equation	Description	Hematite Site Values
$Cost_V$	N-4	Waste disposal cost, $/m^3	low estimate = $1,100/m^3 high estimate= $2,800/m^3
$Cost_R$		Remediation Cost	$4/m^2 for floor washing (NUREG-1757, N.1.4) $50/m^2 for floor scabbling (NUREG-1757, N.1.4)
T_A	N-5	Worker time required for remediation, worker-hr	*1.62 person-hr/m^3 of waste, monitor, packaging and handling.
D_T	N-6	Distance traveled for waste transportation, km	Round trip: Festus, MO to Clive, UT = 2,840 miles = 4,571 km
V_{Ship}	N-6	Volume of rail shipment, m^3	65m^3/rail car *10 railcars/shipment = 650 m^3/shipment (Reference 7.7)
F_W	N-5	Workplace accident fatality rate, /hr	* 4.2 E-8 /hr

Table 7-4 (continued)

ALARA Cost Analysis Parameters For Building Surfaces

Parameter	NUREG 1757 Volume 2 Equation*	Description	Hematite Site Values
F_T	N-6	Transportation fatal accident rate, /km	9.96 E-9/km Rail Accident Fatality Rate 8-yr average (Reference 7-7)
r	N-2, N-8	Monetary discount rate	0.07 /yr for buildings [parameter from NUREG/BR-0058, Revision 4]
N	N-2, N-8	Number of years of exposure	* 70 years for buildings

Note: * values from NUREG-1757, Volume 2, Table N.2 “Acceptable Parameter Values for Use in ALARA Analysis”.

Table 7-5

ALARA Calculations - Building Surfaces

Estimated Waste Volume (m^3)	Estimated Waste Cost Per Unit Volume ($\$/m^3$)	$Cost_{WD}$	$Cost_R$	$Cost_{Acc}$	$Cost_{TF}$	$Cost_T$	Discount, r=0.03 $\frac{Conc}{DCGL_W}$	Discount, r=0.07 $\frac{Conc}{DCGL_W}$
ALARA Analysis– Washing Building Surfaces								
0.227	**$1,100**	$250	$400	$0.05	$0.05	$650	0.49	1.02
0.227	**$2,800**	$636	$400	$0.05	$0.05	$1,036	0.79	1.62
1	**$1,100**	$1,100	$400	$0.20	$0.21	$1,500	1.14	2.35
1	**$2,800**	$2,800	$400	$0.20	$0.21	$3,200	2.43	5.02
ALARA Analysis – Scabbling Building Surfaces								
0.318	**$1,100**	$350	$5000	$0.06	$0.07	$5,350	0.51	1.05
0.318	**$2,800**	$890	$5000	$0.06	$0.07	$5,891	0.56	1.15
1	**$1,100**	$1,100	$5000	$0.20	$0.21	$6,100	0.58	1.20
1	**$2,800**	$2,800	$5000	$0.20	$0.21	$7,800	0.74	1.53

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ACRONYMS AND ABBREVIATIONS

ACM	Asbestos-containing Material
AHA	Activity Hazard Analysis
ALARA	As Low As Reasonably Achievable
ARARs	Applicable Or Relevant And Appropriate Requirements
BMPs	Best Management Practices
BPA	Burial Pit Area
CAA	Controlled Access Area
CCIS	Criticality Control Inventory System
CDBS	Collared Drum Buffer Store
CFR	Code of Federal Regulations
DCGL	Derived Concentration Guideline Level
Dec.	December
DNAPL	Dense Non-Aqueous Phase Liquid
DOSA	Drum Over-pack Storage Area
DOT	Department of Transportation
DP	Decommissioning Plan
EEMP	Effluent And Environmental Monitoring Plan
EH&S	Environmental Health And Safety
FCSA	Field Container Storage Area
FMSA	Fissile Material Storage Area
FNMCP	Fundamental Nuclear Material Control Plan
FR	Federal Register
FSS	Final Status Survey
GAC	Granulated Active Carbon
gpm	gallons per minute
GWS	Gamma Walkover Survey
HASP	Health And Safety Plan
HDP	Hematite Decommissioning Project
HEPA	High Efficiency Particulate Air
HRGS	High Resolution Gamma Spectroscopy
LLRW	Low Level Radioactive Waste
MAA	Material Assay Area
MC&A	Material Control And Accounting

ACRONYMS AND ABBREVIATIONS
(continued)

MDNR	Missouri Department Of Natural Resources
MTSC	Material Transit And Storage Container
NCS	Nuclear Criticality Safety
NCSA	Nuclear Criticality Safety Assessment
NPDES	National Pollutant Discharge Elimination System
NRC	U. S. Nuclear Regulatory Commission
OCA	Owner Controlled Area
OSHA	Occupational Safety And Health Administration
POC	Project Oversight Committee
PPE	Personal Protective Equipment
PSP	Physical Security Plan
QA	Quality Assurance
RA(s)	Restricted Area(s)
RASS(s)	Remedial Action Support Survey(s)
RACM	Regulated Asbestos-containing Material
RCRA	Resource Conservation and Recovery Act
rem	Roentgen equivalent man
RG(s)	Remediation Goal(s)
RP	Radiation Protection
RPP	Radiation Protection Plan
RR	Railroad
RSO	Radiation Safety Officer
RWP	Radiation Work Permit
SDS	Storm Drain System
SNM	Special Nuclear Material
SVE	Soil Vapor Extraction
SWPPP	Storm Water Pollution Prevention Plan
SWTP	Sanitary Wastewater Treatment Plant
TSDF	Transportation Storage Disposal Facility
U-235	Uranium-235
UCSA	Un-assayed Container Storage Area
VOCs	Volatile Organic Compounds
VOCTA	VOC Treatment Area

ACRONYMS AND ABBREVIATIONS
(continued)

WCA	Waste Consolidation Area
WEA	Waste Evaluation Area
WHA	Waste Holding Area
WMP	Water Management Plan
WMTP	Waste Management And Transportation Plan
WTS	Water Treatment System

8.0 PLANNED DECOMMISSIONING ACTIVITIES

Decommissioning activities will be performed pursuant to Title 10 of the Code of Federal Regulations (CFR), Part 70 (Reference 8-1), The Hematite Decommissioning Plan (DP), and in accordance with applicable federal, state and local requirements. Table 8-1 contains a listing of site plans that will be used to control the activities described in this chapter.

The DP provides the site-specific basis for the completion of decommissioning activities as presented in this chapter. This chapter was prepared using guidance from NUREG-1757, "Consolidated Decommissioning Guidance, Decommissioning Process for Materials Licensees" (Reference 8-2). Figure 8-1 depicts the current status of the developed portion of the site.

The scope of remaining decommissioning activities to the point of license termination in accordance with 10 CFR 20 (Reference 8-3), include the following:

- Install site infrastructure including: temporary utilities, security equipment, rail spur and loading pad, soil treatment facility, water treatment system, equipment and soil staging areas, and temporary haul roads;
- Decontaminate structures, systems and equipment intended to remain at the time of license termination;
- Perform radiological surveys of buried and embedded piping to assess nuclear criticality safety requirements, design remediation "cut-plans", and develop waste disposition strategies;
- Demolish and package for off-site disposal, site buildings and infrastructure not designated for unrestricted release. Table 8-2 provides a listing of the all existing Hematite structures and a description of the previous uses of the structures. Table 8-3 provides a listing of buildings, foundations and paved areas intended to be removed during decommissioning;
- Excavate soil, buried waste, and concrete foundations within impacted areas while segregating soil that is acceptable for re-use as backfill;
- Treat soil and waste potentially containing volatile organic compounds (VOCs);
- Package and coordinate transportation for disposal of radioactive, hazardous and mixed waste;
- Perform Final Status Surveys (FSS) and sampling within structures, systems and soils that will remain at the time of license termination;

- Backfill excavations using a combination of on-site material determined to be suitable for re-use (e.g., excavated overburden), and material from a pre-evaluated source;
- Submit FSS Data Summary Reports, optionally in a logical sequence as phases of the decommissioning are completed; and,
- Disposition of recoverable special nuclear material.

A Gantt chart schedule is provided as Figure 8-2. The identified activities are intended to provide an overview of the remaining activities and an estimated time schedule for each. The time frames for conducting the activities are dependent upon approval of this Decommissioning Plan. Conceptual schedules provided in this section are for general guidance and illustrative purposes only. An updated schedule will be maintained during decommissioning and will be available for review by regulatory agencies.

8.1 DECOMMISSIONING PROGRAMS AND PROCEDURES

Westinghouse will conduct decommissioning activities in accordance with written, approved procedures.

Table 8-1 provides a summary of Hematite plan documents that will be used for support of activities described in this chapter. Compliance with work plans and procedures for decommissioning activities is described in detail in Chapter 9.0.

Decommissioning activities will be reviewed by the Hematite Decommissioning Project (HDP) project staff in accordance with the work control program prior to implementation. Certain activities will be presented to the Project Oversight Committee (POC) for review and approval for implementation. HDP project staff will provide oversight for field work activities using a quality graded approach.

8.1.1 INTEGRATED SAFETY ANALYSIS

Pursuant to 10 CFR 70 (Reference 8-1; specifically, 10 CFR 70.60), the regulations in 10 CFR 70.61 through 10 CFR 70.76, including requirements for an Integrated Safety Analysis, do not apply to decommissioning activities. Westinghouse has been authorized to conduct decommissioning activities pursuant to 10 CFR 70.38 since NRC license amendment 42, dated April 11, 2002.

8.1.2 CRITICALITY SAFETY APPROACH

The HDP criticality control philosophy is based on management of fissile material handling and storage activities and processes, primarily using established container or functional area mass limits. Solid wastes categorized as Nuclear Criticality Safety (NCS)-exempt material will be dispositioned as clean, hazardous, mixed and/or Low Level Radioactive Waste (LLRW), depending on radiological and chemical constituents. NCS-exempt material is not subject to NCS controls or oversight. The criticality safety approach for concrete, piping systems and soil areas of the Hematite Site relies on identification of fissile material, or suspect fissile material, by comprehensive in-situ radiological surveys and visual inspection followed by careful extraction, containerization and segregation from other fissile material. Extracted, containerized fissile materials, or suspect fissile materials, will be measured to establish Uranium 235 (U-235) content. Based on evaluation results, materials will be controlled as either fissile material or NCS-exempt material.

Materials are evaluated against two concentration levels. Material that is less than the lower concentration is classified as NCS-Exempt Material and is not subject to additional criticality controls. Material with concentrations greater than the lower limit but less than the upper concentration is classified as Fissile Material and will be exhumed using the controls

summarized in this section and detailed in Chapter 10.0 and the Nuclear Criticality Safety Assessments. Material with concentrations above the upper concentration triggers a Stop Work in the area. In the event that actual concentrations identified in the material exceed the established controls, work will stop while additional controls are established.

Criticality safety controls will be maintained during remediation activities until suspect physical materials have been removed, and radiological information indicates the NCS-Exempt Materials Limit has been met in the removal area. Criticality safety controls are required in accordance with governing Nuclear Criticality Safety Assessments (NCSAs), and are planned for the following areas during remediation activities and excavations:

- Buried Waste and Contaminated Soil Remediation;
- Water Collection and Treatment;
- Collared Drum Transit, Staging and Buffer Storage;
- Waste Evaluation and Assay;
- Collared Drum Repacking Area Operations;
- Sub-surface structures; and
- Fissile Material Storage Area Operations.

A detailed discussion of criticality safety controls to be employed during remediation is provided in Chapter 10.0.

8.1.3 SECURITY REQUIREMENTS

The requirements for physical security, and protection of remediation activities and excavated materials, are provided in the Hematite Physical Security Plan (PSP, Reference 8-4). The Hematite PSP is based on the requirements of 10 CFR 73.67 (Reference 8-5), for fixed site and in-transit requirements for the physical protection of special nuclear material of low and moderate strategic significance, with a contingency for Special Nuclear Material (SNM) of other strategic significance.

8.1.4 TYPICAL RADIATION PROTECTION AND SAFETY CONTROLS

RP methods and controls, as discussed in Chapter 10.0, will be employed during remediation of each room or area of contaminated structures. The discussion that follows represents an overview of RP methods to be employed.

Areas and rooms will be controlled and posted based on current radiological conditions. Access control will be maintained using physical barriers, roped areas and access control points that include step-off pads for donning and doffing Personal Protective Equipment (PPE).

Radiation Work Permits (RWPs) will be established and will specify the requirements for PPE and dosimetry. RWPs will specify the occupational monitoring to be performed, such as: Health Physics Technician coverage; personal dosimeters for external exposure; air monitoring for internal exposure; and, RWPs will provide guidance for maintaining personnel exposures As Low As Reasonably Achievable (ALARA).

Engineering controls will be employed to prevent the spread of contamination and minimize airborne radioactive materials. Controls may include application of water misting, sealants or encapsulants to minimize the spread of contamination, and HEPA-vacuum-fitted mechanical grinders or hooded scabblers. Shields fitted to mechanical equipment can protect from airborne releases and reduce spreading particulates to adjacent areas. Water mists may be applied to control dust. Equipment may be fitted with HEPA-filtered vacuum attachments, with sufficient flow to capture dust and fine particulates. HEPA-ventilated enclosures will be used as necessary.

The Hematite Health and Safety Plan (HASP, Reference 8-6), contains the requirements for protection of project personnel and the general public form industrial and chemical hazards. A task-specific activity hazard analysis (AHA) will be performed to identify the industrial, physical, chemical and biological hazards associated with remediation of contaminated structures. The AHA will provide engineering, administrative and PPE control measures, and industrial hygiene monitoring requirements. Where practicable, RWP and AHA requirements will be integrated to optimize PPE and monitoring requirements.

8.2 SITE PREPARATION

Site preparation for soil remediation may include temporary shelters, structures, facilities and infrastructure. Temporary structures and facilities will be removed by the conclusion of the project, and any impacted or disturbed areas will be restored to an acceptable condition. Figure 8-3 shows a conceptual layout of temporary facilities and infrastructure, as detailed below:

- Temporary haul roads will be constructed by removing vegetation and grading the route, followed by placing, spreading and compacting road materials. Temporary haul roads will be maintained throughout the project, as required; then removed prior to FSS of the areas underlying the temporary haul roads. A temporary haul road was constructed along the northern portion of the site. The haul road connects the main parking lot with the Controlled Access Area and the newly constructed Laydown Area. A portion of the haul road is located next to the Waste Evaluation Area and provides access to a truck scale for weighing conveyances. The Haul Road also provides access to a Box Counter system for the performance of radiological assessment of conveyances containing materials originating from the Burial Pit remediation;

- Temporary fencing may be installed and relocated as necessary, as work proceeds to define areas where radiological controls for access have been established;

- Civil surveys will be performed to establish baseline topography and create control points for FSS survey units;

- Underground utilities will be located, marked and isolated, as necessary. A natural gas pipe line crosses the impacted portion of the site adjacent and parallel to the existing site railroad line (see Figure 8-3). The natural gas pipe line is anticipated to remain in service for the duration of the decommissioning activities. The location and depth of this gas line below the existing ground surface has been clearly defined by excavating at intervals of approximately 40 feet and recording location coordinate data. Prior to performing excavation activities at this location, precautions and control measures for excavation activities in areas affected by the natural gas pipe line will be detailed in the appropriate work control documents;

- A railroad spur and railcar loading pad was constructed near the southeast side of the impacted area of the site to support bulk loading of waste from the Waste Holding Area (WHA). The railcar loading pad was constructed of materials purchased from approved offsite sources. The Loading Pad is immediately adjacent to the Rail Spur, and will serve as an operating surface for heavy equipment used to load material into railcars. The Loading Pad is 50 by 150 feet

area that abuts the WHA. The WHA is where waste is received into two or more bins for final disposal characterization prior to being loaded into the railcars;

- A Water Treatment System (WTS) has been installed within the southeast corner of Building 230. The WTS includes storage and settling tanks, pumps, water lines, bag filters, and vessels containing granulated activated carbon (GAC) and ion exchange resin. The WTS will be decommissioned and removed prior to conducting the FSS of this portion of Building 230;

- Monitoring wells within or adjacent to planned excavation areas, will be identified and marked in the field to reduce the potential for inadvertent damage. Wells that exist within areas to be excavated will be identified for abandonment during remediation. On an area-by-area basis, post-remediation monitoring wells will be installed and developed during the first quarter following remediation and will be sampled for laboratory analysis during the second quarter following remediation.

- Storm water management controls will be implemented to prevent surface water run-on and run-off from planned excavation areas. Storm water management controls include, but are not limited to: grading; trenching; berming; and, installing sump pumps and lines within excavations, as needed. Collected storm and ground water will be sampled and processed through the Water Treatment System (WTS), if required;

- If onsite treatment for VOC is conducted a VOC Treatment Area (VOCTA) and system will be installed to prepare soil containing VOC for disposal. Currently, the VOCTA is planned for installation at the northeast end of the northern-most slab on the footprint of Building 255. The VOCTA will include a staging area for soil having concentrations that meet the DCGL, and a physically separate staging area for soil having concentrations that exceed the DCGL. The VOC treatment system design will comply with Resource Conservation Recovery Act (RCRA) requirements, incorporating a liner, secondary containment, and leachate collection. Barriers will be installed to prevent inadvertent damage to the system by heavy construction equipment; and,

- A high resolution gamma spectroscopy (HRGS) system will be installed to measure radioactivity in soil as a part of the measurement process for qualifying soil for re-use as backfill. Currently, a location north of Building 110 and inside of the Controlled Access Area (CAA) fence has been identified for the installation of the HRGS and the associated truck scale.

- Temporary shelters and facilities will be constructed, as necessary, to support demolition and removal of the FCSA, FMSA, WEA, HRGS Station, Waste Holding Areas (WHA) and Equipment Decontamination Facility.

A number of locations and facilities have been identified as necessary to implement the requirements of the nuclear criticality safety program. The following text is provided to identify the primary elements that will be required for implementation, and to familiarize the reviewer with the purpose of each in the context of the overall process. A detailed technical discussion of the basis for the nuclear criticality safety controls to be employed during remediation is provided in Chapter 10.0.

- A Collared Drum Buffer Store (CDBS) will be constructed east-southeast of the HRGS to stage empty containers;

- A Fissile Material Storage Area (FMSA) will be constructed within the CAA Barrier. It is anticipated the FMSA will consist of a 'sea-land' container configured in accordance with the NCSA;

- A Waste Evaluation Area (WEA) has been constructed within Building 115. If necessary, another appropriate area may be used as a WEA upon approval of the HDP Project Director, RSO, and Security Manager;

- A WHA will be constructed northeast of the HRGS area. The WHA will be graded, compacted, lined and bermed to provide adequate drainage, and water run-off and run-on control;

- A Waste Consolidation Area (WCA) will be constructed adjacent to the Burial Pit Area. This area will be graded, compacted, lined and bermed, as necessary, to provide adequate water run-on and run-off control; and,

- A Drum Over-pack Storage Area (DOSA) will be established south of the FMSA.

8.3 CONTAMINATED STRUCTURES

The locations of the buildings and paved surfaces are illustrated on Figure 8-1, and a brief description of each is provided in Table 8-2.

8.3.1 STRUCTURES TO BE DEMOLISHED

The structures to be removed have prior approval for demolition in the U.S. Nuclear Regulatory Commission (NRC) License No. SNM-33 (Docket No. 70-36) (Reference 8-7). The buildings to be demolished, and paved or concrete surfaces that will be removed are illustrated on Figure 8-4. Table 8-3 provides a listing of buildings, foundations, and paved areas that will be removed during the decommissioning.

The floor slabs and foundations of the structures will be removed as part of the Decommissioning Plan. Process drains within the foundation slab of buildings that will be demolished (Building 240, Building 253, Building 254, Building 255, Building 256 and Building 260) will be surveyed in conjunction with removal of the building floor slabs and foundations. Concrete foundations, slabs and paved areas may be decontaminated prior to removal, or removed and prepared for off-site disposal. Water management methods associated with remediation of concrete foundations, slabs and paved areas are discussed in Section 8.6.

- Contamination surveys of surfaces of concrete slabs will be performed prior to and following decontamination efforts. Samples of processed concrete and asphalt will be analyzed to determine compliance with appropriate release criteria, or waste acceptance criteria for a waste disposal facility.

- Decontamination of concrete slabs, foundations and paved surfaces will be performed in accordance with approved work instructions and hazard control measures. Typical decontamination techniques include wiping, High Efficiency Particulate Air (HEPA)-vacuuming, mechanical grinding, scabbling, chipping, saw-cutting, chemical stripping and power-washing surface areas. Surfaces that cannot be decontaminated to levels below DCGL will be removed.

- Breaking and sectioning (sizing) concrete foundations, slabs and paved areas will be performed using an excavator equipped with a hydraulic breaker, or other concrete processing equipment which allows breaking of concrete and asphalt into pieces of manageable size. Concrete rebar will be removed as necessary, and sections that can not be readily decontaminated will be segregated.

- Handling and processing broken concrete slabs, foundations and materials from paved areas to allow contamination surveys to be performed on all surfaces.

- Segregation and preparation of concrete slabs, foundations and materials from paved surfaces to meet off-site disposal requirements.

- Remedial Action Support Surveys (RASS) will be performed periodically during decontamination to gauge the effectiveness of method, and to determine when DCGLs have been met.

8.3.2 STRUCTURES THAT MAY REMAIN OR BE DEMOLISHED

Note that the Fire Pump House (Building 115), Building 235, and Sanitary Wastewater Treatment Plant Shed may remain at the time of license termination depending upon the outcome of an evaluation of the cost-benefit for decommissioning and/or survey of the system piping versus removal and disposal. Figure 8-4 depicts structures where their future disposition is yet to be determined. Depending on final evaluation, these structures may remain; they will be addressed as described in Section 8.3.1 or 8.3.3 depending on final decision.

8.3.3 STRUCTURES TO REMAIN

For the buildings and paved areas that are planned to remain at the time of license termination (Figure 8.4), this section summarizes the tasks that are required to decontaminate buildings to levels that do not exceed the Derived Concentration Guideline Levels (DCGLs).

Remediation tasks and techniques that may be performed for concrete foundations, slabs and paved areas, include:

- RASS will be performed to confirm the location of residual contamination. Locations identified as contaminated will be marked to assist in guiding the decontamination effort. Additionally, RASS will be performed on the floors, floor trenches and floor seams/cracks to determine if further remediation will be required. Remediation tasks include cleaning surfaces with chemical, physical and mechanical methods, or physical removal of contaminated surfaces. Techniques for decontamination include wiping, HEPA-vacuuming, chemical stripping, scabbling, chipping, saw-cutting, power-washing and mechanical grinding.

- Removal of portions of floors, trenches or cracks/seams will be performed using techniques such as saw cutting and jack hammers. Refer to Section 8.2.1 for additional information.

8.3.4 RADIATION PROTECTION AND SAFETY CONTROLS

Radiation protection methods and control procedures applicable to excavation and open land areas are described in Chapter 10.0 and Section 8.1.4. In general, selected decontamination techniques will utilize equipment and materials with demonstrated effectiveness and a safe performance record. None of the areas requiring remediation presents unique remediation or safety concerns.

8.4 CONTAMINATED SYSTEMS AND EQUIPMENT

Depending on the results of radiological surveys and estimates of the contained Uranium in section of the process drains materials such as latex paint, expanding foam or low density flowable-fill (grout) may be added to the piping to stabilize the contents prior to removal.

8.4.1 SANITARY WASTEWATER TREATMENT PLANT (SWTP) AND STORM DRAIN SYSTEM (SDS)

For buildings designated to remain on site following license termination (Building 110, Building 230 and Building 231), contaminated systems and equipment that may require remediation, include the SWTP and the SDS (which includes exterior building drains for precipitation for Building 110 and Building 230). Figure 8-5 shows the location of the SWTP and SDS. These systems will be surveyed and evaluated from a cost-benefit perspective to determine if components may be decontaminated and remain in place, or need to be disposed of as radioactive waste.

Drain piping that cannot be accessed for survey may require removal based on an evaluation of historical information, and the information obtained from similar drain components during decommissioning. If this information is incomplete or inadequate to form a reasonable basis that the drain and surrounding soil meet the DCGL, then the drain will be removed and surrounding soil evaluated by radiological surveys and sampling.

Remediation tasks for the SWTP and the SDS include: locating and stabilizing contamination, as necessary for contamination control; excavation, removal and segregation of soil and debris for disposal; *in-situ* Gamma Walkover Surveys (GWS); VOC screening; and, visual inspection. In general, excavations will proceed along the length of marked utilities, and expand and progress forward as soil and debris are removed. FSS will be performed in stages along the length of the excavations, with sufficient buffer zones and physical barriers installed to prevent re-contamination of remediated areas.

- RASS will be performed on the interior surfaces of drain systems to determine if remediation will be required. Contaminated drain systems will be remediated to levels that do not exceed the DCGLs that are approved for small office and large warehouse building surfaces; or will be physically removed and packaged for disposal at an off-site facility; or will be remediated to levels that do not exceed the DCGLs that are approved for buried piping and filled with grout.

- Access will be established for contaminated drains and piping, which will be decontaminated or removed as necessary, utilizing approved work instructions and hazard control measures; and

- Decontamination techniques may include mechanical decontamination such as brushing, grinding, and stripping. Techniques for physical removal of contaminated systems and equipment may include concrete or asphalt saw cutting, and jackhammer or breaking of concrete and asphalt surfaces.

Management of surface and groundwater during excavation, removal, interim stabilization and remediation of the SWTP and SDS, is discussed in Section 8.6. Generally, remediation tasks and techniques for the SWTP and the SDS will be performed in the following sequence:

- Civil surveys, underground utility location surveys and survey marking will be performed to identify the location of the SWTP and SDS;

- In-situ radiological surveys will be performed within contaminated piping. Survey results will be compared to actions levels that define the requirements for nuclear criticality safety during removal; used to characterize the piping for disposal; or used to demonstrate that the piping meets the DCGL. The DCGL for piping to remain in place are those approved for building surfaces, or those specifically defined for buried piping provided the piping is filled with grout;

- The SWTP and SDS will be isolated from existing services, and existing services will be redirected or abandoned. Residual contamination in piping systems will be stabilized using materials such as latex paint, expanding foam or low density flowable-fill (grout) prior to removal;

- Excavation and removal of overburden, waste soil and debris will proceed as described in Section 8.5.

Excavation and removal will continue until RASS, as described in Chapter 14.0, and chemical sampling activities indicate the applicable DCGL have been met. Excavation sites will then be prepared for the FSS.

Physical barriers will be installed and buffer zones maintained to protect portions of excavations available for FSS. Buffer zones will be transitioned to areas available for FSS, as excavation proceeds along the length of underground utilities and systems. Open excavations will be maintained throughout the FSS, and until restoration is authorized. Restoration of excavations will include placement of clean fill from an approved source, or site material that meets the criteria for re-use as backfill, followed by grading and re-vegetation.

8.4.2 VENTILATION SYSTEM

Ventilation systems will be decontaminated or removed, utilizing approved work instructions and hazard control measures. Decontamination techniques may include HEPA-filtered vacuuming.

8.4.3 RADIATION PROTECTION AND SAFETY CONTROLS

Radiation protection methods and control procedures applicable to excavation and open land areas are described in Chapter 10.0 and Section 8.1.4. The discussion that follows represents an overview of the RP methods applicable to excavations and open land areas.

Areas surrounding open excavations associated with removal of the SWTP and SDS will be controlled and posted in accordance with the radiological conditions. Contamination control surveys will be performed for personnel, equipment and materials leaving the posted area, as necessary. Additional controls may include: covering waste stockpiles; constructing temporary berms around waste staging and handling locations; and, constructing lay-down areas so storm water will drain into one area for collection and discharge.

Engineering controls will be employed to prevent the spread of contamination and minimize airborne radioactive materials, including: dust suppression using water; covering waste piles; and, water management practices. Prior to removing embedded or buried drain piping systems, residual contamination in piping will be stabilized using an aerosol (e.g., latex paint), expanding foam or low density flowable-fill (grout).

Chapter 11.0 of this plan describes the elements of the environmental monitoring and protection program, which are intended to meet the requirements of Missouri Department of Natural Resources (MDNR) regulations and National Pollutant Discharge Elimination System (NPDES) Permit No. MO-0000761 (Reference 8-8); and, ensure releases of radioactive material comply with 10 CFR Part 20 (Reference 8-3) and are maintained ALARA.

A unique safety and remediation issue related to excavation and removal of the SWTP is the potential for site workers to be exposed to biological hazards in the form of raw sanitary waste. Personnel assigned to perform this work will receive specific training regarding the biological hazard and Environmental Health and Safety (EH&S) personnel will monitor potential exposures.

With respect to preventing the spread of contamination by the wind, the primary method that will be employed to prevent the spread of contamination during material handling will be the use of water mist. After application of water mist, temporary stockpiles (e.g., those that remain until the next workday) may also be tamped using the flat side of the excavator bucket or similar piece of heavy equipment to consolidate the surface of the material thus reducing the potential for

erosion. Additives may also be added with the water mist that form a thin crust-like layer, (e.g., a dilute non-hazardous adhesive), or those that posses hygroscopic properties to sustain the effectiveness of water application. (e.g., calcium chloride). To gauge the effectiveness of contamination control measures, the results of general area and breathing zone air samplers will be evaluated to identify outliers or trends in concentration that suggest appropriate actions be taken to mitigate airborne radioactivity. With respect preventing the spread of contamination by precipitation, see Section 8.6

8.5 CONTAMINATED SOIL

Figure 8-6 shows the site layout and general location of these remediation areas. The sections below detail the tasks and techniques for remediation of each of the surface and subsurface soil remediation areas.

8.5.1 EXCAVATION AND REMOVAL OF SOIL AND BURIED OBJECTS

This section describes techniques to remove or remediate surface and subsurface soil including ground-surface surveys, and subsequent excavation and removal of soil in approximately 1 foot cut depths, in accordance with the governing nuclear criticality safety assessment(s). Excavation and removal will typically be performed using heavy construction equipment.

Following is the general order of techniques to be employed in removing soil from designated areas:

- Soil will be evaluated using in-situ GWS, VOC monitoring (Photo-Ionization detector) and visual inspection of the exposed surface, repeated for each newly exposed surface. If elevated radioactivity measurements indicating amounts in excess of the NCS Exempt Material Limit are encountered prior to or during excavation, the detector response will be evaluated and the appropriate excavation depth determined. An analysis shall be performed that establishes the detector response that corresponds to the NCS Exempt Material Limit (defined in Section 8.5.2.1);

- Soil will be excavated and removed in nominal 1-ft lifts;

- Excavated surface and subsurface soil will be segregated, based on: visual inspection; radiological and chemical survey/screening; supplemental sampling and analysis; the appropriate DCGLs; Remediation Goals (RGs); or, the NCS-Exempt Material Limit for potentially fissile material. Excavated soil will be stockpiled at a safe distance adjacent to the excavation, or loaded into a haul truck for transfer to the WCA and further visual inspection; and,

- Soil overburden transferred to the WCA will be consolidated, inspected and loaded into a haul truck for evaluation at the HRGS. Material confirmed to be acceptable for re-use will be transferred to a lay-down area; material exceeding the re-use criteria will be disposed of as radioactive waste.

- When objects are encountered in the soil, excavation will generally be performed using heavy equipment; however, more precise methods and equipment may be used as deemed necessary. It is anticipated some hand-shoveling will be

necessary to preclude damage. When the object is an intact or damaged drum or other container, the following apply:

- Since identification and recovery of containers may introduce unknown hazards. Work Packages, AHAs and RWPs will specify additional controls required to assess and disposition intact drums and containers.

- Excavated drums or containers will be prepared and placed into an over-pack for subsequent transfer to the WEA, or the DOSA for transport to the WEA. Smaller containers will be transferred into Field Containers as soon as possible; and,

- Over-packed drums and Field Containers will be transferred to the Material Assay Area (MAA) for assay and evaluation. The results of the evaluation will determine the appropriate handling, storage and final disposition of drums or containers.

- The excavation process will include sloping and benching, as required, and will continue until visible wastes are removed, and in-process surveys and sampling activities indicate the applicable DCGLs and RGs have been met.

- Dust and erosion controls to be employed during remediation include non-hazardous surfactants and tarpaulins. The excavation will be isolated and maintained throughout FSS, and until restoration is authorized.

8.5.2 SORT AND SEGREGATE WASTES

8.5.2.1 Fissile Material Including Recoverable Material

Relative to removal of buried wastes, contaminated soils and sub-surface structures (e.g., concrete slabs, buried piping) in areas where it is determined that the presence of fissile materials is a reasonable possibility based on characterization data and historical knowledge, HDP has developed consistent generic screening and handling approaches which have been analyzed from a NCS perspective in NCSAs specific to buried waste exhumation and contaminated soil remediation, and sub-surface structure decommissioning. Screening for fissile materials during remediation is the initial goal. Screening typically will involve duplicate performance of radiological surveys, using sodium iodide scintillation detectors, of defined volumes of material to ensure that NCS limits have not been exceeded. The objective of the in-situ radiological surveys is to identify materials that do not satisfy the NCS exempt material criteria. The in-situ radiological surveys are complemented by visual inspection of the survey area with the aim of identifying:

1) Items with the potential to contain fissile material (e.g., a process filter);

2) Items that resemble intact containers;

3) Bulky objects with linear dimensions exceeding the permitted excavation '*cut depth'*; and,

4) Metallic items.

Unless otherwise defined and justified within a nuclear criticality safety evaluation, NCS Exempt Material is conservatively defined as material containing ^{235}U with an average nuclide fissile concentration not exceeding 0.1 g^{235}U/L, or material that comprises no greater than 15 g^{235}U and is enclosed within a container with a volume of at least 5 liters. Refer to Chapter 10.0 for further details on NCS and handling of fissile material.

8.5.2.2 VOC-Contaminated Soil

Four types of VOC-contaminated soil are expected to be encountered:

- Mixed waste, having a radioactive component above the appropriate DCGLs, and a component above the threshold defining Hazardous Waste under RCRA;
- Radioactive soil above the DCGL and above the RG, but below the threshold defining Hazardous Waste under RCRA;
- Non-radiologically contaminated Hazardous Waste soil above the threshold defining Hazardous Waste under RCRA; or
- Non-radiologically contaminated soil above the RG , but below the Hazardous Waste threshold.

If the VOC contaminants cannot be successfully stabilized by VOC treatment, the soil will be handled and packaged for off-site disposition. Soil below the RG, and above the appropriate DCGL, will be handled as LLRW.

If onsite treatment for VOC is conducted VOC treatment will be performed by *ex-situ* Soil Vapor Extraction (SVE). The SVE system will be designed, constructed and operated to meet applicable requirements of 40 CFR 262.34, 40 CFR 265.40 and 40 CFR 266.230 (Reference 8-10, Reference 8-11 and Reference 8-12, respectively) for tanks and tank systems.

Onsite VOC treatment is conducted by *ex-situ* SVE by which a vacuum is induced by a mechanical blower and the VOCs are stripped and volatilized into the air stream. The exhaust air is then treated to remove particulates and VOCs before it is emitted to the atmosphere. The

VOC treatment will be designed to comply with the air emission standards for process vents at 40 CFR 265.1032(a), requiring total VOC emissions of no more than 3 pounds per hour or emissions control of greater than 95 percent efficiency. Hazardous waste being accumulated into the units/tanks for treatment that is not rendered and confirmed to be non-hazardous will be removed from the site for disposal within 90 days. For units containing exempt LLMW, the accumulation and treatment time will be as short as necessary to achieve the required degree of treatment, but may extend longer than 90 days.

The majority of VOC treatment required during the remedial action will be treatment of VOC impacted soil to meet reuse or disposal criteria. Transfer and consolidation of VOC-impacted soils will occur throughout the remediation of the localized areas of VOC contamination. Piles will be covered during the consolidation phase.

Techniques for treatment and handling of VOC-contaminated soil exceeding the RG include:

- Stockpiling or direct loading soil into haul trucks for transfer to the WCA, and subsequent visual inspection at the WCA;
- Loading VOC-contaminated soil into a transfer truck for assay in the HRGS; and,
- Treatment of VOC-contaminated soil in one of two VOC treatment areas: the radioactive side (> DCGL); or, the 'clean' side (< DCGL). Upon verification that treated soil is below the RG and the backfill DCGL, transfer to a lay-down area for re-use. Upon verification that treated soil is below the RG, but remains classified as LLRW, transfer to the WHA for final disposition.

8.5.2.3 Low Level Radiological Waste (LLRW)

This section describes the techniques used to manage LLRW. Field screening will be performed to establish an initial material classification, followed by sampling and analysis to validate the field screening classification.

There are three primary soil staging areas; the Waste Consolidation Area (WCA) located at the edge of the burial pit location, the Waste Holding Area (WHA) near the railcar loading pad, and the Laydown Area located northeast of the central tract, as described below:

- Waste Consolidation Area and Waste Holding Area

  The construction of the WCA and WHA consists of compacted road base material at an average thickness of approximately six inches. Concrete jersey barriers are then placed to form "waste bins." The use of jersey barriers allow for material sorting and segregation, while maintaining the long-term integrity of the physical dividers. The WCA is comprised of four separate bins, allowing the segregation of multiple streams originating from the Burial Pit remediation. The WHA is

designed with at least two separate waste bins for stockpiling prior to railcar loading.

- Laydown Area

  The Laydown Area is a portion of the site property located northeast of the Burial Pit area, across the Northeast Site Creek, that has been deforested to create the area. The Laydown area is surrounded by a security fence which has gated access to the Controlled Access Area. The Laydown Area allows for the segregation of soil, that is suitable for re-use as backfill and for off-site borrow, by the use of solid barriers. Surface water barriers are placed at the edges of the portion of the Laydown Area to control sediment in stormwater runoff and to minimize soil erosion. The stormwater will be direct to Outfall #006.

The general sequence for excavation, removal and handling of LLRW will be as follows:

- Excavated LLRW will be loaded directly into haul trucks for transfer to the WCA, or stockpiled until a sufficient quantity is available for transport to the WCA, for final visual inspection and assay at the HRGS; and,
- LLRW will be sent to the WHA for stockpiling, loading, verification of compliance with waste acceptance criteria (WAC), and subsequent transportation to off-site disposal facilities. Transportation for off-site disposal will generally be by gondola cars; however, alternate conveyances which meet the requirements of the Waste Management and Transportation Plan (WMTP, Reference 8-9) may also be utilized.

8.5.2.4 Hazardous Waste

VOC-contaminated soil is addressed in Section 8.5.2.2. Other hazardous waste components will be segregated from other waste types.

- A hazardous waste treatment plan which meets RCRA requirements will be developed, if necessary, based on characterization results. Treatment options (see Chapter 12.0) effective in reducing hazardous waste characteristics include:

  1. Chemical neutralization for low- and high-pH corrosive waste;

  2. Use of stabilization agents to bind hazardous constituents, to meet Transportation Storage Disposal Facility (TSDF) waste acceptance criteria; and,

  3. Processing Hazardous Waste for transport to a permitted TSDF.

8.5.3 SPECIFIC ASPECTS

In addition to the general approach discussed in Sections 8.5.1 and 8.5.2 for soil remediation, specific aspects applicable to each of the areas of soil remediation are discussed in the following Sections.

8.5.3.1 Burial Pit Soils

Excavation and removal of Burial Pit soil will likely begin at the northwest corner and continue towards the east and south. Excavation and removal of soil will be performed in multiple burial pits concurrently. This approach provides sufficient space for heavy equipment to operate and maximizes the material available for re-use, without cross-contamination.

The majority of materials buried in the Burial Pits are anticipated to be contaminated soil and trash; some laden with VOCs, floor tiles, glass wool and laboratory glassware. Minor components of the buried waste volume are anticipated to include: acid-insoluble residue; filters; metallic debris; and, metallic oxides. These buried materials may result in multiple waste streams requiring specific management strategies.

Historical records for the Burial Pit Area suggest that regulated asbestos-containing material (RACM) may be present within the sub-surface soil. The excavation and removal of potentially RACM will be performed in accordance with Asbestos NESHAP (40 CFR 61, Subpart M) and Missouri Solid Waste Management requirements.

Excavation and removal will continue until all buried wastes have been removed, based on: visual inspection and survey; NCS criteria; FSS criteria; and RG are satisfied.

8.5.3.2 Soils Southeast of the Process Buildings and Surrounding Areas

8.5.3.2.1 Evaporation Ponds

Diversions and berms will be installed to isolate the area from water run-on and run-off and to minimize standing water until excavation is performed. Remediation of the evaporation pond area will require pumping the water from the ponds, and treatment and/or sampling the water prior to discharge. The water removed from the evaporation ponds will be collected and processed according to the WMP (Reference 8-13). These activities will be followed by excavation and removal of contaminated sediment, limestone, and adjacent soil.

- Dense non-aqueous phase liquids (DNAPLs) may be encountered during excavation, and will be collected and packaged for off-site treatment and disposal. DNAPL handling tasks will be performed according to specific work instructions, and a task-specific RWP and AHA, as applicable, and

- The location of the natural gas pipe line running along the south side of the property line will be verified, and its impact on remediation of the evaporation ponds will be evaluated.

  The natural gas pipe line, existing railroad line running along the south side of the site property line, and project-installed rail spur will remain in place following license termination. In the event that work activity is required within the natural gas company easement area, the natural gas company will be contacted prior to work to ensure proper safety precautions are employed. If excavation within 5 feet of the natural gas pipe line is necessary, it will be performed in accordance with the requirements of 29 CFR 1926.651, Special Excavation Requirements (Reference 8-14). Shoring or similar protective systems will be used, as necessary.

  In the event of a mishap, the remediation of soil within close proximity to the gas line could present significant hazards to workers and the potential for disrupting local utility service. The degree of this risk is dependent upon the depth and amount of soil that will require removal to meet the DCGL. If it is determined at the time of the work that additional excavation may introduce an unacceptable risk to the workers, the environment, or the public, Westinghouse may propose that an independent dose assessment be considered as a basis for achieving the remedial goal in this area.

8.5.3.2.2 Former Leach Field

Remediation of the Former Leach Field will include decontamination and processing the concrete slab and asphalt, followed by excavation and removal of contaminated soil and piping associated with the abandoned system.

Residual contamination surveys, cleaning and decontamination will be performed in the same manner as described in Section 8.3.1 for concrete slabs and paved areas. The concrete and asphalt covering the Former Leach Field will be managed to meet the release criteria for off-site disposal and recycle, or disposed of as radioactive waste.

8.5.3.3 Soils Beneath and Surrounding the Barns, Cistern Burn Pit and Red Room Roof Burial Area

Historical data applicable to the Red Room Roof Burial Area suggests RACM may be present in surface and sub-surface soil. The removal and excavation of potential RACM will be performed in accordance with Asbestos NESHAP (40 CFR 61, Subpart M) and Missouri Solid Waste Management requirements. A State-of-Missouri-licensed Asbestos Abatement Contractor will

perform supervision, removal and packaging of suspected RACM, and provide characterization data.

Residual radiological contamination exceeding the DCGLs has been identified within the footprint of the Wood Barn and in the Cistern Burn Pit area. Remediation for these areas will involve removal of concrete and paved surfaces (Wood Barn), followed by excavation and removal of soil and debris (Wood Barn and Cistern Burn Pit area). Residual contamination surveys, cleaning and decontamination will be performed in the same manner as described in Section 8.3.1 for concrete slabs and paved areas. The concrete and asphalt covering the paved surfaces will be managed to meet the release criteria for off-site disposal and recycle, or disposed of as radioactive waste.

8.5.3.4 Site Pond, Site Creek, and Surrounding Soils and Sediment

Remediation of the Site Pond and Surrounding Area will require draining and diverting inflow to the Site Pond, followed by excavation and removal of sediments and soil:

- The non-impacted spring and surface water originating from the north side of State Road P and then entering the Site Pond will be diverted around the Site Pond and the portion of the Site Creek below the dam that is to be remediated. This diversion will discharge sufficiently downstream of the remediation area within the Site Creek to avoid interference with remediation.

- The current location of Outfall #001 is along the eastern bank of the Site Creek, immediately downstream of the Site Pond Dam. Outfall #001 effluent consists of waste water from the Sanitary Treatment System. There is a composite sampler at the location of Outfall #001. While the Site Creek is undergoing remediation, the discharge at the location of Outfall #001 will be diverted sufficiently downstream of the Site Creek remediation area to avoid interference with remediation. The diverted discharge at the location of Outfall #001 will continue to be sampled by a composite sampler.

- The current location of Outfall #003 is along the eastern bank of the Site Pond. Note that Outfall #3 does not discharge directly to the environment. Rather, Outfall #3 discharges to the Site Pond, and is included in the effluent measurement obtained by the composite sampler at Outfall #002 (Site Dam). Outfall #003 effluent consists of effluent from the WTS, stormwater from the parking lot, part of the footprint of the former processing building, barn area, and building roof drains. While the Site Pond is undergoing remediation, discharge at Outfall #003 will be diverted either to the location of Outfall #001 or sufficiently downstream of the Site Creek remediation area to avoid interference with remediation, depending upon the timing of Site Creek remediation. The diverted

Outfall #003 discharge will be sampled by a composite sampler during the diversion.

- Once the inflow-bypass system is operational and diverting the inflow of the Site Spring and Outfall #003, the Site Pond will be drained. Site Pond water will be drawn down in accordance with site discharge permits, sampled, and processed as necessary for discharge under the Water Management Plan (WMP, Reference 8-13).

- Site Pond sediments will be allowed to dry prior to performing remediation of the Site Pond and Surrounding Area. Excavation and removal of sediments and soil to meet the appropriate DCGLs and RGs follows, and will be managed according to the WMTP.

- Restoration will be accomplished with placement of material that meets regulatory criteria for use as backfill, followed by grading. Compacting will be to a standard proctor, and grading will be performed to achieve pre-remediation contours, to the maximum extent practical. Upon completion of grading, the water inflow-bypass will be abandoned.

- Remediation of the concrete dam will involve decontamination or removal, to meet the appropriate DCGLs and RGs. If necessary, a new dam will be constructed during site restoration, and prior to removal of the water inflow-bypass.

8.5.3.5 Soil Beneath and Surrounding the Process Buildings

- The WCA and VOCTA will be temporarily relocated to an open area adjacent to the WEA, or similar location, to allow for excavation and removal of building slabs, foundations and underlying soil; and

- DNAPLs encountered during excavation will be removed and packaged for off-site treatment and disposal.

8.5.4 RADIATION PROTECTION AND SAFETY CONTROLS

Radiation protection methods and control procedures applicable to excavation and open land areas are described in Chapter 10.0 and Section 8.1.4.

Unique safety and remediation issues associated with remediation of surface and subsurface contaminated soil are associated with excavation, removal and handling of soil and wastes from each remediation area. Following is a summary of the unique safety and remediation issues, including identified hazards and control measures:

- Excavation and removal of subsurface contaminated soil, waste and debris may involve the recovery, handling, containerizing, staging, evaluating, characterizing, packaging and transferring of fissile material. This unique issue is relevant to buried waste in the Burial Pit Area, the Red Room Roof Burial Area and the Cistern Burn Pit Area. Section 8.1.2 and Chapter 10.0 describe handling SNM during buried waste exhumation and contaminated soil remediation activities. The nuclear criticality safety assessment provides requirements for managing soil/waste that warrant nuclear criticality safety controls. As appropriate, those controls have been reflected throughout Section 8.5;

- Excavation and removal of soil introduces the possibility of cave-ins, depending on prevailing weather conditions, soil type, depth and width of the excavation. In general, excavations will be benched in 4-foot increments of depth below ground surface, while ensuring compliance with 29 CFR 1926 Subpart P, Appendix B, "Sloping and Benching," (Reference 8-15). An AHA will be prepared to address issues such as the design of ingress/egress, controlling the excavation to protect against falling loads, and performing daily inspections of the excavation, adjacent structures and protective systems. These safety measures will be adhered to as long as an excavation remains open; and,

- Uranium metal has pyrophoric properties and exposure of Uranium metal shavings to air has the potential to generate a localized fire. Two documented events indicate fires in the Burial Pits may have resulted from the spontaneous combustion of Uranium. Uranium metal in the Burial Pits is expected to have an oxidation layer, due to exposure to moisture in soil and groundwater, which mitigates the potential for a fire; however, intact containers exhumed from the Burial Pits could potentially contain un-oxidized Uranium metal. Therefore, intact containers exhumed from the Burial Pit Area will be sorted and evaluated in the WEA. Due to the potential for spontaneous combustion, a Class D fire extinguisher will be staged in the WEA as a precaution.

8.6 SURFACE WATER AND GROUNDWATER

The Hematite DP addresses management and controls for surface water and groundwater, to be utilized during remediation tasks. DP Chapter 4.0 provides a discussion regarding the interpretation of the surface water and groundwater data, and the basis for the determination that surface and groundwater require no remediation.

Surface water sources include:

- Storm and surface water run-on and run-off for active excavations;
- Storm water collection in active excavations and remediation areas;
- Draining of the evaporation ponds and the Site Pond;
- Wastewater associated with dust suppression;
- Wastewater associated with decontamination of concrete surfaces, foundations and paved areas; and,
- Wastewater associated with decontamination equipment used in remediation.
- Groundwater sources, incidental to site remediation activities that enter active excavations.

Sources of surface water and groundwater, incidental to site remediation activities, will be managed according to the WMP (Reference 8-13) and storm water pollution prevention measures within work packages, procedures, or other site documents. Incidental surface water and groundwater that is potentially contaminated will be handled according to the WMP. Sources of water that are not potentially contaminated or have been diverted around active excavations will be handled in accordance with the Storm Water Pollution Prevention Plan (SWPPP).

The perimeter of contaminated areas will be sloped inward or curbed to contain and direct potentially contaminated surface water to sumps or other low-lying areas that will be used as collection points within the contaminated area. All impacted water encountered during the remedial actions will be sampled and discharged providing effluent release criteria are met with due consideration of maintaining release concentrations ALARA, or will be collected and processed through the Water Treatment System, as appropriate.

To reduce the amount of surface water run-on into contaminated areas (which would create additional water requiring collection and processing), diversion features (e.g., curbs) will be constructed at up gradient locations to direct precipitation around contaminated areas.

Specifically, berms, either soil or other impermeable material, will be installed along the eastern side of the waste handling area to control suspended solids associated with sheet flow in this direction. The berm will be used to direct surface water flow to the subsurface drain at the loading pad. The subsurface drain discharges to the former evaporation pond, which has been lined with impermeable material and now serves as a collection sump. The subsurface drain was designed to handle a nominal 10 year rainfall event. If a larger rainfall event occurs during the remediation of the site, the area surrounding the subsurface drain would have minor ponding, but be limited to the impacted area of the site. This area is planned to be excavated near the end of the remediation work so remediated areas are not re-impacted by potentially contaminated surface water. In addition, the drain design will be re-evaluated if precipitation overwhelms the drain capacity three times during remediation work.

Storm sewer inlets in the areas of active remediation will be protected with a combination of straw bales and/or silt fencing. Storm sewer inlets may also be blocked with impermeable material (e.g., neoprene/EDPM and ballast material). Water collected will be pumped to the WTS, or discharged through a permitted outfall depending on the results of sample analysis. Batch sampling or composite sampling of the effluent will be performed.

8.6.1 POTENTIALLY CONTAMINATED WATER

The following tasks summarize the process and control measures for handling potentially contaminated water associated with remediation of areas with residual contamination greater than the DCGLs, RGs, the requirements of SNM-33 (Reference 8-7), or the MDNR NPDES Permit (Reference 8-8).

- Physical barriers such as berms, slopes for excavations, temporary drainage ditches and silt fences will be used to manage and control water within remediation areas. Inspection of control measures will be performed periodically to ensure operational and physical integrity;

- Remediation areas will be graded such that water will drain to designated capture points and sumps. Collected water will be pumped into settling and holding tanks;

- Water collected in settling and holding tanks will be treated and/or sampled and processed and the analytical results compared to the liquid effluent discharge limits. If discharge limits are exceeded, water may be re-processed through the WTS;

- Collected and/or processed water will be discharged in accordance with Chapter 11.0 of this DP, implementing procedures, and the NPDES Permit (Reference 8-8); including the Applicable or Relevant and Appropriate Requirements (ARARs) waiver (Reference 8-16); and,

- Sediments and solids collected within excavations, settling and holding tanks, the WTS, and filtration media will be assessed for radiological characteristics prior to recovery and placement in field containers or consolidation with similar waste streams. Wastes associated with water management and control measures as well as any leachate present in the VOCTA will be managed according to Chapter 12.0, the WMP and nuclear criticality safety assessment controls.

Table 8-4 provides a summary of the tasks and techniques for water management for each remediation area.

8.6.2 UNCONTAMINATED WATER

Handling uncontaminated surface and subsurface water will be conducted in accordance with construction Best Management Practices (BMPs), throughout the decommissioning process. The most appropriate BMPs will be implemented, including: land contouring and grading; sediment traps/basins; surface water diversions; and/or flow-bypass systems.

The primary method for managing uncontaminated water around excavations is diversion prior to the water contacting the excavations. Diversion techniques will be utilized in the following remediation areas: Burial Pit Area, Leach Field, Wood Barn, Red Room Roof Burial Area, Cistern Burn Pit Area, Slabs, Soil Beneath Slabs, the Soil Remediation Area, and project support areas such as waste handling and loading areas.

Sediment traps, soil berms and basins will be deployed as necessary when remediating the SWTP and SDS.

Water diversion systems, consisting of a headwall and flexible pipe or lined trench, will be installed to divert surface water flowing through the Northeast Site Creek and the Site Pond. The natural grade of diversions may be enhanced to ensure gravity flow of diverted water to designated water collection areas.

8.6.3 RADIATION PROTECTION AND SAFETY CONTROLS

Radiation protection methods and control procedures applicable to excavation and open land areas are described in Chapter 10.0 and Section 8.1.4.

The primary collection points for radiological contamination will be within filter media in WTS components and sediment or sludge in water holding tanks. These WTS components and holding tanks will be radiologically surveyed or their contents sampled and analyzed prior to removal of media, sediment or sludge from the WTS components and the holding tanks.

Unique safety and remediation issues associated with management of potentially contaminated surface and subsurface water include:

- The potential exists for a storm event to overwhelm active earthen sumps during remediation. Weather conditions will be monitored during remediation and additional pumps and/or holding tanks will be on-site and added, if necessary, to manage water from storm events;

- The potential exists for accumulation of fissile material in the Burial Pit Area or Leach Field earthen sumps. These extracted materials will be managed in accordance with the controls stated in Chapter 10.0 and the WMTP;

- The potential exists for asbestos-containing materials (ACM) in sediments and sludge from the Burial Pit Area, Red Room Roof Burial Area and Wood Barn soil remediation area. Sampling will be performed prior to processing water through the WTS; if asbestos is verified as present, then filtration will be added to the WTS to assure treated and/or discharged water meets the requirements of SNM-33 (Reference 8-7) and the NPDES Permit (Reference 8-8);

- The potential exists for the presence of sanitary waste during Leach Field and SWTP remediation. Additional precautions, such as supplemental pre-filters for the pump heads, will be considered to mitigate the potential of sanitary waste entering the water treatment stream; and,

- The potential exists for the need to make confined space tank entries per existing site procedures governing confined space entries.

Unique safety and remediation issues associated with the management of uncontaminated surface and subsurface water include:

- Water diversion systems, consisting of a headwall and flexible pipe or lined trench, will be installed to divert surface waters flowing through the Northeast Site Creek and the Site Pond; and,

- The potential exits for a storm event to overwhelm active earthen sumps, dikes, trenches or sediment basins during remediation. Weather conditions will be monitored during remediation, and water that enters an active excavation will be

managed in accordance with potentially contaminated water techniques and control measures.

8.7 FINAL STATUS SURVEY (FSS)

Upon completion of the RASS and confirmation that the applicable DCGLs and RGs have been met, excavations will be prepared for FSS as described in Chapter 14.0 of this plan.

Open excavations will be maintained throughout the FSS process until restoration is authorized. Depending on the season and prevailing weather, excavations may be covered with tarps to preserve the surface and limit erosion or the potential for generation of dust.

8.8 SITE RESTORATION

In general, site restoration will include backfilling excavated areas, compacting to a standard proctor, spreading topsoil, reseeding and removing temporary features that impede final site restoration. Excavated soil determined to be below the appropriate DCGLs, and meeting other regulatory requirements for re-use, will be used as backfill material. Additional off-site backfill material will be imported from an approved off-site source(s), as needed, and tested to ensure it meets site cover requirements for radiological and chemical constituents.

Excavations will be restored using backfill obtained from on-site material determined to be suitable for re-use (e.g., excavated soil overburden), and/or backfill material from a pre-approved source.

Compacting will be to a standard proctor, and grading will be performed to achieve pre-remediation contours, to the maximum extent practical. Adjustments will be made to the grade to mitigate the potential for surface water to pool over the remediated site. Reseeding of backfilled areas will be performed with a MDNR approved seed mixture, to limit the potential for erosion. Site restoration activities will be performed iteratively, i.e., the restoration of one excavation area may begin while other excavation areas are undergoing remediation activities.

Topsoil will be placed above backfill material in areas to be seeded, and will be cultivated and graded to ensure a smooth, uniform grade with positive drainage towards wetland areas. Winter rye seed and/or other MDNR approved cover will be utilized.

8.8.1 DEMOBILIZATION

Temporary fixtures, trailers, generators, fuel tanks, supplies and equipment will be surveyed, decontaminated as necessary for unconditional release, and removed from the site.

Techniques for decontamination include dry methods, such as using impregnated fabric wipes or dry wipes; or wet decontamination using a high-pressure wash, with or without surfactant, as appropriate. Surfaces will be allowed to dry before final release surveys are performed on equipment and material decontaminated by wet methods.

Additional tasks that may be performed during demobilization include:

- Removing temporary haul roads, staging areas and gravel footing; and,
- Grading to pre-remediation contours, to the maximum extent practical; with adjustments made to the grade, to mitigate the potential for surface water to pool over the remediated site.

8.9 SCHEDULES

The identified activities provided in the Gantt chart schedule (Figure 8-2) are intended to provide an overview of the remaining activities and an estimated time schedule for each. The time frames for conducting the activities are dependent upon NRC approval of this Decommissioning Plan. However, project circumstances may change during decommissioning. An updated schedule will be maintained during decommissioning and will be available for review by regulatory agencies.

REFERENCES FOR CHAPTER 8.0

8-1 Code of Federal Regulations, Title 10, Part 70, "Domestic Licensing of Special Nuclear Material."

8-2 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated Decommissioning Guidance, Decommissioning Process for Materials Licensees," Volume 1, Revision 2.

8-3 Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

8-4 Westinghouse Electric Company Document, "Physical Security Plan" (PSP), Dated July 28, 2011.

8-5 Code of Federal Regulations, Title 10, Part 73, "Physical Protection of Plants and Materials."

8-6 Westinghouse Electric Company Document No. HDP-PO-EHS-001, "Health and Safety Plan."

8-7 U.S. Nuclear Regulatory Commission, License No. SNM-33 (Docket No. 70-36).

8-8 Missouri Department of Natural Resources, Water Protection Program, Water Pollution Branch, NPDES Permit Number MO-0000761, February 24, 2006.

8-9 Westinghouse Electric Company Document No. HDP-PO-WM-900, "Waste Management and Transportation Plan" (WMTP).

8-10 Code of Federal Regulations, Title 40, Part 262.34 "Accumulation time."

8-11 Code of Federal Regulations, Title 40, Part 265, "Interim Status Standards for Owners, and Operators of Hazardous Waste Treatment Storage and Disposal Facilities."

8-12 Code of Federal Regulations, Title 40, Part 266.230 "What conditions must you meet for your LLMW to qualify for and maintain a storage and treatment exemption?"

8-13 Westinghouse Electric Company Document No. HDP-PO-EM-004, "Water Management Plan."

8-14 Code of Federal Regulations, Title 29, Part 1926.651, "Specific Excavation Requirements."

8-15 Code of Federal Regulations, Title 29, Part 1926, Subpart P "Sloping and Benching," Appendix B.

8-16 Missouri Department of Natural Resources, "MO-ARAR013, *Applicable or Relevant and Appropriate Requirements (ARARs) Discharges to Waters and Groundwater of the State in Section 19, R5E, Jefferson County, MO*," November 7, 2003.

Table 8-1

HDP Plans

Title
Effluent and Environmental Monitoring Plan (EEMP)
Final Status Survey Plan (FSSP)
Radiation Protection Plan (RPP)
Fundamental Nuclear Material Control Plan (FNMCP)
Waste Management and Transportation Plan (WMTP)
Water Management Plan (WMP)

These plan documents are supported by Hematite procedures and contractor generated documents.

Table 8-2

Hematite Structures

Building Number	Description	Previous Use	Radiological Contamination
101	Tile Barn*	Emergency operations center; clean and contaminated equipment storage	Yes
110	Office and Security	Security, conference rooms, offices	Yes
115	Fire Pump House	Generator and fire pump	No
120	Wood Barn*	Clean and contaminated equipment storage	Yes
230	Rod Loading*	Fuel rod loading plant	Yes
231	Warehouse	Shipping container storage	No
235	West Storage Building	Storage Building	Yes
240	Process Building*	Uranium recycle and recovery, laboratories, maintenance shop, laundry, and ventilation room	Yes
245	Well House*	Potable water well and chlorination	Yes

Table 8–2 (continued)

Hematite Structures

Building Number	Description	Previous Use	Radiological Contamination
252	South Storage Building*	Storage Building	Yes
253	Process Building*	Uranium storage and processing, site utilities, office space, and decontamination facility	Yes
254	Process Building*	Uranium fuel pellet processing	Yes
255	Pellet Plant*	Erbia fuel pellet processing	Yes
256	Process Warehouse*	Warehouse space and fuel pellet drying	Yes
260	Oxide Building*	Conversion of UF6 gas into Uranium Oxide (UF6 enrichment 5 percent or less)	Yes

*Dismantled down to slabs and foundations.

Table 8-3

Structures to be Demolished

Concrete Slabs and Foundations
Process Buildings
Building 240 Building 253 Building 254 Building 255 Building 256 Building 260 Building 261
Non-Process Buildings
Building 101 Building 120 Building 115* Building 235* Building 245 Building 252
Concrete and Asphalt Paved Areas
Loading Pad - South of Building 230 (Leach Field) Sanitary Wastewater Treatment Plant - Unit Pad* Section of Asphalt Parking Lot - SDS Outfall 3 Concrete and Asphalt Pad - East of Buildings: 255, 260, 256 * May remain

Table 8-4

Remediation Areas, Sources, Tasks And Techniques For Water Management

Remediation Area and Source	Tasks and Techniques
Burial Pit Soils • Storm and surface water diversion from excavation • Storm, surface, and groundwater within excavation • Dust suppression • RACM	• BMP – sand bags, berms, silt fences, ditches to divert surface and storm water around the excavation area toward outfalls, i.e., Implement SWPPP controls • Grade excavation to low point for collection, pump and transfer to holding tanks, i.e., Implement WMP controls • Sampling for direct discharge or WTS treatment prior to permitted release, i.e., Implement WMP controls • Prefilter reuse water if used for dust suppression • Modify WTS to meet the requirements of the NPDES issued by the MDNR, Permit No. MO-0000761 for RACM, as necessary (Reference 8-8)
Former Leach Field • Same sources as Burial Pit Area • Residual sanitary wastewater from former leach field	• Implement SWPPP controls • Implement WMP controls • Prefilter collected water to minimize potential for residual sanitary wastewater
Red Room Roof Burial Area, Cistern Burn Pit Area, and Wood Barn • Same sources as Burial Pit Area, with no groundwater • RACM from Red Room Roof Burial Area excavation	• Implement SWPPP controls • Implement WMP controls • Modify WTS to meet the requirements of the NPDES issued by the MDNR, Permit No. MO-0000761 for RACM, as necessary (Reference 8-8)

Table 8–4 (continued)

Remediation Areas, Sources, Tasks And Techniques For Water Management

Remediation Area and Source	Tasks and Techniques
Site Pond and Surrounding Area, and Evaporation Ponds • Same sources as Burial Pit Area, no groundwater for Site Pond • Water from ponds, Site Creek, and Outfall #3	• Implement SWPPP controls • Implement WMP controls • Drain ponds, transfer water to holding and settling tanks, sample, direct discharge or WTS treatment prior to permitted discharge • Install water diversion from Site Creek and Outfall #3. Application of non-hazardous surfactants to minimize suspension of drying pond sediments
Slabs And Soil Beneath Slabs • Same sources as Burial Pit Area, minimal groundwater • Waste water from decontamination of concrete and paved areas	• Implement SWPPP controls • Implement WMP controls • Supplemental berms and collection sumps around concrete and paved surfaces during decontamination • Plugging drains, sumps, and other utility access points on concrete slabs, foundations, and paved surfaces
SDS and SWTP • Same sources as Burial Pit Area	• Implement SWPPP controls • Implement WMP controls • Prefilter collected water from excavation & removal of the SWTP to minimize potential for residual sanitary wastewater

Table 8–4 (continued)

Remediation Areas, Sources, Tasks And Techniques For Water Management

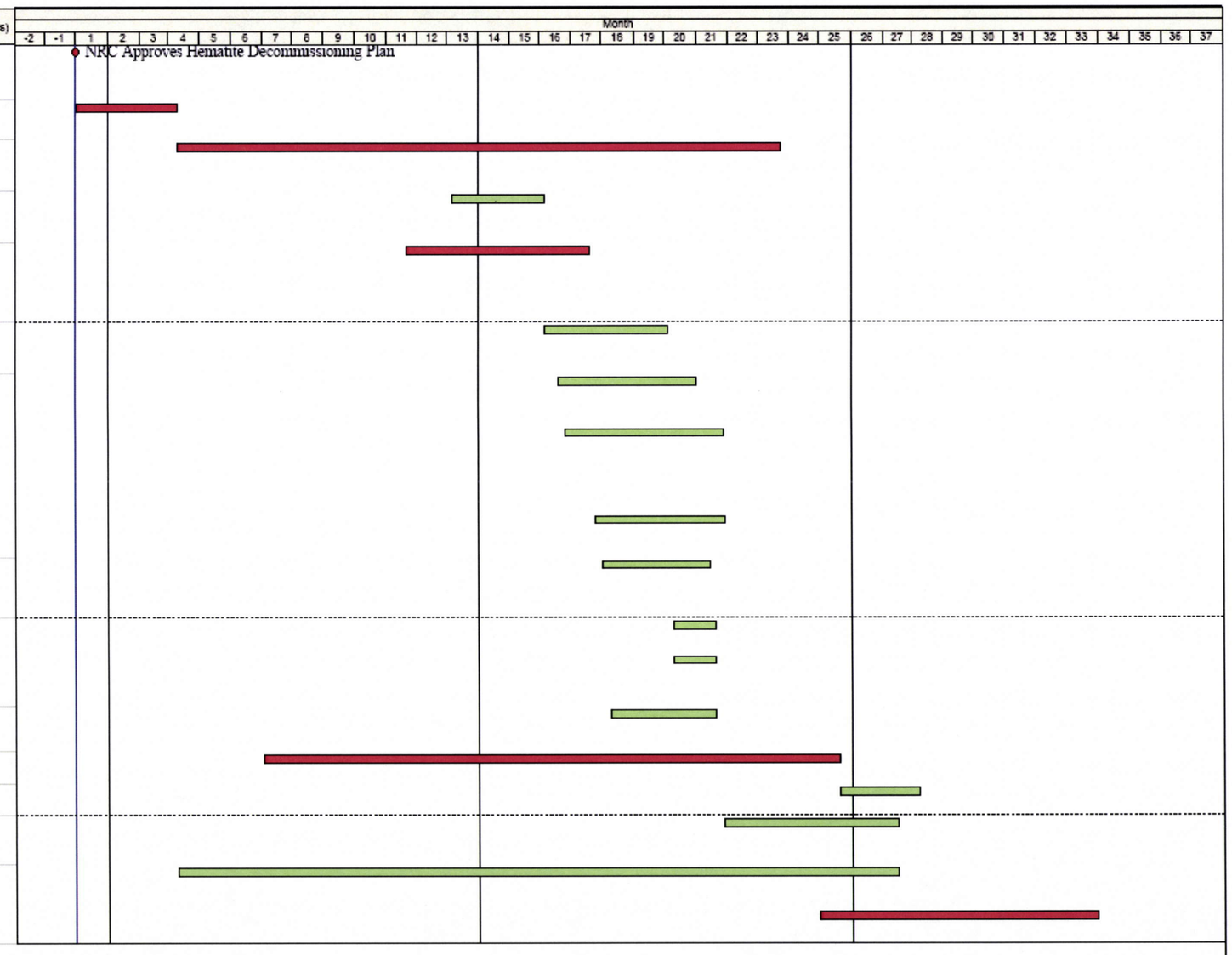
Remediation Area and Source	Tasks and Techniques
Soil Remediation Areas • Same sources as Burial Pit Area, with no groundwater intrusion	• Implement SWPPP controls • Implement WMP controls
Laydown Area • Storm and surface water	• BMP – silt fence, straw bales, waddles, berms, or ditches to divert surface and storm water towards outfalls, i.e., Implement SWPPP controls.
Waste Staging Area • Storm and surface water	• BMP – sand bags, berms, silt fences, or ditches to divert surface and storm water towards subsurface drain for water collection, sample analysis, and treatment as necessary.

Figure 8-1

Major Structures At Hematite

Figure 8-2
Proposed Schedule For Hematite Decommissioning

Activity ID	Activity Name	Duration (Work Days)
A1000	NRC Approves Hematite Decommissioning Plan .	0
A1010	Remediation Site Preparation	70
A1020	Burial Pits Remediation & Restoration	423
A1030	Former Leach Field Remediation & Restoration	65
A1040	Red Room Roof, Cistern Burn Pit & Barns Soil Remediation & Restoration	130
A1050	Site Pond Surrounding Areas Remediation & Restoration	87
A1060	Evaporation Ponds Remediation & Restoration	97
A1070	Slabs & Soils Beneath Buildings to be Demolished & Areas Around Building Exteriors, Remediation	110
A1080	Storm Drain System Remediation	92
A1090	Sanitary Wastewater Treatment System Remediation	76
A1100	Soil Remediation Areas	30
A1110	Remediation & Release of Remaining Structures	30
A1120	Demolition of other Structures	74
A1130	Final Status Surveys	405
A1140	Final Site Restoration	55
A1150	Decontamination & Demobilization	123
A1170	Water Treatment System	505
A1180	License Termination	196



Remaining Work
Milestone
Critical Remaining Work

Hematite Decommissioning Project

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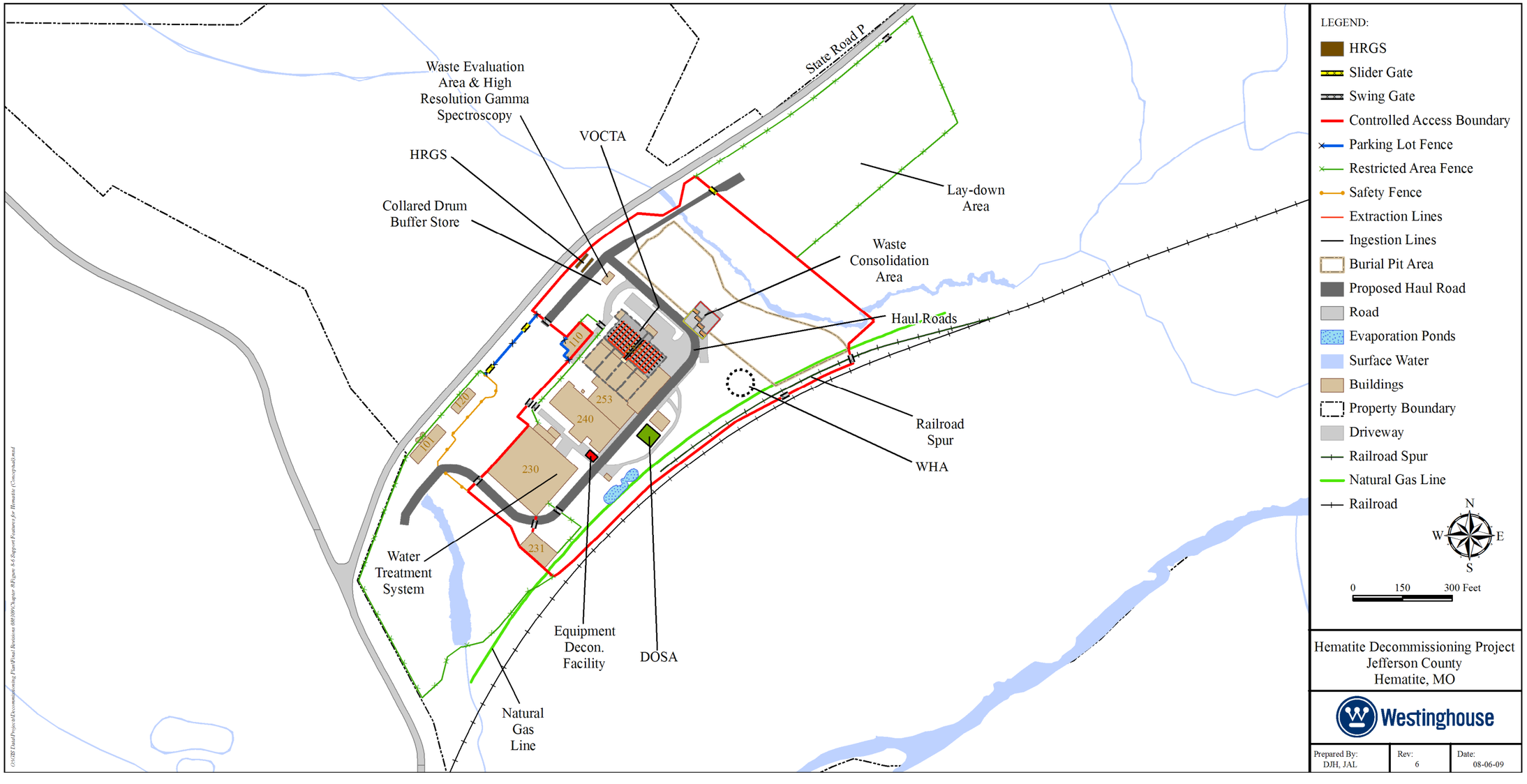
Figure 8-3

Support Features For Hematite (Conceptual)



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Building Demolition Scope For Structures At Hematite

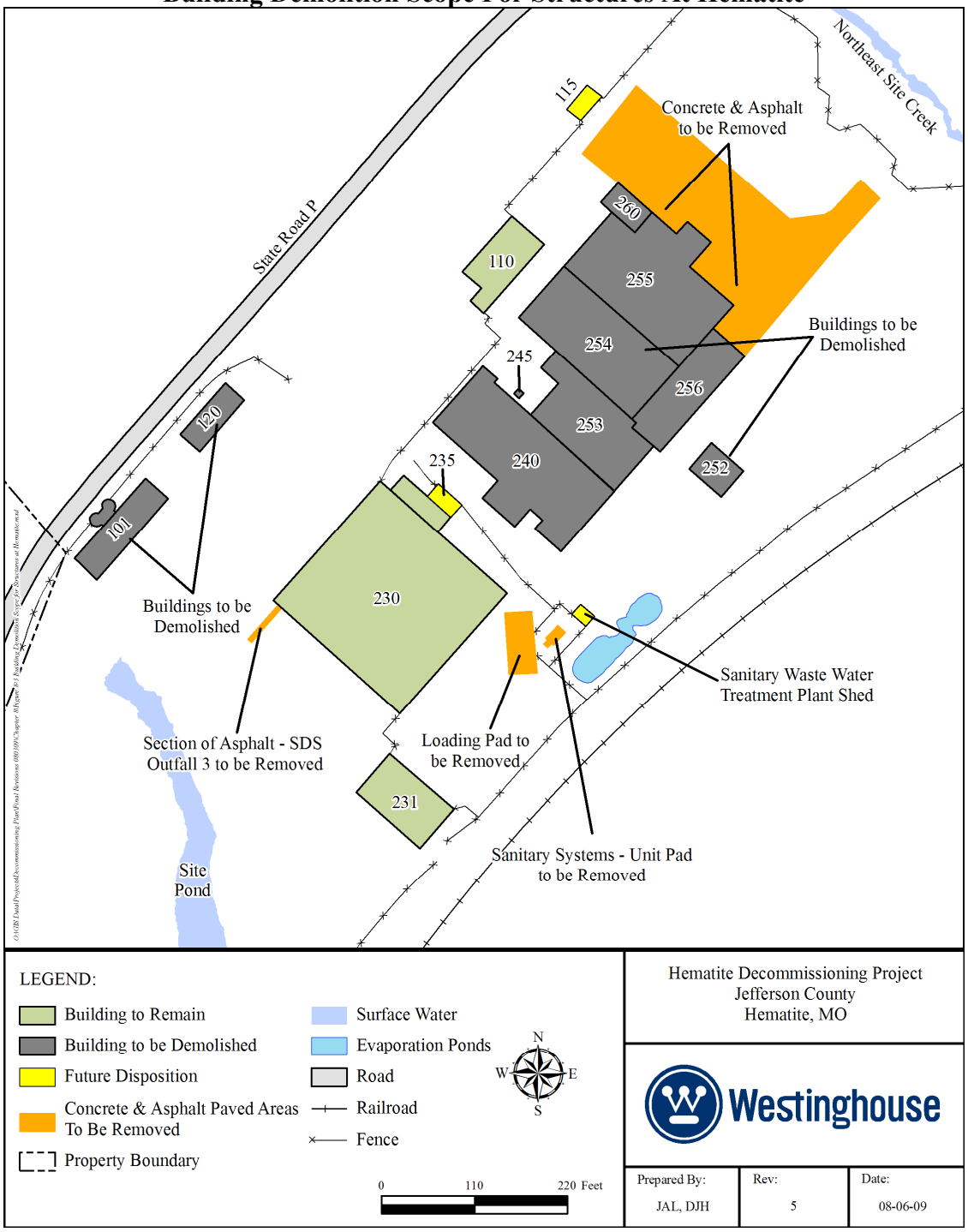


Figure 8-5

Sanitary Wastewater Treatment Plant And Storm Drain System For Hematite

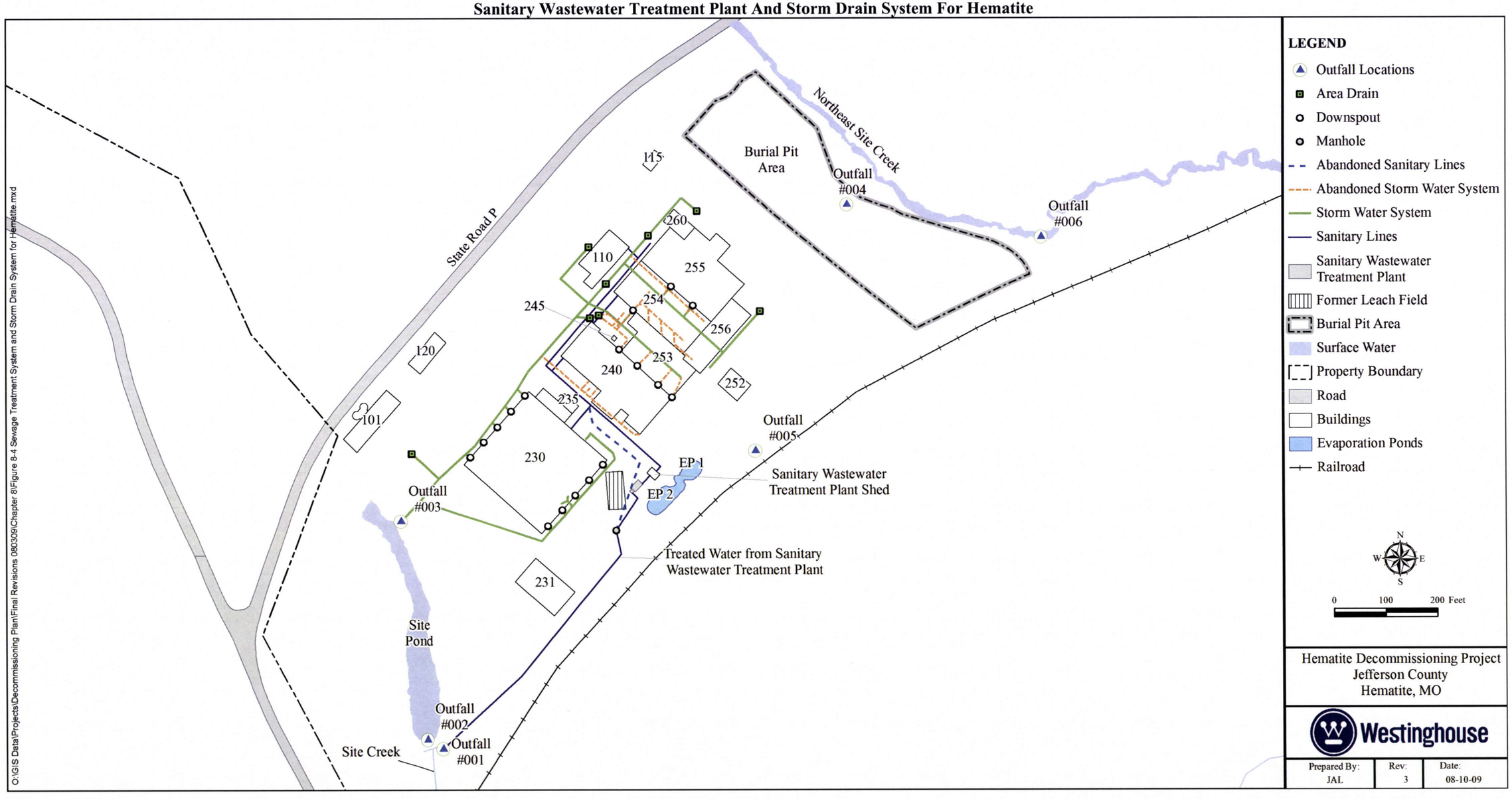
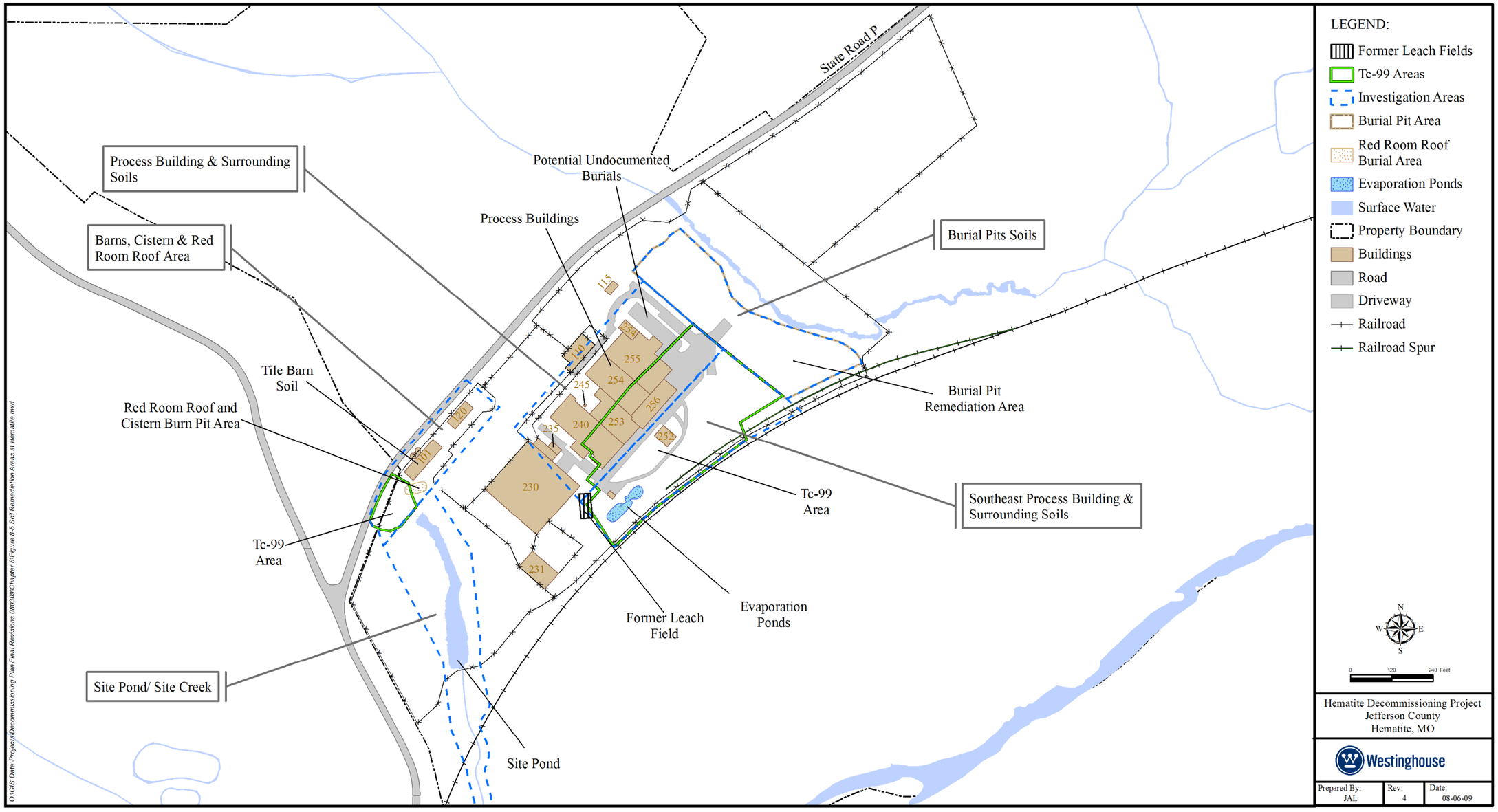


Figure 8-6

Soil Remediation Areas At Hematite



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ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
DOT	U.S. Department Of Transportation
EH&S	Environmental Health And Safety
GERT	General Employee Radiation Training
HDP	Hematite Decommissioning Project
MC&A	Material Control And Accounting
NRC	U. S. Nuclear Regulatory Commission
POC	Project Oversight Committee
QA	Quality Assurance
RWT	Radiation Worker Training
RSO	Radiation Safety Officer
RWP	Radiation Work Permit
SNM	Special Nuclear Material

9.0 PROJECT MANAGEMENT AND ORGANIZATION

9.1 DECOMMISSIONING MANAGEMENT ORGANIZATION

9.1.1 HEMATITE DECOMMISSIONING PROJECT (HDP) ORGANIZATION

The Westinghouse Electric Company LLC (Westinghouse) Hematite Decommissioning Project (HDP) organization is led by the Project Director and a staff of functional area managers. A description of each functional area responsibilities and authorities are provided in Table 9-1 of this chapter.

As the decommissioning of the site progresses, it is expected that changes to the organization will be made by the HDP Project Director. The Project Director will assign or re-assign responsibilities as necessary to ensure safety and compliance during execution of the HDP. An individual manager may be responsible for more than one management area. An individual manager, as is appropriate, may assign a designee to fulfill specified functions in the manager's absence or when necessary to support decommissioning activities. Changes to the organization will be in accordance with the requirements of the site License SNM-33 (Reference 9-1). An example of the HDP functional organization is provided as Figure 9-1 of this chapter.

9.1.2 PROJECT SAFETY (STOP WORK)

In addition to the stated responsibilities of site personnel to perform decommissioning activities safely and in accordance with site procedures, when anyone at HDP identifies a potentially unsafe condition, an imminent danger, a procedure step that cannot be performed as specified, or a condition that is not compliant with applicable requirements, they have the authority to stop work. The authority granted to anyone at HDP to stop work provides an approach that helps ensure decommissioning activities are conducted in a safe manner.

The RSO has access to all areas of the facility to effectively oversee the implementation of the radiation protection program. The RSO has specific "Stop Work" authority to ensure that work activities are conducted in accordance with the radiation safety requirements contained in work plans and procedures. For Stop Work events initiated by the RSO, the RSO shall also approve the proposed resolution and work restart with concurrence from the Project Director.

After a person declares Stop Work, personnel in the area immediately put their work in a safe condition and stop work. The individual declaring the Stop Work informs the Supervisor or Manager in the Department that has overall lead for the work (typically Operations or Project Engineering Departments). That Supervisor or Manager, informs the Project Director and the Manager of the appropriate safety discipline (e.g., EH&S, NCS, radiation safety) of the Stop Work. The responsible HDP Manager shall:

- Initiate a review into the basis for the Stop Work,

- Determine how to resolve the issue so work may be safely resumed in compliance with applicable requirements,
- Confirm with the individual who declared the Stop Work, that the proposed resolution resolves the safety concern, and
- Obtain approval from the affected safety discipline manager(s) for the proposed resolution.

The Project Director approves and ensures implementation of the proposed resolution and issues the approval to restart. For Stop Work events initiated by the RSO, the RSO shall also approve the proposed resolution and restart.

9.2 DECOMMISSIONING TASK MANAGEMENT

9.2.1 PROCEDURES

Decommissioning activities are managed through policies and procedures which establish the constraints under which a specific program or plan (e.g., Radiation Protection Plan) will operate. Work is accomplished by procedures that implement the requirements of regulations and License SNM-33 (Reference 9-1).

Functional area managers are responsible for the subject matter covered by a program or plan. The functional area manager is responsible for ensuring that other organizations impacted by the document are given the opportunity to review them, and any revisions, before issuance. Prior to issuance or revision, the following classifications of documents require the minimum functional management approvals indicated:

- Radiation Protection – Radiation Protection and Project Director;
- Environmental, Health and Safety – EH&S and Project Director;
- Material Control and Accounting (MC&A) – Manager of MC&A and Project Director;
- Criticality Control – Nuclear Criticality Safety Specialist, Manager of Nuclear Criticality Safety and Project Director;
- Waste Management – Waste Management, Radiation Protection and Project Director;
- Quality Assurance – Quality Assurance and Project Director; and
- Physical Security – Security, Licensing and Project Director.

9.2.1.1 Control of Procedures

Proposed changes to procedures will be evaluated by qualified individuals prior to implementation to ensure they are consistent with license and regulatory requirements.

Procedures will be reviewed biennially from the date of the last revision to ensure applicability to current site activities. Updating of procedures is the responsibility of the applicable functional area.

Changes to procedures will be documented and available for periodic review. Documentation will include a description of the change, the effective date of the change and the appropriate approvals of the change.

9.2.2 SITE WORK CONTROL

Individual decommissioning tasks are evaluated through the site work control process. This process provides the formal structure in which proposed work is evaluated to ensure it is compliant with the site license and this Decommissioning Plan. The proposed work is evaluated by a committee of individuals from the various site functional areas who have the appropriate technical knowledge base to evaluate the requested work activity. At a minimum, the committee will consist of a representative from the Operations, Radiation Protection, Environmental Health & Safety, and the Licensing functional areas. Representatives from other functional areas of the site may participate in the screening of work requests. The committee will screen the work request and determine the appropriate level of evaluation, reviews and approvals, including prior NRC approval, necessary to plan and perform the work activity.

To ensure the proper level of control of site work activities, work will be categorized as follows:

- Routine Work - Typically repetitive work that does not require the specific guidance provided by a procedure or work package to complete the work. Examples of this type of work are grounds maintenance and housekeeping activities. These types of work activities are considered to be within the skill of the craft. When performing this work, all applicable site procedures, such as wearing the correct PPE, complying with security requirements or radiation work permit requirements, shall be followed. These work activities are typically verbally authorized by a supervisor and are not quality related;
- Work Controlled By Procedure - A work activity that requires sufficient directive detail to perform the work. Work controlled by a procedure can be of a routine repetitive nature or it can be work that is infrequently performed. Examples of work controlled by a procedure would be operation of a system or inspecting heavy equipment. Procedures are approved documents that ensure qualified individuals perform the work in an approved manner or sequence. Procedures implement requirements and directives from plans and programs;
- Work Controlled By Work Package - Typically a one time work activity that incorporates requirements of the various site functional areas into a work plan. The work activity may be performed multiple times. Examples include building demolition or soil remediation. Work Packages provide the mechanism to review, approve, implement and control work activities within the Site Work Control process. Work Packages are approved by the appropriate functional areas prior to performing the work.

9.2.2.1 Site Work Control Change Management

When it becomes necessary to make a change to an approved work activity, the proposed change is evaluated to ensure that the change is appropriate, and is within the scope of the work as

originally authorized. Based upon the scope and impact of the proposed change, the appropriate functional area shall be identified for review of the proposed change. If the change is not within the scope of the work originally authorized, then it shall be evaluated by Licensing to determine if prior NRC approval is required. Site work control change management requirements are summarized below for each category of work:

- Routine Work - Changes required to routine work activities are implemented by direction of supervision;
- Procedures - Changes required to work activities within a procedure are implemented through a procedure revision; and,
- Work Package - Changes to a Work Plan that has been approved requires the use of a Work Package Change Notice.

9.2.3 RADIATION WORK PERMIT

Radiation Work Permits (RWPs) are used to control radiological work and are developed, evaluated and authorized in accordance with the Radiation Protection Plan (Reference 9-2). RWPs are integrated into routine work activities, activities controlled by procedure and activities controlled by a work package. RWPs are further discussed in detail in Chapter 10.

9.3 DECOMMISSIONING MANAGEMENT POSITIONS AND QUALIFICATIONS

This section provides a description of the minimum qualifications and responsibilities of the key functional positions of Hematite Decommissioning Project Director and Radiation Safety Officer. Within 30 days after a change of any individual in one of these positions the licensee shall submit to the NRC written notification of the change. This notification shall include a summary of the new individual's experience and qualifications, and an evaluation that verifies that the individual's experience and qualifications meet the minimum requirements for the position.

The functional area of Nuclear Criticality Safety shall be assigned to a safety-related organizational unit, i.e., Licensing; Radiation Protection; or Environmental, Health and Safety. Nuclear Criticality Safety shall be independent from production related organizations, such as Operations or Project Engineering. The manager for the safety-related organizational unit containing NCS shall have:

- Direct access to the Project Director and be a member of the Project Oversight Committee,
- Direct the activities of the NCS program including approving and reviewing operations and procedures, and establishing NCS controls,
- A BS or equivalent, three years of previous management experience in a safety discipline, and three years of nuclear work experience,
- "General Employee Training" and "Fissile Material Training for Supervisors and Managers (FMTSM)."

9.3.1 DIRECTOR, HEMATITE DECOMMISSIONING PROJECT

The HDP Director has overall responsibility to ensure safety and compliance during the decommissioning of the Hematite Site while complying with applicable laws and regulations. The Project Director is responsible for hiring personnel who meet the minimum qualifications described in this chapter.

At a minimum the Project Director will meet the following qualifications:

- Bachelor's degree in an appropriate discipline or an equivalent combination of education and experience;
- Previous managerial experience in project management of large complex projects; and
- Training in nuclear safety.

9.3.2 PROJECT OVERSIGHT COMMITTEE

The Chairperson is the head of the POC. The Chairperson is responsible for the following; chairing the POC; designating in writing appropriate sub-committees, with the concurrence of a majority of the POC members; determining which committee members shall attend each meeting according to the topics to be covered; and recommending the committee members (at a minimum, Radiation Protection, Licensing and Operations shall have representatives).

At a minimum the Chairman will meet the following qualifications:

- Bachelor's degree in an appropriate discipline or an equivalent combination of education and experience; and
- Competency in nuclear safety.

9.3.3 OPERATIONS

The Operations functional area manager is responsible for the safe and efficient execution of work performed by HDP Staff, identifying the personnel and physical resources required to complete work tasks, and interfacing with Project Engineering and Project Controls on work assigned to contractors.

At a minimum the Operations functional area manager will meet the following qualifications:

- Bachelor's degree in an appropriate discipline or equivalent combination of education and experience;
- Previous managerial experience; and
- Nuclear criticality safety training.

9.3.4 LICENSING

Licensing responsibilities include those site activities necessary to ensure compliance with the License SNM-33 (Reference 9-1). Licensing activities include interacting with the NRC and other regulators as assigned, preparing license amendments, and reviewing planned work activities to ensure compliance with License SNM-33 (Reference 9-1).

At a minimum, the Licensing functional area manager will meet the following qualifications:

- Bachelor's degree in an appropriate discipline or an equivalent combination of education and experience;
- Previous managerial experience in the environmental and safety discipline;
- Two years of experience in licensing, or regulatory affairs, or equivalent; and
- Strong skills in written and oral communication and organizational management.

9.3.5 RADIATION PROTECTION

The Radiation Safety Officer (RSO) directs and manages Radiation Protection staff and resources in the accomplishment of job responsibilities. The RSO is responsible for the establishment and guidance of radiation protection programs. As such, the RSO is responsible for ensuring that activities involving the use of radioactive material are conducted safely and in accordance with applicable regulatory requirements. The RSO also evaluates potential and/or actual radiation exposures, establishes appropriate control measures, approves written procedures, and assures compliance with pertinent procedures and regulations.

Under the RSO's direction, health physics personnel collect samples, perform analyses, take measurements, maintain records, and assist in performing the technical aspects of the radiation protection program. The Project Director has also assigned the RSO responsibilities for waste management and hazardous material transportation.

The RSO has access to all areas of the facility to effectively oversee the site radiation protection program. The RSO has specific "Stop Work" authority to ensure conflicts over the appropriate manner to conduct site activities with respect to radiation safety are properly resolved. Additional responsibilities of the RSO include:

- Conducting audits and inspections on activities involving radioactive material.
- Monitoring activities involving radioactive material and storage areas for radioactive material.
- Ensuring radiation workers are properly trained.
- Maintaining a source control procedure that includes the requirements for leak testing sealed sources.
- Monitoring and evaluating radiation worker exposure.
- Responding to and investigating incidents and accidents involving radioactive material.
- Ensuring radiological records are maintained in accordance with Quality Assurance and regulatory requirements.
- Development and implementation of final status survey requirements.
- Other duties as assigned.

In general, the RSO will have the knowledge and ability necessary to respond effectively to the radiation safety needs of the HDP. The RSO will have a background of training and experience and a maturity of judgment sufficient to recognize the need for expert assistance at an early stage in the development of potential radiation safety problems involving disciplines outside of his or

her area of expertise. Specifically, the RSO will have, or have access to, individuals with the skills and knowledge, as necessary, to address the following areas and implement the radiation protection program per applicable regulatory, program, and procedural requirements:

- Developing, reviewing and revising radiological protection procedures for approval.
- Monitoring and surveying of areas in which radioactive material is used.
- Overseeing ordering, receipt, surveys, and delivery of licensed material.
- Packaging, labeling, surveys, etc., of shipments of licensed material.
- Monitoring of personnel, including determining the need for and evaluating bioassays, monitoring personnel exposure records, and developing corrective actions for those exposures approaching maximum permissible limits.
- Monitoring and controlling radiological effluent and monitoring the environment, including the need for evaluating air, liquid and soil effluents and developing corrective actions for those effluents approaching maximum permissible limits.
- Radiological training of personnel.
- Inventory and leak testing of sealed sources.
- Responding to and investigating incidents and accidents involving radioactive material.
- Maintaining required radiological records.
- Having experience in Emergency Management Operations.
- Having previous managerial experience.

At a minimum the RSO will meet the following criteria. An acting RSO shall be designated when the named RSO is not present on-site. The acting RSO shall meet the first three bullets listed below:

- A B.S. in the physical sciences, industrial hygiene, or engineering from an accredited college or university, or an equivalent combination of training and relevant experience in radiological protection. Two years of relevant experience are considered equivalent to one year of academic study. (Required for acting RSO.)
- At least three years of work experience in applied health physics, industrial hygiene, or similar work relevant to radiological hazards associated with site remediation. This experience should involve actually working with radiation detection and measuring equipment. (Required for acting RSO.)
- A thorough knowledge of the proper application and use of health physics equipment used for the radionuclides present onsite, the analytical procedures

used for radiological sampling and monitoring, and the methodologies used to calculate personnel exposure to radionuclides present at the site. (Required for acting RSO.)

- Strong skills in written and oral communication and organizational management.
- Previous managerial experience.

9.3.6 DECOMMISSIONING COMMITTEES

Decommissioning Committees have been established to provide management oversight of HDP decommissioning activities, with responsibility for:

- Evaluating the effectiveness of project policies and programs developed to protect the health and safety of project employees, contractors and the general public; and
- Providing direction, guidance and overview to ensure compliance with site decommissioning licenses, permits and regulations.

Specific HDP committees and their responsibilities are described below.

9.3.6.1 Project Oversight Committee (POC)

The goal of the POC is to promote and continuously improve work place safety for the HDP. The POC's purpose is to evaluate the effectiveness of and recommend improvements to HDP safety rules, policies and procedures. The POC also reviews first-of-a-kind project activities and changes to the intent of existing programs.

9.3.6.2 Work Control Committee

As described in section 9.2.2, the Work Control Committee provides evaluation of specific decommissioning tasks and work activities to ensure that all work is performed in accordance with the license and decommissioning plan.

9.4 TRAINING

The site Training Plan (Reference 9-3) establishes the administrative controls necessary to ensure project and contractor personnel are adequately trained and qualified to perform their assigned duties, safely and in accordance with applicable requirements. The Training Plan specifies training to ensure personnel are adequately informed of the hazards, preventative measures and procedures associated with performing their task. All on-site personnel receive radiation safety training, ranging from awareness of radiological postings for escorted visitors to radiation exposure reduction methods for Radiation Workers. Safety training is reinforced in plan-of-the-day and toolbox topics briefing for individuals assigned to perform or oversee on-site physical work. This briefing covers selected components of one or more of the following topics: safety items, radiological protection, contamination control, criticality safety, As Low As Reasonably Achievable (ALARA), emergency response and other activities as dictated by the current work processes. Plan-of-the-day and toolbox topics alert personnel to accidents or "near misses" that have occurred on the project, review project and industry lessons learned, and discuss employee safety concerns.

Individuals performing activities affecting quality will receive appropriate training to familiarize them with the quality requirements of the quality assurance program. The training required will be determined by functional areas working with training personnel, and will be based on individuals' past experience (e.g., work history, past qualifications, certifications, etc.), education and job description.

The Training Plan addresses initial training, re-qualification and continuing training, and training documentation (e.g., lesson plans, instructor qualifications and examination results). Training Plan categories include:

- Visitor Access Training includes general safety, emergency response, radiation controls and security instruction;

- General Employee Training (GET) is provided to Non-Radiation Workers. GET includes the following elements; basic knowledge of radiation hazards and protective policies, including: risks of exposure to radiation and radioactive material (including pre-natal exposure); basic radiological fundamentals and radiation protection concepts; individual rights and responsibilities relating to the HDP Radiation Protection Program; individual responsibilities for implementing ALARA measures; and individual exposure reports that may be requested. All general employees are also provided the following training:

    - General Employee Criticality Safety Training provides indoctrination into general elements, requirements and basic fundamentals of criticality safety.

- General Employee Safety Training is provided to ensure employees are aware of the safety elements of the project.

- Emergency Training is provided to ensure employees recognize emergency alarms and react appropriately to them.

- Radiation Worker Training (RWT) is designed to meet the training requirements for individuals entering a radiological Restricted Area to perform work activities. RWT topics are the same as those for GET but include detailed instructions for radiological area ingress/egress, radiological controls and PPE, etc.;

- Fissile Material Handler Training is provided for RWT-trained individuals that require in depth knowledge of criticality safety, fundamentals of atomic structure, nuclear fission, subcritical and critical reactions, conservative nuclear safety factors, change management and accidents. This training provides a thorough working knowledge of criticality safety factors and how they are used to safely handle fissile material;

- Health Physics Technician Training is provided to ensure suitably experienced personnel with information necessary, such as operational health physics program and Final Status Survey responsibilities, and to verify qualification commensurate with project Health Physics Technician job requirements;

- Safeguards Information Training, commensurate with an individual's clearance and their "need to know" regarding site nuclear material security measures;

- Plan-of-the-Day and Toolbox Training is intended for individuals assigned to perform physical work on-site or are assigned direct oversight over physical work;

- Emergency responders are trained on the specific aspects of their responsibilities in an emergency; and

- Contractor personnel receive equivalent training to the training that would be required for HDP personnel performing the same task. If used, contractor training shall be reviewed to ensure it meets HDP training requirements.

9.5 CONTRACTOR SUPPORT

Management of decommissioning activities will be performed by a combination of Westinghouse personnel and staff augmentation by qualified contractors. Decommissioning activities may be performed by technically competent and qualified contracted resources. Contractors at HDP will comply with applicable Westinghouse policies and procedures and the license requirements of HDP.

HDP personnel monitor contractor operations to ensure that they are compliant with License SNM-33 (Reference 9-1), HDP procedures and applicable regulations.

9.6 REFERENCES FOR CHAPTER 9.0

9-1 U.S. Nuclear Regulatory Commission, License No. SNM-33 (Docket No. 70-36).

9-2 Westinghouse Electric Company Document No. HDP-PO-HP-100, "Radiation Protection Plan."

9-3 Westinghouse Electric Company Document No. HDP-PO-GM-002, "Training Plan."

Table 9-1

Functional Area Responsibilities and Authorities

Functional Areas	Responsibilities and Authorities
Radiation Protection	Establishes and manages all radiation protection programs. Ensures that activities involving radiation or radioactive material are conducted safely and in accordance with applicable regulatory requirements. Monitors and evaluates potential and/or actual radiation exposures. Establishes appropriate control measures, including source controls. Approves written procedures involving radioactive material. Assures compliance with pertinent procedures and regulations. Performs radiation measurements, sample collection, and analysis, including environmental monitoring for radiological parameters. Responds to and investigates incidents and accidents involving radiation or radioactive material. Maintains radiological records in accordance with quality assurance and regulatory requirements. Reviews and approves subcontractor health physics procedures.
Waste Management	Identifies, controls, and prepares waste for disposal. Provides proper documentation for packages and for shipments by certified, licensed, and/or permitted carriers. Ensures recipients of shipments have proper authorization to receive the shipment.
Project Engineering/ Management	Develops technical requirements for the safe and efficient conduct of physical work and planning how physical work will be performed. Coordinates and manages remediation, maintenance, or repair work performed by subcontractors. Creates and maintains drawings and graphical information systems.
Operations	Conducts remediation, maintenance, or repair work performed by HDP staff. Assists with overseeing remediation, maintenance, or repair work assigned to subcontractors.

Table 9-1

Functional Area Responsibilities and Authorities

Functional Areas	Responsibilities and Authorities
Environmental Health and Safety (EH&S)	Maintains the environmental, health and safety compliance program in accordance with applicable regulations. Implements the Health and Safety Plan, the emergency action program, the storm water control program, and the environmental monitoring program for non-radiological attributes. Develops and ensures compliance with environmental permits and actions related to chemical cleanup. Coordinates with non-NRC regulators. Manages the study of subsurface water conditions.
Training	Coordinates and documents training and qualifications of site personnel.
Community Relations	Coordinates with the press, public, and community leaders.
Licensing	Ensures compliance with the License SNM-33 and NRC regulations. Coordinates with the NRC, including submittals, license amendments, and Decommissioning Plan. Reviews planned work activities to ensure compliance with SNM-33.
Material Control & Accounting (MC&A)	Ensures the control and management of special nuclear material, including the tracking of its location and quantity and providing overall direction to the project for compliance with the HDP Fundamental Nuclear Material Control Plan.
Nuclear Criticality Safety	Performs a nuclear criticality safety assessment before a new or modified operation is started when the operation may involve sufficient contamination to require defense-in-depth controls. Reviews field operations to ensure they follow nuclear criticality safety defense-in-depth controls.

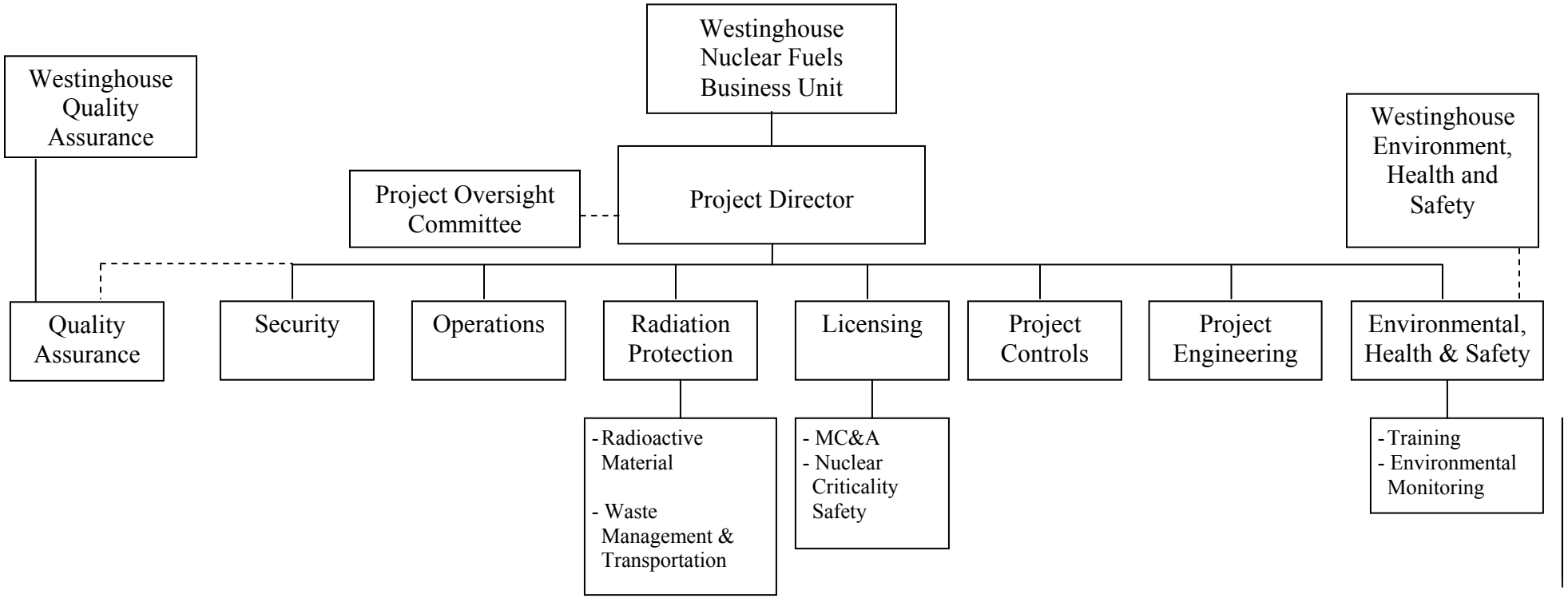
Table 9-1

Functional Area Responsibilities and Authorities

Functional Areas	Responsibilities and Authorities
Quality Assurance	Performs audits, surveillances, receipt inspections, and trends their results. Coordinates the corrective action and continual improvement programs. Maintains controls for measuring and test equipment. Evaluates subcontractor quality programs and procedures. Establishes document control and records management requirements.
Security	Provides physical security for HDP. Controls access to protected areas and protected information.
Project Controls	Provides financial, accounting and managerial controls. Coordinates office facilities, informational systems, and administrative support functions. Develops and maintains the HDP schedule. Procures products, materials and services.

Figure 9-1

Hematite Decommissioning Project Organization Chart



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LIST OF TABLES

ACRONYMS AND ABBREVIATIONS

ADM	Simple Administrative Controls
AEC	Active Engineered Controls
ALARA	As Low As Reasonable Achievable
ALI	Annual Limit On Intake
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOA	Area of Applicability
BZ	Breathing Zone
CAM	Continuous Air Monitor
CDBS	Collared Drum Buffer Store
CDE	Committed Dose Equivalent
CDRA	Collared Drum Repacking Area
CDs	Collared Drums
CDSA	Collared Drum Storage Area
CEDE	Committed Effective Dose Equivalent
CFR	Code Of Federal Regulations
cm	centimeters
CSC	Criticality Safety Control
DAC	Derived Air Concentration
DCDs	De-collared Drums
DDE	Deep Dose Equivalent
EADM	Enhanced Administrative Controls
EDMS	Electronic Document Management System
EH&S	Environmental Health And Safety

ACRONYMS AND ABBREVIATIONS
(continued)

FCSA	Field Container Storage Area
FMSA	Fissile Material Storage Area
FNMCP	Fundamental Nuclear Material Control Plan
g	gram
GA	General Area
HASP	Health And Safety Plan
HDP	Hematite Decommissioning Project
HEU	High Enriched Uranium
HEPA	High Efficiency Particulate Air
HP	Health Physics
HSA	Historical Site Assessment
ISOCS	In-Situ Object Counting System
L	Liter
MAA	Material Assay Area
MC&A	Material Control And Accounting
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
NCS	Nuclear Criticality Safety
NCSA	Nuclear Criticality Safety Assessment
NCSS	Nuclear Criticality Safety Signs
NIOSH	National Institute for Occupational Safety and Health
NIST	National Institute Of Standards And Technology
NMMSS	Nuclear Materials Management And Safeguards Systems
NP	Negative Pressure
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
OSHA	Occupational Safety And Health Administration
PA	Perimeter Air

ACRONYMS AND ABBREVIATIONS
(continued)

PAPR	Powered Air Purifying Respirator
pCi	picoCuries
PEC	Passive Engineered Controls
PEL	Permissible Exposure Level
PF	Protection Factor
PLHCP	Physician Or Other Licensed Health Care Professional
POC	Project Oversight Committee
PPE	Personal Protective Equipment
RIS	Reporting Identification System
RPP	Radiation Protection Plan
RSO	Radiation Safety Officer
RWP	Radiation Work Permit
SCBA	Self Contained Breathing Apparatus
SNM	Special Nuclear Material
SSC	Structures, Systems And Components
SSNM	Strategic Special Nuclear Material
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TLV	Threshold Limit Value
U	Uranium
U-235	Uranium-235
USL	Upper Subcritical Limit
VOCs	Volatile Organic Compounds
WEA	Waste Evaluation Area
WHA	Waste Holding Area
WTS	Water Treatment System

10.0 HEALTH AND SAFETY PROGRAM DURING DECOMMISSIONING

10.1 RADIATION SAFETY CONTROLS AND MONITORING FOR WORKERS

In accordance with 10 CFR 70.38 (g) (4) (iii), "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas," (Reference 10-1) the Decommissioning Plan must provide a description of the methods used to ensure the protection of workers and the environment against radiation hazards encountered during decommissioning. This chapter, utilizing the criteria established in NUREG-1757, Volume 1 Revision 2 - Final Report, "Consolidated Decommissioning Guidance - Decommissioning Process for Materials Licensees," (Reference 10-2) discusses the methods used to ensure the protection of the workers from ionizing radiation during decommissioning.

The Decommissioning Plan, the Radiation Protection Plan (RPP) (Reference 10-3) and the associated implementing procedures are intended to ensure that decommissioning activities will be performed in compliance with 10 CFR 19, "Notices, Instructions And Reports To Workers: Inspection and Investigations," (Reference 10-4) and 10 CFR 20, "Standards For Protection Against Radiation," (Reference 10-5).

The RPP provides standards and requirements to minimize the potential risk of harm or injury to workers, visitors, the public and the environment from ionizing radiation. The RPP programmatic elements are implemented through approved site procedures to ensure activities are conducted in accordance with 10 CFR 20. The program elements as included in the RPP are provided in Table 10-1.

In accordance with the RPP, annual individual occupational doses for employees should not exceed the administrative occupational exposure limit of 2,000 mrem TEDE. The action level for investigation and possible work restrictions shall be 1,000 mrem for DDE. Approval by the Project Director and RSO is required for an employee to exceed the administrative limits. Individual doses to minors or members of the general public and visitors shall not exceed 100 mrem per calendar year from Hematite Decommissioning Project (HDP) operations involving licensed radioactivity, and the radiation dose in any unrestricted area from external sources is less than 2 mrem in any one hour. Exposures for the embryo/fetus shall not exceed 500 mrem for the duration of the gestation period following declaration of pregnancy by a female facility worker. Declared pregnant females likely to receive a Committed Effective Dose Equivalent (CEDE) in excess of 100 mrem or a Deep Dose Equivalent (DDE) in excess of 100 mrem shall be monitored for internal and/or external exposure as appropriate. Internal dose to the embryo/fetus shall be calculated as provided in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus," (Reference 10-6).

10.2 WORKPLACE AIR SAMPLING PROGRAM

Workplace air sampling shall be performed to comply with 10 CFR 20.1501 and to enable appropriate work controls so radiation exposure and radiological effluents are as low as reasonably achievable (ALARA). Sampling shall be performed as necessary to measure concentrations of airborne radioactivity in the work place, to estimate the corresponding internal exposure, to gauge the effectiveness of engineered controls to minimize airborne radioactivity, and to serve as a comparator to the regulatory limits and administrative action levels for concentrations in air effluents.

For the decommissioning activities described for HDP, the extent of occupational exposure to airborne radioactivity shall be assessed using the Derived Air Concentration (DAC) values specified in 10 CFR 20 Appendix B, "Annual Limits on Intake and Derived Air Concentrations of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage." The DAC may be based on the most conservative radionuclide present or on a calculation weighting the radionuclides. As Uranium (Class Y) is conservatively assumed to be the predominant radionuclide contributing to dose, the occupational DAC values most relevant to HDP are 2.0 E-11 µCi/ml for gross alpha radioactivity and 6.0 E-8 µCi/ml gross beta radioactivity (based upon Th-234).

Work place sampling shall be performed when airborne radioactivity concentrations are likely to exceed 2 percent of the occupational DAC values in general areas. Sampling that is representative of the concentrations within the breathing zone shall be performed when airborne radioactivity concentrations are likely to exceed 10 percent of the occupational DAC values in the breathing zone, or when airborne radioactivity concentrations are likely to exceed 2 percent of the occupational DAC values in the breathing zone of a declared pregnant female. Sampling shall also be performed when respirators are worn for the purpose of protecting individuals from exposure to airborne radioactivity.

Airborne monitoring requirements are implemented through the site workplace air sampling program, as established in the RPP, the Health and Safety Plan (HASP), (Reference 10-7), and site approved implementing procedures. The workplace air sampling program is designed to implement the guidance of 10 CFR 20, NRC Regulatory Guide 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing And Fuel Fabrication," (Reference 10-8), and NRC Regulatory Guide 8.25, "Air Sampling In The Work Place," (Reference 10-9).

10.2.1 EVALUATING THE NEED FOR AIR SAMPLING

Concentrations of airborne radioactive materials are measured within work areas where radioactive materials are handled, within adjacent areas to assess occupational exposure, and at locations that are representative of the concentrations released to the environment. The locations for these measurements are appropriate for the intended use of the data. The measurement results are used as the basis to confirm or update the requirements for radiological posting, gauge

the effectiveness of engineered controls and work practices, and to prescribe the requirements for respiratory protective devices.

Based upon estimates of airborne radioactivity, the Radiation Work Permits (RWP) will specify the type and frequency of air sampling required for the monitoring of occupational exposure from airborne radioactivity while performing work in a radiological environment at HDP. Air sampling shall be performed using a combination of low-volume, high-volume and lapel air samplers to confirm the results of the hazard assessment. The type, frequency and location of the air sample shall depend on the type, duration, and radiological conditions of the work being performed.

The direction for generating, issuing and maintaining RWPs shall be implemented in accordance with the RPP and its associated implementing procedures. RWPs shall describe the radiological hazards (including internal and external dose estimates) associated with work activities, and specify appropriate engineered controls, radiological work practices, personal protective equipment, radiological surveys and air sampling requirements as appropriate, as well as any special radiological instrumentation or personnel monitoring requirements.

As directed through approved procedures, a Pre-Job ALARA Evaluation is completed during preparation of a RWP. Process knowledge, historic, current and expected radiological data are utilized to complete the evaluation. The evaluation provides the basis for an estimate of the Total Effective Dose Equivalent (TEDE) for the work, a determination of any appropriate engineered and/or administrative controls, requirements for respiratory protection, and the type of air sampling required during the work activity.

The Radiation Safety Officer (RSO) shall approve the requirements specified in each RWP.

10.2.2 TYPES OF AIR SAMPLING

The types of air samples that will be collected for laboratory analysis to determine the concentrations of airborne radioactivity include:

- Personal Air Samples (BZ);
- General (Work) Area Air Samples (GA); and
- Perimeter Air Samples (PA).

10.2.2.1 Personal Air Sampling

When monitoring is required for the purpose of determining occupational exposure, representative sampling is accomplished through the use of a personal air sampler (lapel pump), or a portable low volume air sampler. The air filter is located within approximately 12 inches of

the worker's head during sample collection, increasing the probability of being representative of the concentration in the worker's breathing zone."

10.2.2.2 General Area Air Sampling

GA sampling is performed when work activities are likely to cause airborne concentrations in excess of 2 percent of an occupational DAC value. The samples are collected using a portable low volume air sampler, and the results of sampling are used to establish the requirements for posting and controls, for assessing the effectiveness of engineered controls, or for assessing the effectiveness of contamination controls. The sample data from GA air samples may be used for DAC-hour tracking if determined to be representative of the airborne concentrations breathed by the worker.

10.2.2.3 Perimeter Air Sampling

PA sampling is an element of the environmental monitoring program, and is performed when work activities are likely to cause airborne concentrations at the work area boundary in excess of 10 percent of the air concentration values listed in 10 CFR 20, Appendix B, Table 2. The samples are collected using a portable low volume air sampler. The number and location(s) of the perimeter air sampler(s) shall be selected with consideration for the location and nature of the work activities, and environmental conditions such as wind direction.

10.2.3 AIR SAMPLING EQUIPMENT AND CALIBRATION

Air sampling equipment and calibration systems shall be used in accordance with the RPP and the associated implementation procedures. Air samplers used for quantitative measurements shall be flow tested with a calibrated standard before and after an air sample is collected to determine air volume sampled. Calibration methods shall meet the requirements of Regulatory Guide 8.25.

10.2.3.1 Personal (Lapel) Air Sampler

For BZ air sampling, the MSA Model Escort LC Lapel Air sample pump or equivalent is utilized. The unit has an adjustable air flow rate capacity ranging from 0.5 liters/minute to 3 liters/minute. The lapel air sampler will be flow tested before and after each sample is collected. The flow test of the lapel air sampler will be accomplished using a Gillian Model Gilibrator-2 Diagnostic Calibration System or equivalent.

10.2.3.2 Portable Low Volume Air Sampler

For GA air sampling, the RADēCO Model AVS-28A constant flow air sampler or an equivalent unit will be utilized. This constant flow air sampling unit has an adjustable flow rate capacity ranging from 20 liters/minute to 100 liters/minute. Constant flow air samplers will be flow

tested before and after each sample is collected utilizing a flow meter calibrated as specified in Chapter 13.0.

10.2.3.3 Alarming Continuous Air Monitor

Continuous Air Monitors (CAM) with alarm capability will be used during work activities likely to generate airborne radioactivity concentrations exceeding five times the occupational DAC values as provided in 10 CFR 20, Appendix B, and workers are likely to exceed 40 DAC-hrs in one day after credit is taken for respiratory protective devices.

The expected radiological conditions do not suggest that the use of a CAM is warranted; however, should there be a demonstrated need for a CAM, the RSO will be responsible to ensure that the procurement of an adequate supply of CAMs is available for use prior to the commencement of work activities that would require their use.

10.2.4 AIR SAMPLING

The Health Physics (HP) staff is responsible for performing measurements of airborne radioactivity concentrations. The HP staff, in accordance with the requirements of the RPP and associated implementing procedures, determines the type, frequency, and appropriate location of sampling to be performed for the activity to be monitored.

The selection of the type of air sampler considers the sample flow rate, the duration of the work activity and the volume of the sample that can be obtained given these two conditions. The sampler selection will best-afford the ability to obtain the sample volume necessary to achieve the target minimum detectable concentration (MDC). Minimum air volumes necessary to identify activity at the action levels will vary based on the type of air sample, the radionuclides present and the analytical minimum detectable activity (MDA) for the counting instrument used. Lower sample volumes may be acceptable depending on the instrumentation and/or methods used for air sample analysis.

10.2.5 AIR SAMPLE EVALUATION

The HDP maintains on-site ability to perform air sample analysis for gross alpha and gross beta radioactivity. If necessary, samples can be sent to a qualified laboratory for additional analysis to identify the contributors to the gross alpha or gross beta radioactivity. The on-site counting system is calibrated and operated in accordance with the site procedure for the system.

10.2.5.1 Counting System Requirements

The Stationary Alpha Counting System has a nominal counting efficiency of 25 percent of 4 pi geometry with a target minimum detectable activity of 20 disintegrations per minute. The

Stationary Beta-Gamma Counting System has a nominal counting efficiency of 30 percent of 4 pi geometry with a target minimum detectable activity of 200 disintegrations per minute.

10.2.5.2 Analysis Results

Samples are counted in accordance with the operating procedure for the system. Sample results are then reported on the appropriate air sample report and utilized for the intended purpose, such as worker intake, DAC-hour tracking, work area evaluation or evaluation for posting of the area.

10.2.5.3 Action Levels

An area shall be posted as an Airborne Radioactivity Area when an area is accessible to individuals and when observed airborne radioactivity levels exceed the DAC values specified in 10 CFR 20, Appendix B (2.0 E-11 µCi/ml for gross alpha radioactivity or 6.0 E-8 µCi/ml for gross beta radioactivity) or, when observed airborne radioactivity levels are present to such a degree that an individual present in the area without respiratory protective equipment could exceed, during a period of one week, an intake of 0.6 percent of the Annual Limit on Intake (ALI) or 12 DAC-hours.

10.3 RESPIRATORY PROTECTION PROGRAM

The HDP Respiratory Protection Program must be capable of addressing both radiological and non-radiological hazards and will be designed and implemented to comply with both NRC and U.S. Occupational Safety and Health Administration (OSHA) requirements as specified by 29 CFR 1910.134, "Occupational Safety and Health Standards," (Reference 10-10).

The program will be administered by both the RSO and the Environmental Health and Safety and (EH&S) Manager. The implementing procedures will incorporate the guidance provided in NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," (Reference 10-11). Table 10-2 presents the list of program elements that will be incorporated into the HDP Respiratory Protection Program.

To the extent practical, the project will use engineered or administrative controls as necessary to reduce the concentration of radioactive material in air and to minimize the potential for inadvertent spread of contamination. When it is not practical to apply such controls, respiratory protective devices may be used. The respiratory protection equipment will used and maintained in accordance with written implementing procedures and appropriate training as required by 10 CFR 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas," (Reference 10-5).

10.3.1 CONTROLS

The primary method that will be employed at the HDP to protect workers from occupational exposure to airborne contaminants resulting from site decommissioning activities will be through the use of engineered controls. The typical engineered controls that may be utilized include the use of HEPA ventilation, fixatives, dust suppression by misting, and the use of enclosures. Administrative controls will also be utilized to reduce exposure to airborne contaminants. Administrative controls such as the review and implementation of appropriate work practices, stay times, and rotation of personnel will be applied as necessary. Other engineered and administrative controls may also be implemented as determined appropriate by evaluation.

When an evaluation of applied engineered and administrative controls concludes that worker protection from exposure to airborne contaminants is not adequate, then respiratory protective devices will be selected and utilized as necessary based upon the contaminant and the desired protection factor.

10.3.2 ALARA REQUIREMENTS

The primary mode of occupational exposure from radiological hazards at the HDP is attributed to internal exposure resulting from the inhalation of radioactive materials.

To ensure that worker exposure remains ALARA, decommissioning work activities shall be evaluated as part of the RWP generation process and estimates of the internal and external exposures shall be documented. The Pre-job ALARA Evaluation will document this assessment, including the basis for the appropriate use of engineered and administrative controls, and/or respiratory protective devices. For non-radiological contaminants, the evaluation will be performed as part of the Personal Protective Equipment (PPE) hazard assessment process.

10.3.3 EQUIPMENT

The assigned respiratory protection factors listed in 10 CFR 20, Appendix A "Assigned Protection Factors for Respirators," shall be used when respirators are required to protect against airborne radiological contaminants. The assigned respiratory protection factors listed in 29 CFR 1910.134, Table 1, shall be used when respirators are required to protect against airborne non-radiological contaminants. At HDP, the respiratory protection equipment approved for use includes the full-face Negative Pressure (NP) respirator and the full-face Powered Air Purifying Respirator (PAPR). In accordance with 10 CFR 20, Appendix A, a full-face NP respirator has a protection factor (PF) of 100 and the PAPR has a PF of 1,000 for radiological contaminants.

Respiratory protection equipment shall be approved by the National Institute for Occupational Safety and Health (NIOSH). All respirators will be inspected before each use and during cleaning. Respirators will be stored in bags or containers to protect them from damage, contamination, dust, sunlight, extreme temperatures, excessive moisture and damaging chemicals, and stored in a manner to prevent deformation of the face piece and exhalation valve.

Minor repairs that do not impact the integrity of the respirator will be conducted onsite by a qualified technician. If the respirator requires complex repair, then the respirator will be repaired by a qualified off-site vendor or removed from use and properly disposed.

10.3.4 TRAINING

Workers shall receive training on the proper use and care of respirators prior to initial use, and annually thereafter while participating in the respiratory protection program. Respiratory protection training will incorporate the guidance of Regulatory Guide 8.15 and 29 CFR 1910.134. Training will include instruction on inspecting a respirator prior to use, performing negative and positive pressure checks, the potential consequences of improper respirator use, proper donning and doffing of the respirator, and the actions to be taken in the event of respirator failure or user distress for both radiological and non-radiological contaminants. As a component of the training, participants must successfully complete a written examination to demonstrate understanding of these requirements.

10.3.5 MEDICAL EVALUATION

Prior to wearing a respiratory protection device, personnel are required to undergo a medical evaluation by a Physician or other Licensed Health Care Professional (PLHCP). The medical evaluation shall determine if the worker is able to wear a full-face NP respirator without undue stress to the cardiovascular or respiratory system. This evaluation typically consists of a general physical examination, an electrocardiogram, an audiogram, blood work, a pulmonary function test and a chest X-ray. A medical re-evaluation will be conducted annually thereafter.

10.3.6 FIT TESTING

Personnel that have been medically qualified and trained to wear respiratory protection shall be quantitatively fit-tested for the brand, model, and size of respirator to be worn. Fit-testing shall be performed prior to initial respirator use and annually (at a minimum) while participating in the respiratory protection program. The respirator fit test is a quantitative test utilizing a TSI Portacount Respirator Fit Tester or equivalent and the same make, model, style and size respirator the individual will wear in the field. The fit test must demonstrate a fit factor of 1,000 or more to be considered successful.

10.3.7 NON-RADIOLOGICAL AIRBORNE HAZARDS

The Historical Site Assessment (HSA) and other site documents chronicle the historical use of chemicals in the materials used to construct the facility, the waste streams generated at the facility, and the various different processes that were employed during the operational period of the facility. The non-radioactive possible hazards identified include metals, chemicals and particulates.

The program administrator for EH&S shall evaluate the physical state and chemical form of the potential chemical contaminants when evaluating the need for respiratory protection. The evaluation shall include the Threshold Limit Value (TLV) and Permissible Exposure Level (PEL), where appropriate, when determining the need for respiratory protection and the type of cartridge or canister required for protecting the worker. When respiratory protection is deemed to be required, the PPE Hazard Assessment performed as part of the work control process will identify the minimum required respiratory protection.

10.3.8 SAFETY

Based upon the current physical and radiological conditions of the HDP, it is not anticipated that it will become necessary for personnel to enter an area where the airborne physical and radiological condition has not been assessed. Should this improbable scenario arise, as required by OSHA guidance (29 CFR 1910.134, Reference 10-10), entry into the area shall only be accomplished with a self-contained breathing apparatus (SCBA).

Should the work planning and the TEDE Evaluation for a specific work activity determine the need for the use of supplied air hoods or suits, personnel will be assigned as standby rescue persons while the equipment is in use. Standby rescue personnel will ensure direct communication is maintained with the worker(s) and immediately assist them in case of failure of the air supply or any other reason that causes distress. Standby rescue personnel will be supplied with the necessary equipment to perform a rescue.

If utilized, breathing air systems will meet the requirements of Grade D air for breathing air systems as defined in CGA G-7.1-1997, "Commodity Specification for Air," (Reference 10-12).

10.4 INTERNAL EXPOSURE DETERMINATION

Individuals who are likely to receive, in one year, an intake in excess of 10 percent of the ALI (100 mrem CEDE for declared pregnant female) shall be monitored for occupational exposures to radioactive material. For individuals likely to receive an intake in excess of 10 percent of the ALI, internal dose will be assessed by taking suitable and timely measurements of concentrations of radioactive materials in the air of the work area; or quantities of radionuclides in the body; or quantities of radionuclides excreted from the body; or any combination of these measurements.

Although internal exposure is not expected to exceed 10 percent of the ALI, the potential for internal exposure during decommissioning shall be monitored. The primary method of monitoring and subsequently calculating internal exposure is based on measurements of the radioactivity concentrations in air. In the event of an unplanned intake, air sampling shall be supplemented by bioassay sampling for the evaluation of occupational exposure.

10.4.1 DETERMINATION OF INTERNAL EXPOSURE BY DAC-HOUR TRACKING

The results of personal air samplers or other means to obtain representative samples of the airborne radioactivity concentration in the breathing zone are the primary method used for DAC-hour tracking. The results of general area sampling may be used for DAC-hour tracking with the approval of the RSO if airborne radioactivity greater than the monitoring threshold is identified and personal air sample results are not available.

For the purpose of implementation, the air sample concentration is divided by the DAC value for the radionuclides of concern from 10 CFR Part 20, Appendix B to obtain a DAC fraction. This DAC fraction is multiplied by the time of exposure (in hours) to obtain the number of DAC-hours. The DAC-hours are then multiplied by a value of 2.5 mrem per DAC-hour to provide the (stochastic) CEDE for the potential intake. Alternatively, when the breathing zone sampling is performed, the internal exposure (CEDE) may be calculated directly, based on the air filter activity.

10.4.2 DETERMINATION OF INTERNAL EXPOSURE BY BIOASSAY SAMPLING

In the event of an unplanned intake, bioassay sampling shall also be performed for the evaluation of occupational exposure. Urinalysis will be the primary bioassay method. The methods and frequency, and duration of sampling will be determined by the RSO based on the radiological and chemical nature of the materials involved.

NRC Regulatory Guide 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program,” (Reference 10-13) and other applicable regulatory guidance will provide the basis for determining worker intake by bioassay. The analysis of bioassay samples shall be performed by a qualified analytical laboratory that meets the performance criteria recommended in ANSI N13.30, “Performance Criteria for Radiobioassay,” (Reference 10-14). Results will

then be interpreted in accordance with the equations and examples of calculations presented in Regulatory Guide 8.9 to extrapolate the resultant CEDE in mrem for the estimated intake from the single bioassay sample result or for the estimated intake for the entire event.

For the embryo/fetus, internal dose shall be calculated as provided in NRC Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus," (Reference 10-6). NRC Regulatory Guide 8.36 describes two methods for calculating equivalent dose to the embryo/fetus. The Simplified Method as provided in Revision 1 of NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses--Interim Recommendations," (Reference 10-15) or the Gestation-Time Dependent Dose method as provided in Revision 2 of NUREG/CR-5631.

10.4.3 ACTION LEVELS

These air samples shall be collected at least weekly and an investigation shall be conducted whenever any sample result exceeds 1 DAC or whenever a lapel sampler indicates an exposure in excess of eight DAC-hours per day.

Special bioassay measurements may be performed to evaluate intakes of radioactive material. Examples of circumstances that might warrant special bioassays include:

- Air concentration that exceeds 1 DAC, or an exposure in excess of eight DAC-hours in one day;
- Operational events with a reasonable likelihood that a worker was exposed to unknown quantities of airborne radioactive material;
- Known or suspected incidents of a worker ingesting radioactive material;
- Incidents that result in contamination of wounds or other skin absorption;
- Evidence of damage to or a failure of respiratory protection or the presence of facial and/or nasal contamination; and
- An administrative limit is reached or exceeded.

10.5 EXTERNAL EXPOSURE DETERMINATION

An external exposure monitoring plan shall be maintained consistent with the requirements of 10 CFR 20.1502(a). At a minimum, external exposure to radiation shall be monitored for individuals likely to receive 10 percent of the annual occupational dose limits specified in 10 CFR 20. If an individual exceeds 1 rem DDE within a calendar year, then an investigation will be performed into the reason and work restrictions for the individual will be considered.

Based upon the HSA and other investigations, the primary HDP radionuclides of concern are Uranium (U-234, U-235, U-236 and U-238), Thorium (Th-232), Technetium-99 (Tc-99), Americium-241 (Am-241), Plutonium-239/240 (Pu-239/240) and Neptunium-237 (Np-237). Radium-226 (Ra-226) is also considered to be a radionuclide of concern and is found primarily within the elevated Ra-226 area identified in the burial pits. Additionally, based on the HSA and in consideration of the conservative assumptions that form the basis for the nuclear criticality safety program, exposure to neutron and gamma radiation resulting from a criticality event is not credible. In consideration of the radioactive source term, and experience gained from previous decommissioning work at HDP, it has been determined that the external radiation exposure of any individual performing decommissioning activities at the HDP is unlikely to exceed 100 mrem in a calendar year.

The primary and secondary dosimetry devices that may be used to perform external radiation monitoring consist of thermoluminescent dosimeters (TLD) and self-alarming dosimeters. Primary dosimetry shall be processed by a qualified vendor accredited by the National Voluntary Laboratory Accreditation Program (NVLAP), typically at a quarterly frequency.

Secondary dosimetry usage will be implemented in accordance with the RPP and associated implementing procedures. The requirements as presented in the implementing procedures which utilize the guidance of NRC Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters," (Reference 10-16), and NRC Regulatory Guide 8.28, "Audible-Alarm Dosimeters," (Reference 10-17).

- Monitored personnel shall wear dosimetry on the chest or other portion of the whole body as directed by the RWP or HP Staff so that the primary dosimetry device is located on the portion of the whole body likely to receive the highest exposure.
- If it is uncertain which part of the whole body may receive the highest exposure (due to worker movement or multiple radiation sources), monitored personnel may wear additional dosimetry devices on those portions of the whole body that could receive the highest exposure. The RSO will determine which dosimeter will be used to assign the worker's dose when multiple dosimeters are worn.
- For work situations in which extremity exposures are expected to be at least five times greater than the whole body exposures, or if extremity exposures are expected to exceed

1,250 millirem per calendar quarter, the RSO shall specify additional dosimetry devices for the extremities to measure and control extremity dose.

Dosimetry will be utilized that is capable of detecting beta, gamma, x-ray and neutron radiation, such as the Global Dosimetry Model 760 TLD. The Model 760 has an energy response of 0.766 MeV to 5 MeV for beta, 5 keV to 6 MeV for photons, and thermal to 6 MeV for neutrons.

Radiation surveys will be performed to determine and verify the radiation levels within general work areas and at specific work locations to identify localized areas of elevated radiation levels, and to serve as a comparator when determining radiological posting and external radiation monitoring requirements. Radiation survey instrument selection and method of calibration will be based on the type, energy, and amount of the radiation expected to be encountered.

10.6 SUMMATION OF INTERNAL AND EXTERNAL EXPOSURE

It has been concluded that the primary radiological hazard for individuals performing decommissioning activities at the Hematite Site is internal dose from the inhalation of radioactive material. Due to the radioactive material inventory remaining on the site and nature of the work activities, it is unlikely that the external radiation exposure of any individual performing decommissioning activities at the HDP Site will exceed 100 mrem in a calendar year. However, when dosimetry is issued at the HDP Site for the monitoring of external exposure, then the summation of internal and external exposure will be accomplished in accordance with 10 CFR 20, utilizing the guidance as provided in NRC Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," (Reference 10-18).

Occupational exposures for a declared pregnant female shall be monitored when it is likely that her occupational dose will exceed 10 percent of the federal limit. In addition, DAC-hour assignment will also be initiated to track occupational exposure from airborne radioactivity to a declared pregnant female when measured airborne radioactivity concentrations in the areas that she may access exceed 2 percent of the DAC values presented in 10 CFR 20, Appendix B. Internal dose to the embryo/fetus shall be calculated as provided in NRC Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus," (Reference 10-6).

10.6.1 REPORTING OF OCCUPATIONAL EXPOSURES

The results of all monitoring of occupational exposure as required under 10 CFR 20.1502 (a) or 10 CFR 20.1502 (b) will be reported in accordance with the requirements of 10 CFR 20.2106, Records for Individual Monitoring Results. Individuals shall be provided with the results of their occupational exposure monitoring upon request.

10.7 CONTAMINATION CONTROL PROGRAM

The contamination control program for the Hematite Site is presented by the RPP and implemented through compliance with site approved procedures. The contamination control program is intended to demonstrate compliance with 10 CFR 20, implement the ALARA philosophy and utilize the guidance of Regulatory Guide 8.24 and Regulatory Guide 8.15.

Radioactive contamination on surfaces (both fixed and removable) shall be identified to prevent the inadvertent spread of contamination to non-contaminated surfaces, to identify locations where controls are appropriate to minimize the potential for creating airborne radioactivity, and ultimately, minimize internal and external radiation exposures. Radiological postings will be used to alert personnel to the presence of radioactive materials. To enter a posted radiological area, a person must meet and comply with all training and access requirements applicable to that area.

The access requirements to areas where radiological contamination has been identified will be conveyed through the use of radiological postings, and the detailed instructions contained on the RWP. The RWP, in conjunction with applicable survey data, will communicate radiological conditions to the worker, and establish the requirements for personnel protective equipment, proper work practices, and the requirements for radiological monitoring and sampling. A RWP shall be issued for all work inside Restricted Area(s) at the Hematite Site.

10.7.1 SURVEYS OF SURFACE CONTAMINATION

Contamination surveys are performed to determine the level of total and removable radioactivity in the form of surface contamination. The level of total contamination is typically determined by direct measurement of the surface. The level of removable contamination is typically determined by wiping a defined surface area with an absorbent material and then analyzing the material to determine the amount of removed radioactivity per unit surface area. For operational surveys, the gross results of alpha and/or beta surface contamination are converted to standard units of activity (no surface material background subtracted, results are corrected to account for ambient background). The results are compared to limits specified in the HDP Materials License and Health Physics operational implementation procedures as well as to define radiological posting requirements, establish effective radiological controls to prevent the spread of contamination, and prescribe the type of personal protective equipment required based on the type of work to be performed in the area.

Contamination control surveys are performed on a regular basis to:

- Confirm the absence of radioactive material within areas where radiological controls are not applied;

- Assess the degree of hazard in the work place from occupational exposure to radiation and radioactive materials;
- Determine the appropriate radiological posting and labeling requirements;
- Provide input to the requirements for engineered controls, appropriate work practices, and protective equipment and clothing requirements;
- Gauge the effectiveness of implemented engineered controls and work practices; and
- Demonstrate compliance with applicable federal and state regulations for the packaging and transportation of radioactive materials.

In establishing the frequency of routine contamination control surveys in an area, the contamination levels in the area, the extent that the area is occupied, and the probability of significant personnel exposures are considered. The following minimum frequency for contamination control will be applied to areas when personnel access is required:

Area	Survey Frequency	Action Level [(a)] (Removable Contamination)
Contamination areas	Monthly	None
Non-contaminated areas	Monthly	200 dpm/100cm^2 (alpha or beta)
Step-off pad areas	Daily	200 dpm/100cm^2 (alpha or beta)

(a) An investigation that may include additional surveys and decontamination shall be conducted when removable surface contamination exceeds the action levels specified above.

Work planning and job coverage contamination control surveys are performed as necessary during the planning process and during the course of the work to confirm anticipated conditions and the adequacy of control measures. The frequencies for work planning and job coverage contamination control surveys are defined by the RWP applicable to the decommissioning task.

The basis for the survey of surface contamination and limits as provided through the site contamination control and radiological monitoring procedures for the HDP Site are provided in Table 10-3, which is reproduced from NRC Policy and Guidance Directive FC 83-23, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," (Reference 10-19). For a mixture of radionuclides with differing limits, the effective contamination limit may be derived by using the most conservative radionuclide present or by

the sum of the fractions reflecting the relative contributions of the radionuclides present. The RSO shall approve the sum of the fractions calculation.

10.7.2 SURVEYS OF AIRBORNE CONTAMINATION

Concentrations of airborne radioactive materials are measured within work areas where radioactive materials are handled, within adjacent areas to assess occupational exposure and at locations that are representative of the concentrations released to the environment. Refer to Section 10.2 for a detailed discussion.

10.7.3 PERSONNEL CONTAMINATION SURVEYS

Appropriate PPE shall be worn when entering or working in posted radiological areas where loose contamination levels exceed 1,000 dpm/100 cm^2, and when working in areas where the concentrations in soil are sufficient to result in skin or clothing contamination. The PPE worn shall be appropriate for the anticipated radiological conditions and will be specified by the applicable RWP. A contamination survey meter shall be provided at the exit of the contamination area.

Contamination monitoring requirements for personnel exiting radiologically posted areas are implemented through the site contamination control and radiological monitoring procedures. At a minimum, all personnel exiting a contamination area will be required to monitor their hands, feet, personal hand carried items, and to monitor other body surfaces and personal clothing as appropriate based on the potential for contamination. HP personnel will provide assistance as required to perform radiological surveys and assist with decontamination in the event that personnel contamination is identified, as well as to perform radiological surveys prior to the release of tools, equipment, and materials from contamination areas.

10.7.3.1 Personal Protective Clothing

The personal protective clothing used at the Hematite Site may be of the reusable or disposable type. If reusable clothing is used, laundering will be done by a commercial vendor licensed to receive radioactive material. Contaminated disposable protective clothing will be dispositioned to a licensed waste vendor for processing/disposal. The contamination limits for receipt of laundered protective clothing are listed below:

SURVEY	LIMIT
Re-usable Launderable PPE	5,000 dpm/100 cm^2 (Fixed)

10.7.4 EQUIPMENT AND FACILITY SURVEYS FOR UNRESTRICTED RELEASE

Surface contamination surveys will be performed for both removable and fixed contamination prior to unrestricted release of equipment and facilities. The unrestricted release limits for the Hematite Site are established under a special authorization to License SNM-33 (Reference 10-20). The surface contamination limits for HDP are provided in Table 10-3.

10.7.5 RADIOACTIVE MATERIAL PACKAGE SURVEYS

Radiological survey for the shipment, receipt and opening of packages containing radioactive material will be accomplished utilizing the approved site implementing procedures. These procedures incorporate the requirements of 10 CFR 20.1906 and the requirements of 49 CFR, "Transportation," (Reference 10-21). Radioactive Material Shipment records will be retained in accordance with Westinghouse and site quality assurance requirements. Table 10-4 presents the contamination limits for packages containing radioactive material that are prepared for shipment.

10.7.6 LEAK TEST OF SOURCES

The RSO shall have responsibility for the use of non-exempt sealed sources for training and instrument calibration. When not in use, non-exempt sealed sources will be stored in a labeled container in a manner that prevents unauthorized removal or use. Radioactive sources shall be leak tested prior to use after receipt and every six months thereafter. The leak test shall be capable of detecting the presence of 0.005 μCi of removable activity on the test sample. Any source with removable activity equal to or greater than 0.005 μCi shall be immediately withdrawn from service. The RSO shall prepare a report to the NRC for submittal within five days of determining that a source has exceeded the 0.005 μCi leak test action limit.

10.7.6.1 Non-exempt Sealed Source Accountability

Non-exempt sealed sources will be assigned an individual tracking number which will be maintained in a radioactive source log and inventory. A physical inventory will be conducted at least every six months and documented in accordance with the radioactive source control procedure. Any discrepancies will be reported to the RSO.

10.7.7 BACKGROUND RADIATION SURVEYS

10.7.7.1 Background Radiation Levels

TLDs will be placed at the perimeter of the impacted area as an indicator of the potential exposure to the public as a part of the environmental monitoring program. The observed response of these TLDs will be corrected for the contribution from naturally-occurring sources of radioactivity. This correction will be the result of subtracting the response of a TLD placed at a

location unaffected by the licensed activities in order to obtain an indication of the potential exposure that could occur within area in close proximity to the decommissioning.

TLDs that are used to measure occupational exposure will be corrected for the response due to sources of radioactivity when not being worn, and any inadvertent exposure to radiation during transportation. Control TLDs will be maintained at the TLD storage location outside of the restricted area, and the response of these TLDs will be used to correct the response of those worn by workers. This is done so that the TLDs for the purpose of measuring occupational exposure reflect only the exposures that occurred while engaged in activities associated with decommissioning.

10.7.7.2 Background Concentrations in Air

An air sampler is placed at a location unaffected by the licensed activities in order to obtain background radioactivity concentrations in air. The final analysis of air samples collected as a part of the environmental monitoring program, or for assessing occupational internal exposure may be delayed for a period of time sufficient for the radioactive decay of Radon and Thoron daughter products.

10.7.7.3 Background Survey for Soil

Background information for soil is discussed in detail in DO-08-003, "Hematite Radiological Characterization Report," (Reference 10-22).

10.7.8 SURVEY RECORDS

Contamination survey records will be prepared in accordance with approved site procedures utilizing the same units as provided in 10 CFR 20. Contamination survey records will be retained in accordance with Westinghouse and HDP quality assurance requirements.

10.8 INSTRUMENTATION PROGRAM

The HDP Site maintains an adequate number of radiological instruments of sufficient accuracy and sensitivity to ensure the radiation monitoring and measuring requirements of the RPP are met. The RPP provides guidance on the use, calibration and maintenance of radiological instrumentation and the guidance is implemented through approved site procedures. Consistent with ANSI-N323A-1997 "Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments," (Reference 10-23), radiological instrumentation, including flowmeters, velometers, rotameters, and orifices shall be calibrated annually at a minimum.

Calibration is also required after maintenance, repair, or adjustment likely to affect the primary calibration, or as recommended by the instrument manufacturer. Sources used for instrument calibration or to establish instrument efficiency shall be National Institute of Standards and Technology (NIST) traceable.

For the purpose of implementing the HDP operational health physics program, the instrument efficiencies are based on a 4π geometry. The instrument efficiency values are determined using NIST traceable sources having energies that are comparable, or conservative, with respect to the energies of the radionuclides that are present. These methods are consistent with ANSI N323A-1997, *American National Standard Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments* (Reference 10-23).

10.8.1 INSTRUMENT CALIBRATION

10.8.1.1 Quality Assurance

Quality assurance requirements related to radiation detection instruments are provided in DP Chapter 13.0.

Uncertainty bounds are established at ± 3 standard deviations of the mean source count for bench counters and ± 20% of the expected source response for friskers, portable scalars, and dose rate meters.

10.8.1.2 Portable Instrumentation

HDP does not maintain a calibration facility on-site. Portable instrumentation requiring calibrations are calibrated at a qualified vendor facility within the required calibration frequency for the instrument.

10.8.1.3 Stationary Instrumentation

The stationary counting system is calibrated and operated in accordance with the site procedure within the required calibration frequency.

10.8.1.4 Air Sampling Equipment

Air sampling equipment requiring calibration is calibrated at a qualified vendor facility.

10.8.2 INSTRUMENT INVENTORY

HP maintains an inventory log of available instruments, instruments out of service, instruments sent to calibration and instrument calibration due dates. Table 10-5 provides a listing of the anticipated types of radiological instrumentation that may be required to support the planned decommissioning activities. Table 10-6 provides a listing of the different types of radiological instrumentation currently maintained at the Hematite Site.

10.8.3 INSTRUMENT STORAGE, CALIBRATION AND MAINTENANCE FACILITY

The HDP Site does not maintain an on-site calibration facility for instrumentation or an on-site counting laboratory with sufficient capabilities to accommodate the types and quantity of radiological sample analysis necessary to support the planned decommissioning activities. Subsequently, HDP will procure the services of qualified off-site vendors to supplement the instrument calibration and radiological sample analysis needs of the project.

HDP does maintain an on-site counting laboratory with sufficient capabilities to accommodate the stationary counting system. HDP also maintains a storage facility of adequate size to accommodate the storage and maintenance of the radiological instrument inventory. Access to the radiological instrument storage is locked and admittance to the area is controlled by HP Supervision.

10.8.4 MDC AND MDA CALCULATIONS

Equipment and materials being surveyed for unrestricted release shall be surveyed with radiological instrumentation capable of achieving a MDA sufficient to detect activity at the limits specified in NRC Directive FC 83-23. Air samples will be taken with sufficient sample volume to achieve a sufficient MDC and analyzed with radiological instrumentation with a MDA sufficient to detect airborne radioactivity at the established airborne monitoring limits. The equations used to calculate MDC and MDA at a 95 percent confidence level are as follows:

Air Sample MDC **(10-1)**

$$\text{Air Sample MDC (uCi/ml)} = \frac{3 + 3.29\sqrt{(R_b)(T_g)\left(1 + \frac{T_g}{T_b}\right)}}{(\varepsilon_i)(\varepsilon_c)(T_g)(V_s)(2.22E^9)}$$

Where;	V_s	=	sample volume (liters)
	ε_i	=	instrument efficiency-intra-
	ε_c	=	collection efficiency (default 0.99)
	R_b	=	background count rate (cpm)
	T_b	=	background count time (minutes)
	T_g	=	gross count time (minutes)
	$2.22E^9$	=	conversion factor (dpm to uCi and liters to ml)
	3	=	derived constant based on Type I and Type II errors of 0.05 (NUREG-1507, Sect 3.1)
	3.29	=	derived constant based on the 95 percent confidence level (NUREG-1507, Sect 3.1)

For operational surveys, the MDC formulations equivalent to those in Section 14.4.4.2.5 are used with the exception that the efficiency term represents the instrument efficiency (ε_i) as opposed to a total weighted efficiency (ε_t).

The protocol for establishing the MDC for bench counter and scalar instrumentation for operational (non-FSS) surveys includes the following formulas for MDC calculation:

MDC for a Portable Counter (timed count) **(10-2)**

$$\text{MDC (dpm/100cm2)} = \frac{3 + 3.29\sqrt{(R_b)(T_g)\left(1 + \frac{T_g}{T_b}\right)}}{\frac{DA}{100}(\varepsilon_i)(T_g)}$$

Where;	DA	=	detector area (cm^2)
	ε_i	=	instrument efficiency (c/d)
	R_b	=	background count rate (cpm)
	T_b	=	background count time (minutes)
	T_g	=	gross count time (minutes)
	3	=	derived constant based on Type I and Type II errors of 0.05 (NUREG-1507, Sect 3.1)
	3.29	=	derived constant based on the 95 percent confidence level (NUREG-1507, Sect 3.1)
	100	=	conversion factor (detector area (cm^2) to 100 cm^2)

Bench Counter Smear MDC **(10-3)**

$$\text{Smear MDC (dpm/100cm2)} = \frac{3 + 3.29\sqrt{(R_b)(T_g)\left(1 + \frac{T_g}{T_b}\right)}}{(\varepsilon_i)(T_g)}$$

Where;	ε_i	=	instrument efficiency (c/d)
	R_b	=	background count rate (cpm)
	T_b	=	background count time (minutes)
	T_g	=	gross count time (minutes)
	3	=	derived constant based on Type I and Type II errors of 0.05 (NUREG-1507, Sect 3.1)
	3.29	=	derived constant based on the 95 percent confidence level (NUREG-1507, Sect 3.1)

10.9 NUCLEAR CRITICALITY SAFETY

10.9.1 NUCLEAR CRITICALITY SAFETY PROGRAM

The Nuclear Criticality Safety (NCS) program for the HDP complies with NRC regulations. The regulations applicable to NCS at the HDP Site are 10 CFR 70, "Domestic Licensing of Special Nuclear Material," (Reference 10-24). In addition to compliance with applicable NRC regulations, the HDP NCS program recognizes the following guidance documents, as applicable:

- Regulatory Guide 3.71, "Nuclear Criticality Safety Standards For Fuels And Material Facilities," (Reference 10-25); and
- ANSI/ANS 8 national standards.

Specific ANSI/ANS 8 national standards observed by HDP are outlined in Section 10.9.1.4.

10.9.1.1 Organization And Administration

The HDP NCS program fosters ownership of nuclear criticality safety by the HDP organization. The key organizational positions related to criticality safety are detailed in Section 10.9.1.1.2.

The HDP NCS program requires all personnel to report, without delay, defective NCS conditions directly to or through a designated supervisor.

10.9.1.1.1 NCS Organization

The NCS organization is independent of operations to the extent practical and is responsible for implementing NCS practices of ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," (Reference 10-26), and ANSI/ANS-8.19-2005, "Administrative Practices for Nuclear Criticality Safety," (Reference 10-27). The NCS organization has the authority and responsibility to shut down potentially unsafe HDP operations. Specific responsibilities of the NCS organization include the following:

- Establish the HDP NCS program, including any design criteria, procedures, and training;
- Produce documented Nuclear Criticality Safety Assessments of HDP operations to establish the NCS control modes (parameters), barriers (controls and requirements) and limits on modes/barriers necessary to ensure criticality safety under all normal and credible abnormal conditions;
- Produce Criticality Safety Calculations, as needed, to support NCSAs;

- Provide NCS support to operations;
- Review and approve HDP operations and operating procedures that involve fissionable material;
- Review, and approve as appropriate, any proposed change(s) to operations or Structures, Systems and Components (SSCs) relied on for safety, to establish whether the proposed change(s) impact the criticality safety basis of HDP;
- Assess the effectiveness of the NCS program through the audit/assessment program;
- Identify NCS posting requirements to support administrative controls in applicable work areas;
- Maintain NCS programs for HDP in accordance with applicable regulatory guides and industry standards;
- Be the single point of contact for nuclear criticality issues with internal and external groups or agencies;
- Provide incident analysis;
- Support and participate in the corrective action program; and
- Support emergency response planning and events.

The nuclear criticality process requires that upon identification of a defective NCS condition, the HDP organization take no further action not specified by approved written procedures or work packages, until an NCS Specialist has analyzed the condition and provided appropriate direction.

10.9.1.1.2 Key Personnel

The NCS organization is staffed by qualified engineers or technical staff with experience at nuclear facilities involving Special Nuclear Material (SNM).

The functional area manager for the NCS organization has the authority and responsibility to assign and direct activities for the NCS function.

NCS Specialists have the authority and responsibility to conduct activities assigned to the NCS function. The minimum qualifications for a NCS Specialist are: a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety.

10.9.1.2 Management Measures

The management practices for the HDP NCS program recognize the guidance on administration and technical practices provided in Reference 10-26.

10.9.1.2.1 NCS Training

The NCS practices and associated procedures comply with regulatory requirements and subscribe to the training requirements of Reference 10-27, and ANSI/ANS-8.20-1991, "Nuclear Criticality Safety Training," (Reference 10-28). The training is appropriately tailored to the decommissioning scope of the HDP and the function of the various HDP staff.

The NCS training shall be appropriate to the work conducted by the person, such as the following three levels: Basic Concepts in General Employee Training; Fissile Material Handler Training (FMHT); and Fissile Material Training for Supervisors and Managers. The Basic Concept training provided in General Employee Training would be provided to all project personnel to ensure understanding of criticality safety controls and postings. FMHT would be provided to Personnel at the project whose job function requires them to handle fissile material in quantities requiring Nuclear Criticality Safety control measures. FMTSM would be provided to Supervisors and Managers of personnel assigned to plan or perform work and all persons involved in planning work associated with fissile materials in quantities requiring Nuclear Criticality Safety control measures.

10.9.1.2.2 Audits and Inspections

Westinghouse Electric Company LLC (Westinghouse) uses an Audit and Inspection process to evaluate the effectiveness of the HDP NCS program and other programs to ensure that operations conform to criticality safety requirements and controls in accordance with Reference 10-27.

Assessments are management directed evaluations, within their area of responsibility, to assess the adequacy, programmatic compliance, and implementation effectiveness of the NCS program and other management measures.

Key elements of the management audit, inspection and assessment processes are summarized below:

1. Annual Audits

   Annual audits, in which the results of previous inspections or audits are reviewed, are conducted as an evaluation of the effectiveness of the HDP NCS program. To the extent practical, the person(s) performing NCS program audits will not have direct responsibility for the function and areas being audited.

The annual audit of the NCS program is conducted and documented by a formal report to the Project Director. The functional area manager shall assign responsibility regarding follow-up for recommendations made by the audit team.

2. Inspections

Planned and documented quarterly criticality safety inspections, including walk-down of areas or activities involving fissile material operations, are conducted by representative(s) of the NCS organization in accordance with approved written procedures.

Inspection findings and recommendations are communicated in accordance with the corrective action program as described in Chapter 13.0.

10.9.1.2.3 Procedures

Procedures are established and implemented in accordance with the Project Quality Plan and site requirements. NCS procedures also utilize guidance as provided in Reference 10-27.

10.9.1.2.4 Posting and Labeling

When specifically identified as necessary by the governing Nuclear Criticality Safety Assessment (NCSA), and to ensure personnel protection and awareness of criticality safety requirements, designated work and storage areas where SNM is recovered, handled, processed, or stored are posted with Nuclear Criticality Safety Signs (NCSS) applicable to that area. When so required, postings shall be approved by representative(s) of the NCS organization. The NCS Organization shall maintain the current list and control distribution of NCSS.

When specifically identified as necessary by the governing NCSA, SNM containers (containers used in the internal transport, handling, or storage of special nuclear material on the HDP Site) shall be labeled with an identification code and the amount and enrichment of SNM contained.

10.9.1.2.5 Change Management

HDP management shall review proposed changes to SSCs, hardware, software, processes and procedures to ensure that proposed facility changes are managed to maintain the integrity of the facility's safety basis and to ensure that proposed changes receive the appropriate level of NCS review. The NCS review assures that the ability of the Criticality Safety Controls (CSCs) to perform their function when needed is maintained.

The NCS organization reviews, and approves as appropriate, proposed changes to operations or SSCs relied on for safety, to establish whether the propose changes impact the criticality safety basis of HDP.

10.9.1.3 Methodology And Technical Practices

10.9.1.3.1 Nuclear Criticality Safety Assessment (NCSA)

As part of the endorsement of the NCS practices of Reference 10-26, the NCS organization is responsible for producing documented NCSAs of all HDP operations, before implementation, to establish:

- The NCS control parameters, barriers (controls and requirements) and associated limits necessary to ensure criticality safety under all normal and credible abnormal conditions; and

- To demonstrate that HDP operations incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality event could occur.

NCSAs are documented with sufficient detail and clarity such that an independent reviewer could reconstruct the analysis and bases for the conditions presented. NCSAs include the following documentation:

- A description of the operations and processes covered by the NCSA;
- A description of all assumptions important to the scope of the NCSA (i.e., important to the criticality safety basis);
- A description and assessment of normal conditions;
- A description and assessment of abnormal (i.e., unanticipated) conditions;
- Identification of controlled parameters, and their associated limits;
- Determination of CSCs, and their limits, necessary to provide double contingency protection against credible abnormal events, including assurance of resilience to common mode failure;
- Identification of any practicable defense-in-depth measures that provide further risk reduction, thus ensuring that the risk of criticality is acceptable; and
- A summary of any explicit calculations used in support of the assessment.

Features that ensure that the CSCs identified in the NCSAs are sufficiently available and reliable are provided through implementation of management measures (procedures, training, surveillance, etc.).

Computational methods will be validated in accordance with guidelines of ANSI/ANS-8.1-1998. Validations shall comply with the requirements of ANSI/ANS-8.24-2007, “American National

Standard, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations," (Reference 10-29).

Validation reports will be prepared, reviewed, and approved by qualified individuals for each combination of computational method (e.g., code), cross-section library, computer platform, and analytical area of applicability (e.g., homogenous UO_2 versus heterogeneous UO_2), as appropriate. In all cases, each validation report, or the calculation note documenting an analysis using a specific computational method, shall include the following:

(1) Demonstration of the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of $k_{eff;}$

(2) Demonstration that the calculation of k_{eff} is based on a set of variables whose values lie in a range for which the methodology used to determine k_{eff} has been validated; or demonstration that trends in the bias support the extension of the methodology to areas outside the areas of applicability;

(3) A description of the specific validation method used, including reference to input data, area of applicability, and discussion of the applicable uncertainties; and

(4) A description of data outliers rejected shall be based on inconsistency of the data with known physical behavior, and not on statistical rejection methods alone.

All NCSAs are subject to an independent documented review by a qualified NCS specialist reviewer.

10.9.1.3.2 Nuclear Criticality Safety Control Philosophy

The NCS control philosophy established for the HDP is based on the following hierarchy, in order of preference:

- Alternative Processes – In all instances, consideration is given to the removal of identified hazards by the use of other processes;
- Passive Engineered Controls (PEC) – These are design features which achieve a certain safety function through their inherent characteristics. An example of a PEC is a fixed physical barrier to provide separation of safe mass units. No human intervention is required for PEC, except for maintenance and inspection;
- Active Engineered Controls (AEC) – Controls that use active hardware to sense conditions and automatically place a system in a safe state or mode. Actuation and operation of these controls do not require human intervention;
- Enhanced Administrative Controls (EADM) – Controls that rely on human judgment, training, and actions governed by procedures for implementation, but are augmented by an active warning device (audible or visual) that prompts specific human actions; and

- Simple Administrative Controls (ADM) – Controls that rely solely on human judgment, training, and actions governed by procedures for implementation.

Decommissioning processes at the Hematite Site are selected according to the above order of preference, as reasonably achievable. For example, in instances where the selection of alternate processes would not remove an identified hazard, reliance is placed on equipment design that uses passive engineered controls (i.e., PEC), rather than on administrative controls (i.e., EADM or ADM).

10.9.1.3.3 Nuclear Criticality Safety Control Parameters

As part of the assurance of criticality safety of HDP operations under all normal and credible abnormal conditions, NCS control parameters are analyzed to determine the extent of their control and their associated limits. The determination of subcritical parameter limits employs standards, handbook values, or validated calculational methods. Specifically, the subcritical values provided in ANSI 8.1 may be used and safety factors may be applied to critical values provided in handbooks such as Los Alamos Report LA-10860, "Critical Dimensions of Systems Containing U-235, Pu-239,and Pu-233," (Reference 10-30), Los Alamos Report LA-12808, "Critical Safety in Processing Plants," (Reference 10-31) and Los Alamos Administrative Resource Handbook ARH-600. "Criticality Handbook," (Reference 10-32). The NCS control parameters analyzed for the HDP include the following:

- Geometry control;
- Interaction control;
- Mass control;
- Isotopic control;
- Moderation control;
- Density control;
- Concentration control;
- Reflection control;
- Neutron absorber control;
- Volume control; and
- Heterogeneity control.

Geometry-Geometry control involves the use of passive engineered devices to control geometry within ensured tolerances. Geometry controls are established in a manner that ensures an adequate margin of subcriticality, including margins to protect against uncertainties in process variables. Geometry control is used for HDP wherever possible, particularly for storage areas containing potentially large quantities of fissile material (for which mass control alone is not viable).

Geometry controls implemented for HDP are established cognizant of the following principles:

- Tolerances on credited design dimensions are treated conservatively.

- Dimensions of geometry controlled features are verified prior to their use for HDP operations, and routinely inspected under the HDP NCS program.
- Credible means of displacing fissile materials from a favorable geometry to a less favorable geometry are identified and evaluated, and controls established to ensure that such displacements are precluded.
- Possible mechanisms for changes to credited geometry are identified and evaluated, and controls established as necessary. Examples of mechanisms that could potentially result in changes to a credited geometry may include bulging, deformation or loss of structural integrity.

Whenever the possibility of neutron interaction with other fissile units exists, mass and/or interaction control may also be used, in conjunction with geometry control.

Interaction-Interaction control involves the use of spacing to limit neutron interaction between fissile units.

Interaction controls implemented for HDP are established cognizant of the following principles:

- When maintaining physical separation between fissile units, passive engineered features (i.e., spacers or other passive geometrical means) are used to the extent practical.
- Conditions that could potentially compromise the structural integrity of any engineered features used for maintaining physical separation between fissile units are identified and evaluated, and controls established as necessary.
- When the physical separation between fissile units is controlled by procedure, it is demonstrated that no single event can cause a spacing upset that can result in a possible criticality.
- When evaluating the criticality safety of units in an array or pairs of arrays, the spacing limits in ANSI/ANS-8.7-1998, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials," (Reference 10-33) are used, or spacing is based on computational methods described below.

Mass-Mass control involves constraining the SNM mass associated with a single fissile unit to limit subcritical multiplication of the system.

Mass control may be used for HDP operations to ensure the safety of SNM identified and recovered during remediation of the site, particularly SNM discovery and exhumation operations where recovered materials containing SNM may initially encompass a wide range of geometric and volumetric conditions.

When mass is controlled, mass balance logs will be established and maintained for the respective

isolated functional area(s) of the site. Mass limits applicable to designated (and isolated) SNM storage areas may also be based on container mass limits, as opposed to a functional area total SNM mass limit.

Mass controls implemented for HDP are established cognizant of the following principles:

- SNM mass estimates used by fissile material handlers in complying with mass control limits are based on measurements using calibrated instruments and may include gamma assay, surface activity measurements or weight.
- The mass limits are based on a spherical geometry, unless geometry is controlled, with optimum moderation and credible full reflection conditions.
- Mass limits are established with a consideration of credible over-batching.
- The possibility of accidental accumulation of SNM is evaluated, and controls established, as necessary, to ensure that accidental accumulation of SNM is precluded.

Isotopic (enrichment)-Isotopic control involves taking credit for controls that assure a maximum, non-bounding, enrichment of SNM. Unless specific and independently confirmed isotopic analysis is available to conservatively identify a bounding enrichment for a specific application, the HDP NCSAs will conservatively assume that all identified SNM is associated with Highly Enriched Uranium (HEU) with an isotopic distribution of 100 wt. % 235U/U. Consequently, isotopic controls are not normally implemented for the HDP.

Moderation-Moderation control involves taking credit for controls that assure non-optimal SNM moderation.

SNM identified and recovered during remediation of the Hematite Site may be commingled with a wide variety of materials; potentially including materials with efficient moderating properties (e.g. hydrocarbon based materials). Furthermore, saturation of recovered SNM with water may be possible due to the subsurface environment of buried process wastes on the Hematite Site. Based on these considerations, the HDP NCSAs conservatively assume bounding optimal moderation conditions for SNM, including moderation by hydrocarbons, if credible. Consequently, moderation controls are not implemented for HDP.

Owing to the bounding moderation assumptions used in the HDP NCSAs, external sources of moderation from the process (including fire suppression) and external sources of moderation from the environment (such as ground water, rain and snow) are not controlled, although precautions are taken to minimize the presence of moderators, where practicable.

Density-Density control involves taking credit for controls that assure a maximum, non-optimal, SNM density. The HDP NCSAs assume conservative SNM densities when evaluating the safety of waste materials generated and exhumed during remediation of the site. Conditions that could

potentially result in more onerous SNM densities are identified and evaluated, and controls established as necessary.

Concentration-Concentration control involves taking credit for controls that assure a maximum, non-optimal, SNM concentration. The HDP NCSAs assume conservative SNM concentrations when evaluating the safety of waste materials generated and exhumed during remediation of the site. Conditions that could potentially result in more onerous SNM concentrations are identified and evaluated, and controls established as necessary.

Reflection-Reflection control involves taking credit for controls that assure a maximum, non-bounding, degree of neutron reflection. The HDP NCSAs assume conservative reflection conditions when evaluating the safety of individual fissile units and arrays. Conditions that could potentially result in an increase in reflection of individual fissile units or arrays are identified and evaluated, and controls established as necessary.

Neutron Absorber-Neutron absorber control involves the use of material(s) with a significant neutron absorption cross-section to limit subcritical multiplication of a single fissile unit or to limit neutron interaction between multiple (spaced) fissile units.

Neutron absorber controls are not implemented for HDP, however the neutron absorption properties of materials may be considered in validated calculational methods to assess the degree of conservatism in limits on other parameters.

Volume-Volume control involves constraining the volume occupied by a single fissile unit to limit subcritical multiplication of the system. Volume control may also be applied to arrays of fissile units when the combined volume of the individual fissile units is less than the maximum safe subcritical volume documented in standards, handbook values, or validated calculational methods. The maximum safe subcritical volume used for HDP operations is based on bounding optimum moderation, spherical geometry and full reflection conditions.

For arrays with a combined fissile unit volume exceeding the maximum safe subcritical volume, and for which the possibility of neutron interaction with other fissile units exists, interaction control may also be used in conjunction with volume control.

Heterogeneity-Heterogeneity control involves taking credit for controls that assure a non-bounding distribution of fissile material.

Heterogeneity controls are not implemented for HDP; however, fissile material distribution may be considered in validated calculational methods to assess the degree of conservatism in limits on other parameters (i.e. margins of safety).

10.9.1.3.4 Nuclear Criticality Safety Calculations

To establish that a system is subcritical under normal and credible abnormal conditions, it is necessary to establish acceptable sub-critical limits for the operation, and then show that the proposed operation will not exceed the established subcritical limit. This demonstration may require criticality safety calculations employing validated computational methods with verified software, including use of a documented acceptable margin of sub-criticality.

Criticality safety calculations performed in support of HDP NCSAs employ bounding assumptions for parameters that are not controlled. For example, calculations may consider:

- Optimum moderation, including moderation by hydrocarbons, if credible;
- Spherical geometry;
- Close fitting full reflection; and
- Bounding 100 weight percent Uranium-235 (U-235) / Uranium (U) enrichment.

10.9.1.3.5 Computational Methods

Criticality safety calculations performed in support of HDP NCSAs may employ validated computational methods. The MCNP 5 computer code system is the primary software used for this purpose. MCNP 5 is a three dimension neutron transport code that is well suited to the determination of the effective neutron multiplication factor (k_{eff}) of systems containing fissile material. The MCNP 5 code software is available from the Radiation Safety Information Computational Center.

Note that other validated computation methods (e.g., the SCALE code package) may be used for HDP criticality safety calculations provided they meet project requirements outlined in this chapter.

Criticality safety for a single planar array of drums, waste storage boxes, and shipping packages may be determined by using the areal density method. The maximum mass per container must not exceed 45% of the minimum critical mass and the maximum areal density shall not exceed 0.19 $g^{235}U/cm^2$.

10.9.1.3.5.1 <u>Criteria To Establish Subcriticality</u>

The HDP NCSAs demonstrate that under normal and credible abnormal conditions, nuclear processes are subcritical, including use of an approved margin of subcriticality for safety. When computational methods are used to establish subcriticality of a configuration, the computed maximum k_{eff} plus calculational uncertainties is compared to the configuration-specific Upper Subcritical Limit (USL).

The USL takes into account bias, uncertainties, and administrative and/or statistical margins, such that the calculated configuration is subcritical with a high degree of confidence. In equation notation, the use of the USL is defined as follows:

$$k_{eff} + \Delta k_{eff} \leq USL \quad \textbf{(10-4)}$$

$$USL = k_c - \Delta k_c - \Delta k_m \quad \textbf{(10-5)}$$

where:

k_{eff} = the calculated effective neutron multiplication factor for the system/configuration analyzed.

Δk_{eff} = a value to account for the uncertainty in the value of k_{eff}. Sources of uncertainty include:
- statistical and convergence uncertainties in the computation of k_{eff};
- material and fabrication tolerances; and
- uncertainties due to the geometric or material representations used.

k_c = the mean k_{eff} value obtained from the calculation of applicable benchmark critical experiments using a specific calculation method and data.

Δk_c = a value to account for the uncertainty in the value of k_c. Sources of uncertainty include:
- uncertainties in critical experiments;
- statistical and/or convergence uncertainties in the computation;
- extrapolation outside the range of experimental data; and
- limitations in the geometric and/or material representations used.

Δk_m = an administrative margin to ensure subcriticality.

10.9.1.3.5.2 <u>Upper Subcritical Limit (USL)</u>

The USL applied to computational methods performed in support of HDP NCSAs is documented and established in a validation report developed for HDP, which describes:

- Selection and justification of the administrative margin to ensure subcriticality; and
- Determination of the calculation bias and uncertainties for each Area of Applicability.

The administrative margin used for computational methods are selected consistent with the guidance provided in ANSI/ANS-8.24-2007 (Reference 10-29). Based on these guidance documents, a conservative 0.05 administrative margin is used for all for computational methods performed in support of HDP NCSAs.

The USLs and associated Areas of Applicability (AOAs) relevant to use of the MCNP 5 code for computational methods performed in support of HDP NCSAs are summarized in Table 10-7.

10.9.1.4 Observed Guidance Documents

Regulatory Guide 3.71 (Reference 10-25), endorses specific NCS standards drafted by Subcommittee ANS-8 (*Fissionable Materials Outside Reactors*) of the ANS Standards Committee.

The HDP NCS program includes use of ANSI/ANS standards endorsed by Regulatory Guide 3.71 as guidance documents for the HDP NCS program. The standards utilized by the HDP are listed in this section. The ANSI/ANS-8 series of Standards are applicable to HDP operations. The HDP NCS program will be set up in a manner consistent with the following standards, but will only implement requirements directly applicable to HDP operations as specifically invoked in the governing NCSA.

ANSI/ANS-8.1-1998 (Reference 10-26), provides general guidance addressing administrative and technical practices, as well as single-parameter and multiparameter control limits for systems containing U-235.

The scope of guidance provided in ANSI/ANS-8.1-1998 is applicable to HDP operations.

ANSI/ANS-8.7-1998 (Reference 10-33), provides general guidance for technical practices related to SNM storage as well as parameter limits and conditions related to SNM storage.

The technical practices outlined in ANSI/ANS-8.7-1998 are generally applicable to HDP operations. The HDP criticality safety calculations and NCSAs provide explicit basis for parameter limits and controls related to SNM storage.

ANSI/ANS-8.19-2005 (Reference 10-27), provides criteria for the administration of a NCS program for operations outside reactors, for which there exists a potential for criticality events.

The scope of guidance provided in ANSI/ANS-8.19-2005 is applicable to HDP operations.

ANSI/ANS-8.20-1991 (Reference 10-28), provides detailed guidance for NCS training for personnel associated with (non-reactor) operations where a potential exists for criticality events.

The scope of guidance provided in ANSI/ANS-8.20-1991 is applicable to HDP operations.

ANSI/ANS-8.24-2007 (Reference 10-29), provides guidance for criticality code validation.

The scope of guidance provided in ANSI/ANS-8.24-2007 is applicable to HDP operations.

10.9.2 DECOMMISSIONING NUCLEAR CRITICALITY SAFETY BASIS

The NCS basis for the decommissioning activities described in Chapter 8.0 of the Decommissioning Plan for the Hematite Site is summarized in this section. The following structure is used:

- Section 10.9.2.1 provides an overview of the explicit decommissioning activities;
- Section 10.9.2.2 provides a description of NCS parameters, including a summary of the extent of their control for the decommissioning activities outlined in Section 10.9.2.1; and
- Section 10.9.2.3 lists and summarizes the criticality hazards evaluated in the HDP NCSAs and provides a summary of the hazard evaluation results and controls established to ensure criticality safety.

10.9.2.1 Overview Of Decommissioning Activities

A detailed description of the explicit decommissioning activities to support final decommissioning of the Hematite Site is provided in Chapter 8.0. An overview of these activities, from a NCS perspective, is provided below.

10.9.2.1.1 Buried Waste and Contaminated Soil Remediation

Relative to removal of buried wastes, contaminated soils and sub-surface structures (e.g., concrete slabs, buried piping) in areas where it has been determined the presence of fissile materials is a reasonable possibility based on characterization data and historical knowledge, HDP has developed consistent generic screening and handling approaches. These approaches have been analyzed from a NCS perspective in NCSAs specific to buried waste exhumation, contaminated soil remediation, and sub-surface structure decommissioning. Screening for fissile materials during remediation is the initial goal. Screening typically will involve duplicate performance of radiological surveys, using sodium iodide scintillation detectors, of defined volumes of material to ensure that NCS limits have not been exceeded.

The objective of the in-situ radiological surveys is to identify materials that do not satisfy the NCS exempt material criteria. Unless otherwise defined and justified within a nuclear criticality safety evaluation, NCS Exempt Material is conservatively defined as material containing U-235 with an average nuclide fissile concentration not exceeding 0.1 g U-235/L, or material that comprises no greater than 15 g U-235 and is enclosed within a container with a volume of at least 5 liters.

The in-situ radiological surveys are complemented by visual inspection of the survey area with the aim of identifying the following items, which are treated as non-NCS exempt material unless proved otherwise by sampling, radiological survey, or other method(s) permitted in an approved NCSA:

1) Items with the potential to contain fissile material (e.g., a process filter);
2) Items that resemble intact containers;
3) Bulky objects with linear dimensions exceeding the permitted excavation '*cut depth'*; and
4) Metallic items.

During fissile material screening, the secondary remedial action objectives of identifying hazardous materials (e.g., Volatile Organic Compounds or VOCs), and verifying radioactivity concentrations of potential backfill soils will also be completed. Excavation will continue in areas of suspected fissile materials to a depth where historical knowledge, and/or visible and radiological evidence indicate that suspect materials have been removed. Once fissile material screening has determined that fissile materials are not present in the remediation area in excess of the NCS Exempt Material Limit, NCS controls are then curtailed. By making this the initial goal, remaining remediation, Health Physics, and Final Status Survey activities can proceed unencumbered by NCS controls. Identified items that exceed NCS limits will be segregated into designated Field Containers, which are placed in transport containers such as Collared Drums (CDs). Individual designated containers will he handled and stored in accordance with NCSA requirements in the work areas described below. Where discussed here and below, Field Containers are used when practical.

10.9.2.1.2 Water Collection and Treatment

Soil and ground water at the site have been contaminated with constituent from former processing operations such as VOCs and radioactive materials (Uranium and Technetium). During excavation and recovery of contaminated solid wastes from the burial pits, a considerable amount of water is expected to intrude into the open excavations, including ground water seepage and rainwater. This water will be evacuated from the excavations to allow recovery of the buried wastes. Water removed during this process will be treated to remove entrained and soluble contaminants and tested prior to release to the water outfall.

Water will be generated from the following sources:

- Ground water associated with buried waste (leachate);
- Ground water that may seep into the excavation from surrounding soil;
- Precipitation that falls directly into an excavation; and
- Precipitation that falls onto waste and adjacent outside waste-processing areas.

Treatment of the water collected from, in and around the burial pit excavations involves a number of collection and treatment stages:

- Collection of leachate, ground, and storm water from an excavation or wells;
- Settling of course solids in holding tanks;
- Filtration and volatiles adsorption; and
- Filtration and ion exchange polishing.

Water treated for release to the environment is strictly regulated and requires removal of radiological contamination to very low levels. The water pumped from the burial pit excavations or wells is not expected to entrain significant (e.g., g/L level) concentrations of fissile contamination and previous routine sampling programs support this view. The low concentration of fissile material entrained in liquids collected during burial pit remediation activities is an underlying basis for safety of the Water Treatment System (WTS). The waste water is expected to contain pCi/L levels of contamination. At such a small concentration, millions of liters of water must be processed before even a minimum critical mass of 760 grams U-235 (for optimum conditions) could pass through the treatment equipment. Simple periodic cleanout of the holding tanks and filter media is expected to reduce and maintain any fissile contamination from the ground water to minimal levels such that potential accumulation of fissile material over time, as settled out/captured solids in holding tanks and filtration / treatment media, will not present a criticality risk. Also, the equipment will be periodically inspected, sampled, or scanned to provide further assurance that significant fissile material quantities/concentrations are not accumulated.

The waste water treatment equipment installed for the excavation activities will also collect entrained fissile material as solid wastes in various filtration steps. The solids generated by these processes will require recovery, containerization, handling, characterization and storage as a SNM package, or disposition as NCS exempt material, which is not subject to NCS controls.

The underlying criticality control philosophy for the collection and containerization of solids from the WTS (e.g., holding tank sediment and used treatment media) is based on limitation of SNM mass to within established maximum safe mass limits. This is achieved by collecting solids with an average concentration exceeding 0.1 $g^{235}U/L$ from the WTS into CDs, which together with established maximum SNM concentration (from radiological surveys or sampling and analysis), will ensure that an unsafe mass of SNM cannot be collected into a CD. Following collection of solids into a CD, the CD is sealed and transferred to the Waste Evaluation Area (WEA) for evaluation or directly to a Material Assay Area (MAA) for assay, to establish the total U-235 content.

NCS controls for the WTS are described in an NCSA specific to water collection and treatment activities.

10.9.2.1.3 Collared Drum Transit, Staging and Buffer Storage

Depending on the generation rate of Field Containers during remediation work, it may be operationally advantageous to stage CDs loaded with Field Containers in a protected area local to the excavation site(s). Each Collared Drum Storage Area (CDSA) will be comprised of a clearly delineated area that has an even and level terrain and is protected against accident disturbance (e.g., impact from moving excavation equipment or waste transfer vehicles) by a robust physical barrier (e.g., concrete barricades similar to the type used to provide median barriers for road traffic). Due to material security requirements (i.e., shelter from precipitation and the

environment) all CDs staged in a CDSA during an operation shift are removed and placed into a Collared Drum Buffer Store (CDBS) prior to completion of the operations shift.

Depending on the generation rate of Field Containers during remediation work and the availability of WEAs and MAAs, it may be necessary to interim store CDs loaded with Field Containers in a CDBS. CDBSs may also be used for interim storage of any other CDs loaded with containers with un-assigned fissile mass content. For example, CDs loaded with Assay Containers may be stored on an interim basis in a CDBS in the event that a MAA is not available.

Each CDBS will be established within an enclosed area to protect the CDs within their environs against accident disturbance and to afford access control and shelter from precipitation and the environment.

NCS controls for the CDSA are described in an NCSA specific to collared drum staging, buffer storage and transit.

10.9.2.1.4 Waste Evaluation and Assay

All fissile materials, suspect fissile materials, items resembling intact containers identified during operations in an HDP remediation area will be containerized in CDs (no more than 350g U-235 total per CD unless otherwise stated in an NCSA) and ultimately transferred to a WEA for evaluation of fissile content. In addition, solids recovered from the WTS, bulky objects and/or metallic items with an average concentration exceeding 0.1 g U-235/L will be containerized in CDs and may be transferred to a WEA for evaluation of fissile content. Each WEA is equipped with one sorting tray and gamma survey instrument(s) and is bounded by a clearly defined perimeter.

When specifically identified as necessary by the governing NCSA, SNM containers (containers used in the internal transport, handling, or storage of special nuclear material on the HDP Site) shall be labeled with an identification code and the amount of fissile nuclides U-235 contained.

Following WEA operations, fissile materials are transferred to a MAA for radiological counting to establish U-235 mass content. The general aim of waste evaluation operations is to identify and extract Uranium associated with the material content of an incoming CD that would be of concern from a NCS perspective. Identification of Uranium is achieved by spreading/disassembling the material content of the CD across the surface of a sorting tray, and using hand-held gamma survey instruments (e.g., sodium iodide scintillation detectors) and visual inspection to locate the Uranium.

NCS controls for the waste evaluation and assay activities are described in an NCSA specific to waste evaluation and assay activities.

10.9.2.1.5 Collared Drum Repacking Area (CDRA) Operations

Depending on the generation rate of Fissile Material Containers during remediation work and their fissile loading, it may be desirable to consolidate the content of CDs. This could be particularly likely in the event of storage of a large number of CDs in a Fissile Material Storage Area (FMSA), with the majority of CDs containing only small quantities of fissile material. CDRAs may be used for these CD repacking operations. For efficiency and to reduce CD handling, loaded CDs identified as potentially requiring repacking may also be interim stored in a CDRA.

Any CDRAs used for HDP operations will be established within an enclosed structure (e.g., a building) to protect the CDs within their environs against accident disturbance and to afford access control and shelter from precipitation and the environment.

NCS controls for the CDRAs are described in an NCSA specific to collared drum repacking areas.

10.9.2.1.6 Fissile Material Storage Area Operations

FMSAs are used to store CDs or De-collared Drums (DCDs) with known fissile content. Operations conducted in a FMSA concern only CD receipt, de-collaring, storage, re-collaring (as necessary) and export. FMSAs employed for HDP operations will be established within an enclosed structure (e.g., a building) to protect the CDs within their environs against accident disturbance and to afford access control and shelter from precipitation and the environment.

NCS controls for FMSAs are described in an NCSA specific to fissile material storage areas.

10.9.2.2 NCS Parameters

As part of the assurance of criticality safety of HDP operations under all normal and credible abnormal conditions, NCS control parameters are analyzed to determine the extent of their control and their associated limits. The NCS control parameters analyzed for the HDP include:

- Geometry Control;
- Interaction Control;
- Mass Control;
- Isotopic Control;
- Moderation Control;
- Density Control;
- Concentration Control;
- Reflection Control;
- Neutron Absorber Control;
- Volume Control; and
- Heterogeneity Control.

The results of the assessment of criticality safety parameters for HDP operations outlined in Section 10.9.2.1 are summarized in the following Sections. The parameters are grouped according to two categories; controlled and not controlled parameters.

10.9.2.3 Controlled Parameters - Summary

This section summarizes the criticality safety parameters which are controlled, as applicable, for the following activities (see Section 10.9.2.1):

- Buried waste and contaminated soil remediation;
- Sub-surface structure decommissioning;
- Collared Drum transit, staging and buffer storage;
- Waste evaluation and assay;
- Collared Drum repacking; and
- Fissile material storage.

There are no controlled criticality safety parameters for water collection and treatment activities, because no credible criticality accident sequences have been identified for the associated operations.

Of the NCS parameters listed in Section 10.9.2.2, only those which are controlled are listed below. For each controlled criticality safety parameter, safety assessment results are summarized only for the applicable activities.

10.9.2.3.1 Geometry Control

The safety assessment of CD staging, buffer storage, and transit activities credits the geometry of CDs.

Supporting criticality safety calculations have demonstrated that DCDs and CDs stored in a planar array, within a CDRA or a Fissile Material Storage Area (FMSA), are safely subcritical. However, since these calculations (which are used in the safety assessment of CD repacking activities) credit the dimensions of De-Collared Drums (DCDs) and CDs, the geometry of DCDs and CDs is important to the safety basis of CDRAs.

10.9.2.3.2 Interaction Control

The safety assessment credits administrative CSCs to ensure safe interaction between fissile materials during the following activities: buried waste exhumation; contaminated soil remediation; sub-surface structure decommissioning; and waste evaluation and assay.

The safety assessment credits the physical design of CDs for CD staging, buffer storage and transit activities. The CD design incorporates a collar that provides a fixed distance between

adjacent CDs; this collar remains in place any time the CD is being used, except when secured in a FMSA or CDRA, or other such approval by an NCSA.

The safety assessment credits administrative CSCs that prohibit CD and DCD stacking during CD repacking or fissile material storage activities; thus, preventing an unanalyzed neutron interaction in a CDRA or FMSA.

10.9.2.3.3 Mass Control

The safety assessment establishes numerous administrative CSCs where mass control is employed during the following activities: buried waste exhumation; contaminated soil remediation; sub-surface structure decommissioning; waste evaluation and assay; CD repacking; and fissile material storage.

The safety assessment credits a limiting fissile mass content for any loaded CD that is unlikely to be exceeded during CD staging, buffer storage and transit activities.

10.9.2.3.4 Concentration Control

The safety assessment credits administrative CSCs to ensure maximum safe concentration limits are not exceeded in a Waste Holding Area (WHA) during the following activities: buried waste exhumation; contaminated soil remediation; and sub-surface structure decommissioning.

The safety assessment does not directly credit concentration control for CD staging, buffer storage and transit activities. However, concentration control is indirectly credited, since it is unlikely any loaded CD would exceed a limiting fissile mass content.

The safety assessment credits administrative CSCs to ensure that fissile material is identified during waste evaluation operations and assay activities.

10.9.2.3.5 Volume Control

The safety assessment credits administrative CSCs that ensure fissile materials will be limited in volume for the following activities: buried waste exhumation; and contaminated soil remediation.

The safety assessment credits administrative CSCs that ensure fissile materials will be packaged into limited-volume Field Containers during sub-surface structure decommissioning activities.

The safety assessment does not directly credit volume control for CD staging, buffer storage and transit activities. However, volume control is indirectly credited, since it is unlikely any loaded CD would exceed a limiting fissile mass content.

The safety assessment credits administrative CSCs that ensure loose fissile materials evaluated in a WEA, at any one time, will be limited to an analyzed safe volume; additionally, the safety assessment credits administrative CSCs that ensure loose fissile materials loaded into an Assay Container, following evaluation in a WEA, will be limited in volume. These administrative CSCs apply to waste evaluation and assay activities.

10.9.2.4 Parameters Not Controlled - Summary

This section summarizes the criticality safety parameters which are not controlled, as applicable, for the following activities (established in the HDP NCSAs):

- Buried waste and contaminated soil remediation;
- Sub-surface structure decommissioning;
- Water collection and treatment;
- Collared Drum transit, staging and buffer storage;
- Waste evaluation and assay;
- Collared Drum repacking; and
- Fissile material storage.

The basis for not controlling the criticality safety parameters listed below is provided. Bounding conservative assumptions and consideration of the practicality, suitability and viability of establishing control over a particular parameter were used to establish the basis for not controlling parameters.

Only those criticality safety parameters identified in Section 10.9.2.2 which are not controlled are listed below. For each criticality safety parameter that is not controlled, safety assessment results are summarized only for the applicable activities.

10.9.2.4.1 Geometry Control

Geometry control is considered impractical for operations in HDP remediation areas and WEAs. Therefore, the safety assessment does not credit geometry control, or specific dimensions of equipment or containers for the following activities: buried waste exhumation; contaminated soil remediation; sub-surface structure decommissioning; and waste evaluation and assay.

The safety assessment of the individual WTS equipment is conservatively based on idealized, simple geometries.

10.9.2.4.2 Interaction Control

The safety assessment of WTS equipment is conservatively based on collection and retention/deposition of 100 percent of the Uranium contaminants in the respective equipment.

10.9.2.4.3 Mass Control

Mass upsets with the potential to exceed a maximum safe mass in a single location have been established as not credible for water collection and treatment activities.

10.9.2.4.4 Isotopic Control

The safety assessment is conservatively based on sub-critical limits derived for Uranium with 100 weight percent U-235/U enrichment; therefore, isotopic controls are not established for any of the activities listed in this Section. Allowance is provided for use of bounding enrichments less than 100 weight percent based on analysis results and defined work scopes in subsequent NCSAs.

10.9.2.4.5 Moderation Control

The safety assessment is conservatively based on sub-critical limits derived for Uranium-water mixtures at optimum concentration for the following activities: buried waste exhumation; contaminated soil remediation; water collection and treatment; and waste evaluation and assay.

The potential for the presence of hydrogenous solutions with moderating properties more effective than water has been evaluated in the HDP NCSAs, and it has been determined that a criticality incident due to co-mingling of contaminated solids or buried wastes with hydrogenous solutions other than water is not credible.

The safety assessment is conservatively based on bounding, credible moderation conditions for the following activities: sub-surface structure decommissioning; Collared Drum repacking; and fissile material storage.

The safety assessment does not credit moderation control for CD staging, buffer storage and transit activities.

10.9.2.4.6 Density Control

The safety assessment is conservatively based on sub-critical limits derived for Uranium metal at maximum theoretical density for the following activities: buried waste exhumation; contaminated soil remediation; sub-surface structure decommissioning; CD staging, buffer storage and transit; waste evaluation and assay; CD repacking; and fissile material storage.

10.9.2.4.7 Concentration Control

Concentration upsets with the potential to exceed maximum safe concentration limits have been established as not credible for water collection and treatment activities.

The safety assessment accounts for worst-case concentrations of Uranium, assuming the Uranium concentration associated with fissile material in a CDRA or FMSA is at its bounding optimum value for the following activities: CD repacking; and fissile material storage.

10.9.2.4.8 Reflection Control

The safety assessment conservatively uses sub-critical limits, based on full thickness (i.e., 30 cm) close-fitting water reflection conditions, for assessment of fissile material outside the confinement of a CD (i.e., fissile material not in transit, or not within the environs of a CDSA). This assumption is considered bounding for the following activities: buried waste exhumation; contaminated soil remediation; water collection and treatment; and waste evaluation and assay.

The safety assessment conservatively uses sub-critical limits, based on credible conservative or bounding reflection conditions, for the following activities: sub-surface structure decommissioning; CD staging, buffer storage and transit; CD repacking; and fissile material storage.

10.9.2.4.9 Neutron Absorber Control

The safety assessment does not credit fixed neutron absorbers for the following activities: buried waste exhumation; contaminated soil remediation activities; sub-surface structure decommissioning; water collection and treatment; CD staging, buffer storage and transit; and waste evaluation and assay.

No neutron absorbers are credited in the analysis of the following activities: CD repacking; and fissile material storage.

10.9.2.4.10 Volume Control

The safety assessment does not credit volume control for activities performed in a CDRA or FMSA.

Volume control is not viable for the WTS, due to the very large ground water feed-rate to the WTS, and the consequent need for large volume vessels and tanks for water collection and treatment.

10.9.2.4.11 Heterogeneity Control

The safety assessment is conservatively based on sub-critical limits derived for homogeneous Uranium-water mixtures (with 100 weight percent U-235/U enrichment) for which sub-critical limits are smaller than equivalent heterogeneous Uranium-water mixtures. This assessment is applicable to all activities listed in this Section.

The potential impact of U-235 content under-estimation during *in-situ* radiological surveys of HDP remediation areas, due to the potential variation in Uranium distribution and particle size, is evaluated in the HDP NCSAs.

10.9.2.5 NCS Hazard Evaluation Results

Table 10-8 through Table 10-13 provide a summary of credible criticality hazards established in the HDP NCSAs, and includes a description of the hazards, their potential causes, their potential consequences if unmitigated, the criticality parameter(s) that are potentially challenged, and a summary of the criticality safety barriers that ensure criticality safety.

The credible criticality event sequences and criticality safety barriers presented in Table 10-8 through Table 10-13 apply to the scope of operations evaluated in the NCSAs. Adjustments to operations may result in the revision of existing NCSA(s) which could result in a change to the credible criticality event sequences and criticality safety barriers presented in Table 10-8 through Table 10-13.

10.10 HEALTH PHYSICS AUDITS, INSPECTIONS AND RECORDKEEPING

10.10.1 PROJECT OVERSIGHT COMMITTEE

The Project Oversight Committee (POC) provides management oversight and review of operations associated with the HDP. The POC is chartered with monitoring activities to ensure they are being performed safely and according to regulatory requirements. The POC ensures that appropriate measures are taken to maintain radiation exposures ALARA through administrative and procedural controls, in addition to the design and control of radiological facilities and equipment.

10.10.1.1 Radiation Protection Program

On an annual basis, the RSO will provide a comprehensive written report of the Radiation Protection Program content and implementation to the POC. The POC will then be tasked with assessing the effectiveness of the program.

10.10.1.2 Meeting Frequency

The POC will meet at least quarterly, or more frequently, at the discretion of the Committee Chairman.

10.10.1.3 Meeting Minutes

The Committee will maintain a written record of the minutes of each meeting. Meeting minutes will include the date and time of the meeting, members present, a summary of the deliberations and discussions, recommendations, and actions taken or needed.

10.10.2 HEALTH PHYSICS AUDIT PROGRAM

10.10.2.1 Annual Report

The annual report conducted by the RSO will include a review of audits, inspections, and radiological measurements performed during the past calendar year, with emphasis on data collected from the following areas:

- worker exposure;
- bioassay results;
- airborne radioactivity; and
- environmental monitoring.

10.10.2.2 RSO Audits

A list of the audits performed by the RSO are provided in Table 10-14.

10.10.2.3 Surveillance

Table 10-15 presents a schedule of surveillances that may be performed by HP. The schedule and surveys may be changed as appropriate to ensure surveillances are commensurate with actual performance of decommissioning activities.

10.10.3 CORRECTIVE ACTIONS PROCESS

Westinghouse tracks and resolves issues of non-compliance or conditions adverse to quality using a computer based corrective action process. This corrective action process is described in Chapter 13.0.

10.10.4 RECORDKEEPING

All records of audits, inspections and surveillances are maintained in accordance with the Project Quality Plan (Reference 10-34).

10.11 HEALTH AND SAFETY PLAN

The HASP describes the procedural and equipment requirements for protection of project personnel and the general public from industrial hazards and non-radioactive hazardous material. It is the basis for compliance with 29 CFR 1904, "Recording and Reporting Occupational Injuries and Illnesses," (Reference 10-35), 29 CFR 1910 and 29 CFR 1926, "Safety and Health Regulations for Construction," (Reference 10-36) as well as Westinghouse EH&S policies. Within the scope of the plan, compliance with applicable regulations is the objective. One of the project's primary goals is that no harm should come to project personnel or the public as a result of remediation activities.

The HASP requires regular inspections of site equipment, work practices, and the facility as part of daily routine. EH&S performs further inspections on a regular basis as outlined in the EH&S Oversight procedure. Site personnel are in engaged in accident prevention, near miss reporting and investigation. Reporting is actively promoted through regular safety meetings, near miss and safety suggestions, posted safety information, and enforcement of site rules. An integral part of identification, prevention, and corrective actions is the corrective action process.

10.11.1 HAZARD ANALYSIS

Health and safety hazards are evaluated during the planning phase for each aspect of the project and in the course of lessons learned throughout the project using the site work control process. Engineered controls are the preferred means of addressing such hazards, followed by administrative controls, then by PPE. This process is implemented by placing conservative limits, investigation, and/or action levels to ensure occupational health and safety hazards and exposures are kept ALARA.

10.11.2 STOP WORK AUTHORITY

Project personnel have the authority to stop work without risk of reprimand if someone is in danger, or if continuation of a task could result in injury or violation of a regulatory parameter.

10.11.3 REPORTING

NRC reportable events will be reported to the NRC. The Licensing Manager is responsible for such reports.

Worker injuries or illness will be reported to the individual's supervisor. The EH&S Manager and the Operations Manager will initiate an investigation upon receiving a report of an injury, illness, or incident. Such investigations should determine cause(s) and prevent recurrence of the incident. Corrective actions arising from investigations will be tracked to resolution. Subcontractor personnel shall report injuries/illnesses and incidents to project supervision as well as their employer. OSHA 300 Logs will be maintained in compliance with 29 CFR 1904.

10.12 REFERENCES FOR CHAPTER 10.0

10-1 Code of Federal Regulations, Title 10, Part 70.38, "Expiration and Termination of Licenses and Decommissioning of Sites and Separate Buildings or Outdoor Areas."

10-2 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated Decommissioning Guidance, Decommissioning Process for Materials Licensees," Volume 1, Revision 2, Final Report.

10-3 Westinghouse Electric Company Document No. HDP-PO-HP-100, "Radiation Protection Plan."

10-4 Code of Federal Regulations, Title 10, Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

10-5 Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

10-6 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus."

10-7 Westinghouse Electric Company Document No. HDP-PO-EHS-001, "Health and Safety Plan."

10-8 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication."

10-9 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.25, "Air Sampling in the Work Place."

10-10 Code of Federal Regulations, Title 29, Part 1910, "Occupational Safety and Health Standards."

10-11 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

10-12 Compressed Gas Association (CGA), G-7.1-1997, "Commodity Specification for Air."

10-13 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

10-14 American National Standards Institute, ANSI N13.30, "Performance Criteria for Radiobioassay."

10-15 U.S. Nuclear Regulatory Commission, NUREG/CR-5631, “Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses--Interim Recommendations.”

10-16 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.4, “Direct-Reading and Indirect-Reading Pocket Dosimeters.”

10-17 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.28, “Audible-Alarm Dosimeters.”

10-18 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.”

10-19 U.S. Nuclear Regulatory Commission, Policy and Guidance Directive, “Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material,” April 1993.

10-20 U.S. Nuclear Regulatory Commission, License No. SNM-33 (Docket No. 70-36).

10-21 Code of Federal Regulations, Title 49, “Transportation.”

10-22 Westinghouse Electric Company Document No. DO-08-003, “Hematite Radiological Characterization Report.”

10-23 American National Standards Institute, ANSI-N323A-1997, “Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments.”

10-24 Code of Federal Regulations, Title 10, Part 70, “Domestic Licensing of Special Nuclear Material.”

10-25 U.S. Nuclear Regulatory Commission, Regulatory Guide 3.71, "Nuclear Criticality Safety Standards For Fuels And Material Facilities."

10-26 American National Standards Institute, ANSI/ANS-8.1-1998, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.”

10-27 American National Standards Institute, ANSI/ANS-8.19-2005, “Administrative Practices for Nuclear Criticality Safety.”

10-28 American National Standards Institute, ANSI/ANS-8.20-1991, “Nuclear Criticality Safety Training.”

10-29 American National Standards Institute, ANSI/ANS-8.24-2007, "American National Standard, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations."

10-30 Los Alamos Report LA-10860-MS, "Critical Dimensions of Systems Containing U-235, Pu-239,and Pu-233," 1986.

10-31 Los Alamos Report LA-12808-MS, "Nuclear Criticality Safety Guide," 1996.

10-32 Los Alamos Administrative Resource Handbook ARH-600, "Criticality Handbook," June 28, 1968.

10-33 American National Standards Institute, ANSI/ANS-8.7-1998, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."

10-34 Westinghouse Electric Company Document No. HDP-PO-QA-001, "Project Quality Plan."

10-35 Code of Federal Regulations, Title 29, Part 1904, "Recording and Reporting Occupational Injuries and Illnesses."

10-36 Code of Federal Regulations, Title 29, Part 1926, "Safety and Health Regulations for Construction."

Table 10-1

Programmatic Elements Of The Radiation Protection Plan

ALARA Policy	Exposure Control for Personnel
Occupational Exposure Monitoring	Radiation Protection Work Controls
Radiological Limits	Regulatory Notifications and Reports
Routine Radiological Surveys	Contamination Control
Radiological Instrumentation	Radioactive Material Controls
Radioactive Waste Management	Records Management
Qualifications and Training	Decommissioning Survey Program
Audits and Inspections	

Table 10-2

Programmatic Elements Of The Respiratory Protection Program

Monitoring	Supervision of the Program
Program Audits	Minimum Qualifications
User Training	Fit Testing
Respirator Selection	Maintaining Breathing Air Quality
Inventory and Control	Storage and Issuance
Maintenance, Repair and Testing	Record-keeping
Medical Evaluations	ALARA Evaluations
Routine/Non-Routine/ Emergency Use	

Table 10-3

Surface Contamination Limits

NUCLIDES[a]	AVERAGE[b,c,f,1]	MAXIMUM[b,d,f,1]	REMOVABLE[b,e,f,1]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α	15,000 dpm α	1,000 dpm α
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, and I-129	100 dpm	300 dpm	20 dpm
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, and I-133	1,000 dpm	3,000 dpm	200 dpm
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5,000 dpm β-γ	15,000 dpm β-γ	1,000 dpm β-γ

[1] The units for all limits are expressed on a 100 cm^2 basis (i.e., dpm/100 cm^2).

[a] Where surface contamination by both alpha and beta-gamma-emitting nuclides exist, the limits established for alpha and beta-gamma-emitting nuclides should apply independently.

[b] 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.

[c] Measurements of average contaminant should not be averaged over more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination of objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Table 10-4

Radioactive Material Package Survey Limits

Non-Fixed External Radioactive Contamination Limits For Packages	Maximum Permissible Limits (dpm/100 cm^2)
Beta and gamma and low toxicity alpha emitters	22,000
All other alpha emitting radionuclides	2,200

Non-Fixed External Radioactive Contamination Limits For Packages Exclusive Use Shipment	Maximum Permissible Limits (dpm/100 cm^2)
Beta and gamma and low toxicity alpha emitters	220,000
All other alpha emitting radionuclides	22,000

From 49 CFR 173.443

Table 10-5

Radiological Instrumentation Anticipated For Decommissioning Activities

Type	Criteria
Alpha Counting System (Stationary)	Nominal counting efficiency: 25 percent - (4 pi) geometry Target MDA 20 dpm
Beta-Gamma Counting System (Stationary)	Nominal counting efficiency: 30 percent - (4 pi) geometry Target MDA 200 dpm
Alpha Survey Meters (Portable)	Nominal counting efficiency: 10 percent - (4 pi) geometry Target MDA 200 dpm
Beta-Gamma Survey Meters (Portable)	Nominal counting efficiency: 10 percent - (4 pi) geometry Target MDA 400 dpm
Gamma Survey Meters (Portable)	Range: 0 – 5 rem/hr
Neutron Survey Meter (Portable)	Range: 0 – 100 rem/hr
Lapel Samplers	Average $\geq$ 2 liters/min
Air Samplers (General Area)	10-100 liters/min

Table 10-6

Radiological Instrumentation Currently Maintained At HDP Site

Type	Radiation Detected	Detection Method	Intended Use
Laboratory Counter (Automatic)	Alpha, Beta	Gas flow proportional	Smear, Air sample Analysis
Scaler	Alpha, Beta	Scintillation	Smear Analysis
Portable Survey Meter Up to 5,000 μR/hr	Gamma	Scintillation	Radiation Survey
Portable Survey Meter Up to 5 rem/hr	Beta Gamma	Ionization	Radiation Survey
Portable Survey Meter Up to 100 rem/hr	Neutron	BF_3	Radiation Survey
Portable Survey Meter Up to 500,000 cpm	Beta-Gamma	Geiger-Muller	Contamination Survey
Portable Survey Meter Up to 500,000 cpm	Alpha	Scintillation	Contamination Survey
In-Situ Object Counting System (ISOCS)	Gamma	Germanium	Isotopic Analysis

Type	Flow Rate	Intended Use
Lapel Sampler	Avg. ≥ 2 liters/min	Breathing Zone
General Area Air Sampler	20-100 liters/min	Breathing Zone, General Area
Continuous Air Monitor	20-100 liters/min	General Area

Table 10-7

USLs And AOAs Relevant To Use Of The MCNP 5 Code For The HDP

Area of Applicability (AOA)	AOA Key Parameters and Definition	Upper Subcritical Limit (USL)
AOA (1)	Homogeneous Uranium mixtures consisting of: • Geometry – Spheres, Cylinders, Slabs & arrays • Moderators – Water, Plexiglas, Paraffin, Fluorine, Polyethylene, Hydrogen, HNO_3 • Physical Form – Solid and Solution • Enrichment - 2 to 93 wt.% $^{235}U/U$ • Reflectors – Water, Plexiglas, Paraffin, Polyethylene, Concrete, Depleted Uranium, Unreflected • Absorbers – Aluminum, Steel, Stainless Steel, Boral & Teflon • $H/^{235}U$ – 0.07 to 1437.51 • Average Neutron Energy Causing Fission – 0.006 to 0.800 MeV	0.9340
AOA (2)	Heterogeneous Uranium mixtures consisting of: • Geometry – Spheres, Cylinders, Slabs & arrays • Moderators – Water • Physical Form – Solid • Enrichment – 2.35 to 93 wt.% $^{235}U/U$ • Reflectors – Water & Acrylic • Absorbers – Aluminum, Steel, Stainless Steel, Copper, Cadmium, Boroflex, Boral, Boron & Zircaloy • $H/^{235}U$ – 33 to 693 • Average Neutron Energy Causing Fission – 0.061 to 0.249 MeV	0.9393

Table 10 - 8

Credible Criticality Hazards For Buried Waste And Contaminated Soil Remediation Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
BWCSR-C-01 Mass and/or Concentration	<u>Hazard</u> Fissile materials exhumed from a HDP Remediation Area are transferred to a Waste Transfer Vehicle. <u>Potential Causes</u> • Not surveying HDP Remediation Areas to identify fissile material. • Failure to segregate identified fissile material. • Misdirection of fissile material. <u>Potential Consequences</u> • Potential to exceed a maximum safe mass of U-235 in a waste transfer vehicle or within a downstream WHA; and/or • Potential to exceed a maximum safe concentration of U-235 in a WHA	• CSCs require that in-situ radiological survey and visual inspection of HDP Remediation Areas is performed to identify fissile material prior to excavation activities. Any identified fissile material is required to be extracted, containerized, and segregated by placement into a CD. Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to transfer of fissile material from a HDP Remediation Area to a Waste Transfer Vehicle or downstream WHA. • CSCs require that HDP Remediation Area fissile material identification (i.e., in-situ radiological survey and visual inspection), extraction, containerization, and segregation procedures are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of in-situ radiological surveys (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment).	**Yes**

Table 10 – 8 (Continued)

Credible Criticality Hazards For Buried Waste And Contaminated Soil Remediation Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
BWCSR -C-02 Mass and/or Concentration	Hazard There is a failure to identify fissile material during in-situ radiological survey. Potential Causes • Presence of materials with superior attenuation properties relative to the materials accounted for in the survey equipment calibration basis. • Non-uniform spatial distribution of fissile material. • Conglomeration of fissile material or presence of high density fissile material resulting in self-shielding. • Presence of containers, bulky objects or metallic items. Potential Consequences • Potential to exceed a maximum safe concentration of U-235 in an exhumed soil/waste volume or within a downstream WHA soil/waste heap; and/or • Potential to exceed a maximum safe mass of U-235 in an exhumed soil/waste volume or within a downstream WHA soil/waste heap.	• CSCs require that: ➢ radiological surveys performed in support of CSCs use only equipment that is approved and appropriately calibrated to account for credible variation in uranium distribution and particle size, and to account for photon attenuation within the surrounding medium. ➢ visual inspection of HDP remediation areas is performed to identify items not consistent with the calibration basis of the radiological survey equipment used, and that these items are treated equivalently to identified fissile material. ➢ the depth to which non-fissile materials may be exhumed from a HDP remediation area following radiological survey is limited to an approved value supported by the radiological survey equipment calibration basis. Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that failure to identify fissile material during in-situ radiological survey fissile material could result in a criticality accident. • CSCs require that procedures related to the removal of material from a HDP remediation area are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • BWCSR-C-01 criticality safety barriers also apply to this event sequence.	**Yes**

Table 10 - 8 (Continued)

Credible Criticality Hazards For Buried Waste And Contaminated Soil Remediation Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
BWCSR -C-03 Concentration	Hazard There is difficulty in controlling the excavation depth in HDP Remediation Areas. Potential Causes • Excavation techniques and equipment employed. • Uneven and/or saturates terrain. Potential Consequences • Potential to remove an excessive layer of soil or buried wastes, potentially resulting in the transfer burial waste/soil with a U-235 concentration exceeding the NCS Exempt Material limit to a WHA.	• CSCs require that reasonably practicable measures are taken to minimize the potential to exhume a layer of soil/waste with a thickness exceeding the approved value (established in the radiological survey equipment calibration basis). Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that removal of excess material from a HDP remediation area could result in a criticality accident. • CSCs require that procedures related to the removal of material from an HDP remediation area are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 - 8 (Continued)

Credible Criticality Hazards For Buried Waste And Contaminated Soil Remediation Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
BWCSR -C-06 Mass	Hazard Material with high fissile content is identified during in-situ radiological survey. Potential Causes • Undocumented consignment of process waste, with high fissile content, to a burial pit. Potential Consequences • Potential to exhume more than a maximum safe mass of U-235 from a HDP Remediation Area.	• CSCs require that: ➢ in-situ radiological surveys and visual inspection of HDP remediation areas is performed to identify fissile material prior to excavation activities. Any identified fissile material is required to be extracted, containerized, and secured in a CD. It is also required that operations in the affected HDP remediation area cease and that the NCS organization be informed in the event of discovery of exceptionally high concentrations of fissile material. ➢ fissile materials and suspect fissile materials identified during in-situ radiological survey and/or visual inspection of a remediation area are extracted and placed into a limited volume container. Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that failure to identify fissile material during in-situ radiological survey fissile material could result in a criticality accident. • CSCs require that HDP Remediation Area fissile material identification (i.e., in-situ radiological survey and visual inspection), extraction, containerization, and segregation procedures are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of in-situ radiological surveys (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment).	**Yes**

Table 10 - 8 (Continued)

Credible Criticality Hazards For Buried Waste And Contaminated Soil Remediation Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
BWCSR C-09 Interaction	Hazard Fissile material(s) identified during in-situ radiological survey are not segregated following exhumation. Potential Causes • Unavailability of a CDSA, WEA or MAA. • Frequent recovery of fissile material. Potential Consequences • Potential to exceed a maximum safe mass of U-235 at an excavation site.	• CSCs require that: ➢ exhumed containerized fissile materials and suspect fissile materials are secured in a CD and exported from the HDP remediation area to an approved area prior to exhuming additional fissile material. ➢ each CD loaded in a HDP remediation area is limited to containing only one container. Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that failure to segregate fissile material exhumed from a HDP remediation area could result in a criticality accident. • CSCs require that procedures related to the removal of fissile material from a HDP remediation area are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • BWCSR-C-06 criticality safety barriers also apply to this event sequence.	**Yes**

Table 10 - 8 (Continued)

Credible Criticality Hazards For Buried Waste And Contaminated Soil Remediation Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
BWCSR -C-14 Mass and/or Concentration	Hazard Remediation protocols are not applied to all HDP Remediation Areas. Potential Causes • Misjudgment that a HDP Remediation Area does not contain buried waste or significant U-235 contamination. Potential Consequences • Potential to exceed a maximum safe mass of U-235 in a waste transfer vehicle or within a downstream WHA; and/or • Potential to exceed a maximum safe concentration of U-235 in a WHA.	• CSCs require that the surface soils of all non-burial HDP remediation areas insignificantly contaminated with uranium are: ➢ radiologically surveyed for fissile content prior to excavation activities. It is also required that all generic HDP remediation area CSCs be applied to the subject area in the event of discovery of fissile material. ➢ visually inspected prior to excavation activities. It is also required that all generic HDP remediation area CSCs be applied to the subject area in the event of discovery of any foreign items resembling buried wastes. ➢ sampled to a depth equivalent to the anticipated excavation depth and analyzed for fissile content and inspected for buried waste. It is also required that all generic HDP remediation area CSCs be applied to the subject area in the event of discovery of fissile material or foreign items resembling buried wastes within the sample cores. Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that excavation and subsequent handling of soils in a non-burial HDP remediation area insignificantly contaminated with uranium could result in a criticality accident. • CSCs require that procedures related to the removal of material from a non-burial HDP remediation area insignificantly contaminated with uranium are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of in-situ radiological surveys and sampling (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment).	

Table 10 - 9

Credible Criticality Hazards For Sub-Surface Structure Decommissioning Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
SSD-C-01 Mass and/or Concentration	Hazard An unexpected quantity/concentration of fissile material is accumulated as a result of concrete slab extraction. Potential Causes • Collection of debris from scabbling of heavily contaminated surfaces into a localized volume. • Accumulation of heavily contaminated concrete. Potential Consequences • Potential to exceed a maximum safe mass of U-235; and/or • Potential to exceed a maximum safe concentration of U-235.	• It considered at least unlikely that contaminated concrete debris that has been exposed only to infrequent spills of Fissile Material within a controlled facility could absorb sufficient quantities of Fissile Material and for the absorbed Fissile Material to be in sufficient conditions for a criticality accident to be credible. • CSCs require that concrete surfaces are assayed to identify fissile material prior to excavation activities, and that any identified fissile material associated with contaminated concrete is extracted, containerized in a limited volume container, and segregated by placement (singly) into a CD. It is also required that operations in the affected area cease and that the NCS organization be informed in the event of discovery of exceptionally high concentrations of fissile material. Based on these CSCs, the training and qualifications required of operators, redundancy provided by the requirement for two independent operators to follow these CSCs, and the safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to an accumulation of fissile material as a result of concrete slab extraction. • Equipment used by each trained and qualified operator in support of concrete surface assay activities (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment).	**Yes**

Table 10 - 9 (Continued)

Credible Criticality Hazards For Sub-Surface Structure Decommissioning Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
SSD-C-02 Mass and/or Concentration	<u>Hazard</u> Soil contains an unexpected quantity/concentration of fissile material. <u>Potential Causes</u> • Fissile Material was previously discarded into an undocumented burial area. • Unrecognized pipe crack or breach in the vicinity. • Fissile Material previously penetrated concrete slabs and contacted underlying soils. <u>Potential Consequences</u> • Potential to exceed a maximum safe mass of U-235; and/or • Potential to exceed a maximum safe concentration of U-235.	• CSCs require that soils surrounding sub-surface structure decommissioning debris are assayed to identify fissile material prior to excavation activities, and that any identified fissile material associated with contaminated soil is extracted, containerized in a limited volume container, and segregated by placement (singly) into a CD. It is also required that operations in the affected area cease and that the NCS organization be informed in the event of discovery of exceptionally high concentrations of fissile material. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to an accumulation of fissile material as a result of soil extraction. • CSCs require that procedures related to the assay and subsequent removal of soils surrounding sub-surface structure decommissioning debris are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of soil assay activities (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment). • BWCSR-C-03 criticality safety barriers also apply to this event sequence.	**Yes**

Table 10 – 9 (Continued)

Credible Criticality Hazards For Sub-Surface Structure Decommissioning Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
SSD-C-03a Mass and/or Concentration	Hazard An unexpected quantity/concentration of fissile material accumulates when intact subterranean pipe sections are excavated. Potential Causes • Fissile Material was poured into non-fissile drains and pipes during past production operations. • Tilting or breakage of a pipe section when exhumed leads to a significant concentration of Fissile Material in a localized volume. Potential Consequences • Potential to exceed a maximum safe mass of U-235; and/or • Potential to exceed a maximum safe concentration of U-235.	**Prior to exhumation** • CSCs require that subterranean piping is assayed to identify fissile material prior to excavation, and that the results are used to establish sections of piping that contain sufficiently low (subcritical) fissile mass content, which may then be exhumed individually (as intact sections) at one time. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to discovery of an unexpected quantity/concentration of fissile material associated with an exhumed intact section of piping. • CSCs require that immobilizing agents are added to the internal volume of an assayed section of subterranean piping to fix any fissile material contamination prior to exhumation. Based on these CSCs, the training and qualifications required of operators, and the redundancy provided by the requirement for two independent operators to follow these CSCs, it is also at least unlikely that a criticality accident could occur due to discovery of an unexpected quantity/concentration of fissile material associated with an exhumed intact section of piping.	**Yes**

Table 10 - 9 (Continued)

Credible Criticality Hazards For Sub-Surface Structure Decommissioning Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
SSD-C-03a Mass and/or Concentration	Hazard An unexpected quantity/concentration of fissile material accumulates when intact subterranean pipe sections are excavated. Potential Causes • Fissile Material was poured into non-fissile drains and pipes during past production operations. • Tilting or breakage of a pipe section when exhumed leads to a significant concentration of Fissile Material in a localized volume. Potential Consequences • Potential to exceed a maximum safe mass of U-235; and/or Potential to exceed a maximum safe concentration of U-235.	**Following exhumation** • CSCs require that exhumed sections of piping are re-assayed following exhumation and that any subsections classified as containing fissile material are removed and placed (singly) into a CD. It is also required that operations in the affected area cease and that the NCS organization be informed in the event of discovery of exceptionally high concentrations of fissile material. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to an unexpected quantity/concentration of fissile material associated with exhumed piping. • CSCs require that procedures related to the re-assay and disposition of exhumed sections of piping are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of pipe assay activities (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment).	**Yes**

Table 10 - 9 (Continued)

Credible Criticality Hazards For Sub-Surface Structure Decommissioning Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
SSD-C-03b Mass and/or Concentration	<u>Hazard</u> An unexpected quantity/concentration of fissile material accumulates when crushed subterranean pipe sections are excavated. <u>Potential Causes</u> • Fissile Material was poured into non-fissile drains and pipes during past production operations. • Crushing of pipe sections in the ground leads to a significant concentration of Fissile Material in a localized volume. <u>Potential Consequences</u> • Potential to exceed a maximum safe mass of U-235; and/or • Potential to exceed a maximum safe concentration of U-235.	• CSCs require that subterranean piping is assayed to identify fissile material prior to crushing in-situ, and that the results are used to establish the acceptability of crushing the pipe section within the ground. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to an unexpected quantity/concentration of fissile material associated with crushed piping. • CSCs require that crushed piping debris is assayed to identify fissile material prior to excavation activities, and that any identified fissile material associated with the debris is extracted, containerized in a limited volume container, and segregated by placement (singly) into a CD. It is also required that operations in the affected area cease and that the NCS organization be informed in the event of discovery of exceptionally high concentrations of fissile material. Based on these CSCs, the training and qualifications required of operators, and the redundancy provided by the requirement for two independent operators to follow these CSCs, it is at least unlikely that a criticality accident could occur due to an accumulation of fissile material as a result of exhumation of crushed piping debris. • Equipment used by each trained and qualified operator in support of crushed piping debris assay activities (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment). • BWCSR-C-03 criticality safety barriers also apply to this event sequence.	**Yes**

Table 10 - 9 (Continued)

Credible Criticality Hazards For Sub-Surface Structure Decommissioning Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
SSD-C-04 Mass and/or Concentration	Hazard An unexpected quantity/concentration of fissile material is accumulated as a result of septic tank or drain field excavation. Potential Causes • Fissile Material was poured into lab sinks during past production operations. • Removal of Fissile Material during cleaning/washing of personal protective equipment used during past production operations. Potential Consequences • Potential to exceed a maximum safe mass of U-235; and/or • Potential to exceed a maximum safe concentration of U-235.	• CSCs require that septic system decommissioning debris are assayed to identify fissile material prior to excavation activities, and that any identified fissile material associated with the assayed septic system material is extracted, containerized in a limited volume container, and segregated by placement (singly) into a CD. It is also required that operations in the affected area cease and that the NCS organization be informed in the event of discovery of exceptionally high concentrations of fissile material. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that a criticality accident could occur due to an accumulation of fissile material as a result of septic system decommissioning debris extraction. • CSCs require that procedures related to the assay and subsequent removal of septic system decommissioning debris are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of septic system decommissioning debris assay activities (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment). • BWCSR-C-03 criticality safety barriers also apply to this event sequence.	**Yes**

Table 10 - 10

Credible Criticality Hazards For Collared Drum Transit, Staging And Buffer Storage Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDS-C-01 **CDS-C-02** Interaction	Hazard There is neutron interaction between fissile materials within a functional area approved to contain fissile material, or, between fissile materials associated with different functional areas approved to contain fissile material. Potential Causes • Wrong mass per container. • Wrong location. • Wrong container. • Loss of isolation between functional areas on account of failure to establish, and/or demarcate functional area boundaries. Potential Consequences • Potential for excess neutron interaction between fissile materials situated in or between functional areas. • Potential to exceed a maximum safe mass of U-235 due to congregation of fissile material in a functional area.	• The low fissile material mass content of each CD relative to the maximum subcritical mass limit (see BWCSR, SSD and WEA event sequences), and the design dimensions of CDs (without crediting their affixed collars), which ensures the safety of an infinite planar array of loaded CDs in any configuration, ensures that it is at least unlikely that neutron interaction between fissile materials in a CDSA or CDBS could result in a criticality accident. • The physical design of CDs ensures that the content of each CD is effectively neutronically isolated from the content of any number of other CDs by virtue of their affixed collars, which provide a fixed separation distance. This CD physical design feature ensures only negligible neutron interaction between fissile materials associated with CDs in a CDSA or CDBS, irrespective of the quantity and (unstacked) configuration of the CDs involved, and their individual fissile material mass content. This also ensures that it is at least unlikely that neutron interaction between fissile materials in a CDSA or CDBS could result in a criticality accident. Note: The safety of neutron interaction between fissile materials within/between functional areas other than CDSAs and CDBS' is addressed in Buried Waste and Contaminated Soil Remediation, Sub-Surface Structure Decommissioning, WEA, CDRA and FMSA event sequences).	**Yes**

Table 10 - 10 (Continued)

Credible Criticality Hazards For Collared Drum Transit, Staging And Buffer Storage Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDS-C-03 Interaction	Hazard There is a loss of configuration control in a CDSA or CDBS. Potential Causes • CD stacking. • Physical disturbance. • Natural phenomena events. Potential Consequences • Potential for excess neutron interaction.	• A criticality accident in a CDSA or CDBS due to loss of configuration control as a result of CD stacking is prevented by the following criticality safety barriers: ➢ CSCs require that CDs are not stacked. Based on these CSCs, the training and qualifications required of operators, and the inherent difficulty in stacking loaded CDs due to their size, loaded weight, and the unavailability of stacking equipment in CDSAs/CDBSs, loss of configuration control is at least unlikely. ➢ Bounding credible CD stacking upsets in a CDSA or CDBS would remain safely subcritical. • The collars affixed to CDs are designed to account for potential difference in height between adjacent CDs due to possibility of uneven terrain, thus ensuring that their credited safety function (which is to provide a fixed separation distance) is not compromised. • The provision of raised concrete barricades to enclose CDSAs (other than at entrance/exit points) provides a highly robust physical barrier to protect against accidental physical disturbance events. This design feature ensures that physical disturbance events will not compromise the credited safety function of CDs residing within a CDSA. • The physical structure within which CDBS' are enclosed provides a highly robust physical barrier to protect against accidental physical disturbance events. This design feature ensures that physical disturbance events will not compromise the credited safety function of CDs residing within a CDBS.	**Yes** **(for CD stacking)** **N/A** **(for the other scenarios prevented by the highly robust physical design features cited)**

Table 10 - 10 (Continued)

Credible Criticality Hazards For Collared Drum Transit, Staging And Buffer Storage Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDT-C-01 Mass	Hazard There is loss of containment of fissile material. Potential Causes • Spillage during container loading. • Spillage during transfer between functional areas. Potential Consequences • Potential to spill and accumulate a maximum safe mass of U-235 within any discrete location of the site.	• The CSCs related to material exhumation, evaluation, assay, and CD loading activities ensure that each CD will contain, at most, a small amount of fissile material relative to the maximum subcritical mass limit (see BWCSR, SSD and WEA event sequences). This safety margin combined with the low probability of achieving the idealized geometry, moderation and reflection conditions required under low critical mass conditions, ensures that loss of containment of fissile material would be at least unlikely to result in a criticality accident. • CSCs require that fissile materials and suspect fissile materials are containerized and placed within lidded CDs prior to transfer and during staging/storage. Based on these CSCs, the training and qualifications required of operators, and the low probability of spillages from multiple CDs into a single location, it is also at least unlikely that loss of containment of fissile material could result in a criticality accident.	**Yes**

Table 10 - 10 (Continued)

Credible Criticality Hazards For Collared Drum Transit, Staging And Buffer Storage Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDT-C-02 Mass	Hazard Fissile material is misdirected during transfer between functional areas. Potential Causes • Procedure non-compliance. • Incorrect labeling of a container such that the container type is not recognized or misidentified. Potential Consequences • Potential to exceed a maximum safe mass of U-235 within any discrete location of the site, other than an approved functional area.	• CSCs prescribe the approved locations for loaded CDs. Based on these CSCs, the training and qualifications required of operators, and the low fissile material mass content of each CD relative to the maximum subcritical mass limit (see Buried Waste and Contaminated Soil Remediation, Sub-Surface Structure Decommissioning and WEA event sequences), it is at least unlikely that a maximum subcritical mass could be assembled due to the misdirection and receipt of CDs in an area not approved to contain CDs. • CSCs require that the content of CDs are not disturbed upon identification in any unapproved location, and that operations in the affected area cease and not resume until the NCS Organization has been informed and has approved resumption of operations. The former requirements are supported by site wide awareness of CDs and by ensuring that CDs are readily identifiable and are only used for fissile material operations. Combined with the physical design of CDs (see CDT-C-03), it is at least unlikely that a criticality accident could occur due to misdirection and receipt of CDs in an area not approved to contain CDs.	**Yes**

Table 10 - 10 (Continued)

Credible Criticality Hazards For Collared Drum Transit, Staging And Buffer Storage Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDT-C-03 Interaction	Hazard There is neutron interaction between fissile materials in transit. Potential Causes • Wrong mass per container. • Wrong location. • Wrong container. Potential Consequences • Potential for excess neutron interaction between fissile materials in transit.	• CSCs require that fissile materials and suspect fissile materials are containerized and placed within lidded CDs prior to transit between functional areas of the site. Combined with the low fissile material mass content of each CD relative to the maximum subcritical mass limit (see Buried Waste and Contaminated Soil Remediation, Sub-Surface Structure Decommissioning and WEA event sequences), and the design dimensions of CDs (without crediting their affixed collars), which ensures the safety of an infinite planar array of loaded CDs in any configuration, it is at least unlikely that neutron interaction between fissile materials in transit could result in a criticality accident. • The physical design of CDs ensures that the content of each CD is effectively neutronically isolated from the content of any number of other CDs by virtue of their affixed collars, which provide a fixed separation distance. This CD physical design feature ensures only negligible neutron interaction between fissile materials in transit, irrespective of the quantity and (unstacked) configuration of the CDs involved, and their individual fissile material mass content. In conjunction with CSCs that require procedures related to the CD loading and lidding activities to be performed, independently, by at least one other trained and qualified operator, it is also at least unlikely that neutron interaction between fissile materials in transit could result in a criticality accident.	**Yes**

Table 10 - 11

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
WEA-C-01 Mass and/or Concentration	Hazard A collared drum containing a large mass of uranium is received at a WEA. Potential Causes • Undocumented consignment of process waste, with high fissile content, to a burial pit. Potential Consequences • Potential for WEA operations to involve more than a maximum safe mass of U-235 at one time.	• CSCs require that: ➢ the volume of loose material evaluated at any one time on a WEA sorting tray is limited to an approved volume, to ensure the WEA operations will only involve limited (safely subcritical) quantities of fissile material at one time. ➢ any suspect items (e.g., bottles containing liquids, or significant quantities of uranium products such as pieces of metal, pellets/pellet fragments, or uranium powders) are not evaluated beyond the point of discovery without specialist advice from the NCS organization. Based on these CSCs, the training and qualifications required of operators, and safety margin established against their failure, it is at least unlikely that evaluation of the content of a CD with a large mass of uranium could result in a criticality accident. • CSCs require that procedures related to WEA operations are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 - 11 (Continued)

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
WEA-C-02 Mass and/or Concentration	Hazard There is an accumulation of fissile material within a WEA. Potential Causes • Processing of multiple CDs in a WEA at any one time. • Long term accumulation of fissile material within a WEA from waste evaluation activities. Potential Consequences • Potential to assemble more than a maximum safe mass of U-235 in a WEA (excluding the contents of materials sealed in any staged CDs).	• CSCs require that: ➢ prior to unloading the content of any CD in a WEA, the surface of the WEA sorting tray as well as the surrounding area is confirmed to be clear of any materials not contained in a Waste Container or CD. ➢ only the WEA sorting tray is used for WEA operations, which is designed to provide containment of soils/liquids tipped onto its surface. ➢ spillage of the content of a CD onto any surface other than a WEA sorting tray be recovered as soon as is practicable and prior to evaluation of any additional CDs. Based on these CSCs, the WEA sorting tray design, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that accumulation of fissile material in a WEA could result in a criticality accident. • CSCs require that procedures related to WEA operations are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • WEA-C-01 criticality safety barriers also apply to this event sequence.	**Yes**

Table 10 - 11 (Continued)

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
WEA-C-03 Mass and/or Concentration	<u>Hazard</u> Fissile material is transferred from a WEA to a WHA. <u>Potential Causes</u> • Failure to identify fissile material during waste evaluation operations. • Erroneous placement of fissile material (identified during waste evaluation operations) into a Waste Container. • Mislabeling of Assay Containers as waste. • Misdirection of Assay Containers to a WHA. <u>Potential Consequences</u> • Potential to assemble more than a maximum safe mass of U-235 in a WHA.	• CSCs require that radiological surveys and visual inspection of material evaluated in a WEA is performed to identify fissile material, and that identified fissile material is extracted, containerized and assayed for fissile content in a MAA. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that fissile material could be transferred from a WEA to a WHA and result in a criticality accident. • CSCs require that procedures related to WEA operations are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of radiological surveys in a WEA (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment).	**Yes**

Table 10 - 11 (Continued)

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
WEA-C-04 Mass and/or Concentration	<u>Hazard</u> Excess fissile material is transferred from a WEA to a MAA. <u>Potential Causes</u> • Failure to identify fissile material during waste evaluation operations. • Overloading Assay Containers during waste evaluation operations. <u>Potential Consequences</u> • Potential to assemble more than a maximum safe mass of U-235 in an Assay Container.	• CSCs require that each Assay Container loaded in a WEA is limited to materials originating from only a single CD, and to an approved volume of material, which combined with the CSCs in WEA-C-03 ensures limited (safely subcritical) fissile material content. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that excess fissile material could be transferred from a WEA to a MAA and result in a criticality accident. • CSCs require that procedures related to WEA operations are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 - 11 (Continued)

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
MAA-C-01 Mass and/or Concentration	<u>Hazard</u> There is an accumulation of fissile material within a Material Assay Area. <u>Potential Causes</u> • Unloading of multiple CDs within a MAA at one time. <u>Potential Consequences</u> • Potential to assemble more than a maximum safe mass of U-235 in a MAA (excluding the contents of materials sealed in any staged CDs).	• CSCs limit the number of Assay Containers that may outside the environs of a CD to ensure that an unsafe accumulation of fissile material outside the environs of CDs will not occur. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that an accumulation of fissile material within an MAA could result in a criticality accident. • CSCs require that procedures related to Assay Container handling operations in a MAA are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 - 11 (Continued)

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
MAA-C-02 Mass and/or Concentration	<u>Hazard</u> Low fissile mass is assigned to an Assay Container following assay. <u>Potential Causes</u> • Equipment failure. • Incorrect labeling of a container such that the assigned mass is less than the actual mass established by HRGS assay. <u>Potential Consequences</u> • Improper disposition of fissile material as NCS Exempt Material; and/or • Improper transit of a CD with a high U-235 loading to a CDRA or FMSA, resulting in less spacing being applied in a CDRA/FMSA than would be appropriate for the actual fissile mass content of the CD.	• CSCs require that Assay Containers are assayed for fissile mass content in a MAA and are properly dispositioned according to the assigned fissile mass estimate. CSCs also require that Assay Containers are properly labeled following assay. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that erroneously assigning a low fissile mass estimate to an Assay Container could result in a criticality accident. • CSCs require that procedures related to container assay activities in a MAA are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements. • Equipment used by each trained and qualified operator in support of container assay activities in a MAA (when used in support of a CSC) are physically independent (i.e., are not the exact same equipment), or are required to be functionally checked prior to and following a series of container assay measurements, before the assay measurements are regarded as valid.	**Yes**

Table 10 - 11 (Continued)

Credible Criticality Hazards For Waste Evaluation And Assay Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
MAA-C-03 Mass and/or Concentration	Hazard High fissile mass content is assigned to an Assay Container following assay. Potential Causes • Large quantity of U-235 residues within an Assay Container. Potential Consequences • Potential for a single CD in a MAA to contain more than a maximum safe mass of U-235, and for an unanalyzed condition to be realized in the event of transfer of the subject CD to other functional areas of the site.	• CSCs require that Assay Containers are confirmed to contain no greater than the maximum permitted fissile material mass content prior to permitting handling or movement of the Assay Container following assay. It is also required that operations in the affected area cease in the event of discovery of exceptionally high fissile material mass content in any assayed container, and not resume until the NCS organization has been informed and has approved resumption of operations. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that handling of exceptionally high fissile material mass content Assay Containers following assay could result in a criticality accident. • CSCs require that procedures related to container assay and handling activities in a MAA are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 - 12

Credible Criticality Hazards For Collared Drum Repacking Area Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDRA-C-01 Geometry	Hazard Containers other than CDs or DCDs are collected, stored, or used for repack operations within a CDRA. Potential Causes • Empty DCDs are not available. • Smaller containers are easier to store. Potential Consequences • Formation of an unevaluated and potentially unsafe storage configuration.	• CSCs governing fissile material transit between functional areas of the site (see CDT events) require that two trained and qualified operators in the exporting functional area confirm that fissile materials are containerized within lidded CDs prior to their transfer to other functional areas (including to a CDRA). Based on these CSCs and the redundancy provided by the use of two operators, it is at least unlikely that fissile materials within an unapproved container could be transferred to a CDRA. • CSCs require that at least two CDRA operators independently confirm that all fissile materials received at a CDRA are secured within a CD prior to their acceptance. Based on these CSCs, the training and qualifications required of operators, and the redundancy provided by the use of two operators, it is at least unlikely that fissile materials within an unapproved container could be accepted and accommodated within a CDRA. • CSCs require that the entrance/exit points of CDRAs are closed and locked with two locking devices when not in use, with the combination or key of each lock being different and not controlled by a single individual.	**Yes**

Table 10 - 12 (Continued)

Credible Criticality Hazards For Collared Drum Repacking Area Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDRA-C-02 Interaction	Hazard CDs or DCDs are stacked in a CDRA. Potential Causes • Loss of floor space leads to vertical storage. Potential Consequences • Formation of an unevaluated and potentially unsafe storage configuration.	• A criticality accident in a CDRA due to loss of configuration control as a result of CD or DCD stacking is prevented by the following criticality safety barriers: ➢ CSCs require that CDs and DCDs are not stacked. Based on these CSCs, the training and qualifications required of operators, and the inherent difficulty in stacking loaded CDs and DCDs due to their size, loaded weight, and the unavailability of stacking equipment in CDRAs, loss of configuration control due to stacking is at least unlikely. • Bounding credible CD and DCD stacking upsets in a CDRA would remain safely subcritical.	**Yes**

Table 10 - 12 (Continued)

Credible Criticality Hazards For Collared Drum Repacking Area Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDRA-C-03 Mass	<u>Hazard</u> Fissile material Containers are accumulated in the Entry and Repack Zone of a CDRA. <u>Potential Causes</u> • The Entry and Repack Zone is used for storage because the DCD Storage zone is full. • The doors are locked to the DCD Storage zone. • Significant floor space in Entry and Repack Zone as compared to DCD Storage zone leads to storage in Entry and Repack Zone. • High demand rate on operations in a CDRA. <u>Potential Consequences</u> • Potential to assemble more than a maximum safe mass of ^{235}U in the Entry and Repack Zone of a CDRA.	• CSCs require that a maximum of only one Fissile Material Container is permitted within the Entry and Repack Zone of a CDRA at a time, excluding the content of a single DCD retrieved from the DCD Storage Zone for the batching operation. Based on these CSCs, the training and qualifications required of operators, and the safety margin established against their failure, it is at least unlikely that a maximum safe mass of U-235 could be assembled in the Entry and Repack Zone of a CDRA. • CSCs require that procedures related to operations in a CDRA are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 – 12 (Continued)

Credible Criticality Hazards For Collared Drum Repacking Area Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
CDRA-C-04 Mass	Hazard A DCD is over batched in a CDRA. Potential Causes • Lack of available empty DCDs. Potential Consequences • Potential to realize an unevaluated condition in a CDRA due to excess fissile mass content of individual DCD(s).	• CSCs require that the fissile mass content of DCDs is limited to an approved maximum value. The fissile mass content of each Fissile Material Container secured within a CD/DCD is clearly ascribed to the container, ensuring that complying with this requirement is a simple task. This, combined with the training and qualifications required of operators, the redundancy provided by the use of two operators for CDRA batching operations, and the safety margin established against their failure, ensures that it is at least unlikely that a DCD could be double batched. • Bounding credible over-batching upsets of a single DCD would remain safely subcritical. Also, an unlimited number of double-batched DCDs would remain safely subcritical in any unstacked configuration, and also even in any two-high stack configuration (also note that the criticality safety barriers in CDRA-C-02 prevent stacking upsets). • CSCs require that procedures related to operations in a CDRA are performed, independently, by at least one other trained and qualified operator. It is also unlikely that an independent trained and qualified operator would fail to adhere to these requirements.	**Yes**

Table 10 - 13

Credible Criticality Hazards For Fissile Material Storage Area Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
FMSA-C-01 Geometry	Hazard Containers other than CDs or DCDs are stored within a FMSA. Potential Causes • Smaller containers are easier to store. Potential Consequences • Formation of an unevaluated and potentially unsafe storage configuration.	• CSCs governing fissile material transit between functional areas of the site (see CDT events) require that two trained and qualified operators in the exporting functional area confirm that fissile materials are containerized within lidded CDs prior to their transfer to other functional areas (including to a FMSA). Based on these CSCs and the redundancy provided by the use of two operators, it is at least unlikely that fissile materials within an unapproved container could be transferred to a FMSA. • CSCs require that at least two FMSA operators independently confirm that all fissile materials received at a FMSA are secured within a CD prior to their acceptance. Based on these CSCs, the training and qualifications required of operators, and the redundancy provided by the use of two operators, it is at least unlikely that fissile materials within an unapproved container could be accepted and stored in a FMSA. • CSCs require that the entrance/exit points of FMSAs are closed and locked with two locking devices when not in use, with the combination or key of each lock being different and not controlled by a single individual.	**Yes**

Table 10 - 13 (Continued)

Credible Criticality Hazards For Fissile Material Storage Area Activities

Hazard ID And Parameter Challenge	Hazard Description	Criticality Safety Barriers	DCP Compliance
FMSA-C-02 Interaction	Hazard CDs or DCDs are stacked in a FMSA. Potential Causes • Loss of floor space leads to vertical storage. Potential Consequences • Formation of an unevaluated and potentially unsafe storage configuration.	• A criticality accident in a FMSA due to loss of configuration control as a result of CD or DCD stacking is prevented by the following criticality safety barriers: ➢ CSCs require that CDs and DCDs are not stacked. Based on these CSCs, the training and qualifications required of operators, and the inherent difficulty in stacking loaded CDs and DCDs due to their size, loaded weight, and the unavailability of stacking equipment in FMSAs, loss of configuration control due to stacking is at least unlikely. ➢ Bounding credible CD and DCD stacking upsets in a FMSA would remain safely subcritical.	**Yes**

Table 10-14

RSO Audit Schedule

Description	Frequency
POC (including RSO) meeting	Quarterly
ALARA Report to Site Manager	Semiannually
ALARA Report by RSO to POC, for POC annual review	Annually
RSO review of operating procedures that affect radiation protection	Biennial
Audit of ALARA Program	Annually
Manager Self Assessment	Annually

Table 10-15

Health Physics Surveillance Schedule

Activity	Schedule
Oversight of radiation protection activities	Weekly
Contamination survey of contamination areas	Monthly
Posting surveillance	Monthly
Contamination and dose rate survey of non-contaminated areas	Monthly
Radiation and contamination survey of radioactive material areas	Monthly
Inventory of radiological response kits	Quarterly
Inventory of sealed sources	Semiannual
Sealed source leak test	Semiannual
Air and Liquid Effluent Monitoring Report	Semiannual

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ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
CFR	Code Of Federal Regulations
EEMP	Effluent And Environmental Monitoring Plan
HDP	Hematite Decommissioning Project
MDC	Minimum Detectable Concentration
NRC	U.S. Nuclear Regulatory Commission
PQP	Project Quality Plan
QA/QC	Quality Assurance/Quality Control
RSO	Radiation Safety Officer
TLD	Thermoluminescent Dosimeter
WTS	Water Treatment System
µCi/ml	microCuries per milliliter

11.0 ENVIRONMENTAL MONITORING PROGRAM

Decommissioning activities at the Hematite Site will be conducted in a manner that protects the health and safety of the public, employees and the environment from ionizing radiation. Decommissioning activities primarily involve outdoor activities involving the physical removal of soil and buried materials.

The following sections describe the elements of Hematite Decommissioning Project (HDP) environmental monitoring programs as they apply to current and remaining decommissioning activities at the site. The description of HDP environmental monitoring programs contained herein was developed in accordance with the applicable guidance in NUREG-1757 (Reference 11-1).

Information that may aid in understanding the descriptions contained in this chapter may be found in the Effluent and Environmental Monitoring Plan (EEMP) (Reference 11-2). The EEMP is divided into sections that describe the Radiological Effluent Control and Monitoring Program and the Environmental Monitoring Program. The EEMP also contains effluent monitoring requirements pursuant to Missouri Department of Natural Resources regulations and site permits. Collectively, these sections of the EEMP constitute the environmental monitoring programs for the Hematite Site and are designed to ensure releases of radioactive material to the environment comply with 10 CFR Part 20 (Reference 11- 3), and are maintained as low as reasonably achievable (ALARA).

11.1 ENVIRONMENTAL ALARA EVALUATION PROGRAM

This section describes the requirements for establishing and maintaining releases of radioactive materials to levels which are ALARA.

11.1.1 ALARA COMMITMENT

Hematite Site activities will be conducted in a manner that minimizes adverse impact to the health and safety of the public, site employees and the environment. This goal will be achieved through the implementation of programs that maintain the release of radioactive material and exposure to ionizing radiation at levels that are ALARA.

In accordance with Regulatory Guide 8.37, ALARA Levels for Effluents from Materials Facilities (Reference 11-4), every reasonable effort will be made to ensure that decommissioning activities are conducted in accordance with ALARA principles, and that concentrations of radioactive materials in air and liquid effluents are minimized in a manner consistent with the ALARA philosophy.

11.1.1.1 ALARA Goals

The Hematite Site ALARA goal for airborne effluents is based on controlling the generation of dust containing airborne radioactivity during contaminated soil movement and excavation. The Hematite goal for liquid effluents is based on collecting liquids that become contaminated from soil and material being remediated, and then processing those liquids through a Water Treatment System to remove the radioactivity to the maximum extent practical. Section 11.2.4 describes the effluent control program.

The Hematite Site numerical ALARA goals for air and liquid effluent concentrations have been established at 20 percent of the applicable values in 10 CFR 20, Appendix B, and are summarized in Table 11-1. These goals are based on the annual average values contained in Table 2 of 10 CFR 20, Appendix B. These goals are not intended to be applied as a limit, but rather as a standard for comparison during decommissioning work planning and execution. These goals may be adjusted based on decommissioning experience or based on detailed information related to specific work activities.

HDP will constrain airborne radioactivity emissions in accordance with the requirements of 10 CFR 20.1101(d) and will demonstrate compliance with this requirement using methodology contained within Regulatory Guide 4.20 using the data obtained from air monitoring locations defined in Tables 11-1a and 11-1b of this chapter.

11.1.1.2 ALARA Investigation Levels

The investigation levels for air and liquid effluent concentrations have been established at 50 percent of the applicable values in 10 CFR 20, Appendix B. If an individual effluent sample result exceeds an investigation level, a review of the decommissioning activities will be performed to identify appropriate changes to work methods and/or engineering controls to reduce the concentrations as low as reasonably achievable.

11.1.2 PROCEDURES, PROCESS CONTROLS AND ENGINEERING CONTROLS

The outcome of the efforts to achieve the ALARA goals will be realized through implementation of HDP programs and procedures that maintain the release of radioactive material and exposure to ionizing radiation at levels that are ALARA. A summary description of the procedures, process controls and engineering controls to maintain worker exposure and effluent concentrations of radioactive material ALARA, is summarized in Section 11.2.4. In addition to procedures implementing EEMP requirements, specific work plans will be developed for all major decommissioning activities to ensure effective implementation of engineering controls.

11.1.3 ALARA REVIEWS

Environmental ALARA is continually reviewed as analytical data for air and liquid effluent is received and evaluated. In addition, the effectiveness of this continual review is evaluated through the use of periodic reviews, surveillances and/or audits. As listed in Table 10-14, an ALARA report is submitted semi-annually by the Radiation Safety Officer (RSO) to the Project Director, the Project Oversight Committee performs an annual ALARA review, and Quality Assurance performs an annual audit of the ALARA program. Periodic audits of the effluent and environmental monitoring programs and associated procedures will be performed as provided by the site quality assurance plan. Additionally, the RSO may elect to perform periodic surveillances of the environmental effluent control and monitoring program, to ensure ALARA considerations are being effectively applied to specific decommissioning work activities.

11.2 EFFLUENT MONITORING PROGRAM

This section describes the Effluent Monitoring Program for the Hematite Site. Specific requirements of this program are contained in the EEMP (Reference 11-2). Table 11-1 summarizes the ALARA goals, investigation levels and regulatory limits for air and liquid effluents, which are described in the following sections. Implementation of the EEMP requirements described below will be in accordance with site procedures.

11.2.1 AIR EFFLUENTS

Since decommissioning activities involve outdoor soil movement and excavation, these activities do not typically create discrete release points for airborne effluents. Nor do they involve stacks or heat sources that would elevate potential releases over the perimeter. Rather, air releases occur at ground level within the vicinity of the work and tend to be more widespread. Thus, airborne sampling at the perimeter of the central tract area is the best method to assess the concentration in air effluent. Stationary environmental air samples identified in Table 11-1a will be used to assess the concentration in air effluent released to the public. These fixed air monitoring locations are identified on Figure 11-1. In addition, downwind airborne sampling at the perimeter of decommissioning work activities will be performed when work activities are likely to generate, at the perimeter of the work activities, airborne radioactivity concentrations in excess of 20 percent of the values listed in 10 CFR 20, Appendix B, Table 2, Column 1 (Reference 11-3).

This basis is derived from the ALARA constraint in 10 CFR 20.1101 (d) of 10 mrem/yr for public exposure from emissions of airborne radioactive material, which is equivalent to twenty percent of 10 CFR 20, Appendix B, Table 2, Column 1. It should be noted that the location of a stationary environmental or perimeter air sampler is conservative relative to the location of a member of the public. A member of the public would not be at these locations for the duration of the work. Thus, a member of the public would receive less dose than the dose estimated from stationary environmental or perimeter air sample results. Accordingly, these sampling at these locations at an action level equivalent to the ALARA goal for the public, and taking ALARA actions based on exceedances of the ALARA goal, conservatively ensure the ALARA goal is met for the public at a downwind location.

The number and location of perimeter air samplers will be established based on consideration of the location and nature of current work activities, and environmental conditions such as wind direction. In practice, it is Westinghouse's intent to run perimeter samplers during nearly all operations that involve movement of exposed soil, (e.g., excavation, rail car loading), thus portable downwind air samplers will be utilized for many activities that have the potential to generate concentrations that are less than 20 percent of the air effluent limit. When performed, perimeter air sampling will be done on a continuous basis and samples will be analyzed daily, as summarized in Table 11-1. Note that the occupational air concentrations within the work area

will be measured using personal air sampling pumps or additional low-volume portable sampling pumps independently of the perimeter sampling.

Air effluents from the soil vapor extraction will be treated and monitored as discussed in Section 12.4.3.4.

11.2.2 LIQUID EFFLUENTS

Liquid effluent sampling requirements as part of the EEMP are summarized in Table 11-1. Table 11-1 includes ALARA goals, investigation levels and regulatory limits that apply to liquid effluent concentrations, as measured at the site boundary.

11.2.3 BASIS FOR EFFLUENT MONITORING PROGRAM

The followings subsections describe the basis for effluent monitoring at HDP.

11.2.3.1 Effluent Concentrations

Baseline concentrations for air and liquid effluents have been established through historical site effluent monitoring during facility operation and, more recently, the decommissioning period. Since fuel manufacturing operations ended in 2001, and the decommissioning period began, effluent radionuclide concentrations have typically been a fraction of the applicable regulatory limits. Table 11-2 and Table 11-3 provide a summary of typical air and liquid effluent values, respectively, which have been measured during the site decommissioning period. Based on implementation of the work controls, monitoring and action levels described herein, air effluent concentrations are not expected to exceed one-half of the annual average limit as a result of decommissioning activities. Where batch processing is used, the concentration in liquid effluent will be controlled by an evaluation of laboratory results to determine suitability for release, or by process knowledge and retrospective confirmation of the concentrations based on subsequent laboratory analysis. Sample collection and analysis is discussed in Section 11.2.3.4. Process knowledge will be based on the liquid being consistent in terms of source (e.g., same location) and method of generation (e.g., precipitation) as a liquid where previous sample results are known. Methods for water treatment will be implemented when appropriate to further reduce concentrations released to the environment.

11.2.3.2 Physical And Chemical Concentrations

The physical and chemical characteristics of radionuclides that may be discharged in air or liquid effluents during planned decommissioning work activities are not expected to be different than the effluents currently being discharged. Air and liquid effluent samples are routinely analyzed on-site, for both gross alpha and gross beta concentrations. If necessary, effluent samples may be sent to a qualified off-site laboratory for isotopic analysis. Samples sent for off-site

laboratory analysis will be processed in accordance with the site quality assurance plan and applicable site procedures (see Section 11.2.3.5).

11.2.3.3 Discharge Locations

Each decommissioning work area likely to result in discharge air effluent concentrations in excess of 20 percent of the values listed in 10 CFR 20, Appendix B, Table 2 will be monitored by the collection of an air sample. Effluent locations will be established based on consideration of the location and nature of current work activities, and environmental conditions such as wind direction. Air effluents will be monitored as described in Section 11.2.1 and in the EEMP.

If the soil vapor extraction system (SVES) is used, the point source of air effluent from the SVES will initially originate from the equipment location on the eastern portion of the slab of the former process buildings. During remediation of the former process building slab and underlying soil, it is planned that the SVES equipment will be relocated to a location in the Central Tract that does not impact remediation work activities.

Liquid effluents will be controlled such that each work area with the potential to discharge liquid effluents will have a specific point of discharge that can be monitored in accordance with Section 11.2.2 and the EEMP.

11.2.3.4 Sample Collection And Analysis

Effluent sample type, collection frequency and sample analyses will be conducted as specified in Table 11-1. Sample collection and analysis are conducted in accordance with HDP quality assurance requirements, see Section 11.2.3.5.

Perimeter air sampling locations will be established to be representative of the airborne effluents from specific outdoor work activities as described in Section 11.2.1. Samples will be collected during work activities or weather conditions that have the potential to generate airborne effluents. Perimeter air sampling locations are expected to be distributed along the perimeter of the central tract, along the perimeter of the soil staging area, and along the rail spur. These locations are in proximity to the areas with the most potential for airborne dust to leave the site. In addition, air monitors are positioned at fixed locations near the edge of the Central Tract Area, as discussed in Section 11.3. Figure 11-1 shows the locations of the fixed air samplers.

Effluent air samples shall be analyzed for gross alpha radioactivity and gross beta radioactivity. The Minimum Detectable Concentration (MDC) target for gross alpha and gross beta analysis of samples from Stationary Environmental Air Sampling Stations is established at 5 percent of the applicable 10 CFR 20 Appendix B, Table 2, Column 1 limits. The MDC target for gross alpha and gross beta analysis of air effluents of samples from Perimeter Sampling Locations is

established at 50 percent of the applicable 10 CFR 20 Appendix B, Table 2, Column 1 limits (with a goal of 20 percent where practical) [1].

The review of effluent air sample results shall include consideration of whether the isotopic mixture may differ from that previously understood, thus warranting isotopic analysis. Considerations to determine that a change in isotopic mixture may have occurred include isotopic results of soil or other media associated with the origin of the sample.

Isotopic analysis of effluent air samples shall be performed when investigation levels in Table 1, Effluent Monitoring Program, are exceeded. For air samples, the methods of isotopic analysis for uranium and Tc-99 are DOE EML A-01-R and DOE TC-02-RC, respectively, or equivalent. The MDC target for isotopic analysis are 5 percent for samples from Stationary Environmental Air Sampling Stations and 10 percent for samples from Perimeter Sampling Locations.

Liquid effluent sample locations and techniques have been determined to be representative based on historical sample analysis results, an evaluation of remaining decommissioning work activities and potential water flow pathways. Figure 11-1 shows the planned sampling locations of liquid effluents. U.S. Nuclear Regulatory Commission (NRC) License No. SNM-33 (Reference 11-5) historically required a composite sample from the Site Dam and a weekly grab sample from the Sanitary Wastewater Treatment Plant Outfall.

Section 11.2.4 describes typical controls that will be utilized to ensure that effluent limits are not exceeded, and that project ALARA goals are met.

The MDC target for laboratory analysis of gross beta and gross alpha analysis of liquid effluents is established at 5 percent of the applicable 10 CFR 20 Appendix B, Table 2, Column 2 limits. For on-site analysis using a proportional counting system, the MDC target is 25 percent of the applicable 10 CFR 20 Appendix B, Table 2, Column 2 limits.

The MDC targets for laboratory analysis of isotopic uranium and Tc-99 in liquids are 1.0 pCi/L and 3.0 pCi/L, respectively. These targets are less than the guidance in Regulatory Guide 4.16 (Reference 11-6) of 5 percent the applicable limits.

Consistent with Reference 11-6, isotopic analysis of effluent water samples will be performed on selected samples, e.g., quarterly composite samples and individual samples, when radioactivity concentrations are in excess of the Investigation Level. The methods of isotopic analysis for

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[1] The volume of air that can be sampled in a 24-hour period limits the MDC that can be achieved for alpha radioactivity within a reasonable counting time. To achieve an MDC for gross alpha radioactivity at 2.5E-15 µCi/ml would require a 10-hour count time for the sample and 10-hour count time for background radioactivity. Thus, the MDC for samples collected on a daily frequency is set at the ALARA Investigation Limit of 50 percent of the license limit, instead of 5 percent as recommended by Regulatory Guide 4.16.

uranium and Tc-99 in liquids are ASTM 3972-90M and USEPA 906.0M, respectively, or equivalent.

To accurately measure and report the concentration in effluent releases, Westinghouse will collect representative samples of the WTS effluent to determine the quantities and average concentrations of radionuclides discharged using methods based on Regulatory Guide 4.16, Revision 1, December 1985 (Reference 11-7).

Specifically, representative samples will be collected of the liquid release from the WTS for the subsequent determination of the quantities and average concentrations of radionuclides discharged in any liquid effluents that could reach an unrestricted area. For continuous releases from the WTS, representative composite samples will be collected from the release. For batch releases from the WTS, a representative sample of each batch will be collected.

- Batch sampling: Prior to discharge of the liquid effluent a sample is obtained from the WTS tank. The sample, or portion of it, is analyzed on-site and/or sent for laboratory analysis.
- Composite sampling: During continuous discharge of the liquid effluent, a composite sampler is used to obtain a sample that is representative of the effluent and proportional to the volume of the discharge over time. During WTS operations involving continuous releases, this sample will be submitted for laboratory analysis on a weekly frequency. In addition, a daily grab sample will be taken during WTS continuous discharges and analyzed for gross alpha and gross beta radioactivity to ensure the WTS is operating as expected.

Batch sampling will be the primary sampling method during the initial operational phase of the WTS while its performance characteristics (e.g., decontamination factors) are being established. This will allow the retention of effluent until sample results are obtained for comparison against release limits and ALARA goals.

Composite sampling will be the primary sampling method once batch sampling has demonstrated that the WTS system performs in a consistent manner and produces liquid effluent of known quality even when challenged by a range of concentrations. This is a reasonable approach for potentially variable influent concentrations based on a large decontamination factor that will be validated during initial system operation.

As required by 10 CFR 70.59, HDP will submit to NRC, on a semi-annual basis, the results of effluent sample analyses.

11.2.3.5 Quality Assurance Requirements

Chapter 13.0 of the DP addresses general quality assurance requirements for activities conducted at the Hematite Site. Surveillances and audits of effluent and environmental monitoring programs are conducted to verify compliance with 10 CFR 20 requirements as they relate to effluent monitoring and control. As a minimum, an annual audit of the EEMP is conducted at the Hematite Site.

Effluent and environmental samples will be collected and analyzed in accordance with site procedures and the requirements of quality assurance sample collection, chain of custody, handling, storage, instrument maintenance and control, including sample analysis requirements, and vendor laboratory analysis requirements. These procedures contain the appropriate level of detail to ensure that handling, storage, and analysis requirements are conducted in accordance with the HDP Project Quality Plan (PQP) (Reference 11-7). A list of site procedures implementing the requirements of the EEMP is provided in Table 11-4.

Environmental samples that are shipped to vendor laboratories for analysis are handled in accordance with the quality assurance and quality control (QA/QC) programs for the individual laboratory. QA/QC programs for vendor laboratories are reviewed and validated by the HDP quality assurance organization.

11.2.4 EFFLUENT CONTROL PROGRAM

This section describes the effluent controls utilized at the HDP, and the actions necessary to ensure site air and liquid effluents are within applicable regulatory limits and are ALARA.

11.2.4.1 Process And Engineering Controls

Process and engineering controls will be evaluated for each major decommissioning work activity to minimize the quantity of radioactive material in effluents. Examples of process controls include: recycling; leak reduction; and, modification of facilities, procedures and/or operations. Examples of engineering controls include: encapsulation; water mists; filtration; adsorption; containment; and storage. Process and engineering controls will be implemented commensurate with specific work activities and the potential for increasing occupational exposure or effluent concentrations. Effluent controls for various decommissioning activities will be described in written procedures or work instructions/plans for each major decommissioning activity. The primary effluent control measures are expected to be dust suppression, erosion control, and the WTS.

Dust suppression control measures are applicable to demolition, soil remediation, soil excavation, Burial Pit area remediation and waste loading/hauling. Techniques to control the generation of fugitive dust typically include water sprays to suppress dust and reduce dust transport by wind.

Erosion and sedimentation controls will be utilized for work activities impacting site soils, and are intended to protect nearby streams and surface water from potential contaminants. These controls serve to minimize erosion and restrict the transport of sediment within the project area, and typically include: filter fabric (silt fence), erosion control blankets, and channels or barriers to direct storm water flows away from areas of contamination.

These controls may be temporary or permanent, and will be based on Best Management Practices implemented in accordance with the site EEMP. Final or permanent erosion controls will be implemented, as necessary, after work activities in a specific area are completed; these typically involve seeding to establish a vegetative cover.

Potentially contaminated water could result from decommissioning operations, from precipitation that enters work areas, or from excavations that encounter ground water. HDP will use Best Management Practices to divert surface water away from work areas, collect water from work areas (such as open excavations), and to prevent sedimentation run-off.

Examples include:

- Earthen berms to keep surface water from entering impacted areas.
- Coverings/tarps to keep precipitation from soil stockpiles.
- Sumps within excavation areas during remediation.
- A French drain from the railroad loading area to a collection pond.
- Temporary Storage Tanks (Baker Tanks) within/near excavation areas.

Collected water will be directed to the WTS for analysis, and treatment as required. Surface water that pools on-site will be tested. Based upon the concentration of contaminants (by comparison to the ALARA goals) it may either be filtered to remove solids using the bag filters and released to a permitted outfall or processed through the WTS. The treatment will reduce contaminants in the water through the following mechanisms:

- Settling of solids entrained in the liquid. This mechanism is enhanced by the addition of polymer flocculent via a mixer as water enters a series of quiescent settling tanks.
- 10-25 micron bag filters to remove remaining entrained material.
- Granulated activated carbon filters (virgin anthracite media) to remove volatile organics and Technetium 99.
- 5-10 micron bag filters to remove remaining entrained material.

- Ion exchange units (zeolite media) to remove metals, including uranium and the Technetium-99 that may be in solution as uranyl or technetium carbonate complexes and are not removed by the granulated activated carbon filters.

The Temporary Storage Tanks within/near excavation areas will be within a secondary containment and located within/near the excavation(s). The secondary containment will be installed and operated to prevent the migration of wastes or accumulated liquid outside the secondary containment area. The WTS is located inside Building 230 within a lined secondary containment designed to hold the contents of the two largest tanks in the WTS as a conservative measure to meet the minimum requirement of 40 CFR Part 112 (Reference 11-8) for above ground storage tank that it hold 110 percent of the volume of the largest tank . The Temporary Storage Tanks, WTS, and connecting piping are above-ground systems with the exception of underground crossings at on-site travel paths. The underground crossings will be configured so the system piping passes through larger conduit that has above ground openings on both ends to allow visual inspection. Temporary Storage Tanks, WTS, and connecting piping, including underground crossings will be visually inspected on a daily basis during operation of this equipment. Controls and practices such as spill prevention and overfill controls will be employed. Components will be removed from service, repaired or replaced following equipment failure or malfunction resulting in a leak.

While the Sanitary Wastewater Treatment Plant is not designated to receive contaminated liquids, liquid effluent from this system is monitored in accordance with the EEMP. The Sanitary Wastewater Treatment Plant is primarily a below-grade system that is checked on a weekly basis for proper operation. The Sanitary Wastewater Treatment Plant was installed in 1977-78. Consistent with sanitary systems of that era, it does not have a leak detection system. There are no discharges to public sewer systems from the Hematite Site.

The site evaporation ponds are no longer in use for process wastewater; a more complete discussion of these ponds is provided in Section 2.3.2. The primary evaporation pond was modified by: (1) draining the pond and processing the drained water; (2) installing an impervious liner; (3) installing a drain line along the new railroad spur and loading pad to direct surface water to the pond; and (4) installing a pump and piping from the pond to the Water Treatment System. The impervious liner will prevent the collected water from entering the ground water, and should serve to reduce the migration of contaminants in the soil near the primary evaporation pond.

11.2.4.2 Estimate Of Public Dose

An estimate of public dose will be performed by comparing effluent concentration values with the applicable regulatory limits in 10 CFR 20, Appendix B. This comparison will be performed semi-annually and reported to the Project Director. Estimates of public dose are expected to be maintained at less than10 percent of the 10 CFR 20, Appendix B values, applicable to the EEMP.

The potential dose to the public has been evaluated for the recent period beginning year 2005 based on measurements of the radioactivity concentrations in air, liquid effluents, and radiation levels as measured by thermoluminescent dosimeters (TLDs) at the perimeter of the impacted area. These measurements were performed during implementation of the environmental monitoring program. The average and maximum concentrations in air were 4 percent, and 8 percent of the annual average limits specified in 10 CFR 20, Appendix B. The average and maximum concentrations in liquid effluents released to surface water were less than 1 percent of the annual average limits specified in 10 CFR 20, Appendix B.

Since the basis of these concentration limits assumes that a potential exposure to a member of the public would not exceed 100 millirem per year, the potential average annual dose from air and liquid effluents released from the site would be approximately 5 millirem per year during the course of 2005-2008.

The range of the annual net radiation level, as measured at a quarterly frequency, for the period 2006 through the first quarter 2009 was 20 millirem to 22 millirem per year. To obtain the net radiation level, the response of a TLD placed at an off-site location was subtracted from the response of the TLD located at the perimeter.

In consideration of the concentration of radioactivity in waste materials to be excavated and prepared for shipment, from a qualitative perspective it is not anticipated that the potential dose to the public will increase by any measurable amount as a result of decommissioning activities. To ensure this, and consistent with Regulatory Guide 8.37 (Reference 11-4) action levels have been established for comparison to quantitative analysis results of samples.

11.2.5 EFFLUENT LIMITS

Historically, the SNM-33 License Applications committed to using the effluent limits for gross alpha radioactivity based on the U-234 (Class Y) effluent limits from Table 2 of Appendix B to 10 CFR 20. Similarly, the effluent limits for gross beta radioactivity were based on the Th-234 (class Y) effluent limits from Table 2 of Appendix B to 10 CFR 20. These pre-DP effluent limits were based on effluents associated with fuel fabrication processes within the buildings. In consideration of the potential change in the radionuclide mixture in effluents during remediation, DP effluent limits have been derived for remediation work based on the concentrations in buried debris and soil. The derivation is explained in the following subsections, and is based on Notes 1 and 4 to Appendix B of 10 CFR 20.

11.2.5.1 Radionuclides Included in Derivation

Based on DP Section 4.0, the primary radionuclides of concern (ROCs) requiring direct measurement in effluents are U-234, U-235, U-238, Tc-99, Th-232, Ra-226. Data for the balance of the ROCs (Am-241, Np-237, Pu-239/240) have been included in the derivation of Site effluent limits but do not require further measurement.

With the exception of Tc-99, the ROCs primarily emit alpha radiation, so the derivation of the gross alpha radioactivity effluent limits use the limits for these ROCs from Table 2 of Appendix B to 10 CFR 20. In addition, Th-228 is included in this derivation since it is a long-lived daughter.

The derivation of gross beta radioactivity effluent limits use the limits from Table 2 of Appendix B to 10 CFR 20 for the ROC Tc-99 and the daughters of the other ROCs where the daughters have restrictive limits in Table 2 of Appendix B to 10 CFR 20. Specifically, Pa-233 (daughter of Np-237), Pb-210 (daughter of Ra-226), Ra-228 (daughter of Th-232), Th-231 (daughter of U-235), and Th-234 (daughter of U-238) are used in the derivation.

The daughters are assumed to be in equilibrium with the parent ROC, even though sufficient time has not passed in all cases for equilibrium to have been established (e.g., Pb-210). This is a conservative assumption resulting in a more restrictive derivation.

11.2.5.2 Relationship of ROCs in Soil to ROCs in the Air and Liquid Effluents

The concentration of ROCs in soil directly relate to the concentration of ROCs in effluent since remediation disturbs soil, creating the potential for dust (airborne effluent) and suspended solids to be present in water (liquid effluent). Considering the relative solubility of the ROCs, applying the activity fractions of radionuclides in the mixture for the soil as the activity fractions in the water effluent is reasonable since the compounds containing the more restrictive radionuclides (lower limits) are on balance less soluble in water (based on the distribution coefficients identified in DP Table 5-6) than the uranium radionuclides.

11.2.5.3 Equation for Derived Effluent Limits

The comparison of the total amount of gross alpha and beta radioactivity in effluents to the effluent limits for an identified mixture of radionuclides is based on the sum of the ratios of the concentration of each radionuclide in the effluent divided by the radionuclide effluent limit. The result is compared to the limit expressed as the unity value (1.0).

$$\frac{C_A}{EL_A} + \frac{C_B}{EL_B} + \cdots \leq 1 \qquad (11\text{-}1)$$

Where,

C_A = *concentration of radionuclide A*
C_B = *concentration of radionuclide B*
EL_A = *effluent concentration limit for radionuclide A*
EL_B = *effluent concentration limit for radionuclide B*

An effective EL value, EL_{eff}, is determined based on the following equation. The gross alpha and beta radioactivity effluent limits will be effective limits as derived here:

$$\frac{C_A}{EL_A} + \frac{C_B}{EL_B} + \ \dots = \frac{C_A + C_{B+\cdots}}{EL_{eff}}$$

Dividing both sides by $C_A + C_{B+}\cdots$

$$\frac{C_A/(C_A + C_B + \dots)}{EL_A} + \frac{C_B/(C_A + C_{B+\cdots})}{EL_B} + \ \dots = \frac{1}{EL_{eff}}$$

$$Substituting\ f_A = \frac{C_A}{C_A + C_{B+\cdots}}, f_B = \frac{C_B}{C_A + C_B + \ \dots}, \dots$$

$$\frac{f_A}{EL_A} + \frac{f_B}{EL_B} + \ \dots = \frac{1}{EL_{eff}}$$

Where,

f_A = activity fraction of radionuclide A in the mixture
f_B = activity fraction of radionuclide B in the mixture

Regrouping terms yields the final expression

$$EL_{eff} = \frac{1}{\frac{f_A}{EL_A} + \frac{f_B}{EL_B} + \dots} \qquad (11\text{-}2)$$

11.2.5.4 Radionuclide Mixtures

Based on a review of the soil data in the Hematite Radiological Characterization Report (Reference 11-7), the site has three areas where the radionuclide mixture in effluents will be different and warrant independent consideration. These areas are:

- An area of elevated Ra-226 contamination in the north end of the burial pit area (the Derived Effluent Limits for the Elevated Ra-226 Area will apply).
- An area of elevated Th-232 contamination in the south end of the burial pit area (the Derived Effluent Limits for the Elevated Th-232 Area will apply).
- The balance of the Site impacted area (the Derived Effluent Limits for the Balance of the Site will apply).

The Derived Effluent Limits for the Elevated Ra-226 Area and the Derived Effluent Limits for the Elevated Th-232 Area will only apply to effluents associated with soil/debris from the elevated Ra-226 and Th-232 areas, which are shown on Figure 11-2. The Derived Effluent Limits for the Balance of the Site will apply outside of these two areas. The boundaries of these areas are based on the conceptual survey units identified in DP Chapter 14.0.

For each of these areas, the available soil characterization data was used to determine effective air and water effluent concentration limits for gross alpha and beta radioactivity. To avoid

skewing the derivation, soil characterization data at soil depths that was not contaminated was not used. Ra-226 and Th-232 results were corrected for background contribution by subtracting the mean site background concentration value of 0.9 and 1.0 pCi/g, respectively.

11.2.5.4.1 Elevated Ra-226 Area

The radionuclide mixture within the elevated Ra-226 area is tabulated below.

Gross Alpha Constituents		
Radionuclide	**Average Activity** (pCi/g)	**Activity Fraction**
Am-241 (W)	9.90E-03	1.3E-04
Np-237 (W)	3.69E-02	4.9E-04
Pu-239 (W)	4.62E-03	6.2E-05
Ra-226 (W)	5.55E+00	7.4E-02
Th-232 (Y)	0.00E+00*	0.0E+00
U-234(Y)	5.63E+01	7.5E-01
U-235 (Y)	2.76E+00	3.7E-02
U-238 (Y)	1.00E+01	1.3E-01

Gross Beta Constituents		
Radionuclide	**Average Activity** (pCi/g)	**Activity Fraction**
Tc-99 (W)	2.07E+00	1.0E-01
Np-237 [Pa-233] (W)	3.69E-02	1.8E-03
Ra-226 [Pb-210] (W)	5.55E+00	2.7E-01
Th-232 [Ra-228] (Y)	0.00E+00	0.0E+00
U-235 [Th-231] (Y)	2.76E+00	1.4E-01
U-238 [Th-234] (Y)	1.00E+01	4.9E-01

* Average concentration was less than zero, so value was set to zero for calculations.

The Derived Effluent Limits for the Elevated Ra-226 Area are calculated using Equation 11-2 to be as tabulated below. These values are reflected in DP Tables 11-1a and 11-1b.

Media	**Gross Alpha Derived Effluent Limits for Elevated Ra-226 Area** (μCi/ml)	**Gross Beta Derived Effluent Limits for Elevated Ra-226 Area** (μCi/ml)
Air	5.5E-14	2.2E-12
Water	2.3E-07	3.7E-08

11.2.5.4.2 Elevated Th-232 Area

The radionuclide distribution within the elevated Th-232 area is tabulated below.

Gross Alpha Constituents		
Radionuclide	**Average Activity** (pCi/g)	**Activity Fraction**
Am-241 (W)	0.00E+00*	0.0E+00
Np-237 (W)	0.00E+00*	0.0E+00
Pu-239 (W)	0.00E+00*	0.0E+00
Ra-226 (W)	2.85E-01	2.1E-04
Th-232 (Y)	1.23E+02	9.0E-02
U-234(Y)	9.35E+02	6.9E-01
U-235 (Y)	3.75E+01	2.8E-02
U-238 (Y)	1.45E+02	1.1E-01

Gross Beta Constituents		
Radionuclide	**Average Activity** (pCi/g)	**Activity Fraction**
Tc-99 (W)	2.87E-01	9.4E-04
Np-237 [Pa-233] (W)	0.00E+00*	0.0E+00
Ra-226 [Pb-210] (W)	2.85E-01	9.3E-04
Th-232 [Ra-228] (Y)	1.23E+02	4.0E-01
U-235 [Th-231] (Y)	3.75E+01	1.2E-01
U-238 [Th-234] (Y)	1.45E+02	4.7E-01

* Average concentration was less than zero, so value was set to zero for calculations.

The Derived Effluent Limits for the Elevated Th-232 Area are calculated using Equation 11-2 to be as tabulated below. These values are reflected in DP Tables 11-1a and 11-1b.

Media	**Gross Alpha Derived Effluent Limits for Elevated Th-232 Area** (μCi/ml)	**Gross Beta Derived Effluent Limits for Elevated Th-232 Area** (μCi/ml)
Air	2.8E-14	4.9E-12
Water	1.6E-07	1.5E-07

11.2.5.4.3 Balance of the Site

The radionuclide distribution for the balance of the Site (outside of the elevated Ra-226 and Th-232 areas) is tabulated below.

Gross Alpha Constituents		
Radionuclide	**Average Activity** (pCi/g)	**Activity Fraction**
Am-241 (W)	3.27E-02	2.1E-04
Np-237 (W)	1.98E-02	1.3E-04
Pu-239 (W)	1.60E-03	1.0E-05
Ra-226 (W)	5.05E-02	3.2E-04
Th-232 (Y)	0.00E+00*	0.0E+00
U-238 (Y)	9.67E+00	6.2E-02
U-234(Y)	1.40E+02	9.0E-01
U-235 (Y)	6.22E+00	4.0E-02
U-238 (Y)	9.67E+00	6.2E-02

Gross Beta Constituents		
Radionuclide	**Average Activity** (pCi/g)	**Activity Fraction**
Tc-99 (W)	4.06E+01	7.2E-01
Np-237 [Pa-233] (W)	1.98E-02	3.5E-04
Ra-226 [Pb-210] (W)	5.05E-02	8.9E-04
Th-232 [Ra-228] (Y)	0.00E+00*	0.0E+00
U-235 [Th-231] (Y)	6.22E+00	1.1E-01
U-238 [Th-234] (Y)	9.67E+00	1.7E-01

* Average concentration was less than zero, so value was set to zero in calculations.

Based on the above data, derived effluent limits are calculated using Equation 11-2 to be as follows. However, in the following assessment of variability in the soil data, lower derived effluents limits are calculated and will be used instead as the Derived Effluent Limits for the Balance of the Site.

Media	**Gross Alpha Derived Effluent Limits** (μCi/ml)	**Gross Beta Derived Effluent Limits** (μCi/ml)
Air	5.1E-14	3.2E-10
Water	3.0E-07	7.3E-06

11.2.5.4.4 Consideration of Variability in Soil Data

In order to evaluate the effect of variability in activity fractions for the balance of the site, derived effluent limits were developed using the methodology above for the individual conceptual survey units containing previously identified individual elevated thorium samples outside the elevated Th-232 and Ra-226 areas identified above. The results of this investigation, tabulated below, warranted a modification to the derived effluent limits in the preceding section.

Survey Unit	**Air** (μCi/ml)		**Water** (μCi/ml)	
	Gross Alpha	**Gross Beta**	**Gross Alpha**	**Gross Beta**
LSA-02-01	5.0E-14	9.7E-11	2.9E-07	2.3E-06
LSA-02-02	5.1E-14	1.3E-10	2.9E-07	2.4E-06
LSA-02-03	5.1E-14	2.5E-10	2.7E-07	6.3E-06
LSA-05-01	5.2E-14	2.2E-10	2.9E-07	4.9E-06
LSA-08-05	5.1E-14	2.6E-10	2.9E-07	6.2E-06
LSA-08-10	5.2E-14	3.8E-10	3.0E-07	9.5E-06
LSA-08-11	5.1E-14	5.3E-10	2.9E-07	1.8E-05
LSA-08-12	5.1E-14	2.2E-10	2.8E-07	4.7E-06
LSA-08-14	5.0E-14	1.8E-10	3.0E-07	4.4E-06
Minimum	**5.0E-14**	**9.7E-11**	**2.7E-07**	**2.3E-06**

The Derived Effluent Limits for the Balance of the Site will be set to the lowest value for calculated above, as shown in the table below. These values are reflected in DP Tables 11-1a and 11-1b.

Media	**Gross Alpha Derived Effluent Limits for the Balance of the Site** (μCi/ml)	**Gross Beta Derived Effluent Limits for the Balance of the Site** (μCi/ml)
Air	5.0E-14	9.7E-11
Water	2.7E-07	2.3E-06

11.2.5.5 Verification of Mixture

Verification of the isotopic activity fractions for air effluent will be performed in the following manner:

- Individual air samples at Stationary Environmental Air Sampling Stations will be composited on a quarterly basis and submitted for isotopic analysis. The isotopic results will then be used to calculate the effluent concentration directly and to verify the annual activity fractions. Prior to the quarterly composite analysis, the samples from Stationary Environmental Air Sampling Stations will be analyzed weekly for gross alpha and beta radioactivity. These quarterly isotopic results will be used in demonstrating compliance with 10 CFR 20.1101(d).
- Individual air effluent/environmental samples from Perimeter Sampling Locations and from Stationary Environmental Air Sampling Stations that reach or exceed the 50 percent (based on gross alpha or beta results) of the limit will be submitted for isotopic analysis. This isotopic data will be used to verify the annual activity fractions that were used in the derivation of the effective effluent limits.
- Individual air samples will be collected from the air exhaust from the Soil Vapor Extraction System (SVES) on a weekly basis during SVE operations. Sample analysis will include gamma spectroscopy (to detect U-235, U-238, Th-232 and Ra-226) followed by beta analysis for Tc-99, Ra-228, and Pb-210. This isotopic data will be used to verify the annual activity fractions that were used in the derivation of the effective effluent limits.

Verification of the isotopic activity fractions for liquid effluent will be performed in the following manner:

- Weekly composite and quarterly grab liquid effluent samples will be analyzed for gross alpha and beta radioactivity with the results compared against the appropriate effluent limit from Table 11-1b. Individual samples which exceed 50 percent of the effluent limit will be submitted for isotopic analysis (Uranium, Th-232, Ra-226, Ra-228, Tc-99, and Pb-210). This isotopic data will be used to verify the annual activity fractions that were used in the derivation of the effective effluent limits.
- In addition to the weekly analysis for gross alpha and beta radioactivity, isotopic analysis (Uranium, Th-232, Ra-226, Ra-228, Tc-99, and Pb-210) will be routinely performed on quarterly composite samples from the Water Treatment System and from discharges at the locations of Outfalls #001, 002, and 006, unless analyzed for isotopic content during a calendar quarter as a result of exceeding the threshold in the preceding bullet. This isotopic data will be used to calculate the effluent concentration directly and to verify the annual activity fractions that were used in the derivation of the effective effluent limits.

11.2.5.6 Application of Mixture Verification Results

On a quarterly basis, quarterly isotopic air and water effluent data from the Stationary Environmental Air Sampling Stations, SVES, WTS, and the locations of Outfalls #001, 002 and 006 will be used to calculate effective effluent limits. This calculation is required for the Derived Effluent Limits for the Balance of the Site and, if they were in effect, the Derived Effluent Limits for Elevated Ra-226 Area or Derived Effluent Limits for Elevated Th-232 Area. If a quarterly calculated effluent is found to be more than 10 percent different than the effluent limit that was in effect, then that quarterly calculated effluent limit will be applied to effluents for that quarter and on a go forward basis. It should be noted that if the existing effluent limit is more conservative (lower) than the quarterly calculated effluent limit, then no change is required. The RSO shall approve the calculations of the quarterly derived effluent limits and ensure application when the quarterly calculation meets the requirements of this paragraph.

Soil and liquid data (representative of air effluent and liquid effluent, respectively) obtained after remediation of soil in areas requiring the Derived Effluent Limits for Elevated Ra-226 or Th-232 Areas may be used to justify resumption of the Derived Effluent Limits for the Balance of the Site in those post-remediation areas. The RSO shall approve the post remediation justification for using the Derived Effluent Limits for the Balance of the Site instead of the Derived Effluent Limits for Elevated Ra-226 or Th-232 Areas.

11.3 ENVIRONMENTAL MONITORING PROGRAM

The HDP Environmental Monitoring Program is contained in the EEMP. Locations for air particulate, soil, vegetation, ground water and surface water monitoring are established and documented as part of this program. The environmental monitoring locations and the associated sampling parameters are provided in Table 11-5.

Revisions to the Environmental Control and Monitoring Program contained in this document may be made without prior NRC approval for the following conditions, provided POC approval is obtained:

- Adequate justification is provided to discontinue sampling an effluent discharge location, or
- Adequate justification is provided to relocate an effluent monitoring location (e.g., remediation must occur at the original location and effluent flow path altered), and
- Adequate justification is provided that the change does not otherwise necessitate a change to the license application.

Additional or more frequent samples may be taken as required for special studies and evaluations. The quarterly environmental monitoring results shall be reviewed for trends using the non-parametric Mann-Kendall test, or equivalent and a graphical analysis to identify patterns that would otherwise go unnoticed using purely statistical methods. The graphical analysis will include the historical mean plus 3 sigma (calculated under stable, pre-remediation conditions). Measurements which exceed this historical mean plus 3 sigma range may be flagged for further evaluation as outliers. If an adverse trend is identified in the sampling data, the EH&S Manager and RSO will be notified and a review of the associated decommissioning activity(s) will be conducted to identify changes to work methods and/or engineering controls that should be implemented, as appropriate, to reduce effluent concentrations to ALARA levels.

Environmental samples will be collected and analyzed in accordance with approved site procedures (see Table 11-4) and governing programs.

11.4 REFERENCES FOR CHAPTER 11.0

11-1 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," Volume 2, Revision 1.

11-2 Westinghouse Electric Company Document No. HDP-PO-EM-001, "Effluent and Environmental Monitoring Plan."

11-3 Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

11-4 U.S. Nuclear Regulatory Commission, Regulatory Guide 8.37, "ALARA Levels for Effluents from Material Facilities," July 1993.

11-5 U.S. Nuclear Regulatory Commission, "License No. SNM-33 (Docket No. 70-36)."

11-6 NRC Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexaflouride Production Plants" Revision 1, December 1985.

11-7 Westinghouse Electric Company Document No. HDP-PO-QA-001, "Project Quality Plan (PQP)."

11-8 Code of Federal Regulations, Title 40, Part 112, "Oil Pollution Prevention".

Table 11-1a.

Air Effluent Monitoring and Limits

Type	**Frequency**	**Analytical Method**	**Concentration Limits[a]** (ALARA Goals[a] and Investigation Levels[b] are 20 and 50 percent, respectively, of the Limits)
Stationary Environmental Air Sampling Stations[c] (AS-A, AS-B, AS-C, AS-D, AS-F, AS-G, AS-H, AS-I [enabled mobility])	Weekly Media Change-out	Gross Alpha/Beta	Gross Alpha: 5.0E-14 µCi/ml Gross Beta: 9.7E-11 µCi /ml
	Single Elevated Weekly Sample (≥ 50% Limit)	Isotopic Analysis	Analytes: Uranium, Tc-99, Th-232, Ra-226 Limits: Column 1, Table 2, Appendix B, 10 CFR 20
	Quarterly Composite of Weekly Samples (each station)	Isotopic Analysis	Analytes: Uranium, Tc-99, Th-232, Ra-226 Limits: Column 1, Table 2, Appendix B, 10 CFR 20
Perimeter Sampling Locations	Daily[d] Media Change-out	Gross Alpha/Beta	<u>Derived Effluent Limit for Elevated Th-232 Area</u> Alpha: 2.8E-14 µCi /ml Beta: 4.9E-12 µCi /ml <u>Derived Effluent Limit for Elevated Ra-226 Area</u> Alpha: 5.5E-14 µCi/ml Beta: 2.2E-12 µCi/ml <u>Derived Effluent Limit for the Balance of the Site</u> Alpha: 5.0E-14 µCi/ml Beta: 9.7E-11 µCi/ml
	Single Elevated Daily Sample (≥ 50% Limit)	Isotopic Analysis	Analytes: Uranium, Tc-99, Th-232, Ra-226 Limits: Column 1, Table 2, Appendix B, 10 CFR 20
Soil Vapor Extraction System	Weekly Media Change-out (during operations)	Isotopic Analysis	Analytes: Uranium, Tc-99, Th-232, Ra-226, Ra-228, Pb-210 Limits: Column 1, Table 2, Appendix B, 10 CFR 20

[a] Values reflect the annual average concentration.
[b] Values reflect the results obtained from individual samples.
[c] These permanent sampling stations are also a part of the Environmental Monitoring Program.
[d] Daily frequency may be extended up to a week to achieve the necessary MDC for Derived Effluent Limits for Elevated Ra-226 or Th-232 Areas.

Table 11-1b.
Liquid Effluent Monitoring and Limits

Location	Frequency	Analytical Method	Concentration Limits[a] (ALARA Goals[a] and Investigation Levels[b] are 20 and 50 percent, respectively, of the Limits)
At Outfall #001 (Sanitary Wastewater Treatment Plant)	Weekly Composite	Gross Alpha/Beta	Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml
	Quarterly Composite	Isotopic Analysis	Analytes: Uranium, Th-232, Ra-226, Ra-228, Pb-210, and Tc-99 Limits: Column 2, Table 2, Appendix B, 10 CFR 20
At Outfall #002[c] (Site Pond Dam)	Weekly Composite	Gross Alpha/Beta	Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml
	Quarterly Composite	Isotopic Analysis	Analytes: Uranium, Th-232, Ra-226, Ra-228, Pb-210, and Tc-99 Limits: Column 2, Table 2, Appendix B, 10 CFR 20
At Outfall #004 (East Culvert)	Quarterly Grab	Gross Alpha/Beta	Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml
At Outfall #005 (South Culvert)	Quarterly Grab	Gross Alpha/Beta	Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml
At Outfall #006 (Soil Laydown Area)	Weekly Composite	Gross Alpha/Beta	Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml
	Quarterly Composite	Isotopic Analysis	Analytes: Uranium, Th-232, Ra-226, Ra-228, Pb-210, and Tc-99 Limits: Column 2, Table 2, Appendix B, 10 CFR 20
Water Treatment System[e]	Weekly Composite with Daily Grab or Tank Batch	Gross Alpha/Beta	Derived Effluent Limit for Elevated Th-232 Area - Gross Alpha: 1.6E-07 µCi/ml Gross Beta: 1.5E-07 µCi/ml Derived Effluent Limit for Elevated Ra-226 Area - Gross Alpha: 2.3E-07 µCi/ml Gross Beta: 3.7E-08 µCi/ml Derived Effluent Limit for the Balance of the Site - Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml
	Quarterly Composite (not including batch)	Isotopic Analysis	Analytes: Uranium, Th-232, Ra-226, Ra-228, Pb-210, and Tc-99 Limits: Column 2, Table 2, Appendix B, 10 CFR 20

Table 11-1b.
Liquid Effluent Monitoring and Limits

Location	Frequency	Analytical Method	Concentration Limits[a] (ALARA Goals[a] and Investigation Levels[b] are 20 and 50 percent, respectively, of the Limits)
Ponded Surface Water Not Requiring Treatment	Batch Grab Sample (multiple aliquots) Prior to Each Release	Gross Alpha/Beta	Derived Effluent Limit for Elevated Th-232 Area* - Gross Alpha: 1.6E-07 µCi/ml Gross Beta: 1.5E-07 µCi/ml Derived Effluent Limit for Elevated Ra-226 Area* - Gross Alpha: 2.3E-07 µCi/ml Gross Beta: 3.7E-08 µCi/ml Derived Effluent Limit for the Balance of the Site* - Gross Alpha: 2.7E-07 µCi/ml Gross Beta: 2.3E-06 µCi/ml * Runoff from the waste staging area which accumulates in the evaporation pond will be evaluated using Derived Effluent Limits for Elevated Th-232 or Ra-226 Areas, as appropriate, when soil from the radium or thorium areas is present or being loaded at the Loading Pad.
Any	Single Sample Gross Alpha or Beta Activity ≥ 50% Limit	Isotopic Analysis	Analytes: Total U, Th-232, Ra-226 (when gross alpha >50%) Ra-228, Pb-210, and Tc-99 (when gross beta > 50%) Limits: Column 2, Table 2, Appendix B, 10 CFR 20

[a] Values reflect the annual average concentration.
[b] Values reflect the results obtained from individual samples.
[c] Outfall #003 effluent enters the Site Pond, which is monitored at Outfall #002.
[d] The sampling location is within Building 230, even though the discharge is at Outfall #003.

Table 11-2

Summary of Decommissioning Air Effluent Release Concentrations (2002 To 2008)

Reporting Period	Fraction Of Gross Alpha Limit (1)
2002	
1st Quarter(2)	---
2nd Quarter(2)	---
3rd Quarter	1.2E-10
4th Quarter	1.4E-01
2003	
1st Quarter	9.0E-02
2nd Quarter	1.0E-02
3rd Quarter	3.5E-03
4th Quarter	6.2E-04
2004	
1st Quarter	3.4E-03
2nd Quarter	9.9E-04
3rd Quarter	6.6E-04
4th Quarter	3.7E-03
2005	
1st Quarter	4.1E-03
2nd Quarter	7.3E-03
3rd Quarter	9.9E-03
4th Quarter	8.8E-03
2006 (3)	
No air effluent releases	
2007	
No air effluent releases	
2008	
No air effluent releases	

(1) Limit from 10 CFR 20, Appendix B, Table 2 is 5.0E-14 µCi/ml gross alpha.
(2) Prior to July 2002 air effluent values were reported as total Curies released rather than a fraction of the applicable effluent limit.
(3) Air effluents from building stacks ceased in 2006.

Table 11-3

Summary of Decommissioning Liquid Effluent Release Concentrations (2002 – 2008)

Location	Reporting Period Year	Reporting Period Month	Fraction Of Gross Alpha Limit (1)	Fraction Of Gross Beta Limit (1)
Site Dam	2002	Jan-June (2)	---	---
		July-Dec	1.9E-02	1.7E-03
	2003	Jan-June	7.3E-03	1.2E-03
		July-Dec	1.6E-02	2.4E-03
	2004	Jan-June	5.4E-03	1.1E-03
		July-Dec	9.3E-04	1.2E-03
	2005	Jan-June	4.9E-03	8.0E-04
		July-Dec	1.2E-02	1.3E-03
	2006	Jan-June	5.7E-03	1.1E-03
		July-Dec	1.1E-02	1.1E-03
	2007	Jan-June	5.7E-03	9.0E-04
		July-Dec	1.8E-02	3.1E-04
	2008	Jan-June	3.8E-03	8.2E-04
		July-Dec	1.2E-02	1.5E-03
Sanitary Wastewater Treatment Outfall	2002	Jan-June (2)	---	---
		July-Dec	1.3E-01	6.0E-03
	2003	Jan-June	1.2E-01	6.6E-03
		July-Dec	1.0E-01	8.7E-03
	2004	Jan-June	1.5E-01	1.1E-02
		July-Dec	2.5E-01	1.9E-02
	2005	Jan-June	5.2E-02	6.8E-03
		July-Dec	5.7E-02	9.1E-03
	2006	Jan-June	1.6E-01	1.6E-02
		July-Dec	2.2E-01	2.3E-02
	2007	Jan-June	2.8E-01	2.2E-02
		July-Dec	1.0E-01	1.0E-02
	2008	Jan-June	1.8E-01	2.5E-02
		July-Dec	1.5E-01	1.2E-02
Water Treatment System Discharge (3)	2008	Jan-June	0	0
		July-Dec	1.7E-01	7.6E-04

(1) Limits from 10 CFR 20, Appendix B, Table 2 are 3.0E-7 µCi/ml gross alpha and 5.0E-6 µCi/ml gross beta.
(2) Prior to July 2002 liquid effluent values were reported as total Curies rather than a fraction of the applicable effluent limit(s).
(3) Individual sample.

Table 11-4

Effluent and Environmental Monitoring Procedures

Following the prescribed requirements and steps in procedures for the following topics is a key element of obtaining consistent and accurate analysis results.

Subject*	Description
Groundwater Sampling HDP-PR-EM-011 HDP-PR-EM-010	Requirements for the proper collection, handling, and documentation of surface water samples and evaluation of sample results. Samples shipped off-site for analysis.
Surface Water Sampling HDP-PR-EM-003	Requirements for the proper collection, handling, and documentation of surface water samples and evaluation of sample results. Samples analyzed on-site for gross alpha and gross beta radioactivity and shipped off-site for analysis.
Soil Sampling HDP-PR-EM-004	Requirements for the proper collection, handling, and documentation of soil samples and evaluation of sample results. Samples shipped off-site for analysis.
Sediment Sampling HDP-PR-EM-005	Requirements for the proper collection, handling, and documentation of sediment samples and evaluation of sample results. Samples shipped off-site for analysis.
Vegetation Sampling HDP-PR-EM-007	Requirements for the proper collection, handling, and documentation of vegetation samples and evaluation of sample results. Samples shipped off-site for analysis.
Airborne Sampling HDP-PR-HP-301	This procedure ensures the proper collection, handling, analysis, documentation, and evaluation of airborne particulate samples of air effluent and of the ambient environmental air. Samples analyzed on-site for gross alpha and gross beta radioactivity and shipped off-site for analysis.
Environmental Radiation Monitoring HDP-PR-HP-316	Requirements for the proper handling, documentation, and evaluation of environmental TLDs, which are placed around the Westinghouse Hematite Site boundary to verify external dose to public is below Federal Regulations.
Chain Of Custody HDP-PR-QA-006	Requirements for documenting a continuous chain of custody of samples sent for off-site analysis. This procedure is invoked by sampling procedures.
Tennelec Operation HDP-PR-HP-414	Required steps for measuring the amount of gross alpha and gross beta radioactivity in a sample using the Tennelec LB-5100 proportional counting instrument.
Liquid Sample Analysis HDP-PR-HP-313	Required steps for preparing liquid samples for analysis using the Tennelec LB-5100 instrument.
QA/QC Supplier Evaluation HDP-PR-QA-010	Requirements for Quality Assurance to evaluate a vendor's quality program – applies to off-site laboratories.

*The procedure numbers may change in the course of an update.

Table 11-5

Environmental Monitoring Program Sampling

Sample Medium	Sampling Points	Collection Frequency	Sample Type	Type of Analysis[a]
Air	Seven (7) Fixed Locations and one (1) movable around the HDP (AS-A, AS-B, AS-C, AS-D, AS-F, AS-G, AS-H, and AS-I [movable])	Continuous and Analyze Weekly	Particulate	Gross Alpha Gross Beta
Surface Water	Joachim Creek Upstream of Site Creek Outfall (SFW-J)	Monthly	Grab	
	Joachim Creek Downstream of Site Creek Outfall (SFW-M)	Monthly	Grab	
	Joachim Creek and Site Creek Confluence (SFW-H)	Quarterly	Grab	
	Outfall #003 (SFW-C)	Quarterly	Grab	
	Former Outfall #004 (SFW-D)[b]	Quarterly	Grab	
	Former Outfall #005 (SFW-E)	Quarterly	Grab	
	Outfall #006 (SFW-F)	Quarterly	Grab	
	Planned Ditch North of Burial Pits (SFW-K)	Quarterly	Grab	
	Planned Ditch from Laydown Area (SFW-L)	Quarterly	Grab	
Ground Water	On-Site Well (GW-A)	Monthly	Grab	
	Off-Site Well (GW-R)	Quarterly	Grab	
	Three (3) Evaporation Pond Downgradient Monitoring Wells (GW-U[c], GW-G, GW-Y)	Quarterly	Grab	
	Three (3) Process Buildings Downgradient Monitoring Wells (GW-T, GW-Z, GW-S)[c]	Quarterly	Grab	
	Three (3) Burial Pits Downgradient Monitoring Wells (HB-F, GW-BB, GW-W)	Quarterly	Grab	
Soil	Six (6) Site Area Locations (SL-B, SL-E, SL-F, SL-G, SL-H, SL-J)	Quarterly	Grab	Gross Alpha Gross Beta Alpha Spect. Liq. Scint.
Vegetation	Six (6) Site Area Locations (VG-B, VG-E, VG-F, VG-G, VG-H, VG-J)	Quarterly	Grab	Gross Alpha Gross Beta Alpha Spect. Liq. Scint. Gamma Spect.
Sediment	Joachim Creek and Site Creek Confluence Below Site Dam (SD-H)	Annually	Grab	
TLD	Perimeter of Central Tract (TLD-A to TLD-F, TLD-H, TLD-J to TLD-N, TLD-P to TLD-T)	Quarterly Read	N/A	N/A

[a] Analytical procedures that provide equivalent data quality (e.g., gamma spectroscopy, liquid scintillation, kinetic phosphorescence analysis, or alpha spectroscopy) also may be selected based upon the sample medium. For gross alpha radioactivity, uranium (Class Y) may be used as a comparator and for gross beta radioactivity, Thorium-234 may be used.

[b] Until removed as interference to BMPs for burial pit remediation work.

[c] Until removed as interference to remediation, when available downgradient wells will be monitored until replacement wells can be established.

Figure 11-1

Effluent and Environmental Sampling Locations during Decommissioning Operations

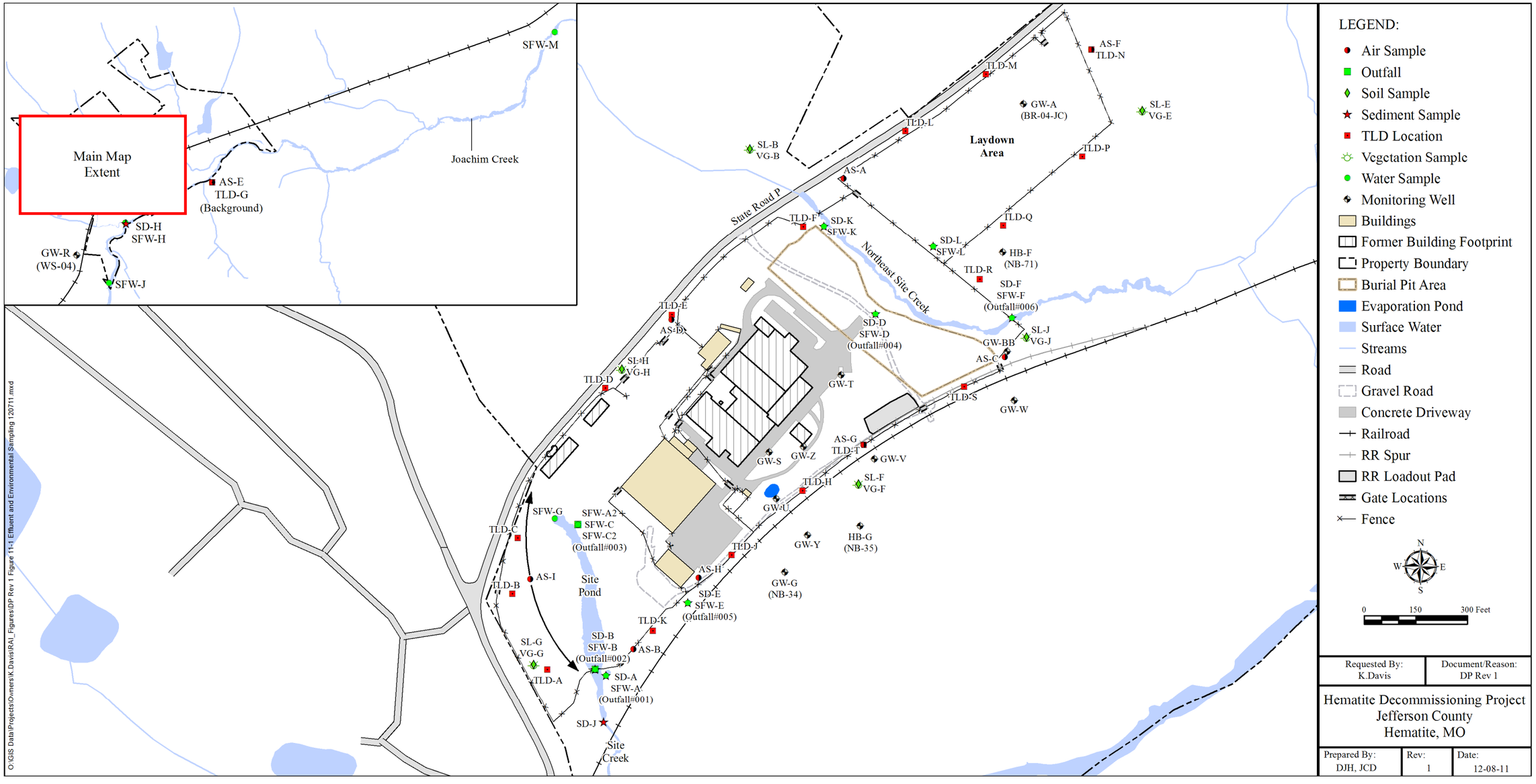
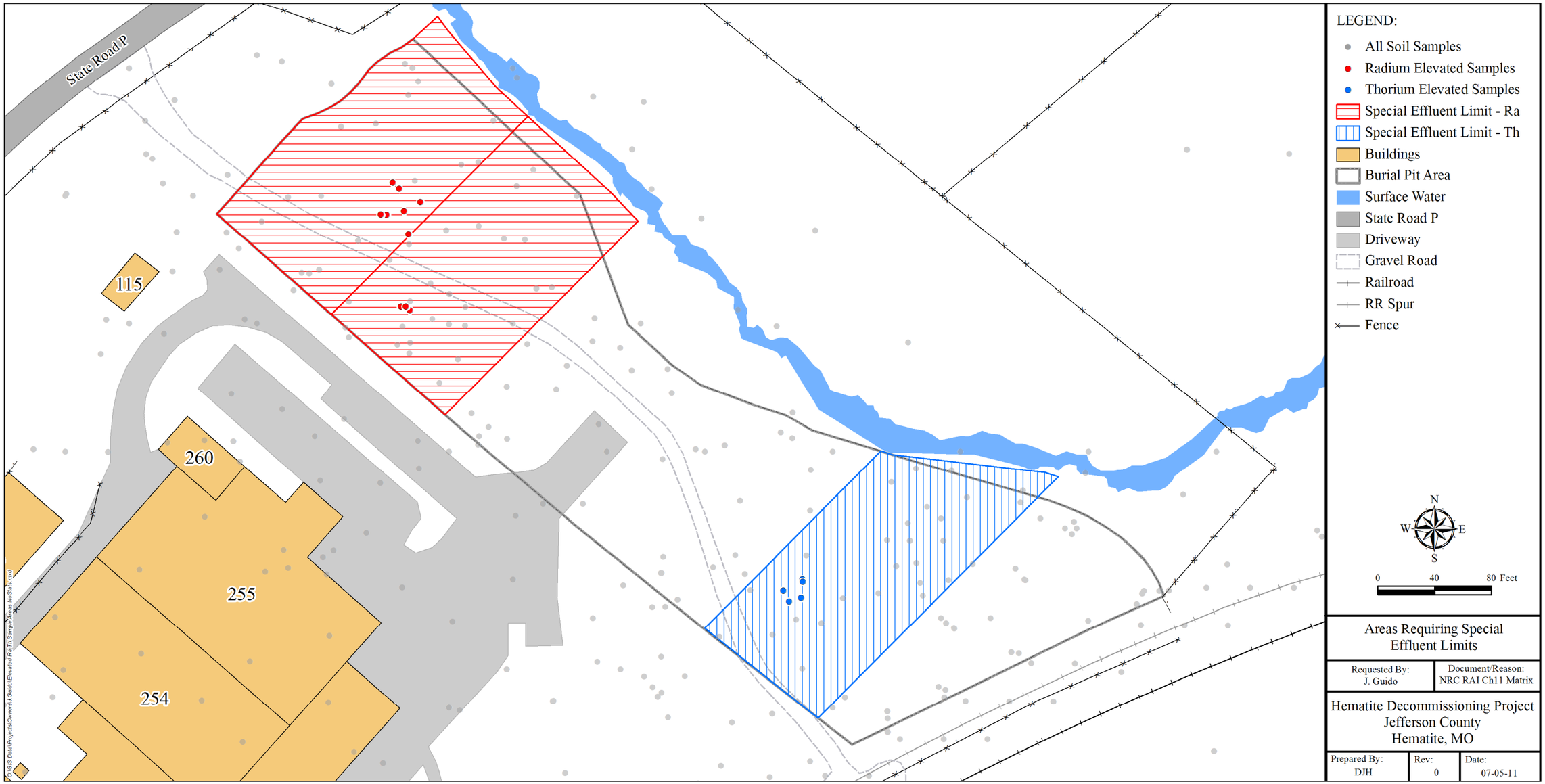


Figure 11-2

Areas Requiring Special Effluent Limits



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ACRONYMS AND ABBREVIATIONS

BMPs	Best Management Practices
CFR	Code Of Federal Regulations
DCGL	Derived Concentration Guideline Level
DNAPL	Dense Non-Aqueous Phase Liquids
DOT	United States Department of Transportation
EPA	United States Environmental Protection Agency
FID	Flame Ionization Detector
HCl	Hydrochloric Acid
HDP	Hematite Decommissioning Project
HF	Hydrofluoric Acid
HNO3	Nitric Acid
HRCR	Hematite Radiological Characterization Report
HSA	Historical Site Assessment
IP-1	Industrial Packaging -1
KOH	Potassium Hydroxide
LDR(s)	Land Disposal Restriction(s)
LLRW	Low Level Radioactive Waste
MC&A	Material Control And Accounting
MDNR	Missouri Department Of Natural Resources
NaI	Sodium Iodide
NCS	Nuclear Criticality Safety
NRC	U.S. Nuclear Regulatory Commission
PCE	Perchloroethylene
PID	Photo Ionization Detector
RCRA	Resource Conservation and Recovery Act
Ra-226	Radium-226
RG	Remediation Goal
ROD	Record Of Decision
SVE	Soil Vapor Extraction
SNM	Special Nuclear Material
Tc-99	Technetium-99
TCE	Trichloroethylene
TCLP	Toxicity Characteristic Leaching Procedure
Th-232	Thorium-232

ACRONYMS AND ABBREVIATIONS
(Continued)

U-234	Uranium-234
U-235	Uranium-235
U-238	Uranium-238
VOC(s)	Volatile Organic Compound(s)
WAC	Waste Acceptance Criteria
WMTP	Waste Management And Transportation Plan
WTS	Water Treatment System

12.0 RADIOACTIVE WASTE MANAGEMENT PROGRAM

12.1 PROGRAM DESCRIPTION

The Hematite Decommissioning Project (HDP) radioactive waste management program will safely control the handling, packaging, transport and disposal of solid, liquid and mixed wastes generated during decommissioning activities. Site activities will be performed in accordance with the Waste Management and Transportation Plan (WMTP), (Reference 12-1). The WMTP is an integrated plan for management of radioactive and non-radioactive waste, and is designed to protect the personnel, public and environment during the generation, handling and transportation of waste.

Based upon the Historical Site Assessment (HSA), (Reference 12-2) and the data from the Hematite Radiological Characterization Report (HRCR), (Reference 12-3), the three major types of radioactive waste expected to be encountered during the decommissioning include:

- Solid radioactive waste, including radioactive asbestos waste;
- Liquid radioactive waste; and
- Mixed waste.

Waste management activities which are designed to address the expected waste types, are integrated into the remediation activities identified in Chapter 8.0 of this plan. As waste is exposed during remediation activities, additional waste characterization and initial segregation steps are implemented. After removal of the waste, final characterization and segregation of waste into the appropriate waste type is completed. Based upon waste type, waste will either be directly loaded into containers or stockpiled awaiting packaging, treatment, or transportation. To prevent the spread of contamination during storage, piles will be maintained utilizing Best Management Practices (BMPs). Packaged waste will be transported using approved, qualified and permitted carriers. If required, based upon the quantity of Special Nuclear Material (SNM) in a package, additional security measures as outlined in Physical Security Plan, (Reference 12-4) will be implemented.

12.1.1 PACKAGING

Packaging will be completed using shipping containers compatible and appropriate for the waste type. The majority of the radioactive waste will be packaged in railcars equipped with lids, or Industrial Packaging-1 (IP-1) flexible bags. Other packaging that may be used includes: intermodal containers, metal boxes, and drums. Packages and packaging used for the shipment of radioactive waste will be inspected before loading, and before shipment, to ensure the package is appropriate for the waste and package integrity has not been compromised during loading and handling.

Radioactive waste activity will be carefully monitored to ensure packages meet the limits associated with the fissile material exemptions in Title 10 CFR Part 71.15, Exemption from Classification as Fissile Material, (Reference 12-5), the receiving facility waste acceptance criteria (WAC), radioactive materials license and/or specifically-granted exemptions.

12.1.2 TRANSPORTATION

HDP intends to use both rail service and truck/trailer as means of transportation of radioactive waste. Regardless of the mode of transportation, verification will be performed to ensure each carrier is permitted to carry the load, and that appropriate security plans are in effect. Pre-transportation checklists from approved procedures will be used to ensure compliance with applicable United States Department of Transportation (DOT) and U.S. Nuclear Regulatory Commission (NRC) regulations. If required, based upon the quantity of fissile material in a conveyance meeting the criteria of SNM of Low Strategic Significance as identified in Reference 12-4, additional security measures as outlined in the Physical Security Plan (Reference 12-4) will be implemented.

12.1.3 QUALITY ASSURANCE REQUIREMENTS

HDP expects that all shipments containing fissile material will meet the fissile exempt requirements as described in Title 49 Code of Federal Regulations Part 173.453, Fissile Material – Exceptions (Reference 12-6). Therefore, an approved quality assurance program in accordance with Title 10 Code of Federal Regulations Part 71, Subpart H, Quality Assurance (Reference 12-7) is not required.

SNM resulting from decommissioning work is expected to be shipped as fissile exempt, using mechanical size reduction as necessary. As a contingency, the U.S. Department of Energy (DOE) or an NRC licensee (other than Westinghouse-Hematite) may be utilized in the unlikely event that a shipment of fissile material is required. For such a contingency, the DOE or NRC licensee would take possession of the material at the Hematite Site. Shipments by the U.S. Department of Energy (DOE) would be performed in accordance with DOE quality requirements. Shipment by another NRC licensee would be in accordance with that licensee's approved quality assurance program in accordance with 10 CFR 71 Subpart H – Quality Assurance (Reference 12-7).

12.2 SOLID RADIOACTIVE WASTE

12.2.1 WASTE GENERATION

The majority of the solid radioactive waste generated during decommissioning will be associated with excavation activities. The two general types of solid radioactive waste expected to be generated are:

- Demolition debris such as concrete rubble, building materials, piping, conduit and exhumed Burial Pit waste; and,
- Volumetrically contaminated material such as soil, sediment, charcoal, resin and limestone.

The estimated volume of each type of solid radioactive waste, and the expected waste classification, is shown in Table 12-1, Estimated LLRW Volumes and Classification. Solid radioactive waste generated by the project is expected to be Class A waste.

The specific isotopes and activity associated with the solid radioactive waste will be dependent on the location where the waste is generated:

- The Burial Pit area is contaminated primarily with uranium isotopes: uranium-234 (U-234), uranium-235 (U-235) and uranium-238 (U-238), including the associated decay daughter products; and to a lesser extent radium-226 (Ra-226), and thorium-232 (Th-232).
- Radiologically impacted soil areas including the barn area, red room roof burial area, site pond, site creek and leach field areas are contaminated primarily with the uranium isotopes. A portion of these areas are also contaminated to a lesser extent with technetium-99 (Tc-99).
- The area southeast of and under the processing buildings is contaminated primarily with uranium isotopes, their associated decay daughter products and Tc-99.

A summary of radioisotopes and the estimated activities for solid radioactive waste are provided in Table 12-2, Summary of LLRW Radioisotopes and Estimated Activity.

Solid waste may be generated from the accumulation of hazardous and radioactive materials on filter media associated with the Water Treatment System (WTS) and soil vapor extraction (SVE) systems.

12.2.2 WASTE HANDLING

Radioactive waste handling is expected to be accomplished to a large extent with mechanical equipment such as excavators, front-end loaders and trucks. There will be occasions when smaller equipment and/or hand shoveling will be required. Materials will be exposed at the excavation site and characterization data will be obtained using field radiological survey instrumentation (e.g., Sodium Iodide (NaI) detectors), hazardous material survey instruments (e.g., Photo Ionization Detector (PID)/Flame Ionization Detector (FID) instruments, X-ray fluorescent detectors) and sampling with subsequent laboratory analysis. Based upon the initial field characterization and assays, materials will be segregated into general categories as discussed below:

- Soils acceptable for re-use, as-is;
- Soils acceptable for re-use, after treatment of volatile organic compounds (VOCs);
- Low Level Radioactive Waste (LLRW) designated for subsequent disposal;
- Mixed Waste, which could exist in three (3) subcategories, corrosive, toxic and reactive; and,
- Items or material requiring Material Control and Accounting (MC&A) procedures or Nuclear Criticality Safety (NCS) controls, such as recoverable SNM or intact containers.

12.2.2.1 Soil Acceptable For Re-Use

For soil to be acceptable for re-use, the soil must meet the appropriate Derived Concentration Guideline Level (DCGL) and the requirements specified in the Record of Decision (ROD) for chemical constituents as approved by the Missouri Department of Natural Resources (MDNR).

Soils which contain levels of VOCs above the remediation goals (RG) stated in the ROD are acceptable for re-use if they are treated to less than the RG, meet the Land Disposal Restrictions (LDRs), and meet the appropriate DCGL for soil. Treatment of soils for VOCs may be performed using SVE, as described further in Section 12.4.3.4.

12.2.2.2 Low Level Radioactive Waste (LLRW)

LLRW is the largest waste stream expected, consisting primarily of radiologically contaminated soil and debris including water treatment media. LLRW will initially be identified at the remediation location, and then transferred to an accumulation area or container. Containers will be used for waste streams such as contaminated asbestos, to minimize handling requirements.

After sufficient material has accumulated, the waste will be assayed and may be transferred to a waste holding area pending loading into rail cars or other containers.

12.2.2.3 Mixed Waste

Mixed waste was not identified during the site characterization process. If mixed waste exists at the Hematite Site, it will most likely originate from material inside intact containers or materials leaking from damaged containers from the Burial Pit Area. The ground water samples and a few of the soil samples taken during the characterization of the site indicated elevated concentrations of some chemicals, but at levels less than the limits for classification as Resource Conservation and Recovery Act (RCRA) hazardous materials.

If mixed wastes are identified during the remediation process, the waste will be segregated and placed in tanks or containers pending treatment. The specific types of mixed waste that could reasonably exist or be generated during remediation, and the associated treatment methods, are identified in Section 12.4.

If an unexpected mixed waste stream is generated during the waste removal and handling process, the waste will be segregated. Appropriate disposition of the waste will be determined. The waste may be treated on-site or packaged and shipped to an appropriately-licensed and permitted facility.

12.2.2.4 Intact Containers

The types of intact containers expected to exist in burial locations include glass, plastic or stainless steel. Due to the oxidizing nature of the burial environment, a limited number of intact carbon-steel containers are expected. Intact containers will be removed from the excavation location, over-packed and transferred to an evaluation area. There they will be opened and the contents evaluated for SNM and hazardous material content, and appropriate treatment and disposition.

12.2.2.5 Nuclear Criticality Safety

Prior to excavation of designated areas, a radiological and visual survey will be performed. Actions specified in section 8.5.1 will be implemented upon identification of an object/intact container or an elevated radioactivity measurement in excess of the NCS Exempt Material Limit.

12.2.2.6 Material Control and Accounting (MC&A)

If recoverable SNM is identified, the SNM will be packaged, transferred to the evaluation area, assayed and placed into the MC&A inventory system. Recoverable SNM may be shipped as re-useable material to an authorized licensee. If the SNM is determined to be not recoverable, a waste disposition plan which includes the appropriate MC&A requirements will be developed.

12.2.3 STORAGE

Storage of solid radioactive waste, either bulk or containerized, will be on the Hematite Site in waste holding areas that are appropriately posted and maintained within the fenced restricted area. Positive control of the Hematite Site is maintained by site security on a continuous basis. Chapter 8.0 provides a conceptual site layout showing the planned waste holding and lay-down areas.

Prior to packaging, solid radioactive waste will be stored in piles near excavation locations, at a package loading site, or at a stockpile location until sufficient material is accumulated for packaging. Erosion, storm water control and contamination control measures (including covering the piles, except when access to the pile is required) will be instituted for storage piles of radioactive material.

12.2.4 WASTE DISPOSITION

Solid radioactive waste will be shipped for processing and/or disposal to an appropriately licensed or permitted facility. The following facilities may be used for solid radioactive waste disposition:

- Studsvik, Inc., Memphis, TN;
- Energy Solutions, Inc., Oak Ridge, TN;
- Energy Solutions, Inc., Clive, UT; and
- U S Ecology, Idaho, Grandview, ID.

Regardless of the facility selected for processing and disposal, waste will be prepared for transport in accordance with the receiving facility's WAC, facility license or NRC approved exemption, and the DOT regulations.

12.3 LIQUID RADIOACTIVE WASTE

12.3.1 WASTE GENERATION

The radioactive liquids that are expected to be generated during decommissioning include lubricants, such as oil and hydraulic fluid, from the maintenance of on-site equipment. Table 12-3 summarizes the volumes of liquid radioactive waste expected to be generated during the project. Liquid radioactive waste is expected to meet the criteria of Class A waste. The expected radionuclides and estimated activity of the lubricant waste stream is summarized in Table 12-4.

12.3.2 WASTE HANDLING

Radioactively-contaminated used lubricants associated with equipment maintenance will be collected and accumulated. Typically, no more than six drums of used lubricants will be accumulated prior to shipment for waste disposition. The six drums would be over-packed and shipped in a standard oil bin.

12.3.3 WASTE STORAGE

Storage of liquid radioactive waste will be in waste holding areas that are appropriately posted and maintained. Positive control of the HDP is maintained by site security on a continuous basis.

12.3.4 WASTE DISPOSITION

Liquid radioactive waste will be shipped for processing and/or disposal to an appropriately licensed or permitted facility. The following facilities may be used for waste disposition:

- Permafix of Florida, Inc., Gainesville, FL;
- DSSI, Inc., Kingston, TN;
- EnergySolutions, Inc., Clive, UT;
- EnergySolutions, Inc., Oak Ridge, TN; and
- U S Ecology, Inc., Grandview, ID.

Regardless of the facility selected for processing and disposal, waste will be prepared for transportation in accordance with the receiving facility's WAC, facility license or NRC approved exemption and the DOT regulations.

12.4 MIXED WASTE

Mixed waste will be managed to meet the selected treatment or disposal site's WAC and land disposal restrictions, prior to off-site disposal. A step-wise, conservative process, using the knowledge gained during excavation and handling of waste from each area, will be used for managing potential mixed waste in subsequent excavations. This process strategy consists of the following components:

- Systematically assay excavated material, as necessary, to identify characteristic hazardous waste;
- Prepare the excavated waste materials, as needed, to develop a physical waste form amenable to treatment; and
- Employ on-site, ex-situ treatment technologies, as necessary, to reduce the hazardous waste characteristics. Available technologies include:
  - Vapor extraction for removal of VOCs, or treatment of ignitable waste;
  - Chemical neutralization of low or high pH (corrosive) waste; and
  - Use of stabilization agents to bind hazardous constituents, to meet disposal facility waste acceptance criteria.

To varying degrees, waste treatment technologies may generate by-product waste which also may need to be managed as mixed waste. Treatment systems will be used that, to the extent practicable, minimize the generation of mixed waste. Both vapor-phase and aqueous-phase treatments may be used to separately address suspended particulate radionuclides (e.g., filters) and VOCs (e.g., activated carbon), in order to minimize the generation of mixed waste. By-products (typically filters and activated carbon) from the ex-situ VOC treatment process and the WTS will be treated, as needed, to minimize or eliminate treatment residues exhibiting the characteristics of a hazardous waste. Residual radioactive waste will be packaged and transported to licensed off-site disposal facilities in compliance with the site's WAC.

Mixed waste that is not subject to on-site treatment will be prepared for transport to an appropriately licensed facility for processing to meet the ultimate disposal facilities WAC, site license or NRC approved exemption.

12.4.1 WASTE GENERATION

Industrial chemicals were used and consumed by production operations that continued through termination of plant operations. Based upon the HSA, characteristic hazardous wastes that could reasonably be present on-site, include:

- Corrosives, acids (HCl, HNO3, HF) and bases (KOH);
- Toxic volatile organic compounds (TCE, PCE and their degradation products);
- Toxic Dense Non-Aqueous Phase Liquids (DNAPL);
- Toxic heavy metals (lead, mercury); and,
- Reactive-pyrophoric Uranium fines.

The analysis of characterization sampling did not identify the presence of mixed waste in any location on the site. For the purpose of effective scheduling and planning, as a contingency, it will be assumed that approximately 5,700 cubic feet of solid mixed waste and 1,300 liters of liquid mixed waste will be generated during decommissioning. Table 12-5 summarizes the estimated radionuclides, the activity and waste classification.

Wastes will initially be evaluated for the presence of chemical and radiological contamination at excavation or removal locations. Wastes that show evidence of chemical contamination will be transferred from the excavation location, or process equipment, to a temporary accumulation area or container. After sufficient material is accumulated, the waste will be assayed and transferred to a waste treatment tank or packaged and shipped for offsite treatment and disposal.

12.4.2 WASTE STORAGE

Waste storage areas will be appropriately posted based on the hazards of concern and will be located inside a fenced and controlled area. Approved procedures will be implemented for the routine inspection of containers or tanks in storage or use. The inspection requirements will comply with the requirements of Title 40 Code of Federal Regulations, Part 262, Standards Applicable to the Generators of Hazardous Waste (Reference 12-10).

Except for pyrophoric uranium metal fines which may be discovered in intact containers and discreet items (e.g., lead bricks) and bulk liquids, potentially mixed wastes will be accumulated for testing and processing in “tanks”, built on existing process building slabs. Pyrophoric uranium metal fines will be stored in closed, tight containers, pending processing. Discreet items and bulk liquids will be stored with compatible materials in metal boxes or drums.

12.4.3 WASTE TREATMENT

Although characterization sample analysis did not identify mixed waste at the Hematite Site, provisions will be made to treat or stabilize mixed wastes that could potentially be present on-site. The treatment methods for corrosive, toxic, and reactive mixed waste streams are summarized below. Approved procedures will be utilized to implement the treatment methods used for the HDP.

12.4.3.1 Corrosives-Acids And Bases

Due to the corrosive nature of this type of waste, intact metal containers are not expected to exist in a burial environment that would contain any significant quantity of an acid or base. Due to the burial environment and the variable ground water level at the Hematite Site, corrosives in soil should be attenuated and no longer present. Upon identification, intact plastic or glass containers containing fluids will be evaluated for corrosive material. If acidic or basic liquids are identified, the waste will be neutralized with the appropriate reagent to achieve an acceptable pH in accordance with waste management procedures. The treatment may yield a final waste form that is up to twice the amount of the original waste form. No additional effluents are generated by neutralization.

12.4.3.2 Toxic-Heavy Metal Contamination

Metal contaminated soil could exist in the Burial Pit area, and any required remediation areas are identified in the ROD. Though not specifically identified during the HSA process and also not detected during soil sampling, it is recognized that lead was a commonly used metal for shielding, plumbing, paint and roofing. Mercury was commonly used in various thermometers, gauges and switches for process equipment. Arsenic levels in soil above site background were identified in approximately 10 percent of soil samples from areas co-located with radiological contaminants slated for excavation.

Soils contaminated with metals, at concentrations failing the TCLP test, will be mixed with an appropriate stabilization agent based upon the material of concern. This will bind the hazardous constituents in order to meet TCLP testing requirements and LDRs. The treatment may yield a final waste form that is up to five times the volume of the original waste form. No additional effluents will be generated by stabilization process. Lead forms will be recycled. If the lead cannot be recycled it will be shipped off-site for macro-encapsulation prior to disposal.

12.4.3.3 Toxic-Dense Non-Aqueous Phase Liquid (DNAPL)

DNAPL is a general term for a class of chemicals that do not readily mix with water. Specific examples of DNAPL include chlorinated solvents, coal tar, creosote, polychlorinated biphenyls, mercury and extra heavy crude oil. If present, the material would originate from intact plastic or glass containers containing fluids or from pockets of liquid under ruptured containers. The

liquids will be collected in a compatible container and will be assayed to determine the exact nature of the liquid. Any form of DNAPL that can be incinerated or processed through solvent recovery will be packaged for off-site treatment and disposal at an approved off-site facility.

If the form of the DNAPL is mercury, then the mercury will be treated on-site (e.g. by an amalgamation processes and stabilized to pass TCLP testing) or sent offsite for disposal. The treatment will yield a final waste form that is up to five times the volume of the original waste form. No additional effluents are generated by stabilization.

If other forms of DNAPL waste are encountered that are not amenable to incineration, then absorption or solidification on-site may be required. The treatment may yield a final waste form that is five times the amount of the original waste form. No additional effluents are generated by absorption or solidification.

12.4.3.4 Toxic-Volatile Organic Compounds (VOCs)

Low levels of VOC contamination, below Toxicity Characteristic Leaching Procedure (TCLP) values but above RG values, have been detected in samples from the Burial Pit area and underneath process building slabs. Materials with VOCs that are below the TCLP criteria will be segregated from materials with VOCs that exceed the TCLP criteria. Materials that have VOC contamination in excess of TCLP values will be treated as mixed waste.

If on-site treatment of VOC is conducted, VOC treatment will be in treatment tanks by *ex-situ* soil vapor extraction (SVE). SVE uses a mechanical blower to induce a vacuum, which causes the VOCs to be stripped and volatilized into the air stream. The exhaust air is then treated to remove particulates and VOCs before it is emitted to the atmosphere. The exhaust air treatment consists of condensate trap, heat exchanger, condensate filter separator (condensate sent to water treatment system), HEPA filter, and vapor phase activated carbon filter.

Sampling for Radioactive Emissions.

- During SVE operations, a representative sample will be collected using a continuous sampler and a method consistent with ANSI N13.1-1999.

- The sampling media will include a charcoal adsorber in addition to the particulate filter to account for any radioactivity not collected on the particulate filter. The charcoal medium will be used until sufficient data are compiled to conclude that airborne radioactivity is not in a form requiring collection on a charcoal filter.

- The sample media will be removed and analyzed as defined in Table 11-1a, “Air Effluent Monitoring and Limits.”

Sampling for VOC Emissions.

- Daily grab samples at ports upstream and downstream of the activated carbon filter during operation of the SVE System.

- Analyzed by a photo-ionization detector (PID) for volatile organics (perchloroethylene, trichloroethylene and vinyl chloride). If the PID indicates the sample from the post-activated carbon sample port has concentrations of VOCs at 50 percent or more of the effluent release limits, then additional analysis for volatile organics will be performed according to, EPA Method TO-1 Method for the Determination of VOC in Ambient Air Using TENAX Adsorption and Gas Chromotography/Mass Spectrometry (GC/MS).

Tanks utilized for VOC treatment will be specially constructed cells that meet the tank and tank system definition in Title 40 Code of Federal Regulation Part 265, Interim Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (Reference 12-8). Waste treatment tanks will be lined and covered with a durable, flexible membrane liner, and designed, constructed and operated in accordance with United States Environmental Protection Agency (EPA) Management of Remediation Waste Under RCRA EPA530-F-98-026, October 1998 (Reference 12-9), interim status standards for tank systems. Solid structures will serve as the base/secondary liner (e.g., process building slabs, asphalt pavement) and structural sidewalls (e.g. Jersey barriers). The waste will be accumulated in the treatment tanks for sampling, and until sufficient material is available for effective treatment. Soil that initially was below the TCLP criteria and below reuse DCGLs is expected to be reused as fill material following on-site treatment; non-soil materials that initially were below the TCLP criteria are expected to be shipped for off-site disposal. Materials that exceed the TCLP criteria will be shipped for disposal and/or treatment at an off-site permitted facility. If treated on-site, treated materials will be tested to ensure the disposal facility WAC is met. SVE treatment does not change the volume of the waste form.

12.4.3.5 Reactive - Uranium Metal Fines

Due to the oxidizing nature of the burial environment, reactive pyrophoric-metals (e.g., uranium metal shavings, fines or chips) are not expected to be present; however, if they do exist, they would be expected to be found inside intact containers. The treatment process for reactive pyrophoric-metals involves solidification of the uranium metal fines in a silica concrete mixture. The solidification process will occur in a metal box with the approximate dimensions of 4 ft x 6ft x 2ft. The solidification mixtures will be controlled to limit the package uranium content to less than 180 grams of metallic U-235 per 360 kilograms of concrete if the uranium is special nuclear material, or 15,000 pCi/g U-238 if the metallic uranium is depleted uranium. The treatment may yield a final waste form that is 20 times the amount of the original waste form, but no more than 90 cubic feet is projected to be generated. No additional effluents are generated by

solidification. The treatment process will comply with the disposal site's WAC and will allow shipment for disposal.

12.4.4 WASTE DISPOSITION

Mixed waste may be shipped for processing and/or disposal at an appropriately licensed and permitted facility. The following is a list of facilities that may be used for mixed waste processing and disposal:

- Permafix of Florida, Inc., Gainesville, FL;
- Permafix of Tennessee, Oak Ridge, TN;
- EnergySolutions, Inc., Oak Ridge, TN;
- NSSI, Houston, TX;
- EnergySolutions, Inc. Clive, UT; and
- U S Ecology, Idaho, Grandview, ID.

Regardless of the facility selected for processing and disposal, waste will be prepared for transport in accordance with the receiving facility's WAC, facility license or NRC approved exemption and the DOT regulations.

12.4.5 PERMITTING

The Westinghouse Hematite Facility has registered with the EPA, Identification Number: MOR000012724 and the State of Missouri, as a large-quantity hazardous waste generator. Although treatment of hazardous waste generally requires a permit under EPA, Title 40 Code of Federal Regulations Part 264, Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities, (Reference 12-11) and State of Missouri regulations, there are exemptions to such permitting requirements. For example, a generator of hazardous waste can treat wastes it generates provided treatment occurs within a fully enclosed vessel, and within 90 days of the time the waste was generated.

The EPA has indicated in preambles to the federal hazardous waste rules (Reference 12-12 and Reference 12-13) and numerous interpretive statements in RCRA Permit Policy Compendium documents (Reference 12-14 through Reference 12-19), that hazardous waste regulations authorize on-site generator treatment of hazardous waste in fully-enclosed containers, tanks or containment buildings, without a RCRA permit. Generator requirements, including the storage management standards given in 40 CFR 262 (Reference 12-10), remain applicable.

Generators are not required to obtain a permit prior to storing or treating mixed waste (other than thermal treatment) under Title 40 Code of Federal Regulations 266, Subpart N, Conditional Exemption for Low Level Mixed Waste Storage, Treatment, Transportation and Disposal (Reference 12-20). Westinghouse intends to use this conditional exemption and notify MDNR of its use within 90 days after the storage unit for low-level mixed waste storage and treatment has been placed into service, as allowed in Reference 12-20.

12.5 REFERENCES FOR CHAPTER 12.0

12-1 Westinghouse Electric Company Document No. HDP-PO-WM-900, "Waste Management and Transportation Plan."

12-2 Westinghouse Electric Company Document No. DO-08-005, "Historical Site Assessment (HSA)."

12-3 Westinghouse Electric Company Document No. DO-08-003, "Hematite Radiological Characterization Report (HRCR)."

12-4 Westinghouse Electric Company Document, "Physical Security Plan" (PSP), Dated July 28, 2011.

12-5 Code of Federal Regulations, Title 10, Part 71.15, "Exemptions from Classification as Fissile Material."

12-6 Code of Federal Regulations, Title 49, Part 173.453, "Fissile Material Exceptions."

12-7 Code of Federal Regulations, Title 10, Part 71, Subpart H, "Quality Assurance."

12-8 Code of Federal Regulations, Title 40, Part 265, "Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities."

12-9 U.S. Environmental Protection Agency, EPA530-F-98-026, "Management of Remediation Waste Under RCRA," October 1998.

12-10 Code of Federal Regulations, Title 40, Part 262, "Standards Applicable To Generators Of Hazardous Waste."

12-11 Code of Federal Regulations, Title 40, Part 264, "Standards For Owners And Operators Of Hazardous Waste Treatment, Storage, And Disposal Facilities."

12-12 U.S. Federal Register, Volume 46, Page 2806, January 12, 1981.

12-13 U.S. Federal Register, Volume 51, Page 10186, March 24, 1986.

12-14 U.S. Environmental Protection Agency, RCRA Permit Policy Compendium, PPC 9453.1986(04), "Permitting Of Treatment Activities In A Generator's Accumulation Tanks Or Containers," July 25, 1986.

12-15 U.S. Environmental Protection Agency, RCRA Permit Policy Compendium, PPC 9471.1987(02), "RCRA Subtitle C Exemption for Wastewater Treatment and Elementary Neutralization Units," December 21, 1987.

12-16 U.S. Environmental Protection Agency, RCRA Permit Policy Compendium, PPC 9453.1987(03), "On-Site Treatment By Generators Under 262.34," July 1, 1987.

12-17 U.S. Environmental Protection Agency, RCRA Permit Policy Compendium, PPC 9453.1987(08), "Requested Re-interpretation of On-site Treatment Exemption," December 15, 1987.

12-18 U.S. Environmental Protection Agency, RCRA Permit Policy Compendium, PPC 9453.1991(02), "Treatment Of Hazardous Waste From Large Quantity Generators," September 20, 1991.

12-19 U.S. Environmental Protection Agency, RCRA Permit Policy Compendium, PPC 9451.1992(01), "Treatment In A Generator's 90-Day Containment Building," August 1, 1992.

12-20 Code of Federal Regulations, Title 40, Part 266, Subpart N, "Conditional Exemption for Low-Level Mixed Waste Storage, Treatment, Transportation and Disposal."

Table 12-1

Estimated LLRW Waste Volumes And Classification

Waste Type	Estimated Volume Cubic Feet[1]	Waste Classification
Demolition Debris	173,200	Class A
Volumetrically-Contaminated Material	801,500	Class A
Total :	**974,700**	

1 – Values rounded up to nearest 100 cubic feet

Table 12-2

Summary Of LLRW Radioisotopes And Estimated Activity

Radioisotope	Activity (Curies) [1]
U-234 + daughters	5.0
U-235 + daughters	0.1
U-238 + daughters	0.7
Tc-99	0.4
Ra-226 + daughters	< 3.16 E-03
Th-232 + daughters	< 1.50 E-03
Total :	**6.2**

1 – Values rounded up to nearest 0.1 Curie

Table 12-3

Estimated Liquid Radioactive Waste Volume And Classification

Liquid	Estimated Volume (liters)	Waste Classification
Used lubricants	1,200	Class A

Table 12-4

Summary of Detectable Radioisotopes In Lubricant Waste Stream and Estimated Activity

Radioisotope	Activity (picoCuries)
U-234	1,500
U-235	90
U-238	150
Tc-99	45
Total :	**1785**

Table 12-5

Summary Of Mixed Waste Radioisotopes, Estimated Activity And Waste Classification

Radioisotope	Quantity (Curies) [1]	Waste Classification
U-234 + daughters	0.03	Class A
U-235 + daughters	0.01	Class A
U-238 + daughters	0.01	Class A
Tc-99	0.01	Class A
Ra-226	< 9.48 E-04	Class A
Th-232	< 4.5 E-04	Class A
Total :	**0.06**	

1 – Values rounded up to nearest 0.01 Curie

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ACRONYMS AND ABBREVIATIONS

ASME	American Society Of Mechanical Engineers
CD-R	Compact Disk - Recordable
CFR	Code Of Federal Regulations
EDMS	Electronic Document Management System
FSS	Final Status Survey
HDP	Hematite Decommissioning Project
IAEA	International Atomic Energy Association
ISO	International Organization For Standardization
M&TE	Measuring And Test Equipment
PQP	Project Quality Plan
QA	Quality Assurance
QC	Quality Control
QMS	Quality Management System

13.0 QUALITY ASSURANCE

Westinghouse Electric Company LLC (Westinghouse) has established a Quality Management System (QMS, Reference 13-1) to comply with regulatory and industry quality requirements for materials and services provided by Westinghouse, its contractors, subcontractors and suppliers. The QMS describes the Westinghouse commitments to quality assurance requirements in: ISO 9000 (Reference 13-2); ISO 9001 (Reference 13-3); 10 CFR 50, Appendix B (Reference 13-4); and ASME NQA-1 (Reference 13-5).

The QMS incorporates quality planning, and provides a framework for managing activities that consistently satisfy regulatory requirements. The QMS also provides for continual improvement of the quality management system, by monitoring processes based on their significance, measuring their effectiveness against objectives, and management selection of processes for improvement.

Activities affecting quality are documented in accordance with written manuals, procedures, instructions, specifications and drawings that contain appropriate criteria for determining whether prescribed activities have been satisfactorily accomplished. This documentation is established in the three distinct levels that integrate policies, procedures and working documents as follows (see Figure 13-1):

- Level 1 - Quality Management System;
- Level 2 - Westinghouse and Operational Organization Policies and Procedures; and
- Level 3 - Functional/Department/Plant Procedures and Work Instructions.

The QMS provides the basis for policies and procedures that implement a comprehensive quality management system. Implementing policies and procedures (Level 2 and Level 3) provide details for management, control and work processes.

As part of the QMS, Hematite Decommissioning Project (HDP) management has established Quality Assurance (QA) mandatory policy statements for the Hematite Site. The key policy elements expressed in these statements are as follows:

1. Nuclear safety is the highest priority and shall take precedence over matters concerning Hematite decommissioning and site restoration;
2. The environment, and the health and safety of employees and the public, is the primary consideration in the conduct and support of Hematite decommissioning activities and shall not be compromised; decisions which could impact these considerations shall be made conservatively; and

3. The Hematite Project Quality Plan (PQP, Reference 13-6), is an essential part of the HDP, and program requirements shall be adhered to in the performance of activities covered by the PQP.

Work performed by or for the HDP shall comply with the PQP, and will be implemented using a "graded quality assurance" approach; meaning requirements are applied as necessary, to achieve the level of quality specified. Using this approach, procedures are used to identify and implement control requirements for materials and services based on the complexity of the work, the function of the material or service, and the importance of materials, services, processes and activities.

The quality of work performed by or for the HDP shall be verified by audits, surveillances and inspections.

13.1 ORGANIZATION

The HDP Director is responsible for the overall quality of work through the establishment of requirements and processes. The HDP QA Manager is responsible for managing implementation of the HDP QA Program.

The HDP organizational structure including duties and responsibilities of each unit within the HDP QA organization is described in the PQP. Key personnel in the HDP organization are identified in Chapter 9.0, including a description of their responsibilities and qualifications.

Organizational responsibilities are described in the PQP. The achievement of quality in the performance of quality-related activities is the responsibility of each individual in the HDP organization, including contractors and subcontractors. The organizational structure and responsibility assignments are established such that:

- Quality is achieved and maintained by those responsible for performing work;
- Persons or organizations independent of those performing work will verify the achievement of quality objectives; and
- Individuals or organizations responsible for establishing and implementing PQP requirements may delegate work performed under those requirements to other qualified individuals, but shall retain responsibility for the achievement of quality objectives.

All Hematite project personnel (managers, employees and contractors) are responsible for implementing procedures required by the PQP. Hematite personnel are given authority commensurate with their procedural responsibility, including the authority to stop work that does not conform to established requirements. Stop work authority, including investigation, resolution, completion of corrective actions and authorization for re-starting work, is to be exercised in accordance with approved procedures.

13.2 QUALITY ASSURANCE PROGRAM

The HDP PQP (Reference 13-6) provides the site specific project quality requirements as they apply to the decommissioning site.

The Hematite PQP is a comprehensive Site program that establishes the QA requirements and management controls for safety involving license materials and quality-related activities (e.g., deeds, actions, processes, tasks or work) which influence the achievement or verification of quality requirements and objectives associated with decommissioning the Hematite Site.

The PQP provides measures to ensure:

- Control of decommissioning activities affecting the quality of work performed by or for the HDP (see focus areas below);
- HDP work activities can be performed without undue risk to the environment, and the health and safety of employees and the public;
- Personnel training and qualification;
- Self-Assessments and independence of QA personnel; and,
- Procedures, instructions and documents include quantitative and qualitative acceptance criteria for quality activities.

The Hematite PQP and its implementing procedures define the actions to be taken by Hematite management, employees and contractors during the performance of quality-related activities to ensure QA requirements are consistently met. The QA program is based on line and staff organizations which are held responsible and accountable for the quality of their assigned work, and for verifying the achievement of quality-related objective through audits, surveillances, assessments and reviews.

Contractors performing work are required to comply with the HDP PQP and supporting procedures. Contractor developed procedures that are quality-related will be reviewed and audited in accordance with the PQP.

HDP QA recognizes that there are certain aspects of decommissioning that require particular focus to ensure success, including radiation protection standards found in Code of Federal Regulations 10 CFR 20 (Reference 13-7). When physical work is completed, what remains are the site closure records (e.g., Final Status Surveys-[FSS]) verifying that the property and remaining buildings are suitable for unrestricted use (i.e., 10 CFR 20.1402 radiological criteria for unrestricted use), and that work has been performed in accordance with the appropriate

federal, state and local regulations. Quality control relative to site closure records is essential for successful license termination.

Chapter 14.0 provides a detailed description of the HDP FSS approach, including survey design, implementation, and documentation. Radiation protection and FSS quality assurance measures include:

- Specification of Quality Control (QC) replicate scan/survey rates for FSS measurements;
- Specification of QC requirements for each sample media, sample splits and duplicate rates;
- Specification of quality requirements for instrument calibration, maintenance and operability checks;
- Specification of quality requirements for instrumentation software; and
- Specification of rates for independent reviews of FSS survey packages by QA.

The following activities are also considered focus areas for HDP Quality Assurance:

- Radiological and hazardous material surveys and sampling relative to public and site personnel safety;
- Radiological dosimetry and hazardous material exposure measurements;
- Instrument calibration relative to radiological and hazardous material measurements;
- Training records for personnel performing radiological and hazardous material measurements;
- Survey and manifest documentation for radiological and hazardous material shipments;
- Surveys, measurements and documentation in support of Nuclear Criticality Safety;
- Surveys, measurements and documentation in support of Material Control and Accounting; and
- Effluent and environmental monitoring results and documentation.

13.3 DOCUMENT CONTROL

The HDP has established a document control system in accordance with the PQP, applicable to the following types of documents: quality assurance/quality control manuals; technical reports; procurement documents; non-conformance documents; documents related to computer codes; and QA/QC policies and procedures. Implementation of the HDP document control system includes the following:

- Identification of documents to be controlled;
- Distribution of documents approved for issuance, in accordance with updated and current distribution lists;
- Document control procedures to assure proper documents are accessible and are being used;
- Lists of documents controlled by organizations involved with activities affecting quality;
- Procedural requirements for protection of security related information, when necessary;
- Identifying and assuring proper documents are used in performing activities affecting quality;
- Identifying qualified individuals or organizations responsible for preparing, reviewing, approving and issuing documents, including revisions; and
- Recalling or identifying retired documents.

The HDP document control system includes control of quality records, which are completed documents that furnish evidence of the quality of items, services, and/or activities affecting quality. Quality records will be retained, reviewed, and controlled in accordance with established procedures which will identify the requirements and responsibilities for: records classification, legibility, identification, collection, filing, indexing, storage, distribution, retention, retrieval, and disposition. Records requirements for suppliers of items and services will be specified in procurement documents, as required. Suppliers' records systems will be verified and monitored during surveillance and audits. Quality records will be protected against deterioration, damage and/or loss in accordance with established procedures.

HDP Managers are responsible for ensuring that all activities affecting the quality of materials and services are accomplished in accordance with controlled documents such as quality system manuals, procedures, plans and work instructions. Procedures will be established which provide

for document review, approval, issue and changes, to ensure inclusion of technical, regulatory and quality requirements prior to implementation. Each manager with lead responsibility for a document or document series is responsible for establishing controls that define responsibility, authority, issue, use, and revision and control of the document or document series. All HDP personnel are responsible for ensuring that the correct revisions of HDP documents are used to perform HDP work.

13.4 CONTROL OF MEASURING AND TEST EQUIPMENT

HDP controls have been established and implemented in accordance with the PQP to ensure control and appropriate use of tools, gauges, instruments, and other Measuring and Test Equipment (M&TE) that affect or evaluate the quality of activities. A list of HDP M&TE and measurement reference standards, and their assigned locations or custodians, will be maintained to identify which equipment is controlled under the HDP M&TE program.

Procedures have been established for control of M&TE, to ensure such devices fit the purpose and are of the proper type, range, accuracy and tolerance to accomplish the function of determining conformance to specified requirements. Selection of M&TE will take into account factors that may affect the known measurement uncertainty, including equipment accuracy, environmental effects, skills of personnel using the equipment, and condition of the item being verified.

M&TE will be calibrated and controlled in accordance with established procedures, to ensure the accuracy of measurements. Each device will be properly controlled, calibrated, and adjusted at specified intervals, to maintain its accuracy within the necessary limits. Personnel using M&TE are responsible for ensuring the equipment is appropriately calibrated prior to use.

Documented procedures for calibrating M&TE and measurement reference standards will be used, including published standard practices, written instructions that accompany purchased equipment, or other acceptable instructions. Handling and storage of M&TE will be controlled to ensure the accuracy of the equipment is maintained. A record system including a description of the device, the unique device identifier, calibration intervals, next due date, the calibration standard used, and results of the calibration will be maintained. Systems, procedures and practices have been established to safeguard M&TE from adjustments that would invalidate the calibration settings. Documentation is maintained to support an evaluation of the validity of previous measurements, when M&TE is found to be out of calibration.

13.5 CORRECTIVE ACTION

The Westinghouse corrective action program and the HDP PQP will be used to identify conditions adverse to the quality of materials and services, and to document, analyze and correct such conditions in accordance with established procedures.

Procedures will be established and implemented for identifying and correcting conditions such as: failures, malfunctions, adverse trends, deficiencies (including programmatic), deviations, defective material, design errors, equipment, and non-conformance to specified requirements including policies and procedures. Procedures will address identification of conditions adverse to quality as the result of an event, audit, surveillance, inspection or random observation, and will include the following requirements, as appropriate:

- Prompt remedial actions to correct each reported condition adverse to quality;
- Determination of the cause(s) that lead to the condition;
- Determination of interim and preventative actions to preclude recurrence of the condition;
- Identification of corrective action to address conditions adverse to quality;
- Assignment of responsibility for corrective action;
- Documentation of the cause(s) of the condition, and corrective action taken;
- Implementation, evaluation, and verification of corrective action to prevent recurrence; and
- Reporting to appropriate levels of management.

Any member of the project may initiate an issue report to document a condition adverse to quality that impacts, or has the potential to impact, conformance to applicable requirements, safety, efficiency, schedule or any other aspect of HDP performance or effectiveness. Upon notification or receipt of an issue report, the responsible HDP management shall take the necessary actions to bring the adverse condition under control and investigate the cause(s) of the condition. The controls used by HDP management may include: (1) suspension of all or part of the work, (2) management and supervisory review of work in progress, or (3) other actions deemed appropriate by the management of the organization.

Organizations performing HDP quality assurance functions will participate in evaluating and verifying corrective action implementation. Those organizations have authority to stop work, or ensure adequate controls are in place, until effective corrective action has been taken and any

applicable changes have been incorporated into procedures and communicated to appropriate personnel. The effectiveness of corrective action will be verified by surveillance, audit, trend analysis, or other reviews.

Quality data and audit/surveillance results will be analyzed for trends in materials, services, processes and systems that may require action to eliminate causes of potential conditions adverse to quality. The results of these analyses will be provided to management to determine any action required to prevent recurrence and control measures to ensure the action is effective. Trending and audit/surveillance results will be used to assure corrective measures are implemented effectively, and that actions to prevent recurrence are effective.

13.6 QUALITY ASSURANCE RECORDS

Quality records are completed documents that furnish evidence of the quality of materials, services and/or activities affecting quality and compliance with PQP. Quality records may also include articles such as materials, samples or test specimens when required. Examples of quality records applicable to decommissioning include:

- Quality assurance policies and procedures;
- Site work control instructions;
- Repair and maintenance records;
- Calibration documentation including radiological survey instruments;
- Departmental quality-related instructions, procedures, and drawings;
- Dosimetry records, radiological surveys, air sampling results;
- Final Status Survey and Hazmat site closure records;
- Design-related records;
- Procurement documents;
- Transportation package documentation;
- Training records; and
- Audit, assessment, inspection and test results.

Quality records will be controlled and retained in accordance with the PQP and established procedures. Non-safeguards procedures will identify the requirements and responsibilities for: records classification, legibility, correction, identification, collection, filing, indexing, storage, distribution, retention, retrieval and disposition. Record retention will be administered through the Westinghouse Electronic Document Management System (EDMS) which includes an index of record type, retention period, and storage location. Measures will be established to prevent access to records by unauthorized personnel and to guard against theft and vandalism.

Records requirements for suppliers of materials and services for the HDP will be specified in procurement documents, as required. Suppliers' records systems will be verified and monitored during surveillance and audits.

Records may be stored in electronic media, provided the process for managing and storing data is documented in procedures that comply with applicable regulations. Media used for the retention of records include (but are not limited to): microform, compact disk-recordable (CD-R), and magnetic media including videotape, computer tape, optical disks and hard disk storage.

Quality records will be protected against deterioration, damage, and/or loss in accordance with established procedures and regulatory requirements. Safety-related records requiring long-term storage will be maintained either at an approved single storage facility, or by storage of duplicate copies at separate geographical locations.

13.7 AUDITS AND SURVEILLANCES

The HDP QA organization is responsible for implementing and maintaining an internal audit program to examine and evaluate objective evidence for compliance with the PQP, and for evaluating the effectiveness of implementation. Internal audits and surveillances of activities affecting the quality of materials and services will be scheduled, planned and conducted in accordance with the PQP and established procedures. External audits of suppliers will be performed triennially.

Audit frequency will be based on the status and importance of an activity, results of external audits, and internal quality performance monitoring and indicators. The HDP Quality organization will maintain audit program schedules to ensure adequate oversight is maintained, and is responsible for the validity of external audits used as input to audit schedules. Supplemental audits will be performed when necessary, to verify specific activities, processes, and/or implementation of corrective actions.

Audits will be performed by personnel qualified in accordance with the PQP and independent of the activity being assessed, using written procedures, and/or checklists, as appropriate. Auditors will be trained on quality standards, regulatory requirements and internal practices, and qualified in accordance with applicable standards. Personnel who participate in audits, including Lead Auditors, Auditors, and technical specialists, shall have an education, experience and training commensurate with the scope, complexity, or special nature of the activities to be audited. Qualification records will be maintained by the HDP QA organization.

Reports documenting results will be prepared upon completion of an audit and distributed to appropriate management. HDP organizations are required to respond to audit reports within a specified time, identifying planned corrective actions and a schedule for completion thereof, when applicable. Deficiencies will be documented in audit reports and the corrective action program, for follow-up, trending, and tracking. Audit records include audit plans, checklists, audit reports, written replies, and documentation of completed corrective actions.

13.8 REFERENCES FOR CHAPTER 13.0

13-1 Westinghouse Electric Company Document QMS, "Westinghouse Electric Company Quality Management System."

13-2 International Organization for Standardization, ISO 9000:2005, "Quality Management Systems - Fundamentals And Vocabulary."

13-3 International Organization for Standardization, ISO 9001:2000, "Quality Management Systems - Requirements."

13-4 Code of Federal Regulations, Title 10, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

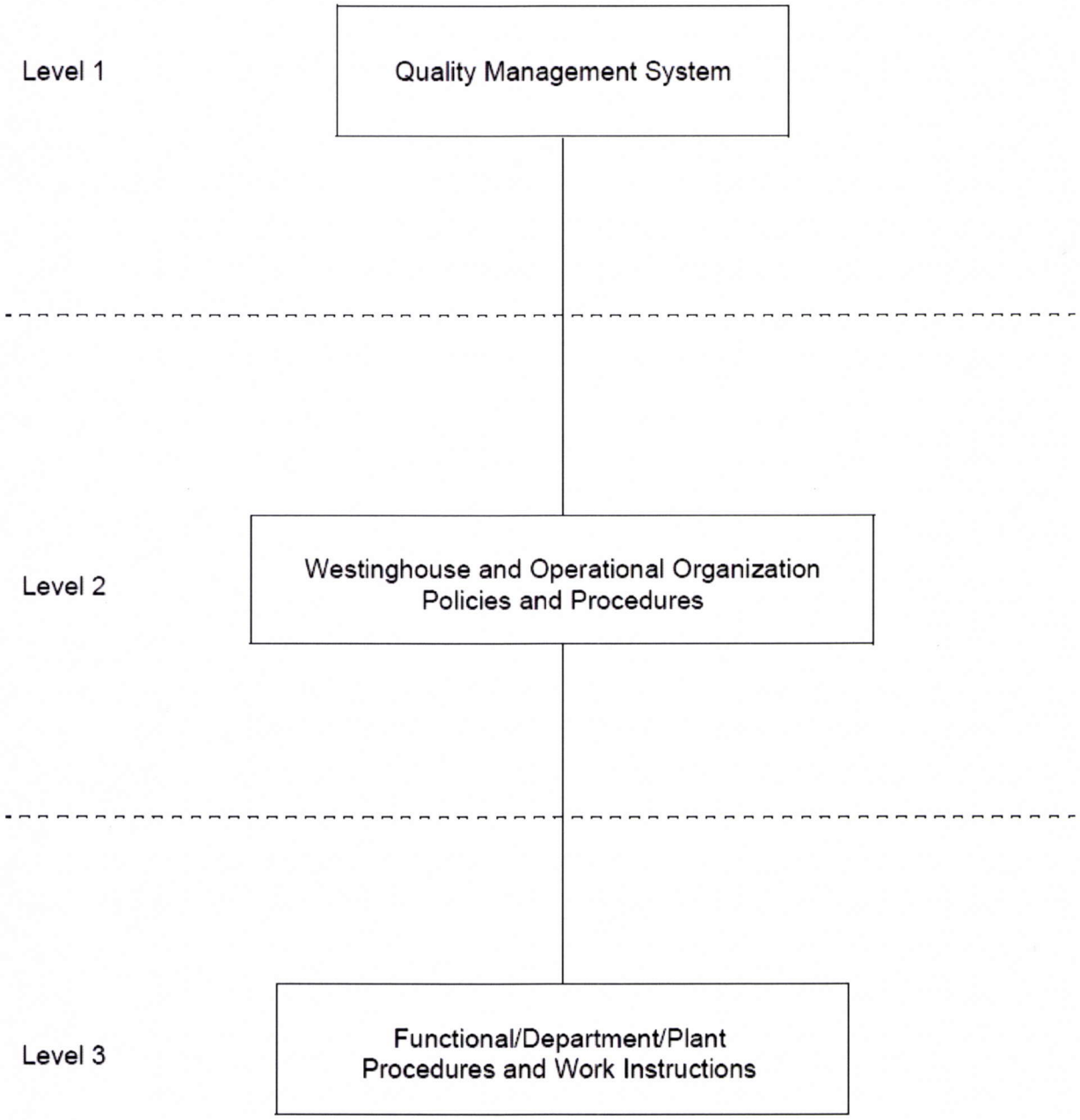
13-5 The American Society of Mechanical Engineers, ASME NQA-1-1994 Edition, "Quality Assurance Requirements for Nuclear Facility Applications."

13-6 Westinghouse Electric Company Document No. HDP-PO-QA-001, "Project Quality Plan."

13-7 Code of Federal Regulations, Title 10, Part 20, "Standards For Protection Against Radiation."

Figure 13-1

Westinghouse Quality Management System (QMS) Policy And Procedure Structure



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ACRONYMS AND ABBREVIATIONS

%	percent
+ C	dose contribution of entire decay chain (progeny) in secular equilibrium is accounted for by the parent
+ D	dose contribution of the short-lived progeny is accounted for by the parent
α	Type I error probability
β	Type II error probability
Δ	delta
σ	standard deviation (sigma)
$\mu Ci/cm^3$	microCuries per cubic centimeter
$\mu R/h$	microRoentgen per hour
AF	Area Factor
ALARA	As Low As Reasonably Achievable
Am	Americium
ASTM	American Society Of Testing Materials
bgs	below ground surface
BTV	Background Threshold Value
CFR	Code Of Federal Regulations
CoC	Chain of Custody
cm	centimeter
cm^2	square centimeter
cpm	counts per minute
cpm/100 cm^2	counter per minute per 100 square centimeters
CSM	Conceptual Site Model
d'	index of sensitivity
DCGL	Derived Concentration Guideline Level
$DCGL_W$	Derived Concentration Guideline Level For Statistical Testing
$DCGL_{EMC}$	Derived Concentration Guideline Level For Elevated Measurement Comparison
DP	Decommissioning Plan
dpm	disintegration per minute
dpm/100 cm^2	disintegration per minute per 100 square centimeters
DQA	Data Quality Assessment
DQO	Data Quality Objectives
EM	Electromagnetic
EMC	Elevated Measurement Comparison

ACRONYMS AND ABBREVIATIONS
(Continued)

EML	Environmental Measurements Laboratory
EPA	U.S. Environmental Protection Agency
Eu	Europium
FSS	Final Status Survey
ft	feet
g	gram
g/cm^3	grams per cubic centimeter
GPR	Ground Penetrating Radar
GPS	Global Positioning System
GWS	Gamma Walkover Survey
h	hour
H_a	Alternate Hypothesis
HEPA	High Efficiency Particulate Air
HEU	High Enriched Uranium
HDP	Hematite Decommissioning Project
H_0	Null Hypothesis
HRGS	High Resolution Gamma Spectroscopy
HPGe	High-Purity Germanium
HRCR	Hematite Radiological Characterization Report
HSA	Historical Site Assessment
HTDR	Hard-To-Detect Radionuclide
in	inch
ISOCS	In Situ Object Counting System
K_d	distribution coefficient
keV	kiloelectron volt
L	liter
LBGR	Lower Boundary Of The Gray Region
LEU	Low Enriched Uranium
m	meter
m^2	square meters
MARSSIM	Multi-Agency Radiation Survey And Site Investigation Manual
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
MDER	Minimum Detectable Exposure Rate

ACRONYMS AND ABBREVIATIONS
(Continued)

mL	milliliter
mrem	millirem
mSv	milliSievert
N	number of systematic measurement and sampling locations (Sign test)
N/2	number of systematic measurement and sampling locations (WRS test)
NAD83	North American Datum 83
NaI	Sodium Iodide
nC	number of composite samples
NCS	nuclear criticality safety
nEMC	number of systematic measurement and sampling locations (EMC test)
NIST	National Institute Of Standards And Technology
Np	Neptunium
NRC	U.S. Nuclear Regulatory Commission
Pa-234m	Protactinium-234 Metastable
Pb	Lead
PCE	Perchloroethylene
pCi/g	picoCurie per gram
Pu	Plutonium
QA	Quality Assurance
QC	Quality Control
R	Roentgen
Ra	Radium
RASS	Remedial Action Support Surveys
RI	Remedial Investigation
RIFS	Remedial Investigation Feasibility Study
ROC	Radionuclide Of Concern
SEA	Surrogate Evaluation Area
SOF	Sum Of Fractions
SSCs	structures, systems and components
TAP	Total Absorption Peak
Tc	Technetium
TEDE	Total Effective Dose Equivalent
Th	Thorium
U	Uranium

ACRONYMS AND ABBREVIATIONS
(Continued)

UF_6	Uranium Hexafluoride
V&V	Verification And Validation
WMW	Wilcoxon Mann Whitney
WRS	Wilcoxon Rank Sum

[1]

[1] MicroShield® is a registered trademark of Yale Security Inc., an ASSA ABLOY Group company.

14.0 FACILITY RADIATION SURVEYS

Following the decision to cease operations, a number of surveys are needed to determine the radiological status of the site, monitor the progress during remediation, and confirm that the site meets the radiological release criteria. This chapter provides detailed discussion on the various radiological surveys performed to support the Hematite Decommissioning Project (HDP) as well as the release criteria that will be used to terminate the site license.

Section 14.1 provides discussion on the site-specific radiological release criteria, referred to as derived concentration guideline levels (DCGLs) including the survey unit average concentrations ($DCGL_W$) for each radionuclide and medium of concern, the applicable values for small areas of elevated concentrations ($DCGL_{EMC}$), the area factors (AF) used to determine the $DCGL_{EMC}$, and the survey methods to be used when multiple radionuclides are present.

Section 14.2 provides a summary of site characterization surveys performed to determine the extent of residual radioactivity on or in structures, systems and components (SSCs) and environmental media. These types of surveys are performed to provide data for planning decommissioning actions, including remediation techniques, projected schedules, costs, waste volumes, and health and safety considerations during remediation.

Section 14.3 provides detailed discussion on Remedial Action Support Surveys (RASS). These measurements are conducted to provide near real-time guidance for remedial actions, and ensure the health and safety of workers and the general public. The precision, accuracy and data quality of these measurements are not in all cases, sufficient to define the final radiological status of the site.

Section 14.4 provides detailed discussion on the Final Status Survey (FSS) process including survey planning, design, implementation and data assessment. The FSS is performed to demonstrate that residual radiological conditions satisfy the predetermined criteria for unrestricted use. The process for obtaining the appropriate number and type of measurements is defined by the Data Quality Objectives (DQO), and serves to provide the basis to demonstrate that all radiological parameters (e.g., total surface radioactivity, radionuclide concentrations in soil or other media) meet the DCGL at a pre-determined level of confidence.

Section 14.5 provides discussion on post-remediation groundwater sampling and analysis.

Finally, Section 14.6 provides discussion on the reporting of FSS results. Survey Unit Release Records are prepared to provide a record of the composition and location of the survey unit; the measurements obtained during the FSS; the number and location of any small areas of elevated concentration; a summary of additional remedial actions necessary to meet the release criteria; and a summary of the data that represents the final radiological condition, including a determination that an individual survey unit meets the release criteria. A FSS Final Report will be prepared to compile the data obtained from the individual survey units, and to serve as the basis for demonstrating that the site meets the radiological criteria for unrestricted use.

14.1 RELEASE CRITERIA

In order to demonstrate that the HDP Site meets the U.S. Nuclear Regulatory Commission (NRC) criterion of 25 millirem (mrem) per year total effective dose equivalent (TEDE) for unrestricted release specified in Code of Federal Regulations (CFR), Title10, Part 20.1402, "Radiological Criteria for Unrestricted Use" (Reference 14-1), DCGLs were defined based on the outcome of exposure pathway modeling. The detailed description of the method used to develop the DCGLs for various media are provided in Chapter 5.0. The additional requirement of 10 CFR 20.1402 that all residual radioactivity at the site be reduced to levels that are as low as reasonably achievable (ALARA) has been satisfied as discussed in Chapter 7.0.

14.1.1 RADIONUCLIDES OF CONCERN

The Historical Site Assessment (HSA, Reference 14-2) and the Hematite Radiological Characterization Report (HRCR, Reference 14-3) identify the radionuclides of concern (ROC) present at the site. In summary, the primary ROCs are Uranium-234 (U-234), Uranium-235 (U-235 + D), Uranium-238 (U-238 + D), and Technetium-99 (Tc-99). The transuranic radionuclides, including Americium-241 (Am-241), Neptunium-237 (Np-237 + D), and Plutonium-239/240 are present in only trace quantities that were introduced by the use of reprocessed Uranium in the gaseous diffusion process.

Thorium-232 is present naturally in background soil, and has been identified at concentration greater than the Background Threshold Value for Th-232 at a limited number of locations within the area of the buried waste. Radium-226 (Ra-226 + C) was identified as a ROC and has been identified primarily at two locations in the Burial Pit Area. Radium-226 was also identified as a ROC in one area containing two burial pits. The elevated Ra-226 was likely introduced into the burial pits with waste as a result of the installation of contaminated equipment into the process operations. Although only low concentrations of Th-232 and Ra-226 have been identified at locations outside of the Burial Pit Area, these radionuclides will be considered ROCs site-wide.

Bismuth-214 was identified in low concentrations in two scale samples from drains in Building 230 indicating the potential presence of Ra-226. However, the concentrations were less than one percent of the Uranium concentrations and the operations conducted in Building 230 did not involve Ra-226. Therefore, Ra-226 was not included as a ROC in buildings. The nomenclature "+ D" indicates that the dose contribution of the short-lived progeny is accounted for by the parent, and "+ C" indicates that the dose contribution of the entire decay chain (progeny) in secular equilibrium is accounted for by the parent.

14.1.2 SITE-SPECIFIC DCGL SUMMARY

Each radionuclide-specific DCGL is equivalent to the level of residual radioactivity in a particular medium (above the background for that medium) that could, when considered independently, result in a TEDE of 25 mrem per year to an average member of the critical group.

These values were subsequently adjusted to account for the dose contribution from all pathways. Additionally, since multiple ROCs are known to be present, the dose contribution from each ROC is accounted for using the sum of fractions (SOF) to ensure that the total dose from all ROCs does not exceed the dose criterion.

14.1.2.1 Building And Structural Surfaces DCGLs

The site-specific building and structural surface DCGLs were derived using the RESRAD-BUILD computer code, Version 3.4, by using the building occupancy scenario for two conceptual site models (CSM) having differing room sizes (Small Office and Large Warehouse CSM). Additional details regarding the dose modeling are discussed in Chapter 5.0.

Table 14-1 presents the site-specific DCGLs for building and structural surfaces which are based on the building occupancy scenario for Small Office and Large Warehouse CSM. The Small Office CSM resulted in the most limiting DCGLs. Considering the very low levels of residual surface contamination present in the buildings to remain at the time of license termination, and the limited effort that should be required to reduce surface contamination to acceptable levels, the DCGLs based on the Small Office CSM will be used for all building surfaces regardless of room size. As discussed in Chapter 7.0, an evaluation was performed and it was determined that the DCGLs for residual surface contamination are ALARA.

14.1.2.2 Soil DCGLs

The site-specific soil DCGLs were derived using the RESRAD computer code, Version 6.4, by modeling the Residential (Resident) Farmer as the critical receptor for the site. The Resident Farmer will be exposed to any residual radioactive contamination left on site through the various dose pathways. The exposure as a function of depth was evaluated within four strata (i.e., Surface, Root, Deep, and Uniform) to account for the source geometry, and differences in the exposure pathways based on depth. These variations on the model were developed to provide flexibility when comparing final conditions to the dose criterion, and in consideration of the requirement to assess the potential dose associated with soil volumes identified for re-use as backfill. DCGLs were also calculated for an Excavation Scenario to evaluate the effects of changing the *in-situ* soil configuration after license termination. These site-specific soil DCGL models are discussed in detail in Chapter 5.0.

Table 14-2 presents the site-specific DCGLs as developed for soil. As presented in Chapter 7.0, an evaluation was performed and it was determined that the DCGLs for soil are ALARA.

14.1.2.3 Buried Pipe DCGLs

In addition to criteria developed for building and structural surfaces and soil, site-specific DCGLs were developed based on a reasonable exposure scenario for buried piping. The gross activity DCGLs for a range of pipe diameters are provided in Table 14-3. The buried pipe

DCGLs are a function of the pipe diameter as the internal surface area increases as a square of the diameter while the interior volume increases as a cube of the diameter. Therefore, the DCGL increases as the pipe diameter increases. Additional details regarding the development of this DCGL can be found in Chapter 5.0.

14.1.3 SOIL DCGL ADJUSTMENT

To derive the soil DCGLs that can be compared directly to the dose criterion, the dose contributions from insignificant ROCs were determined and then subtracted from the TEDE limit of 25 mrem per year. The following sections discuss how the soil DCGLs presented in Table 14-2 were adjusted.

14.1.3.1 Insignificant Radionuclides Of Concern

The characterization data was reviewed and evaluated as documented in Derivation of Surrogates and Scaling Factors for Hard-To-Detect Radionuclides (Reference 14-4) to determine if any of the ROCs were considered insignificant dose contributors. Insignificant dose contributors were determined consistent with the guidance contained in Section 3.3 of NUREG-1757, Consolidated NMSS Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria, Volume 2 (Reference 14-5). The conditions were applied that limit the aggregate dose contribution from radionuclides considered to be insignificant to 10 percent of the TEDE criterion (or 2.5 mrem per year); and the aggregate dose must be included in the accounting when demonstrating compliance with the TEDE criterion. The contribution of insignificant radionuclides was calculated to be 1.7 mrem per year (or 6.8 percent of the TEDE criterion) for Np-237, Pu-239/240, and Am-241 for all soil depths. Details of the calculations are taken from Section 2.2 of Reference 14-4.

14.1.3.2 Calculation Of Adjusted DCGLs

The site-specific soil DCGLs, as shown in Table 14-2, were adjusted (reduced) by a factor of 0.99, illustrated in Equation 14-1 below, to account for the dose contributions from insignificant ROCs.

(14-1)

$$\mathrm{DCGL}_{\mathrm{W-Adjusted}} = \mathrm{DCGL}_{\mathrm{W}} \times \left(\frac{25\,\mathrm{mrem/yr} - 0.266\,\mathrm{mrem/yr}}{25\,\mathrm{mrem/yr}} \right) = \mathrm{DCGL}_{\mathrm{W}} \times 0.99$$

The adjusted site-specific soil DCGLs are presented in Table 14-4.

14.1.4 DCGL MODIFICATION

The guidance provided in NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM, Reference 14-6), Section 4.3.2, was used for DCGL modification. As a general rule, scaling factors are applied where fairly constant radionuclide concentration ratios can be demonstrated to exist. These factors were derived using characterization data collected prior to the FSS. Each scaling factor was evaluated to ensure an appropriate value was selected. Consistent with the derivation of Equation 4-1 of MARSSIM, the scaling factor was defined as the ratio of the inferred to surrogate contaminant concentration.

14.1.4.1 Uranium Radioactivity Fractions And Isotopic Ratios By U-235 Enrichment

An important component in the DCGL modification process is the understanding of the radioactivity fraction and isotopic ratio relationships between the Uranium ROCs as a function of U-235 enrichment. Appendix C, Table C-1 of Reference 14-4 provides the relationships to be used at the site. This table is included as Table 14-5 rather than referenced since the data are used extensively in the subsequent sections.

14.1.4.2 Buildings And Structural Surfaces

Because the isotopes have differing DCGLs and field instrumentation cannot make the isotopic distinction that would be required without assuming some sort of relative contribution to the observed response, a gross radioactivity $DCGL_W$ was calculated for field implementation using Equation 4-4 of MARSSIM. However, before the gross radioactivity $DCGL_W$ was calculated, the fractional radioactivity contribution of each ROC was determined from characterization data and the results presented in Table 14-6. Chapter 4.0 of the Hematite Decommissioning Plan (DP) provides the details of the fractional radioactivity contribution calculations.

Using the radioactivity fractions in Table 14-6 and the gross radioactivity $DCGL_W$ calculations for the Small Office CSM presented in Table 14-7, a gross radioactivity $DCGL_W$ was calculated using MARSSIM Equation 4-4 and is illustrated below.

$$\frac{1}{\frac{8.27E\text{-}1}{20{,}000}+\frac{3.72E\text{-}2}{19{,}000}+\frac{1.27E\text{-}1}{21{,}000}+\frac{2.83E\text{-}3}{1.3E7}+\frac{3.21E\text{-}3}{1{,}200}+\frac{5.57E\text{-}5}{2{,}700}+\frac{2.03E\text{-}6}{3{,}500}+\frac{2.68E\text{-}3}{3{,}400}} = 18{,}925\ \text{dpm}/100\ \text{cm}^2 \qquad \textbf{(14-2)}$$

14.1.4.3 Soil

14.1.4.3.1 Surrogate Radionuclides

For sites with multiple radionuclides, it may be possible to measure one of the radionuclides and infer the amount of other radionuclide(s) when demonstrating compliance with the release criteria through the application of a surrogate relationship. Since the site has multiple ROCs, a surrogate study (Reference 14-4) was performed to determine scaling factors that could be used to demonstrate compliance by inferring the concentration of one or more radionuclides by the measurement of a surrogate radionuclide.

Surrogate relationships have been developed for Tc-99 and U-234 and are presented in Sections 14.1.4.3.2 and 14.1.4.3.3, respectively. However, the Tc-99 surrogate relationship is prohibited from use in the evaluation of analytical results to determine compliance with the final status survey dose criteria. Instead of a surrogate relationship, laboratory analysis for Tc-99 will be performed for all FSS samples.

14.1.4.3.2 Inferring Tc-99

Reference 14-4 documented consistent distribution ratios in soil for the hard-to-detect radionuclide (HTDR) Tc-99. This ROC is considered a HTDR in soil because it does not emit gamma radiation that would be detectable during field scanning of soil using conventional instrumentation. Note that a surrogate is not required when measuring surface contamination on building and structural surfaces using conventional instrumentation. Table 14-8 provides the distribution ratios for the use of U-235 as a surrogate to infer the Tc-99 concentration in soil within three Surrogate Evaluation Areas (SEA). The SEA that showed similar relationships based on the data obtained within each include the Plant Soil SEA, Burial Pit SEA, and Tc-99 SEA and are illustrated in Figure 14-1.

In order for the measurement of U-235 to account for the dose contribution from Tc-99, the U-235 adjusted $DCGL_W$ from Table 14-4 that was adjusted for the contributions from insignificant radionuclides was further modified. This calculation was performed using Equation 4-1 of MARSSIM and the results are provided in Table 14-9. The result for the Surface Soil stratum in the Plant Soil SEA using the distribution ratio of 9.24 (from Table 14-8) is illustrated below.

(14-3)

$$DCGL_{U-235,mod} = \frac{1}{\left(\frac{1}{108.9}\right) + \left(\frac{9.24}{160.4}\right)} = 14.96 \text{ pCi/g}$$

14.1.4.3.3 Inferring U-234

Of the Uranium ROCs shown in Table 14-4, U-234 cannot be detected using conventional field instrumentation during scan survey measurements of soil, or by gamma spectroscopy. The ratio of the U-238 to U-235 concentrations obtained from gamma spectroscopy were used to infer the U-234 to U-235 ratio based on observations of the enrichment in a large number of characterization samples, assumptions regarding the consistency of the enrichment shown by the characterization data, and published values for the enrichment based on isotopic ratios. These relationships are provided in Table 14-5. Figure 14-2 provides a plot of the Uranium radioactivity fractions from Table 14-5. Figure 14-3 provides a plot of the Uranium ratios from Table 14-5.

The following data quality objectives (DQOs) and equations were used to estimate the concentration of U-234 based on the results of analysis by gamma spectroscopy for U-235 and U-238. Alternatively, alpha spectroscopy may be used to quantify the U-234 concentrations.

When U-235 is reported as negative or zero and U-238 is reported as positive, natural Uranium is assumed and the U-234 concentration will be set equal to the U-238 concentration.

(14-4)

$$C_{U\text{-}234}\ (\text{pCi/g}) = C_{U\text{-}238}$$

where:

$C_{U\text{-}238}$ = Concentration of U-238 (pCi/g)

When U-235 is reported as positive and U-238 is reported as negative or zero, highly enriched Uranium is assumed and the U-234 concentration is determined by multiplying the U-235 concentration by 32.50, which is the U-234:U-235 ratio based on the maximum enrichment (100 percent) from Table 14-5.

(14-5)

$$C_{U\text{-}234}\ (\text{pCi/g}) = 32.50 \times C_{U\text{-}235}$$

where:

$C_{U\text{-}235}$ = Concentration of U-235 (pCi/g)

When both U-235 and U-238 data are reported as positive, but the U-238:U-235 ratio for the data is less than 0.0001 (indicating highly enriched Uranium), the U-234 concentration is determined using Equation 14-5.

When both U-235 and U-238 data are reported as positive, but the U-238:U-235 ratio for the data is greater than 155.37 (indicating depleted Uranium), the U-234 concentration is determined by multiplying the U-235 concentration by the minimum U-234:U-235 ratio of 46.31 from Table 14-5.

(14-6)

$$C_{U\text{-}234}\ (\text{pCi/g}) = 46.31 \times C_{U\text{-}235}$$

where:

$C_{U\text{-}235}$ = Concentration of U-235 (pCi/g)

When both U-235 and U-238 data are reported as positive, the U-238:U-235 ratio for the data is used to determine the associated U-234:U-235 ratio from Table 14-5. The U-234 concentration is determined by multiplying the U-235 concentration by the U-234:U-235 ratio.

(14-7)

$$C_{U\text{-}234}\ (\text{pCi/g}) = R_{U\text{-}234:U\text{-}235} \times C_{U\text{-}235}$$

where:

$R_{U\text{-}234:U\text{-}235}$= Estimated U-234:U-235 ratio based on U-235:U-238 ratio using Table 14-5; and,

$C_{U\text{-}235}$ = Concentration of U-235 (pCi/g).

14.1.4.3.4 Sensitivity Analysis For Total Uranium

The calculation of a total Uranium gross radioactivity $DCGL_W$ is required in order to evaluate the sensitivity of gamma surface scans which measure gross gamma radiation since radionuclide-specific measurements typically are not performed with conventional scanning instrumentation. For the sensitivity analysis, it was conservatively assumed that Tc-99 would be inferred from the measurement of U-235. Equation 4-4 of MARSSIM is used to calculate the total Uranium gross radioactivity $DCGL_W$.

(14-8)

$$DCGL_{w,TotU}(pCi/g) = \frac{1}{\frac{f_{U\text{-}234}}{DCGL_{w,U\text{-}234}} + \frac{f_{U\text{-}235}}{DCGL_{w,U\text{-}235}} + \frac{f_{U\text{-}238}}{DCGL_{w,U\text{-}238}}}$$

where:

$f_{U\text{-}234}$ = U-234 radioactivity fraction;

$f_{U\text{-}235}$ = U-235 radioactivity fraction;

$f_{U\text{-}238}$ = U-238 radioactivity fraction;

$DCGL_{W,\ U\text{-}234}$ = U-234 $DCGL_w$ from Table 14-4 for all Strata (pCi/g);

$DCGL_{W,\ U\text{-}235}$ = U-235 $DCGL_w$ from Table 14-9 for all strata (pCi/g);

$DCGL_{W,\ U\text{-}238}$ = U-238 $DCGL_W$ from Table 14-4 for all Strata (pCi/g).

The sensitivity analysis was performed for the Plant Soil SEA, Tc-99 SEA, and Burial Pit SEA and the results illustrated in Figure 14-4, Figure 14-5 and Figure 14-6, respectively. Based upon a review of the soil characterization data, it has been determined that the average enrichment of impacted soil is 4.0 weight percent U-235/U. Subsequently, the fractions from Table 14-5 corresponding to this enrichment were used to calculate DCGLs for total Uranium in impacted soil.

14.1.4.3.5 Summary

The application of the modified U-235 values (and associated total uranium values) from Table 4-19 is restricted to survey design (evaluation of scan sensitivity) and excavation control (remedial action support surveys). Laboratory analysis for Tc-99 will be performed on all final status survey samples and as such, the adjusted U-235 DCGL values shown in Tables14-4, 14-9, and 14-10 will not be used to demonstrate compliance with the final status survey dose criteria.

Table 14-10 presents a summary of the adjusted and modified soil $DCGL_W$ values in a matrix format by SEA, survey type, and strata. The total Uranium $DCGL_W$ values were calculated using Equation 4-4 of MARSSIM and the following inputs:

- Adjusted U-234 and U-238 $DCGL_W$ values from Table 14-4;
- Modified U-235 $DCGL_W$ values from Table 14-9; and

- Radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4 percent.

Because Table 14-10 lists more than one soil $DCGL_W$ value for a given SEA, survey type, and CSM strata, the unity rule must be applied per guidance in MARSSIM.

14.1.5 UNITY RULE

The unity rule will be applied to the data used for the survey planning, and data evaluation and statistical tests for soil sample analyses since multiple radionuclide-specific measurements may be performed or the concentrations inferred based on known relationships. The application of the unity rule serves to normalize the data to allow for an accurate comparison of the various data measurements to the release criteria. When the unity rule is applied, the $DCGL_W$ for planning and evaluation purposes becomes one (1). The use and application of the unity rule will be performed in accordance with Section 4.3.3 of MARSSIM.

14.1.5.1 Sum-Of-Fractions And Weighted Sigma Calculations

Table 14-10 is arranged to include all applicable combinations of SEA, survey type, and CSM strata. The number of measured ROCs varies based on the survey type. Note that when the U-234 concentration is inferred using the U-238 to U-235 ratio rather than by alpha spectroscopy measurement, the inferred value will be used in the unity rule calculations as if it had been measured.

In addition to calculating the SOF, a weighted sigma value must be calculated for planning purposes. When using the Wilcoxon Rank Sum (WRS) test, for each contaminant present in background, the greater of the survey unit and reference area sigma is used in the calculation.

The methodologies that will be used for each survey type are provided below and the applicable $DCGL_W$ values are provided in Table 14-10 by selecting the appropriate SEA, survey type, and CSM strata.

14.1.5.1.1 Sample – Measure Tc-99

When measuring Tc-99, the SOF will be calculated based on the ratio of the radioactivity concentrations (in pCi/g) of U-234, U-235, U-238, Tc-99, Ra-226 and Th-232 (Ra-226 and Th-232 will be corrected for background when calculating dose), and their respective soil $DCGL_W$ values using the following equation, based on Equation 4-3 of MARSSIM.

(14-11)

$$SOF = \frac{Conc_{U\text{-}234}}{DCGL_{w,U\text{-}234}} + \frac{Conc_{U\text{-}235}}{DCGL_{w,U\text{-}235}} + \frac{Conc_{U\text{-}238}}{DCGL_{w,U\text{-}238}} + \frac{Conc_{Tc-99}}{DCGL_{w,Tc-99}} + \frac{Conc_{Th-232}}{DCGL_{w,Th-232}} + \frac{Conc_{Ra-226}}{DCGL_{w,Ra-226}}$$

The weighted sigma value is calculated using the following equation, based on Equation I-17 of MARSSIM.

(14-12)

$$\sigma_{SOF} = \sqrt{\left(\frac{\sigma_{U\text{-}234}}{DCGL_{w,U\text{-}234}}\right)^2 + \left(\frac{\sigma_{U\text{-}235}}{DCGL_{w,U\text{-}235}}\right)^2 + \left(\frac{\sigma_{U\text{-}238}}{DCGL_{w,U\text{-}238}}\right)^2 + \left(\frac{\sigma_{Tc-99}}{DCGL_{w,Tc-99}}\right)^2 + \left(\frac{\sigma_{Th-232}}{DCGL_{w,Th-232}}\right)^2 + \left(\frac{\sigma_{Ra-226}}{DCGL_{w,Ra-226}}\right)^2}$$

Equations (14-13) and (14-14) were deleted in Revision 1.0.

14.1.5.2 Unity Rule Application To Multiple Conceptual Site Models

In the situation where the residual contamination is in a vertical configuration of multiple strata, an extension of the unity rule will be applied to ensure that the TEDE of the survey unit as a whole does not exceed the criterion of 25 mrem per year. This will be accomplished by first evaluating the analytical data for each individual stratum separately, then summing the fraction of the criterion for each stratum. If the SOF is less than or equal to one (1), the survey unit will be considered to meet the criterion. The use of the unity rule in this application is not discussed in the guidance documentation; however, this approach is consistent with the guidance provided in NUREG-1757, Volume 2 and MARSSIM to ensure that the release criterion is met. For a given survey unit utilizing more than one strata, this may be expressed by the following equation.

(14-15)

$$SOF_{Total} = SOF_{CSM\text{-}A} + SOF_{CSM\text{-}B} + \ldots + SOF_{CSM\text{-}n}$$

As an example, assume that both the Surface and Root Strata apply to the configuration of residual contamination in a survey unit. Sampling will be performed for each stratum separately; note that the unity rule will first be applied during the data quality assessment of the survey results to ensure that the criterion of 25 mrem per year for each stratum is met. In this example,

the mean SOFs for the Surface and Root Strata were calculated to be 0.2 and 0.9, respectively. Because the SOF for each stratum was less than 1, the criterion of 25 mrem per year was met for each stratum individually. However, the unity rule needs to be applied a second time. The result in this example of the application of the unity rule for the multiple strata is simply the addition of the individual SOFs using Equation 14-15 which equals 1. Since a SOF of 1 is equivalent to a TEDE of 25 mrem per year, the survey unit in this example would exceed the criterion and would require further remediation. Conversely, if the SOF had been less than or equal to 1, the survey unit would have been considered to meet the criterion.

14.1.6 AREA FACTORS

Section 2.5.1.1 and Section 5.5.2.4 of MARSSIM address the concern of small areas of elevated radioactivity in the survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an AF to account for the small area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. Note that the consideration of small areas of elevated radioactivity applies only to Class 1 survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_W$.

The AFs for building and structural surfaces were developed by using the CSMs and adjusting the size of the contaminated area. Details of the AF development are included in Chapter 5.0. Area factors were determined for surface areas ranging from 1 square meter (m^2) to the maximum size of the floor, 6.5 m^2 for the Small Office CSM. The AFs are provided in Table 14-11 for the building occupancy scenario, which is the most limiting CSM. Note that these AFs will be conservatively applied to any building surface.

The AFs for soil were developed by using the CSMs and adjusting the size of the contaminated zone. The AFs for Surface Soil strata are provided in Table 14-12. Table 14-12 also provides AFs for total Uranium for surface soil in the Plant Soil SEA, Tc-99 SEA, and Burial Pit SEA that correspond to the calculated U-235 enrichment.

The $DCGL_{EMC}$ is also referred to as the required scan MDC, as shown in Equation 5-3 of MARSSIM. The following equation defines the calculation of a $DCGL_{EMC}$:

$$DCGL_{EMC} = AF \times DCGL_W \quad \textbf{(14-16)}$$

The following equation was used to define the calculation of the $DCGL_{EMC,\ TotU}$ using Equation 14-8 and Equation 14-16:

(14-17)

$$DCGL_{EMC,TotU} = \frac{1}{\left(\frac{f_{U\text{-}234}}{AF_{U\text{-}234} \times DCGL_{w,U\text{-}234}} + \frac{f_{U\text{-}235}}{AF_{U\text{-}235} \times DCGL_{w,U\text{-}235}} + \frac{f_{U\text{-}238}}{AF_{U\text{-}238} \times DCGL_{w,U\text{-}238}}\right)}$$

where:

$f_{U\text{-}234}$ = U-234 radioactivity fraction;

$f_{U\text{-}235}$ = U-235 radioactivity fraction;

$f_{U\text{-}238}$ = U-238 radioactivity fraction;

$AF_{U\text{-}234}$ = AF for U-234 from Table 14-12;

$AF_{U\text{-}235}$ = AF for U-235 from Table 14-12;

$AF_{U\text{-}238}$ = AF for U-238 from Table 14-12;

$DCGL_{W,\ U\text{-}234}$ = U-234 $DCGL_w$ from Table 14-4 for all strata (pCi/g);

$DCGL_{W,\ U\text{-}235}$ = U-235 $DCGL_w$ from either Table 14-9 for all strata (pCi/g); and,

$DCGL_{W,\ U\text{-}238}$ = U-238 $DCGL_W$ from Table 14-4 for all strata (pCi/g).

Equation 14-16 was used to define the calculation of the AF_{TotU} shown below.

(14-18)

$$AF_{TotU} = \frac{DCGL_{EMC,TotU}}{DCGL_{w,TotU}}$$

Equation 14-17 and Equation 14-18 were combined and reduced. The following equation was used to calculate the total Uranium AFs presented in Table 14-12.

(14-19)

$$AF_{TotU} = \frac{1}{DCGL_{w,TotU} \times \left(\frac{f_{U\text{-}234}}{AF_{U\text{-}234} \times DCGL_{w,U\text{-}234}} + \frac{f_{U\text{-}235}}{AF_{U\text{-}235} \times DCGL_{w,U\text{-}235}} + \frac{f_{U\text{-}238}}{AF_{U\text{-}238} \times DCGL_{w,U\text{-}238}}\right)}$$

14.2 CHARACTERIZATION SURVEYS

Chapter 4.0 provides a description of the radiological status of the site including summary tables and figures that describe the characterization results. The detailed characterization data is provided in the HRCR. The following sections provide assessments of the characterization data to demonstrate the acceptability of the data for use in decommissioning planning, initial area classification, remediation planning, and final status survey planning.

14.2.1 SURVEY OF IMPACTED MEDIA

The characterization of the site included numerous campaigns as described in the HRCR which included in excess of 2,200 monitoring well water samples, surface water samples, sediment, surface and sub-surface soil samples, as well as samples from drains and measurements of building surfaces. Samples were collected from all site areas and used to refine the delineation between impacted and non-impacted areas provided in the HSA. Additional discussion regarding the impacted and non-impacted areas is provided in Section 14.2.5.

14.2.2 FIELD INSTRUMENT METHODS AND SENSITIVITIES

The descriptions of the scanning and static measurements of building surfaces and gamma radiation scan surveys for soil areas discussed in the HRCR (and documents referenced in the HRCR) were reviewed.

Scanning and static measurements of building surfaces were performed primarily using Ludlum Model 2350-1 data loggers coupled to gas-flow proportional detectors. The data logger/gas-flow proportional detectors had scan MDCs averaging 217 disintegrations per minute per 100 square centimeters (dpm/100 cm^2) for alpha measurements, and 1,200 dpm/100 cm^2 for alpha+beta measurements; the static MDC averaged 105 dpm/100 cm^2 for alpha measurements and 550 dpm/100 cm^2 for alpha+beta measurements.

Scanning of open land areas during the characterization campaigns were performed primarily using 2 inch (in) by 2 in sodium iodide (NaI) detectors with rate meters. A portion of these surveys were performed while collecting coordinate location data using a global positioning system (GPS) unit. All of the scan surveys were consistent with the gamma radiation scanning survey approach discussed in MARSSIM. The sensitivities listed in Table 6.4 of NUREG-1507, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions (Reference 14-7), are therefore reasonable estimations of the actual scan MDCs.

14.2.3 LABORATORY INSTRUMENT METHODS AND SENSITIVITIES

The HRCR, and documents referenced in the HRCR, were reviewed and Table 14-13 provides a list of typical laboratory analysis methods and the associated MDCs. The tables of individual

sample results in the HRCR also provide the MDC for each radionuclide in each sample. Methods used were standard industry methods from the U.S. Environmental Protection Agency (EPA) and the Environmental Measurements Laboratory (EML). Laboratories chosen for analyses were authorized in accordance with quality assurance.

14.2.4 SUMMARY OF SURVEY RESULTS

The characterization data presented in the HRCR was summarized and included in Chapter 4.0 as part of the discussion of the radiological status of the site. In this Chapter, the associated SOF (adjusted) for each soil sample obtained during characterization was calculated using Equation 14-11, and are illustrated on Figures 14-7 through 14-10. Specifically, Figures 14-7, 14-8 and 14-9 illustrate the location and magnitude of the SOF values for samples obtained from the independent strata corresponding to the Surface, Root and Deep CSMs, respectively. Figure 14-10 illustrates the resulting SOF (total) based on a summation of the contribution from each strata at each sampling location in order to account for the contribution to dose in the vertical column of soil across the three CSMs. Additionally, shaded contours based on the magnitude of the SOF have been included on the figures to readily identify areas that require remedial action.

14.2.5 IMPACTED AND NON-IMPACTED AREAS

Activities with special nuclear materials (SNM) were conducted within an approximately 10-acre Central Tract area of the site. The Central Tract area is bounded by State Road P to the north, the land adjacent to east bank of the Northeast Site Creek, the Union-Pacific Railroad to the south and the Site Pond to the west. Approximately 3.8 acres along the Site Creek downstream to Joachim creek and along Joachim Creek to the location of sample SW-14-SS are considered potentially impacted based on site characterization data; and 7.1 acres to be used as a soil staging area near the Northeast Site Creek are expected to become impacted as result of the decommissioning activities. Additionally, a 20 foot wide area immediately south of the railroad in the central tract, an area west of the Site Pond, and an area between the Northeast site creek and the Lay-down area are also considered as impacted (total of about 10.1 acres). The remaining portions of the 228-acre Hematite Site are considered to be non-impacted as illustrated on Figure 14-11.

14.2.6 JUSTIFICATION FOR NON-IMPACTED AREAS

MARSSIM defines non-impacted areas as those areas where there is no reasonable possibility of residual contamination. Based on the findings presented in the HSA, there is an absence of licensed activities on site land areas outside of the impacted areas defined above and, therefore, the following areas may be considered non-impacted: land on the north side of State Road P, land east of the Lay-down area and a line running southwest from the Lay-down area to the Northeast Site Creek, land 20 feet south of the railroad tracks (not including the site stream and the portion of Joachim Creek discussed above).

Sufficient survey coverage and an adequate number of samples were obtained in the areas subsequently designated as non-impacted to serve as the basis for this classification. The survey measurements and laboratory data from the samples obtained from areas designated as non-impacted did not show detectable Tc-99 activity or concentrations of licensed radioactivity as statistically distinguishable from background. The "statistically distinguishable from background" determination used ProUCL v4.00.005 for Th 232 and total Uranium consistent with the statistical process described in Appendix A of the HRCR, which included two-sample hypothesis testing performed using the Quantile and Mann-Whitney U tests (referred to as Wilcoxon-Mann-Whitney in ProUCL) in parallel.

Both tests concluded that Th-232 data from non-impacted areas were indistinguishable from the background data. For total Uranium, the Mann-Whitney U test concluded that the data from non-impacted were indistinguishable from the background data, but the Quantile test concluded that the data from non-impacted areas were distinguishable from the background data. A review of the summary statistics and a graphical review of the data were performed and showed that the non-impacted data and background data had very similar means and medians (as demonstrated by the Mann-Whitney U test), but the non-impacted data had greater variability. The non-impacted data, while having greater variability, was not skewed compared to the background data. The Quantile test specifically looks at the upper tails of the two distributions and does not consider the lower tails and therefore it is expected that the Quantile test would fail in this situation. It is also an acceptable result that the non-impacted data had greater variability since the data was collected over a larger geographically-sized area than the background data.

Lastly, analysis of the uranium data from the non-impacted area where uranium was detected outside the error band of the MDC shows that only one sample, NB-71-01-SL, exceeded the background threshold value (BTV) of 2.4 pCi/g established in DP Section 4.3.5. Sample NB-71-01-SL had a result of 2.6 pCi/g and was taken within the top 1 foot of soil. This single data point at the surface that exceeds the BTV is reasonable considering that the BTV is selected such that some non-impacted total uranium results would exceed it.

14.2.7 ADEQUACY OF THE CHARACTERIZATION

The site characterization included the information that should be collected per the guidance in NUREG-1757, Volume 1, Consolidated NMSS Decommissioning Guidance, Decommissioning Process for Materials Licensees, Appendix D, XIV.b (Reference 14-8) and is discussed in detail in Section 5.0 of the HRCR. Extensive characterization and monitoring have been performed at the site. Samples taken in each area, along with the historical information, provide a clear picture of the residual radioactive materials and its vertical and lateral extent at the site. Using appropriate DQOs, monitoring well water samples, surface water, surface soil, sediment, and sub-surface soil have been collected to provide the profile of the residual radioactivity at the site. Samples have been analyzed for the ROC with detection limits that provide the level of detail

necessary for decommissioning planning. Buildings have also received characterization sufficient to understand the nature and extent of contamination.

14.2.8 INACCESSIBLE OR NOT READILY ACCESSIBLE AREAS

Areas at the site that are not readily accessible included the drain piping within the buildings that will remain after site closure. Floor drains were evaluated by direct survey of the drain surface and sampling and analysis of residue within the drain traps. The storm drain system and the Sanitary Wastewater Treatment Plant have not been extensively characterized directly by radiological surveys and sampling, however since process knowledge and laboratory analytical data of the liquids within these systems show that radioactivity is likely to be present, these systems are determined to be Class 1 (per the HRCR). Additional characterization of these systems will be performed at the time of decontamination and/or removal to ensure nuclear criticality safety (NCS), to demonstrate that the components meet the release criteria, or to confirm the appropriate method of disposal.

Buried piping and equipment that will remain in place after site closure that had a potential for radioactive contamination above $DCGL_W$ (based on site operating history) or known contamination above $DCGL_W$ (based on previous radiation surveys or surveys performed during decommissioning) will be designated as Class 1 for the purpose of Final Status Survey.

14.2.9 RATIOS OF RADIONUCLIDES

14.2.9.1 Building And Structural Surfaces

Section 14.1 provides the fractional radioactivity contribution of each ROC as determined from characterization data. As described in Section 3.3.5.1 of the HRCR, the survey strategy was to identify locations of elevated radiation for sampling to determine fractional radioactivity contributions for use in FSS planning from the site buildings that are planned to remain after license termination (Building 110, Building 230 and Building 231). Because of the lack of significantly contaminated surfaces, the approach described in the HRCR was the most appropriate method for empirically determining the ratios of radionuclides.

The fractional radioactivity contribution of Am-241, Np-237, and Pu-239/240 are consistent with the trace radionuclides commonly present in enriched Uranium that has been blended with recycled fuel. The fractional radioactivity contributions of U-234, U-235, and U-238 are reasonable because they are consistent with those expected to be present in Uranium enriched to 4.5 weight-percent, and this enrichment is consistent with the site history as the HSA indicates that only low enriched Uranium was processed after the time that Building 230 and Building 231 were constructed. The HSA did not note any radiological processes that occurred in Building 110. In addition, the building and structural surface $DCGL_W$ values presented in Table 14-1 do not vary significantly between the three isotopes and therefore the gross radioactivity calculation (discussed in Section 14.1) is not sensitive to the enrichment. For

Tc-99, the fractional radioactivity contribution is not consistent and is much lower than those presented in Section 5.0 of Reference 14-4; however, this result is conservative because of the much larger (three orders of magnitude) $DCGL_W$ for Tc-99 compared to those for U-234, U-235, and U-238.

14.2.9.2 Soil

Reference 14-4 provides surrogate relationships and justifications for inferring concentrations of U-234 and Tc-99 in soil. Implementation of the surrogates was discussed in detail in Section 14.1.

14.3 REMEDIAL ACTION SUPPORT (IN-PROCESS) SURVEYS

Remedial Action Support Surveys (RASS) are conducted to: 1) guide remediation activities; 2) determine when an area or survey unit has been adequately prepared for the FSS; and, 3) provide updated estimates of the parameters (e.g., variability, and in some instances, a verification of the isotopic mix) to be used for planning the FSS. During soil excavation, the RASS will also serve to assess the potential concentration and amount of U-235 for comparison to the NCS Exempt Material Limit.

RASS of soil areas will rely principally on direct radiation measurements using gamma sensitive instrumentation described in Table 14-14. In addition to direct radiation measurements, the RASS will include the collection of samples of soil, sediment and surface residue for laboratory analysis.

RASS of the surface of building or structures and systems to be remediated, or where there is a potential for residual surface contamination, the RASS will be performed using surface contamination monitors, augmented with sampling for removable surface contamination.

14.3.1 DESCRIPTION OF FIELD SCREENING METHODS AND INSTRUMENTATION

14.3.1.1 Field Screening – Capability Of Detection At DCGL

Table 14-14 shows typical field instruments that will be used for performing final status surveys. The same or similar instruments will be used during the performance of the RASS. The typical MDCs provided in Table 14-14 are sufficient to measure concentrations at the $DCGL_W$ for field instruments used for scanning.

Analytical capability for soil sample analysis will supplement field scanning techniques to provide radionuclide-specific quantification, achieve lower MDCs, and provide timely analytical results. The on-site laboratory will include a gamma spectroscopy system calibrated to various soil sample geometries. The system will be calibrated using NIST-traceable mixed gamma standards or intrinsic calibration routines. Count times will be established such that the DQOs for MDC will be achieved. Methods analysis of Tc-99 in soil samples will include beta spectroscopy by liquid scintillation, and will be performed at an approved off-site laboratory. Likewise, alpha spectroscopy will be performed at an approved off-site laboratory. Table 14-13 provides a list of typical laboratory analysis methods and the associated MDCs.

14.3.2 FIELD SCREENING METHODS FOR THE RASS OF SOIL DURING EXCAVATION

A gamma walk-over survey (GWS) will be performed of the excavated surface, typically using a 2 inch by 2 inch NaI gamma scintillation detector. Appropriate scanning speed and scanning distance will be implemented to ensure the MDCs listed in Table 14-14 are achieved. Locations of elevated count rate will be identified for additional scanning and/or the collection of biased

soil samples to determine if the elevated count rate indicates the presence of soil concentrations in excess of the $DCGL_W$. The information obtained during the RASS (GWS and the analytical data from any associated soil samples) will be used to categorize soil/debris into one of four categories:

- Potentially exceeding the NCS Exempt Material Limit and requiring additional evaluation and/or handling methods (described in Chapter 10.0);
- Potentially containing radioactivity concentrations above the applicable $DCGL_W$ and requiring further excavation;
- Expected to contain radioactivity concentrations that are less than the $DCGL_W$, but requiring removal in order to access additional soil/debris having radioactivity concentrations above the applicable $DCGL_W$. Potentially acceptable for re-use as backfill; or,
- Expected to contain radioactivity concentrations that are less than the $DCGL_W$, and not requiring removal.

If the survey instrument scan MDC is less than the applicable $DCGL_W$ for the stratum (elevation) in which the soil resides, then scanning will be the primary method for guiding the remediation. The average net count rate corresponding to the $DCGL_W$ will be determined based on surveyor experience in correlating the count rate observed in the field to the results of subsequent laboratory analysis of samples, and then used to identify the locations requiring additional remediation. Once the scan surveys and the laboratory data obtained from any biased soil samples that may have been collected indicate residual concentrations are less than the $DCGL_W$, the area will be considered suitable for FSS.

If the scan MDC is greater than the $DCGL_W$, the GWS will still be used to initially guide remediation; however, as the levels are reduced to the range of the $DCGL_W$ an additional number of biased soil samples may be required to ensure the area is suitable for FSS.

A summary discussion regarding the performance of RASS in areas requiring nuclear criticality controls is provided below. This discussion is followed by two additional examples RASS for the remediation of areas of contaminated surface soil, and for the removal of overburden soil in order to gain access to contaminated sub-surface soil.

14.3.2.1 Survey Methodologies For Removal Of Soil And Commingled Materials Potentially Containing Enriched U-235

Gamma scanning will consist of a combination of scans to assess the soil for compliance with the appropriate DCGLs for the exposed lens of soil (e.g., shallow, root or deep strata) in conjunction with scan surveys to specifically identify U-235 concentrations or amounts above the NCS

Exempt Material Limit as discussed in Chapters 8.0 and 10.0. The following sections describe the implementation method for each scan.

14.3.2.1.1 Scans For Nuclear Criticality Safety

Gamma scans to address the requirements for NCS will consist of two independent surveys performed by two technicians using different instruments. Depending upon the anticipated source term, the instruments may be setup in a single channel mode using the predominant gamma energy associated with U-235 decay (185.72 KeV). It is understood that this protocol may not be effective in the portion of the Burial Pit Area where Ra-226 was identified due to the similar decay energy of 186.2 KeV. In this area, the scan surveys will be validated using either a portable gamma spectroscopy system or the collection of soil samples at locations of confirmed elevated count rates. The surveys will be performed for each exposed cut depth of soil (nominally 1 foot). These scan surveys will be implemented at the start of the surface excavation and will continue to be performed until no visible debris is observed in the excavation. Materials exceeding the NCS Exempt Limit will be dispositioned as described in DP Chapter 10.0.

In situations where subsequent FSS results indicate the residual U-235 concentration exceeds the NCS Exempt Material Limit, these controls will be re-initiated during the removal of the material exceeding the NCS Exempt Material Limit. This protocol will be employed in the following site areas: Burial Pit Area; the soil east of the process complex extending to the documented Burial Pit Area (areas suspect for undocumented burial of materials containing enriched U-235); below the process complex slabs; evaporation ponds; Red Room Roof burial area; and any other excavation area where buried waste is discovered during the remediation process.

14.3.2.1.2 Scans For DCGL Compliance

In conjunction with the scans for NCS to verify U-235 concentrations are below the NCS Exempt Material Limit, information will be obtained that is needed to determine compliance with the applicable DCGL. The scan survey will typically be performed with the survey instruments setup to detect any gamma emitting radionuclide (open window). This scan survey will be performed as described in Section 14.3.1 above.

14.3.2.2 <u>Survey Methodologies During Removal Of Surface Contaminated Soil</u>

Prior to remediation, the location of contaminated soil, as identified by characterization surveys and sampling will be visually marked in the field (e.g., civil land survey stakes, spray paint). Gamma scanning will be performed during excavation to confirm, or redefine the lateral and vertical extent of contamination, and to identify soil concentrations that likely exceed the remedial goal.

As soil is excavated, gamma scans will be used to guide the remediation and to support the segregation of soil for potential re-use as backfill. When gamma scans indicate that the concentrations in the remaining soil are likely below the remedial goal, and an adequate number of soil samples verify radioactivity concentrations below the $DCGL_w$, the area will be deemed suitable for FSS.

14.3.2.3 Survey Methodologies During Removal Of Soil Intended To Be Used As Backfill

The objectives of the gamma scan surveys performed during the excavation of soil potentially suitable for re-use as backfill (e.g., overburden in the Burial Pit Area) include the identification of discrete locations of elevated concentrations (as indicated by count rate) for segregation from the balance of the soil. These surveys also serve to confirm that the count rates associated with the remaining soil intended for re-use as backfill are relatively uniform, and below those typically associated with soil containing concentrations in excess of the DCGL.

Due to limitations regarding criticality safety (discussed in Chapter 10.0) and the practicality of soil excavation (discussed in Chapter 8.0), approximately 1 foot of soil (30 cm) will be surveyed and subsequently excavated (lifted) at one time.

One of the methods described below will be used for further evaluation of soil intended for re-use as backfill, dependent on whether High Resolution Gamma Spectroscopy (HRGS) is utilized.

14.3.2.3.1 Survey Methodologies Utilizing HRGS

Analysis of the soil may be completed by use of a gamma spectroscopy box counter, or equivalent configuration, in conjunction with soil sampling and analysis. If the box counter does not have adequate sensitivity such that an MDC is greater than the applicable $DCGL_W$ for the stratum where the material will be placed as backfill, this approach will not be used.

a. Prior to the excavation, a gamma scan survey of the subject surface area will be performed and areas of elevated count rate will be flagged for segregation. As an alternative to flagging the area of an elevated count rate, a soil sample may be obtained from the depth of 0 cm - 30 cm below ground surface for subsequent laboratory analysis and evaluation. Soil subject to this protocol will originate in Class 1 survey units; therefore, gamma scan surveys will be performed over 100 percent of the exposed surface of each exposed lens of soil. The scan survey will also be used to document the uniformity of the soil prior to measurement by the HRGS.

   Note, the ability of surface scans to detect the gamma emissions below depths of 15 cm is diminished, but compensated for, by the use of the HRGS since the field of view will include a portion of the soil below 15 cm once placed in container.

b. This process will be repeated for each one-foot lift of material.

c. The removed soil will be loaded into a container (e.g., dump truck with a twenty (20) cubic yard capacity) and then assessed with an appropriately calibrated gamma spectroscopy system that achieves an MDC that is less than the applicable $DCGL_W$ for the stratum where the material will be placed as backfill.

d. The material will be transported to the material lay down area and dumped. As an added measure of assurance that the soil is suitable for re-use as backfill, a gamma scan survey will be performed of the surface of the pile to identify any locations of elevated count rate for subsequent removal.

e. Following the scan survey, a composite sample, consisting of four or more aliquots collected at random, will be submitted for laboratory analysis. The laboratory analyses will meet the applicable DQO for FSS.

f. Dependent on the results of the gamma scan survey and/or laboratory analysis of the composite sample, the pile will then be relocated to the appropriate stockpile as discussed below.

g. Final evaluation of the excavated area (when remediation is believed to be completed) will be performed as discussed in Section 14.4.4.

Note that the sequence of this approach for evaluating soil for re-use as backfill will provide for: (1) a gamma scan survey of 100 percent of the surface prior to excavation; (2) spectral analysis of the entire volume of soil intended for re-use as backfill; (3) a gamma scan survey of the soil in a second configuration; and, (4) the results of the laboratory analysis is based on a representative sample as the soil is being accumulated.

14.3.2.3.2 Survey Methodologies When HRGS Is Not Utilized

One of the two following approaches will be used when a gamma spectroscopy box counter is not utilized. To compensate for the lack of use of the HRGS, a GWS will be performed followed by systematic and biased sampling as follows:

<u>Approach 1:</u>

a. A gamma scan survey will be performed over 100 percent of the exposed surface of each lens of soil, and areas of elevated count rate will be flagged for segregation. Alternatively, a soil sample may be obtained from the depth of 0 cm - 30 cm below ground surface for subsequent analysis and evaluation.

b. Systematic and biased soil sampling will be performed using methods based on FSS protocols described in Section 14.4.

c. Once the soil has been determined to meet the requirements for re-use as backfill, the soil will be removed in approximately 1 foot lifts, and stockpiled.

d. This process will be repeated for each one-foot lift of material.

e. Final evaluation of the excavated area (when remediation is believed to be completed) will be performed as discussed in Section 14.4.4.

Approach 2:

a. A gamma scan survey will be performed over 100 percent of the exposed surface of each lens of soil, and areas of elevated count rate will be flagged for segregation. Alternatively, a soil sample may be obtained from the depth of 0 cm – 30 cm below ground surface for subsequent analysis and evaluation.

b. The soil will be removed in approximately 1 foot lifts and taken to an interim laydown area where it will be spread.

c. A gamma scan survey will be performed over 100 percent of the spread pile and areas of elevated count rate will be flagged for segregation.

d. Systematic and biased soil sampling will be performed using methods based on FSS protocols described in Section 14.4.

e. Once the soil has been determined to meet the requirements for re-use as backfill, the soil will be stockpiled.

f. Final evaluation of the excavated area (when remediation is believed to be completed) will be performed as discussed in Section 14.4.4.

14.3.2.4 Soil Segregation

Independent of the method employed to survey and demonstrate that the excavated soil meets the applicable $DCGL_W$ values for the stratum where the material will be placed as backfill, the soil will be segregated dependent on survey results and consigned to the appropriate interim stockpile as follows:

- If the survey results indicate the soil is ≤ the Uniform stratum DCGL, then the material will be placed in the stockpile designated for use as backfill within any strata;

- If the survey results indicate the soil is > the Uniform stratum DCGL and ≤ the Root stratum DCGL, then the material can be placed in the stockpile designated for use as backfill in the Root or Deep strata;

- If the survey results indicate the soil is > the Root stratum DCGL and ≤ the Excavation DCGL, then the material can only be placed in the stockpile designated for use as backfill in the Deep stratum; and,

- If the survey results indicate the soil exceeds the Excavation DCGL, then the material will be placed in the stockpile designated for disposal as radioactive waste.

For each stockpile of soil, the average concentration of the stockpile will be calculated and accounted based on a weighted average of each lift or container as the material is added to the stockpile. This average value will then be used to evaluate the dose impacts of using that particular stockpile of soil as backfill. This application of the unity rule is discussed in more detail in Section 14.1.

14.3.3 FIELD SCREENING METHODS FOR THE RASS OF STRUCTURES, SYSTEMS AND COMPONENTS

For SSCs to be remediated, or where there is a potential for residual surface radioactivity, operational type surveys with surface contamination monitors will be performed (see Table 14-14). Surface scanning will be performed to identify any areas of residual radioactivity that exceed the gross radioactivity $DCGL_W$. The count rate that corresponds to the gross radioactivity $DCGL_W$ will be determined for the instrument used and the surveyor will mark areas exceeding this value with paint, a marker, or other identifying means.

Following remediation, the area will be rescanned. When the area has been effectively remediated, a post-remediation survey will be documented. The results will be evaluated to determine suitability of the SSC for turnover for FSS. Once the SSC has been determined to be ready for FSS, isolation and control measures will be established as described in Section 14.4 to ensure the area does not become further impacted by the surrounding remediation efforts.

14.4 FINAL STATUS SURVEY DESIGN

The objective of the FSS is to demonstrate that the dose from residual radioactivity at the HDP Site does not exceed the annual dose criterion for license termination for unrestricted use specified in 10 CFR 20.1402 (Reference 14-1), and that the levels of residual radioactivity are ALARA. The additional requirement of 10 CFR 20.1402 that all residual radioactivity at the site be reduced to levels that are ALARA is addressed in Chapter 7.0. An FSS will be performed on all impacted open land areas and SSCs that are to remain at the time of license termination. The following describes the major elements of the FSS process and provides a general roadmap on how the FSS will be implemented.

The final status survey process described in this section adheres to the guidance of MARSSIM for the design of final status surveys. The guidance as contained in the following regulatory documents was used in the development of the FSS design:

- NUREG-1757, Volume 2, Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria (Reference 14-5);
- NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Reference 14-6);
- NUREG-1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions (Reference 14-7); and,
- NUREG-1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys (Reference 14-9).

Buried piping and equipment that will remain in place after site closure that have had a potential for radioactive contamination above the $DCGL_W$ (based on site operating history) or known contamination above the $DCGL_W$ (based on previous radiation surveys or surveys performed during decommissioning) will be designated as Class 1 for the purpose of Final Status Survey. Pipe crawlers or other specialty conveyance devices will be deployed using conventional instrumentation. If advanced technology instrumentation, such as in-situ gamma-spectroscopy, is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. The method for final status surveys of piping will be submitted for NRC review and approval, with approval received prior to implementation of final surveys of piping

14.4.1 OVERVIEW

The final status survey provides data to demonstrate that all radiological parameters in a specific survey unit satisfy the established guideline values and conditions. The primary objectives of the FSS are to:

- select/verify survey unit classification;
- demonstrate that the potential dose from residual radioactivity is below the release criterion for each survey unit; and,
- demonstrate that the potential dose from small areas of elevated radioactivity is below the release criterion for each survey unit.

The final status survey process consists of four principal elements:

- Planning (Section 14.4.2);
- Design (Section 14.4.3);
- Implementation (Section 14.4.4); and,
- Data Assessment (Section 14.4.5)

The Data Quality Objective (DQO) and Data Quality Assessment (DQA) processes are applied to these four principal elements. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions (as is the case in FSS). The DQA process is an evaluation method used during the assessment phase of the FSS to ensure the validity of survey results and demonstrate achievement of the sampling plan objectives (e.g., to demonstrate compliance with the release criteria in a survey unit).

Survey planning includes review of the HSA, the HRCR, and other pertinent characterization information to establish the radionuclides of concern and survey unit classifications. Survey units are fundamental elements for which final status surveys are designed and executed. The classification of a survey unit determines how large it can be in terms of surface area. If any radionuclides of concern are present in background, the planning may include establishing appropriate reference areas to be used to establish baseline concentrations for these radionuclides and their variability. Reference materials are specified for establishing background instrument responses for cases where gross radioactivity measurements were made and to allow replication of survey efforts if necessary.

Before the FSS process can proceed to the design phase, concentration levels that represent the maximum annual dose criterion of Reference 14-1 must be established. These concentrations are established for either surface contamination or volumetric contamination. They are used in the survey design process to establish the minimum sensitivities required for the available survey instruments and techniques and, in some cases, the spacing of total surface contamination measurements or samples to be made within a survey unit. Surface or volumetric concentrations that correspond to the maximum annual dose criterion are referred to as Derived Concentration Guideline Levels, or DCGLs. A DCGL established for the average residual radioactivity in a survey unit is called a $DCGL_W$. Values of the $DCGL_W$ may then be increased through use of area factors to obtain a DCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. The scaled value is called the $DCGL_{EMC}$, where EMC stands for elevated measurement comparison. The $DCGL_{EMC}$ is only applicable to Class 1 survey units. DCGL development is discussed in Chapter 5.0.

Before the FSS process can proceed to the implementation phase, turnover and control measures will be implemented for an area or survey unit as appropriate. A formal turnover process will ensure that decommissioning activities have been completed and that the area or survey unit is in a suitable physical condition for FSS implementation. Isolation and control measures are primarily used to limit the potential for cross-contamination from other decommissioning activities and to maintain the final configuration of the area or survey unit.

Survey implementation is the process of carrying out the survey plan for a given survey unit. This consists of scan measurements, total surface contamination measurements, and collection and analysis of samples. Quality assurance and control measures are employed throughout the FSS process to ensure that subsequent decisions are made on the basis of data of acceptable quality. Quality assurance and control measures are applied to ensure:

- DQOs are properly defined and derived;
- the plan is correctly implemented as prescribed;
- data and samples are collected by individuals with the proper training using approved procedures;
- instruments are properly calibrated and source checked;
- collected data are validated, recorded, and stored in accordance with approved procedures;
- documents are properly maintained; and,
- corrective actions are prescribed, implemented and followed up, if necessary.

The DQA approach is applied to FSS results to ensure the population of the data are complete, the data are valid, and to determine whether the objectives of the FSS have been met. The data quality assessment includes:

- verify that the measurements were obtained using approved methods;
- verify that the quality requirements for the methods were met;
- verify that the appropriate corrections were made to the gross measurements and the data are expressed in proper reporting units;
- verify that the measurements required by the survey design, and any measurements required to support investigation have been included;
- verify that the classification and associated survey unit design remain appropriate based on a preliminary review of the data;
- subject the measurement results to the appropriate statistical tests;
- determine if the residual radioactivity levels in the unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

In some cases, data evaluation will show that all of the measurements made in a given survey unit were below the applicable $DCGL_W$. If so, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the $DCGL_W$ are observed. In these cases, statistical tests must be performed to determine whether the survey unit meets the release criterion. The statistical tests that may be required to make decisions regarding the residual radioactivity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

The statistical tests will include the Sign test, or the Wilcoxon Rank Sum (WRS) test for instances when the measurement results are corrected for the contribution from background radioactivity. The WRS test will used for the evaluation of results obtained within open land surveys. The measurements of surface contamination within buildings will be evaluated using the Sign test.

Survey results will be converted to appropriate units of measure (e.g., dpm/100 cm^2 or pCi/g) and compared to investigation levels to determine if the action levels for investigation have been exceeded. Measurements exceeding investigation action levels will be investigated. If confirmed within a Class 1 survey unit, the location of elevated concentration may be evaluated using the elevated measurement comparison, or the location may be remediated and re-surveyed.

If confirmed within a Class 2 or 3 survey unit, the survey unit, or portion of the survey unit, will typically be reclassified and a re-survey performed consistent the change in classification.

As a survey progresses, reevaluation of a survey unit classification may be necessary based on newly acquired survey data. For example, if contamination is identified in a Class 3 area, an investigation and reevaluation of that area should be performed to determine if the Class 3 area classification is appropriate. Typically, the investigation will result in part or all of the area being reclassified as Class 1 or Class 2. If survey results identify residual contamination in a Class 2 area exceeding the $DCGL_W$ or suggest that there may be a reasonable potential that contamination is present in excess of the $DCGL_W$, then an investigation should be initiated to determine if all or part of the area should be reclassified to Class 1 (see DP Section 14.4.3.6 for details).

Documentation of the FSS will occur in FSS Survey Unit Release Record for each survey unit, and will include a historical record of the FSS process. A FSS Final Report will be prepared to include the Survey Unit Release records as appendices, and will provide a summary of the survey results and the overall conclusions that demonstrate the site, or portions of the site, meets the radiological criteria for unrestricted use. These reports are discussed in detail in Section 4.6.

It is anticipated that the FSS Final Report may be provided to the NRC in phases as remediation and FSS are completed with related portions of the site. The phased approach for submittal is intended to provide NRC with detailed insight regarding the remediation and FSS early in the process, to provide opportunities for improvement based on feedback, and to support a logical and efficient approach for technical review and independent verification.

14.4.2 FINAL STATUS SURVEY PLANNING

14.4.2.1 Data Quality Objectives

The DQO process will be incorporated as an integral component of the data life cycle, and is used in the planning phase for scoping, characterization, remediation and final status survey plan development using a graded approach. Survey plans that are complex or that have a higher level of risk associated with an incorrect decision (such as final status survey) require significantly more effort than a survey plan used to obtain data relative to the extent and variability of a contaminant. The DQO process entails a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions. Furthermore, the DQO process is flexible in that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. The DQO process is iterative allowing the survey planning team to incorporate new knowledge and modify the output of previous steps to act as input to subsequent steps. The appropriate design for a given survey will be developed using the DQO process as outlined in

Appendix D of MARSSIM. The seven steps of the DQO process are outlined in the following sections.

14.4.2.1.1 State The Problem

The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated and the estimated resources. The problem associated with FSS is to determine whether a given survey unit meets the radiological release criterion of 10 CFR 20.1402.

14.4.2.1.2 Identify The Decision

This step of the DQO process consists of developing a decision statement based on a principal study question (i.e., the stated problem) and determining alternative actions that may be taken based on the answer to the principal study question. Alternative actions identify those measures to resolve the problem. The decision statement combines the principal study question and alternative actions into an expression of choice among multiple actions. For the final status survey the principal study question is "Does residual radioactive contamination that is present in the survey unit exceed the established $DCGL_W$ values?" The alternative actions may include no action, investigation, resurvey, remediation and reclassification.

Based on the principal study question and alternative actions listed above, the decision statement for the final status survey is to determine whether or not the average radioactivity concentration for a survey unit results in a SOF less than unity.

14.4.2.1.3 Identify Inputs To The Decision

The information required depends on the type of media under consideration (e.g., soil, water, concrete) and whether existing data are sufficient or new data are needed to make the decision. If the decision can be based on existing data, then the source(s) will be documented and evaluated to ensure reasonable confidence that the data are acceptable. If new data are needed, then the type of measurement (e.g., scan, direct measurement and sampling) will need to be determined.

Sampling methods, sample quantity, sample matrix, type(s) of analyses and analytic and measurement process performance criteria, including detection limits, are established to ensure adequate sensitivity relative to the release criteria.

The following information will be utilized to support the decision:

- radionuclides of concern;

- measuring and/or inferring Tc-99 and U-234;
- minimum detectable concentrations; and,
- measurement and sampling results.

14.4.2.1.4 Define The Study Boundaries

This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame for collecting the data, practical constraints and the scale of decision making. In FSS, the target population is the set of samples or direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest (e.g., soil, water, concrete, and steel) is specified during the planning process. The spatial boundaries include the entire area of interest including soil depth, area dimensions, contained water bodies and natural boundaries, as needed. Temporal boundaries include those activities impacted by time-related events including weather conditions, seasons (i.e., more daylight available in the summer), operation of equipment under different environmental conditions, resource loading and work schedule.

For the site final status survey, the study boundaries include the impacted buildings and systems to remain, and the impacted soil areas of the site to sample depths based on characterization data.

14.4.2.1.5 Develop A Decision Rule

This step of the DQO process develops the binary statement that defines a logical process for choosing among alternative actions. The decision rule is a clear statement using the "If...then..." format and includes action level conditions and the statistical parameter of interest (e.g., mean of data). Decision statements can become complex depending on the objectives of the survey and the radiological character of the affected area.

The decision rule is based on if the radioactivity concentrations of residual radioactivity exceed the established $DCGL_W$ values.

1. If the SOF is less than or equal to any applicable action level and unity (1), then no additional investigation will be performed and the survey unit will be recommended for unrestricted release.
2. If the SOF is greater than unity (1), then the Radiation Safety Officer will be consulted to determine further action. Potential actions included are remediation, reclassification, additional data collection or application of the elevated measurement comparison.

14.4.2.1.6 Specify Limits On Decision Errors

This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided for rejection.

The primary consideration during FSS will be demonstrating compliance with the release criteria. The following statement will be used as the null hypothesis at the site: "The survey unit exceeds the release criteria".

Decision errors occur when the data set leads the decision-maker to make false rejections or false acceptances during hypothesis testing. Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors are considered acceptable. The upper bound corresponds to the release criteria. The Lower Bound of the Gray Region (LBGR) is determined in this step of the DQO process. LBGR is influenced by a parameter known as the relative shift. The relative shift is the $DCGL_W$ minus the LBGR (i.e., the width of the Gray Region). The relative shift is set between (and including) 1 and 3. If the relative shift is not between (or including) 1 and 3, then the LBGR is adjusted. Decision errors are discussed in more detail in Section 14.4.3.1.1.

Sample uncertainty is controlled by collecting a small frequency of additional samples from each survey unit. Analytical uncertainty is controlled by using appropriate instrumentation, methods, techniques, training, and Quality Control. The MDC values for individual radionuclides using specific analytical methods will be established. Uncertainty in the decision to release areas for unrestricted use is controlled by the number of samples and/or measurement points in each survey unit and the uncertainty in the estimate of the mean radionuclide or gross radioactivity concentrations. Table 14-14 provides the MDC values for the field and laboratory instrumentation expected to be used for the FSS.

Graphing the probability that a survey unit does not meet the release criteria may be used during FSS. This graph, known as a power curve, may be performed retrospectively (i.e., after FSS) using actual measurement data. This retrospective power curve may be important when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria) to demonstrate that the DQOs have been met.

14.4.2.1.7 Optimize The Design For Obtaining Data

The first six steps of the DQO process develop the performance goals of the survey. This final step in the DQO process leads to the development of an adequate survey design.

By using an on-site analytical laboratory, sampling and analyses processes are designed to provide near real-time data assessment during implementation of field activities and FSS. Gamma scans provide information on soil areas that have residual radioactivity greater than background and allow appropriate selection of biased sampling and measurement locations. This data will be evaluated and used to refine the scope of field activities to optimize implementation of the FSS design and ensure the DQOs are met.

14.4.2.2 Initial Site Designation

Not all areas of a site will have the same potential for residual contamination and consequently not all areas will require the same level of survey coverage to achieve an acceptable level of confidence that the area satisfies the established release limits. Therefore, to provide an overall planning basis for the FSS, the site has been initially classified into either impacted or non-impacted areas.

The site designations of the impacted areas are based upon the assessment of the HSA, HRCR, and a horizontal and vertical profile review of the characterization results as discussed in Section 14.2. The review followed the guidance as described in Section 4.4 of MARSSIM and Appendix A of NUREG-1757.

14.4.2.2.1 Non-Impacted Areas

Non-impacted areas are defined as areas that have no reasonable potential for residual contamination. These include areas that have no impact from site operations based upon the location(s) of licensed operations, site use, topography, site discharge locations, and other site physical characteristics. These areas include the outlying open land areas of the site and would not require FSS surveys to satisfy regulatory requirements for unrestricted release.

14.4.2.2.2 Impacted Areas

Impacted areas are defined as areas that may contain residual radioactivity from licensed activities. These areas require final status surveys to satisfy regulatory requirements for unrestricted release.

Using the data from the HSA, the HRCR and other previous characterization, impacted site open land areas have been initially designated as impacted or non-impacted as depicted in Figure 14-11. Additionally, the impacted areas were further designated as Class 1, Class 2 or Class 3 open land areas using FSS protocols and are depicted in Figure 14-12.

Some areas of the site that were previously designated as non-impacted will become impacted due to planned decommissioning activities (e.g., the construction of a lay down area northeast of the Burial Pits). These projected decommissioning-impacted areas are depicted in Figure 14-11.

In order to facilitate the scheduling, management and reporting of the FSS, the impacted areas of the Hematite Site have been divided into survey areas as depicted in Figure 14-13. A survey area is comprised of one or more survey units, the bounds of which are defined by existing facility physical features, such as a room, roadway, fencing, intersection of walls, column-and-row layout of a floor elevation, or structural I-beams.

14.4.2.3 Survey Units

To allow a more concentrated survey effort in the areas likely to be contaminated, impacted survey areas are further subdivided into Class 1, Class 2, or Class 3 survey units.

A survey unit is a contiguous area (usually) with similar characteristics and contamination potential. Survey units are assigned only one classification. Survey units are established to facilitate the survey process and aid in the statistical evaluation of the survey data. The site is surveyed and evaluated on a survey unit basis and the decision to release an area is made at the survey unit level. Survey unit shape and size should be consistent with the exposure pathway modeling used to convert residual radioactivity into dose.

The suggested maximum survey unit sizes by classification as recommended by MARSSIM are provided in Table 14-15. Guidance will be taken into consideration when delineating survey units; however, survey units may be increased up to 10 percent in size to account for the impact of physical conditions during the remediation phase. As an example, if an isolated Class 1 open land area has a size of 2,200 m^2, the area will be considered only one survey unit.

Building survey unit delineation will take into consideration the DCGL modeling assumptions. Soil survey units will have compact shapes rather than highly irregular (gerrymandered) shapes unless unusual shapes are practical given appropriate site operational history or site topography. Plant Soil SEAs, Tc-99 SEAs, and Burial Pits SEAs were also taken into consideration when establishing survey units.

A conceptual approach for the configuration of survey units are depicted in Figure 14-14 for open land areas; and Figure 14-15, Figure 14-16 and Figure 14-17 for buildings. An initial classification and description list of the survey areas, and survey units within them, is provided in Table 14-16. It is expected that the conceptual boundaries of these survey units may be altered based on the actual conditions at the time of survey design. This may be especially characteristic of the survey units within open land areas. Examples of the need for this flexibility include the need to complete a portion of an excavation in advance of inclement weather, and challenges associated with water management of ground/surface water and precipitation.

Although these boundaries may be altered, the classification for the purpose of final status survey will not be reduced. If changing the classification of a survey unit from a more restrictive classification to a less restrictive classification (e.g., Class 1 to Class 2), then NRC approval will be required prior to implementing the change.

14.4.2.4 Initial Classification of Survey Units

Classification of a survey unit has a minimum of two stages: (1) initial classification and (2) final classification. Initial classification is performed at the time of identification of the survey unit using the information available. Final classification is performed and verified as a DQO during the final status survey design.

Although it is expected that the existing areas and conceptual survey units will require little modification with regard to classification, the characterization process is iterative. When additional information is obtained during the decommissioning process through additional characterization surveys or remedial action support surveys (performed to track the effectiveness of decontamination techniques), the data will be assessed using the DQO process to verify that the initial classification is appropriate, to guide reclassification of the survey unit, and/or to guide the design of subsequent surveys.

The appropriate classification of a survey unit is critical to the basis of survey design. A classification based on an underestimate of the potential for contamination could result in a survey design that does not obtain adequate information to demonstrate that the survey unit meets the DCGL, and in some cases can increase the potential for making decision errors. Thus, the initial assumption for classifying a survey unit is that the area contains residual radioactivity levels greater than the applicable $DCGL_W$ and, thus is a Class 1 survey unit. Available information is subsequently used to support classification of a survey unit as Class 2, Class 3, or non-impacted. Survey units have been classified using the following definitions:

- Class 1: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the $DCGL_W$. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions; 2) locations where leaks or spills are known to have occurred; 3) former burial or disposal sites; 4) waste storage sites; and, 5) areas with contaminants in discrete solid pieces of material and high specific radioactivity;

- Class 2: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$. To justify changing the classification from Class 1 to Class 2, there should be measurement data that provides a high degree of confidence that no individual measurement would exceed the $DCGL_W$. Other justifications for reclassifying an area as Class 2 may be appropriate based on site-specific considerations. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form; 2) potentially contaminated transport routes; 3) areas downwind from stack release points; 4) upper walls and ceilings of buildings or rooms

subjected to airborne radioactivity; 5) areas handling low concentrations of radioactive materials; and, 6) areas on the perimeter of former contamination control areas; and,

- Class 3: Any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$, based on site operating history and previous radiation surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

A graded approach is applied when defining the requirements for FSS. More emphasis and greater survey efforts are expended within areas that have a higher potential of contamination, while minimizing the survey requirements for areas with lesser or no potential. Class 1 areas receive the highest degree of survey effort because they have the greatest potential for contamination, followed by Class 2 then Class 3 areas. When the available information was not sufficient to designate an area as a particular class, the survey unit was classified as Class 1. Areas that are considered to be on the borderline between classes received the more restrictive classification.

This delineation of the site and proper classification is a critical step in the survey design in the effort to meet the DQOs and to reliably demonstrate that the site meets the requirements for unrestricted release.

A survey unit can have only one classification. Thus, situations may arise where it is necessary to create new survey units by subdividing areas within an existing unit. For example, residual radioactivity may be found within a Class 3 survey unit, or residual radioactivity in excess of the $DCGL_W$ may be found in a Class 2 unit. In such cases, it may be appropriate to define a new survey unit within the original unit that has a lower (more restrictive) classification. Alternately, the classification of the entire unit can be made more restrictive.

14.4.2.5 Background Reference Areas

Background reference area measurements are required when using statistical application of the WRS test; no background correction to soil sample results when performing the WRS test on the sample results. Background reference areas for soil have been identified and sampled with analytical results and resulting background levels provided in Chapter 4.0. The Sign test will be used for surface contamination on building surfaces, and will be based on net FSS results; the net results will be obtained by subtracting the instrument response to ambient conditions from the gross results, but will not include a correction for the response due to naturally-occurring radioactivity in materials of construction.

Background reference areas for open land areas have a soil type similar to the soil type within the site impacted areas. If additional reference areas are required, consideration will be given to selecting reference areas that are most similar in terms of physical, chemical, and geological characteristics. It is not expected that a background reference for building and structural surface survey units will be needed since the contribution from naturally-occurring radioactivity is not significant relative to the DCGL. If a reference area is needed, an area will be selected based on the presence of similar materials of construction.

Should significant variations within the background reference area(s) be encountered, appropriate evaluations will be performed to define the background concentrations. As noted in Appendix A, Section A.3.4 of NUREG-1757, the Kruskal-Wallis test can be conducted in such circumstances to determine that there are no significant differences in the mean background concentrations among potential reference areas. The site may consider this and other statistical guidance options in the evaluation of apparent significant variations in background reference areas.

14.4.2.6 Area Preparation: Isolation And Control Measures

Near the conclusion of remediation activities and prior to initiating the final status survey, isolation and control measures will be implemented. The determination of readiness for controls and the preparation for final status survey will be based on the results of characterization and/or a RASS that indicate residual radioactivity is unlikely to exceed the DCGLs. The control measures will be implemented to ensure the final radiological condition is not compromised by the potential for re-contamination as result of access by personnel or equipment.

These measures will consist of both physical and administrative controls. Examples of the physical controls include rope boundaries and postings indicating that access is restricted to only those persons authorized to enter by health physics. Administrative controls include approved procedures and personnel training on the limitations and requirements for access to areas under these controls.

Isolation and control measures may be implemented for areas such as an entire building or large open areas, for which there should not be any impact from on-going decommissioning activities. In the event that additional remediation is required in an area following the implementation of isolation and control measures, local contamination control measures such as tents, HEPA filters, or vacuums will be employed as appropriate.

Prior to transitioning an area from decommissioning activities to isolation and control, a walk down may be performed to identify access requirements and to specify the required isolation and control measures. The physical condition of the area will also be assessed, with any conditions that could interfere with final survey activities identified and addressed. If any support equipment needed for final survey activities, such as ladders or scaffolding, are in place, it will

be evaluated to ensure that it does not pose the potential for introducing radioactive material into the area. Industrial safety and work practice issues, such as access to high areas or confined spaces, will also be identified during the pre-survey evaluation.

For buildings, measures to prevent against the introduction of radioactive material by persons entering an isolated area may include personnel frisking stations at the entry point, the use of "sticky pads", or other such routine methods. Isolation from airborne material may include sealing off openings, including doors and ventilation ducts. Though not likely to be encountered, if a potential for waterborne material is deemed to exist (e.g., floor drains or penetrations left by decommissioning activities), similar measures will be taken to be sure such sources are sealed off from the isolated area.

For open land areas, access roads and boundaries will be posted (as well as informational notices) with signs instructing individuals to contact health physics personnel prior to conducting work activities in the area. For open land areas that do not have positive access control (i.e., areas that have passed FSS but are not surrounded by a fence), the area will be inspected periodically and any material that has been deposited since the last inspection will be investigated (i.e., scanned and/or sampled).

Isolation and control measures will be implemented through approved HDP procedures and will remain in force throughout final survey activities and until there is no risk of recontamination from decommissioning or the survey area has been released from the license.

14.4.3 FINAL STATUS SURVEY DESIGN PROCESS

The general approach prescribed by MARSSIM for final status surveys requires that at least a minimum number of measurements or samples be taken within a survey unit, so that the non-parametric statistical tests used for data assessment can be applied with adequate confidence. Decisions regarding whether a given survey unit meets the applicable release criterion are made based on the results of these tests. Scanning measurements are used to confirm the design basis for the survey by evaluating if any small areas of elevated radioactivity exist that would require reclassification, tighter grid spacing for the total surface contamination measurements, or both.

The level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Class 3 survey units receive judgmental (biased) scanning and randomly located measurements or samples. Class 2 survey units receive scanning over a portion of the survey unit based on the potential for contamination, combined with total surface contamination measurements or sampling performed on a systematic grid. Class 1 survey units receive scanning over 100 percent of the survey unit combined with total surface contamination measurements or sampling performed on a systematic grid. Depending on the sensitivity of the scanning method, the grid spacing may need to be adjusted to ensure that small areas of elevated radioactivity are detected.

14.4.3.1 Sample Size Determination

Section 5.5 of MARSSIM and Appendix A of NUREG-1757 both describe the process for determining the number of sampling and measurement locations (sample size) necessary to ensure an adequate set of data that are sufficient for statistical analysis such that there is reasonable assurance that the survey unit will pass the requirements for release. The number of sampling and measurement locations is dependent upon the anticipated statistical variation of the final data set such as the standard deviation, the decision errors, and a function of the gray region as well as the statistical tests to be applied.

The methodology in MARSSIM addresses residual radioactivity specifically only in the top 15 cm of the survey unit. Section A.1 of NUREG-1757 discusses the case when residual radioactivity is present sub-surface, or below 15 cm.

> *When there are small amounts of residual radioactivity below 15 centimeters, the MARSSIM survey methods for surface measurements are acceptable. When there are substantial amounts of residual radioactivity below 15 centimeters, the dose modeling and the survey methods should be modified to account for the subsurface residual radioactivity.*

For the site, characterization results identified isolated areas containing sub-surface radioactivity that will require remediation (e.g., the Burial Pit Area). Because of this residual sub-surface radioactivity, sub-surface DCGLs were developed and are summarized in Section 14.1.

In many remediated areas, the remediation effort will be such that surveys and sampling of the surface layer (upper 15 cm) only will be required for demonstrating compliance. In areas where it is not practical to remediate low levels of residual radioactivity (e.g., radioactivity that has leached by rainwater) a sub-surface sample will be collected at each surface sampling location. In this case, the unity rule for each CSM will be applied to demonstrate compliance, as discussed in Section 14.1.

14.4.3.1.1 Decision Errors

The probability of making decision errors is established as part of the DQO process in establishing performance goals for the data collection design and can be controlled by adopting a scientific approach through hypothesis testing. In this approach, the survey results will be used to select between the null hypothesis or the alternate condition (the alternative hypothesis) as defined and shown below.

- Null Hypothesis (H_0) – The survey unit does not meet the release criterion; and,
- Alternate Hypothesis (H_a) – The survey unit does meet the release criterion.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criterion, or false negative. This occurs when the null hypothesis is rejected when in fact it is true. The probability of making this error is designated as "α".

A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criterion, or false positive. This occurs when the null hypothesis is accepted when it is in fact not true. The probability of making this error is designated as "β".

Appendix E of NUREG-1757 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the guidance in NUREG-1757, the decision error rates for final status surveys designed for the HDP Site will be set as follows:

- the α value will always be set at 0.05 unless prior NRC approval is granted for using a less restrictive value; and,
- the β value is nominally set at 0.10, but may be modified, as necessary, after weighing the resulting change in the number of required sampling and measurement locations against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criterion.

14.4.3.1.2 Unity Rule

The unity rule, as discussed in Section 14.1, will be used for the survey planning and data evaluations for soil sample analyses since multiple radionuclide-specific measurements will be performed. As a result, the evaluation criteria and data must be normalized in order to accurately compare and relate the various data measurements to the release criteria.

14.4.3.1.3 Gray Region

The gray region is defined in MARSSIM as the range of values for the specified parameter of interest for the survey unit in which the consequences of making a decision error is relatively minor. This can be explained as the range of values for which there is a potential of making a decision error; however, there is reasonable assurance that the parameters will meet the specified criteria for the rejection of the null hypothesis.

The gray region is established by setting an upper and lower boundary. Values for the specified parameter above and below these boundaries usually result in a "black and white" or "go no go" decision. Values between the upper and lower boundary are within the "gray region" where decision errors apply most. By establishing the decision errors as specified above based on acceptable risk, the number of sampling and measurement locations may be controlled within reason.

14.4.3.1.4 Upper Boundary Of The Gray Region

For the purposes of the FSS, release parameters at or near the release guidelines will typically result in a decision that the survey unit will not meet the requirements for release, with the exception of evaluating elevated areas. As a result, the upper boundary of the gray region is typically set as the $DCGL_W$.

14.4.3.1.5 Lower Boundary Of The Gray Region

The lower boundary of the gray region (LBGR) is the point at which the Type II error (β), or false positive, applies. The LBGR will initially be set at the mean level of residual contamination in the survey unit, if available; otherwise, per MARSSIM, the initial value for the LBGR will be set to one-half of the $DCGL_W$. This value may be adjusted as necessary and may be set as low as the MDC for the specific analytical technique. This will help in maximizing the relative shift and effectively reduce the number of required sampling and measurement locations based upon acceptable risks and decision errors.

14.4.3.1.6 Relative Shift

The relative shift (Δ/σ) for the survey unit data set will be calculated. The shift (Δ) is defined as the upper boundary of the gray region, or $DCGL_W$, minus the LBGR. Sigma (σ) is defined as the standard deviation of the data set. For survey design purposes, sigma values in a survey unit and/or reference area may initially be calculated from preliminary survey and/or investigation data to assess the readiness of a survey area for FSS. Standard deviation values as determined from the characterization data are generally not recommended for Class 1 areas as this will typically contain values in excess of the guidelines and have excessive variability which will not be representative of the conditions at the time of the FSS. The standard deviation at the time of the FSS will be approximated as best as possible to ensure the FSS requirements are not too restrictive. Optimal values for the relative shift range between (and including) 1 and 3.

14.4.3.1.7 Determining Which Test Will Be Used

Appropriate tests will be used for the statistical evaluation of the survey data based on the requirement to correct the gross measurement results for the contribution from background. Tests such as the Sign Test and WRS Test will be implemented using the unity rule, surrogate methodologies, or combinations thereof as described in MARSSIM and Chapters 11 and 12 of NUREG-1505.

If background is a significant fraction of the $DCGL_W$, the WRS test will be used. The WRS Test will typically be used for the open land surveys as the contaminants of interest are present in nature. If the contaminant is not in the background or constitutes a small fraction of the $DCGL_W$, the Sign Test will be used. This Sign Test will be utilized for the building and structural surface surveys.

14.4.3.1.8 WRS Test Sample Size

The number of sampling and measurement locations, N/2, that will be collected from the reference area and survey unit will be determined by establishing the acceptable decision errors, calculating the relative shift, and using Table 5-3 of MARSSIM. The shift (Δ) is the $DCGL_W$ minus the LBGR. In other words, the shift is the width of the gray region.

$$\Delta = DCGL_w - LBGR \tag{14-20}$$

The standard approach is to initially set the LBGR at the anticipated mean radioactivity of the FSS data set. The relative shift must be calculated whether the WRS Test or the Sign Test will be performed.

$$\text{RelativeShift} = \frac{\Delta}{\sigma} \tag{14-21}$$

The value used for σ will be an estimate of the standard deviation expected for the measurements in the survey unit or reference area, whichever is greater. Desirable values for the relative shift are between (and including) 1 to 3. Smaller values substantially increase the number of required sampling and measurement locations, while larger values do little to reduce the required number.

By reading the relative shift from the left side of the Table 5-3 of MARSSIM and cross referencing to the specified decision errors, the number of sampling and measurement locations can be determined. The specified number within the table includes the recommended 20 percent adjustment or increase to ensure an adequate set of data is collected for statistical purposes. Equation 5-1 of NUREG-1575 may alternatively be used to calculate the number of sampling and measurement locations. The result will be rounded up by 20 percent. Note that N/2 locations will be identified in both the survey unit and reference area. The sample size calculations may be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or spreadsheets.

14.4.3.1.9 Sign Test Sample Size

For the Sign Test, the number of sampling and measurement locations that will be required is determined from Table 5-5 of MARSSIM in a similar manner as for the WRS Test, except that a reference area is not used. The specified values within the table also include the recommended 20 percent adjustment or increase to ensure an adequate set of data is collected for statistical purposes. Equation 5-2 of MARSSIM may alternatively be used to calculate the number of

sampling and measurement locations. The result will be increased by 20 percent. The sample size calculations may be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or spreadsheets.

14.4.3.1.10 Excavation Depth Considerations On Sample Size Determination

Remediation activities are described in Chapter 8.0. In limited circumstances after remediation activities are complete, the survey unit excavation may be such that the FSS will need to be conducted on soil surfaces that are at depths that are both less than and greater than 1.5 m deep from the original grade. For example, both the Root stratum and Excavation $DCGL_W$ may be applicable.

A conservative approach of using the most conservative $DCGL_W$ (i.e., the Root stratum $DCGL_W$ in this example) can be used to determine the sample size for the survey unit. In this case, the data assessment process will use the most conservative $DCGL_W$. However, a modification may be made to the DQO process that accounts for the reduced dose from the deeper surface, i.e., appropriately applying the Root stratum and Excavation $DCGL_W$ values for a single survey unit.

First, a modification to the shift (Δ) is required (Equation 14-20). In all cases, the $DCGL_W$ will simply be equal to unity (1) due to measuring multiple ROCs. When it is desired to set the value of the LBGR to the mean concentration in the survey unit, Equation 14-22 will be used to calculate the $LBGR_{SOF}$, normalized to unity, by using the average concentration for each ROC. It is unlikely that the areas of the survey unit at Root stratum and Deep stratum conditions will be equal and therefore the average concentration level in each area will need to be weighted. Also, if actual Tc-99 concentrations are not included in the data set that will be used to determine sample size, then the modified U-235 soil $DCGL_W$ values (Table 14-9), which account for the presence of Tc-99 will be used. The following equation defines this calculation of $LBGR_{SOF}$:

(14-22)

$$LBGR_{SOF} = \sum_{i=1}^{n} \left(f_{SS} \frac{\overline{C}_{i,RZ}}{D_{i,RZ}} + f_{DS} \frac{\overline{C}_{i,DZ}}{D_{i,DZ}} \right)$$

where:

n	=	Number of measured ROCs;
f_{SS}	=	Fraction of survey unit area at Root stratum depth;
$\overline{C}_{i,RZ}$	=	Average concentration of *i*th measured ROC in Root stratum layer;

$D_{i,\,RZ}$	=	Root stratum $DCGL_W$ for the *i*th measured ROC;
F_{DZ}	=	Fraction of survey unit area at Deep stratum depth;
$\overline{C}_{i,\,DZ}$	=	Average concentration of *i*th measured ROC in Deep stratum layer; and,
$D_{i,\,DZ}$	=	Excavation $DCGL_W$ for the *i*th measured ROC.

(Note that the sum of f_{RZ} and f_{DZ} will equal one.)

Last, a modification to the weighted sigma (σ_{SOF}) is also required (Equation 14-23). The concepts describe above in the calculation of the $LBGR_{SOF}$ apply to the modification of the σ_{SOF}. The following equation defines this calculation.

(14-23)

$$\sigma_{SOF} = \sqrt{\sum_{i=1}^{n}\left(f_{RZ}\frac{\sigma_{i,\,RZ}}{D_{i,\,RZ}} + f_{DS}\frac{\sigma_{i,\,DZ}}{D_{i,\,DZ}}\right)^2}$$

where:

n	=	Number of measured ROCs;
f_{RZ}	=	Fraction of survey unit area at Root stratum depth;
$\sigma_{i,\,RZ}$	=	Standard deviation of *i*th measured ROC in Root stratum layer;
$D_{i,\,RZ}$	=	Root stratum $DCGL_W$ for the *i*th measured ROC;
f_{DZ}	=	Fraction of survey unit area at Deep stratum depth;
$\sigma_{i,\,DZ}$	=	Standard deviation of *i*th measured ROC in Deep stratum layer; and,
$D_{i,\,DZ}$	=	Excavation $DCGL_W$ for the *i*th measured ROC.

(Note that the sum of f_{RZ} and f_{DZ} will equal one.)

A reasonable estimation of the area fractions, f_{RZ} and f_{DZ}, can be made by dividing the number of systematic locations in each depth layer by the total number of systematic locations. For

example, if 10 of 15 systematic sampling locations are located at Surface depth, then f_{RZ} will be equal to 10 / 15 = 0.67 and consequently f_{DZ} will be equal to 1 – 0.67 = 0.33.

The modified $LBGR_{SOF}$ and σ_{SOF} values can then be used to calculate the N or N/2 for the Sign and WRS tests, respectively.

14.4.3.1.11 Small Areas Of Elevated Radioactivity

Section 2.5.1.1 of MARSSIM addresses the concern of small areas of elevated radioactivity in the survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an AF to account for the small area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. Note that the consideration of small areas of elevated radioactivity typically applies only to Class 1 survey units since Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_W$. Instances where a measurement obtained in a Class 2 survey unit exceeds the $DCGL_W$ or a measurement obtained in a Class 3 survey unit exceeds 50 percent of the $DCGL_W$ will be evaluated for reclassification per DP Section 14.4.3.6.

The statistical tests that determine if the residual radioactivity exceeds the $DCGL_W$ are not adequate for providing assurance that small areas of elevated radioactivity are successfully detected, as discussed in Section 5.5.2.4 of MARSSIM. Systematic sampling and measurement locations in conjunction with surface scanning are used to obtain adequate assurance that small elevated areas comply with the $DCGL_{EMC}$; however, the number of statistical systematic sampling and measurement locations must be compared to the scan sensitivity to determine the adequacy of the sampling density. The calculation of the $DCGL_{EMC}$ is detailed in Section 14.1.

The comparison begins by determining the area bounded by the statistical systematic sampling and measurement locations. This value is calculated by dividing the area of the survey unit (A_{SU}) by N or N/2 for the Sign or WRS test, respectively.

(14-24)

$$A = \frac{A_{SU}}{n}$$

where:

A	=	Area bounded by samples;
A_{SU}	=	Area of the survey unit; and
n	=	N (Sign test) or N/2 (WRS test).

The bounded area is used to look up an AF from Table 14-11 and Table 14-12 using linear or exponential interpolation as applicable. The AF is then used to calculate the $DCGL_{EMC}$ using Equation 14-16.

The required scan MDC, which is equal to the $DCGL_{EMC}$, is then compared to the actual scan MDC. If the actual scan MDC is *less than or equal to* the required scan MDC, the spacing of the statistical systematic sampling and measurement locations is adequate to detect small areas of elevated radioactivity. If the actual scan MDC is *greater than* the required scan MDC, then the spacing between locations needs to be reduced due to the lack of scanning sensitivity.

To reduce the spacing, a new number of sampling and measurement locations must be calculated. First, a new area factor (AF') that corresponds to the actual scan MDC is calculated as illustrated below.

(14-25)

$$AF' = \frac{Actual\ Scan\ MDC}{DCGL_w}$$

Next, AF' is used to look up a new area (A') from Table 14-11 and Table 14-12 using linear or exponential interpolation as applicable. Finally, using A', an adjusted number of statistical systematic sampling and measurement locations (n_{EMC}) is calculated.

(14-26)

$$n_{EMC} = \frac{A_{SU}}{A'}$$

Therefore, the number of systematic sampling and measurement locations in the survey unit will be equal to n_{EMC} for the WRS test, the number of locations collected in the reference area is not adjusted. When multiple measured radionuclides are present, this process is repeated for each measured radionuclide. The greatest number of systematic sampling and measurement locations determined from the radionuclides will be used for the survey design.

14.4.3.2 Scan Coverage

The purpose of scan measurements is to confirm that the area was properly classified and that any small areas of elevated radioactivity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, the number of total surface contamination measurement locations may need to be increased so the spacing between measurements is reduced.

The amount of area to be covered by scan measurements is based upon the survey unit classification as described in Table 5.9 of MARSSIM and Table A.2 of NUREG-1757 and is summarized in Table 14-17. The emphasis will be placed on a higher frequency of scans in areas of higher risk.

The scan coverage requirements that will be applied for scans performed in support of final status surveys for the site are:

- For Class 1 survey units, 100 percent of the surface will be scanned;
- For Class 2 survey units, between 10 percent and 100 percent of the surface will be scanned depending upon the potential of contamination. The amount of scan coverage for Class 2 survey units will be proportional to the potential for finding areas of elevated radioactivity or areas close to the release criterion in accordance with Section 5.5.3 of MARSSIM. Accordingly, the site will use the results of individual measurements collected during characterization to correlate this radioactivity potential to scan coverage levels; and,
- For Class 3 survey units, judgmental (biased) surface scans will typically be performed on areas with the greatest potential of contamination. For open land areas, this may include surface drainage areas and collection points. For building and structural surfaces such as overhead surveys, this will include overhead horizontal surfaces and air collection systems.

14.4.3.3 Reference Grid And Sampling And Measurement Locations

The survey sampling and measurement locations are a function of the sample size and the survey unit size. The guidance provided in Section 4.8.5 and Section 5.5.2.5 of MARSSIM has been incorporated in this section. For the FSS within open land areas, the current strategy is to utilize civil surveyors and/or GPS based off of the North American Datum 83 (NAD83) State of Missouri East coordinate system, or equivalent coordinate reference system as discussed in Section 6.10.1 of MARSSIM.

14.4.3.3.1 Reference Grid

A reference grid will be used for reference purposes and to locate the sampling and measurement locations. The reference grid may be physically marked during the survey to aid in the collection of samples and measurements. At a minimum, each survey unit will have a benchmark defined that will serve as an origin for documenting survey efforts and results. This benchmark (origin) will be provided on the map or plot included in the final status survey package.

14.4.3.3.2 Systematic Sampling And Measurement Locations

Systematic sampling and measurement locations for Class 1 and Class 2 survey units will be located in a systematic pattern or grid. The grid spacing, *L*, will be determined using Equation 14-27 or 14-28 below based upon the survey unit size and the minimum number of sampling or measurement locations determined.

The spacing to be used in setting up the systematic grid used to establish total surface contamination measurement locations for Class 1 and Class 2 areas will be computed as:

$$L = \sqrt{\frac{A}{0.866N}} \quad \text{for a triangular grid, or} \tag{14-27}$$

$$L = \sqrt{\frac{A}{N}} \quad \text{for a square grid} \tag{14-28}$$

where:

L	=	grid spacing (dimension is square root of the area);
A	=	the total area of the survey unit; and,
N	=	the desired number of measurements.

Once the grid spacing is established, a random starting point will be established for the survey pattern using a random number generator. Starting from this randomly-selected location, a row of points will then be established parallel to one of the survey unit axes at intervals of L. Additional rows will then be added parallel to the first row. For a triangular grid, additional rows will be added at a spacing of 0.866L from the first row, with points on alternate rows spaced mid-way between the points from the previous row. For a square grid, points and rows will be spaced at intervals of L.

The grid spacing may be rounded down for ease of locating sampling and measurement locations on the reference grid. The number of sampling and measurements locations identified will be counted to ensure the appropriate number of locations has been identified. Depending upon the configuration and layout of the survey unit and the starting grid location, the minimum number of sampling and measurement locations may not be identified. In this event, either a new random starting location will be specified or the grid spacing adjusted downward until the appropriate number of locations is reached.

Software tools that accomplish the necessary grid spacing, including random starting points and triangular or square pitch, may be employed during final status survey. When available, this

software will be used with suitable mapping programs to determine coordinates for a GPS. The use of these tools will provide a reliable process for determining, locating and mapping measurement locations in open land areas separated by large distances and will be helpful during independent verification.

For Class 3 survey units, each sampling and measurement location will be randomly selected using a random number generator.

The systematic sampling and measurement locations within each survey unit will be clearly identified and documented for the purposes of reproducibility. Actual measurement locations will be marked and identified by tags, labels, flags, stakes, paint marks, GPS location, photographic record, or equivalent.

14.4.3.4 Investigation Process

14.4.3.4.1 General Approach to Investigation

During the FSS, areas of concern or elevated measurements may be identified that warrant further investigation. Depending upon the results of the investigation, the survey unit may require no action, additional remediation, and/or reclassification and resurvey. The investigation process and levels are described below and are consistent with the guidance in Section 5.5.2.6 of MARSSIM.

During the FSS process, locations with potential residual radioactivity exceeding investigation levels will be marked for further investigation and biased sampling or measurement. For Class 1 survey units, the size and average radioactivity level within the elevated area may be acceptable if it complies with the AFs and other criteria as it applies to the $DCGL_{EMC}$.

Biased sampling and investigations should address:

- The estimated size of the elevated area of contamination;
- The average radioactivity within the elevated area; and
- The effects of summing multiple areas of elevated radioactivity within the survey unit.

Depending upon the results of the investigation, the survey unit may be reclassified, or a portion of the survey unit may be combined with an adjacent area with similar characteristics provided there is sufficient justification. Adequate justification for partial reclassification would include an understanding of the origin of the elevated activity, and a high degree of confidence that a similar condition is unlikely to exist elsewhere within the survey unit.

The results of the investigation process will be thoroughly documented in the survey unit release record for inclusion to the FSS Final Report.

14.4.3.4.2 Specific Investigation Areas

Former Process Buildings Investigation Area

Figure 14-22 shows the investigation area beneath the former Process Buildings in which soil will be sampled and analyzed for Tc-99 and uranium from the surface of the excavation to the top of the sand/gravel layer. Final status survey sampling stations that fall within this Process Building investigation area will be sampled as follows:

- A composite soil sample will be taken from each 5 foot interval of excavated soil down to within 6 inches of the sand/gravel layer; and
- A soil sample will be taken of the remaining 6 inches of soil immediately above the sand gravel layer.

Figure 14-22 shows a conceptual layout of the conceptual final status survey units across the former Process Buildings.

Hybrid Well Investigations

The following actions shall be taken to investigate the potential for a preferential pathway of Tc-99 and uranium along a monitoring well screen that crosses both the Silty Clay Aquitard HSU and the Sand/Gravel HSU (hybrid well), and to determine whether contaminated soil exists in proximity to a hybrid monitoring well:

- When hybrid wells are abandoned they will be over drilled using hollow stem augers of sufficient outside diameter to remove approximately two inches of surrounding soil, the well riser, well screen, and screened filter pack. The auger will continue until reaching refusal, which indicates bedrock. The soil cuttings that are removed during the boring process will be surveyed for indications of elevated radioactivity as a qualitative measure and sampled for laboratory analysis. Within each 5 foot interval, sample(s) of soil indicating elevated concentrations will be collected for laboratory analysis. In the event that an elevated count is not observed, one composite sample of the cuttings collected within each 5 foot interval will be collected for laboratory analysis.
- When completing remediation actions in the area of a hybrid well screen that extends beyond the depth of soil excavation, any water sample taken over the history of that well will be assessed for results that exceed the MDC+Error [2σ] for Tc-99 or exceed the Background Threshold Value for total uranium. For such an exceedance, four borings will be made in close proximity (e.g., approximately equidistant within a 2-4 foot radius) to each monitoring well that is not excavated to the bottom of the well. The borings shall extend down to refusal, which indicates bedrock. Results from the following samples are subject to Section 14.4.5.6 for elevated measurement comparison:
    - From each 5 foot increment of depth to the top of the screened/filtered interval;

- From the increment that is equivalent to the top half of the screened/filtered interval; and
- From the increment that is equivalent to the bottom half of the screened/filtered interval.

Should a sample result from the investigation sampling described in this subsection exceed the applicable DCGL, then remediation of the subsurface soil represented by the sample is required. If remediation was by overboring, then sampling borings as described in the preceding paragraph may be used to demonstrate compliance. If remediation was by excavation, a final status survey (FSS) per Chapter 14.0 will be completed.

14.4.3.5 Investigation Levels

During the FSS, any areas of concern will be identified and investigated. This will include any areas as identified by the technician during the scan survey of soil or SSC surfaces, any areas identified during post-processing and reviewing the gamma scan survey data (if electronically logged), and any results of soil or bulk material analyses, or surface contamination measurements, that exceed the investigation levels. Based on this review, the suspect areas will be addressed by further biased surveys and sampling as necessary. The applicable investigation levels are provided in Table 14-18.

The following actions shall be taken to investigate the potential for a preferential pathway of Tc-99 and total uranium along a monitoring well screen that is across both the Silty Clay Aquitard HSU and the Sand/Gravel HSU (hybrid well):

- When wells are abandoned they will be over drilled using hollow stem augers of sufficient outside diameter to remove approximately two inches of surrounding soil, the well riser, well screen, and screened filter pack. The soil cuttings that are removed during the boring process will be surveyed for indications of elevated radioactivity as a qualitative measure and sampled for laboratory analysis.
- When completing remediation actions in the area of a hybrid well screen that extends beyond the depth of soil excavation, any water sample taken over the history of that well will be assessed for results that exceed the MDC+Error [2σ] for Tc-99 or the Background Threshold Value for total uranium. For such an exceedance, a minimum of two borings will be collected at the well, one upgradient and one down gradient of the monitoring well. The borings shall extend down the length of the well. Samples will be collected from the borings at five foot intervals and from a depth that is approximately equivalent to the screened interval. Since these samples will be obtained at greater than 1.5 meter below the ground surface, the concentrations in the samples will be compared to the Excavation scenario limits.

14.4.3.6 Remediation And Reclassification

Any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated to reduce the residual radioactivity to acceptable levels.

As a survey progresses, reevaluation of a survey unit classification may be necessary based on newly acquired survey data. An investigation should be initiated to determine if all or part of the area should be reclassified when:

- Survey results identify residual contamination in a Class 2 area exceeding the $DCGL_W$ or suggest that there may be a reasonable potential that contamination is present in excess of the $DCGL_W$.
- Survey results identify residual contamination in a Class 3 area exceeding 50 percent of the $DCGL_W$.

Typically, the investigation will involve additional scan surveys and/or sampling and result in part or all of the area being reclassified as Class 1 or Class 2. If the investigation verifies a result exceeds the $DCGL_W$ in a Class 2 or Class 3, then the survey unit will require reclassification of all or part of the survey unit to Class 1. If the investigation verifies a result to be less than the $DCGL_W$ but greater than 50 percent of the $DCGL_W$ in a Class 3 survey unit, then the survey unit will require reclassification of all or part of the survey unit to Class 2. If the investigation fails to verify a result and the variability in population of the individual and average measurement results with respect to the DCGL do not suggest the initial classification was inappropriate, then the survey unit will not be reclassified.

The investigation and the evaluation of the additional information will be thoroughly documented in the release record. If all or part of a survey unit is reclassified, then the reasons for the initial misclassification will be documented in the release record.

Re-classification of a survey unit from a less restrictive classification to a more restrictive classification may be done without prior NRC approval. However, reclassification to a less restrictive classification requires prior NRC approval.

14.4.3.7 Resurvey

If a survey unit is re-classified (in whole or in part), or if remediation is performed within a unit, then the areas affected are subject to re-survey. Any re-surveys will be designed and performed as specified in this plan based on the appropriate classification of the survey unit. That is, if a survey unit is re-classified or a new survey unit is created, the survey design will be based on the new classification.

For example, a Class 3 area that is subdivided due to the unexpected presence of radioactivity will be divided into at least two areas. One of these may remain as a Class 3 area while the other may be a Class 2 area. In order to maintain the survey design Type I and Type II decision error rates in the Class 3 area, additional measurements may be required to be performed at randomly selected locations until the required total number of measurements is met. The new sub-divided Class 2 survey area will then be surveyed using a new survey design. The Type I and II decision error rates used are documented in the final status survey report.

A Class 2 area that is subdivided due to the levels of radioactivity identified will be divided into at least two areas as well. In this case if the original survey design criteria has been satisfied, no additional action is required, otherwise the remaining Class 2 survey unit will be redesigned. The new sub-divided survey unit will be surveyed against a new survey design.

If remediation is required in only a small area of a Class 1 survey unit, any replacement measurements or samples required will be made within the remediated area at randomly selected locations following verification that the remediation activities did not affect the remainder of the unit. Re-survey will be required in any area of a survey unit affected by subsequent remediation activities. Additional guidance regarding the failure and re-survey of a survey unit and is provided in Section 8.5.3 of MARSSIM and Chapter 13 of Decommissioning Health Physics: A Handbook for MARSSIM Users (Reference 14-10).

14.4.4 FINAL STATUS SURVEY IMPLEMENTATION

14.4.4.1 Survey Methods

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable procedure. The techniques for performing survey measurements or collecting samples are specified in approved procedures.

The survey methods to be employed in the final status surveys will consist of combinations of gamma scans, scanning and static measurements of total surface contamination, and soil and sediment sampling. Additional specialized methods may be identified as necessary between the time this plan is approved and the completion of final survey activities. Any new technologies will meet the applicable data quality objectives of this plan, and the technical approach will be documented for subsequent review.

14.4.4.1.1 Scanning

Scanning is the process by which the technician passes a portable radiation detector within close proximity to the surface of a soil volume, or the surfaces of buildings/equipment with the intent of identifying residual radioactivity. Scan surveys that identify locations where the magnitude of the detector response exceeds an investigation level indicate that further investigation is warranted to determine the amount of residual radioactivity. The investigation levels may be

based on the $DCGL_W$, a fraction of the $DCGL_W$, or the $DCGL_{EMC}$, depending upon the detection capability (instrument and surveyor) to identify radioactivity.

One of the most important elements of a scan survey is define the limit of detection in terms of the *a priori* scanning MDC in order to gauge the ability of the field measurement system to confirm that the unit is properly classified, and to identify any areas where residual radioactivity levels are elevated relative to the $DCGL_W$. If the scanning indicates that the survey unit or a portion of the survey unit has been improperly classified, then the survey design process must be evaluated to either assess the effect of reclassification on the survey unit as a whole (if the whole unit requires reclassification) or a new design must be established for the new unit(s) (in the case of sub-division). A new survey design will require a re-evaluation of the survey strategy to decide if it can meet the requirements of the revised survey design. If not, the survey strategy must be revised based on the available instrumentation and methods.

14.4.4.1.2 Total Surface Contamination Measurements

Static measurements of total surface contamination are obtained by stationing the detector in close proximity to the surface, counting for a pre-determined time interval, and recording the reading. Total surface contamination measurements may be collected at random locations within a survey unit, or may be collected at systematic locations. Total surface contamination measurements may also be collected at locations of elevated radioactivity identified by scan surveys as part of an investigation to determine the source of the elevated instrument response, or at locations likely to contain residual radioactivity based on knowledge of operational history and professional judgment.

14.4.4.1.3 Removable Surface Contamination (Smears)

Removable alpha/beta contamination or smear surveys will be performed to verify that the average level of removable surface contamination within a survey unit is consistent with assumption made during dose modeling for structural DCGL development. A smear for removable radioactivity will normally be performed at each direct surface radioactivity measurement location. A 100 cm^2 surface area will be wiped with a circular cloth or paper filter, using moderate pressure. Smear samples will normally only be obtained in building surveys or in areas of hard standing (concrete, asphalt, etc.) in open land areas. Survey units that show average levels of removable contamination in excess of 10 percent of the applicable DCGL will be evaluated on a case-by-case basis to estimate the potential for unaccounted dose and/or to determine the need for additional remediation.

14.4.4.1.4 Volumetric Sampling

Sampling is the process of collecting a portion of a medium as a representation of the locally remaining medium. The collected portion of the medium is then analyzed to determine the

radionuclide concentration. Examples of materials that may be sampled include soil, sediments, and groundwater for open land areas or concrete, or roofing materials for buildings.

Bulk material samples will be analyzed via gamma spectroscopy, alpha spectroscopy or liquid scintillation counting as appropriate.

Trained and qualified individuals will collect and control samples. All sampling activities will be performed under approved procedures. The site will utilize a chain-of-custody (COC) process to ensure sample integrity.

QA requirements for final status survey activities that apply to sample collection (e.g., split samples, duplicates, etc.) and on-site and off-site laboratories employed to analyze samples as a part of the final status survey process will be controlled by approved procedures, in conformance with Chapter 13.0. Performance of laboratories will be verified periodically in accordance with quality assurance.

14.4.4.1.5 Survey Considerations For Buildings, Structures And Equipment

The condition of surfaces following decontamination activities can affect the choice of survey instruments and techniques. Removing contamination that has penetrated a surface usually involves removing the surface material. As a result, the floors and walls of decontaminated facilities can be scarred and uneven. Such surfaces are more difficult to survey because it is difficult to maintain a constant distance between the detector and the surface. In addition, scabbled or porous surfaces may attenuate or scatter radiation, particularly alpha and low-energy beta particles.

Part of the planning for the FSS of a particular survey unit will include an evaluation of the surfaces to be monitored. For conventional instrumentation, surface anomalies will be identified as part of this process and will be taken into account when selecting efficiencies to convert instrument readings to radioactivity and in the calculation of the corresponding MDCs. Conservative values will be chosen based upon surface conditions.

14.4.4.1.5.1 <u>Cracks/Crevices, Wall-Floor Interfaces And Small Holes</u>

Expansion joints, stress cracks, floor/wall interfaces, and penetrations into floors and walls for piping, conduit, anchor bolts, etc., are potential sites for accumulation of contamination and pathways for migration into sub-floor soil and hollow wall spaces. Surface contamination located on or within these irregular structure surfaces (e.g., cracks, crevices, and holes) may be difficult to survey directly. Roof surfaces and drainage points are also important survey locations. In some cases, it may be necessary to core, drill, or use other methods as necessary to gain access to areas for sampling.

Where no remediation has occurred and residual radioactivity has not been detected above background, these surface blemishes may be assumed to have the same level of residual radioactivity as that found on adjacent surfaces. The accessible surfaces are surveyed in the same manner as other structural surfaces and no special corrections or adjustments have to be made.

In situations where remediation has taken place or where residual radioactivity has been detected above background, a representative sample of the contamination within the crack or crevice may be obtained, or an adjustment for instrument efficiency may be made if justifiable. If an instrument efficiency adjustment cannot be justified based on the depth of contamination or other geometry factors, volumetric samples will be collected.

14.4.4.1.5.2 Paint Covered Surfaces

Final status surveys will consider the effect of painted surfaces. Where contamination is suspected on surfaces beneath paint coatings, gross measurements will not be used as the sole basis to assess the radiological condition. The surfaces may be volumetrically sampled and/or the coating removed prior to survey. In general, no special consideration should be required for painted surfaces (e.g., wall, floors, and ceilings) that have not been subjected to conditions that would cause radioactivity to penetrate the painted surface.

14.4.4.1.5.3 Piping And Floor Drains

Compliance with the DCGLs developed for buried piping, and presented in Section 14.1, will be demonstrated by measurements of total surface contamination and/or the collection of sediment samples. The acquisition of direct measurements using "pipe-crawling" technology and/or in-situ gamma-spectroscopy may be utilized provided adequate instrument efficiencies and detection limits can be achieved. If necessary, scaling factors may be applied to establish gross radioactivity levels via radionuclide-specific measurements or other assessments, as appropriate. Radiological evaluations for piping or drains that cannot be accessed directly will be performed via measurements made at traps and other appropriate access points where the radioactivity levels are deemed to either bound or be representative of the interior surface radioactivity levels providing that the conditions within the balance of the piping can be reasonably inferred based on those data.

14.4.4.1.5.4 Ventilation Ducts – Interiors

Measurements of total and removable surface contamination will be obtained at access points, and at locations where radioactivity is most likely to have accumulated (e.g., bends, transitions, filter housings). The measurements of surface contamination will be compared to the limits for surface contamination measurements specified in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct,

Source, or Special Nuclear Material," dated April 1993. Air sampling will be performed at outlets of ventilation ducting remaining on site to directly assess the dose contribution from ventilation ducting. Air sampling locations will be rotated to various ventilation ducting openings of the ventilation systems that will remain. The average of the calculated dose contributions from the air samples associated with the remaining ventilation systems will be added to the dose associated with the surface contamination measurements within each surface and structure survey unit as a final compliance measure to ensure the 25 mrem/yr criterion is met. Measurements of surface contamination obtained from the exterior surfaces of ventilation components will be compared to the DCGLs that apply to the building surfaces.

14.4.4.1.5.5 Building Foundations And Sub-Grade Soil

Building 110, Building 230 and Building 231 are expected to remain at the time of license termination. The HSA and HRCR include a description of the historical use and the analytical data associated with samples and measurements obtained from the structure surfaces and beneath the slabs and foundations of these buildings. Some floor drains in Building 110 and Building 230 indicate the presence of residual radioactivity that may require remediation or removal. A decision will be made whether to remediate or dispose of the drains as waste based upon the approved release criteria and the level of effort necessary to remediate or remove and dispose as waste.

It does not appear that the concentrations in soil beneath Building 110, Building 230 and Building 231 exceed the remedial goal. Section 2.3.13.2 of the HRCR notes that prior to the construction of Building 230, Health Physics sampling was performed to confirm that the building site was less than a 30 pCi/g gross alpha soil concentration guideline. Pre-construction survey results documented that the maximum activity soil concentration was 11.6 pCi/g with an average concentration of 6.2 +/- 2.8 pCi/g gross alpha. Table 5-1 of the HRCR notes that the classification for soil under buildings to be demolished is Class 1, based on the analytical results provided in Table 4-25 of the HRCR. That table shows predominantly low sum-of-fractions (SOF) values with isolated SOF values exceeding unity. Table 5-1 of the HRCR also notes that the soil under the buildings to remain is Class 3. However, it will be necessary to ascertain the radiological conditions of these foundations and sub-soil to demonstrate suitability for unrestricted release.

Measurements of residual radioactivity on surfaces will be obtained using the instrumentation and protocols described previously. Additionally, coring tools may be used to provide access through slabs and foundations to facilitate the collection of soil samples. In addition to obtaining adequate data to evaluate spatial distribution, biased sampling may be performed at locations having a high potential for the accumulation and migration of radioactive contamination to sub-surface soil. The biased locations for sub-slab soil and concrete assessment could include stress cracks, floor and wall interfaces, penetrations through walls and floors for piping, run-off from exterior walls, and leaks or spills in adjacent outside areas, etc.

14.4.4.1.6 Survey Considerations For Open Land Areas

14.4.4.1.6.1 Surface Soil

In this context, surface soil refers to outdoor areas where the soil is, for purposes of dose modeling, considered to be uniformly contaminated from the surface down to a depth of 15 cm (6 in). These areas will be surveyed through combinations of sampling, scanning, and in-situ measurements, as appropriate. Surface soil samples will be collected and prepared in accordance with approved procedures. A GPS reading will be obtained at each surface soil location and a pinned flag or similar will be placed in the ground to mark the location.

Sample preparation includes removing extraneous material and homogenizing and drying the soil for analysis. Separate containers are used for each sample and each container is tracked through the analysis process using a chain-of-custody record. Samples are split when required by the applicable FSS Quality Control requirements.

14.4.4.1.6.2 Sub-surface Soil

Sub-surface soil refers to soil that resides at a depth greater than 15 cm below the final configuration of the ground surface or soil that will remain beneath structures such as building floors/foundations or pavement at the time of license termination.

Sub-surface soil in excess of the remedial goal will be remediated as described in Section 14.3. This process will include scan surveys and the collection of soil samples during excavation to gauge the effectiveness of remediation, and to identify locations requiring additional excavation. The scan surveys and the collection of and subsequent laboratory analysis of soil samples may not be performed in a manner that is intended to meet the DQOs of FSS. For example, the soil samples may be analyzed without drying and homogenization. Although considered to be screening level data with respect to the DQOs, the data are expected to provide a high degree of confidence that the survey unit meets the remedial goal.

One of the following three scenarios will be for the final evaluation of sub-surface soil. Any peripheral portions of an excavated survey unit that are not excavated will follow the sampling protocol outlined in the "Final Evaluation of Residual Radioactivity in Soil for Unpaved Non-excavated Areas or Excavated Areas not Requiring Backfill" scenario. Table 14-24 provides a summary of the three scenarios. Additional information regarding the evaluation of sub-surface soil is provided in Section 14.3.2.

- Final Evaluation of Residual Radioactivity in Soil Prior to Backfill

  Following the collection of the screening level data as described above, physical and administrative controls will be established to prevent the potential for cross-

contamination following remediation. Following implementation of these controls, a RASS will be performed to confirm the effectiveness of remediation, followed by a subsequent FSS, as follows:

A RASS consisting of a gamma walkover survey (GWS) of 100 percent of the surfaces, and the collection of soil samples at biased locations, will be performed of the excavated surfaces (i.e., floor, including sidewalls that will not be subsequently excavated) to provide a basis that the survey unit meets the remedial goal. The GWS will typically be performed and documented in a manner that meets the DQOs of FSS. The data obtained from the collection of soil samples may be of lesser data quality (e.g., collected primarily at biased locations, and analyzed without drying and homogenization), but will nonetheless provide a high degree of confidence that the survey unit meets the remedial goal.

Following the evaluation of data obtained during the RASS, a FSS will be performed. The FSS will consist of a GWS of 100 percent of the excavated surfaces to be included in the survey unit. Note that based on an evaluation of the data obtained during the RASS, the GWS performed during the RASS may fulfill the requirement for the scan survey at the time of FSS, provided that the DQOs of FSS were met. This determination will be documented in the survey instructions. The FSS will also include the collection of soil samples at systematic grid locations, and the collection of additional samples at biased locations from the floor and as applicable, the sidewalls of the excavation, focusing on locations that appear to contain potentially elevated levels of residual radioactivity that were identified during the scan survey. The soil samples will be obtained as follows depending on the depth of the excavation surface where the systematic sample is located:

- Surface Stratum Depth: Follow the sampling protocol outlined in the “Final Evaluation of Residual Radioactivity in Soil for Unpaved Non-excavated Areas or Excavated Areas not Requiring Backfill” scenario.
- Root Stratum Depth (excavation surface is within the Root Stratum): A soil coring will be acquired that extends from the exposed surface, throughout the Root Stratum, and through the upper 15 cm of the Deep Stratum. The portion of the sample soil coring representing the Root Stratum soil (within the range of 15 cm bgs to 1.5 m bgs) will be composited and analyzed. The portion of the sample soil coring representing the top 15 cm of the Deep Stratum (1.5 m bgs to 1.65 m bgs) will be analyzed.
- Deep Stratum Depth: Samples will be taken from the top 15 cm of the exposed surface (1.5m to 1.65 m bgs) and analyzed.

Following data evaluation and management and/or regulatory approval as appropriate, the excavation will be backfilled using soil obtained from an approved off-site borrow location, or using soil originating from the site that has been identified for re-use, tested and determined to meet the remedial goals, and controlled to prevent cross contamination.

Upon completion of backfill, no further FSS samples or measurements are necessary. This is because 1) soil obtained from an approved off-site borrow location was previously tested and determined to be non-impacted, or 2) soil originating from the site that has been identified for re-use has already undergone extensive evaluations (e.g., gamma scans of the soil during excavation in one-foot lifts, analysis by HRGS in a transport container, the collection and laboratory analysis of one composite sample per each 20 yards of soil).

- Final Evaluation of Residual Radioactivity in Soil Following Backfill

The approach to FSS described in this section is envisioned to be applicable to, but not limited to situations where environmental conditions such as groundwater or precipitation pose unreasonable challenges for water management.

Sub-surface soil in excess of the remedial goal will be remediated as described in Section 14.3. This process will include scan surveys and the collection of soil samples during excavation to gauge the effectiveness of remediation, and to identify locations requiring additional excavation. The scan surveys and the collection and subsequent laboratory analysis of soil samples may not be performed in a manner that is intended to meet the DQOs of FSS. For example, the soil samples may be analyzed without drying and homogenization. Although considered to be screening level data with respect to the DQOs, the data are expected to provide a high degree of confidence that the survey unit meets the remedial goal.

A RASS consisting of a gamma walkover survey (GWS) of 100 percent of the surfaces, and the collection of soil samples will be performed of the excavated surfaces (i.e., floor, including sidewalls that will not be subsequently excavated) to provide a basis that the survey unit meets the remedial goal. The GWS will be performed and documented in a manner that meets the DQOs of FSS. Sufficient soil samples will be obtained at biased and random locations from the top 15 cm of the exposed grade of the excavation to ensure adequate coverage. The data obtained from the collection of soil samples may be of lesser data quality (e.g., analyzed without drying and homogenization), but will nonetheless provide a high degree of confidence that the survey unit meets the remedial goal. Additional soil samples, taken to a depth of one meter from the exposed grade of the excavation,

will be obtained at ten percent of the selected sample locations (biased or randomly chosen). An analysis will be performed on the shallowest 85 cm of material (composited) and a separate analysis will be performed on the deepest 15 cm of material. The latter will be used to support that concentrations are decreasing as a function of depth.

Following data evaluation and management and/or regulatory approval as appropriate, the excavation will be backfilled using soil obtained from an approved off-site borrow location, or using soil originating from the site that has been identified for re-use, tested and determined to meet the remedial goals, and controlled to prevent cross contamination.

Following the completion of backfill, a FSS will be performed. In the event that soil identified as re-use is placed as backfill, the FSS will consist of a GWS of 100 percent of the exposed ground surface, and collection and laboratory analysis of soil samples only at biased locations, focusing on locations that appear to contain potentially elevated levels of residual radioactivity that were identified during the scan survey. In the event that soil obtained from an off-site borrow location is placed as backfill, performance of a scan survey or the obtainment of surface soil samples is unnecessary. Sub-surface sampling will consist of coring or drilling through the backfill layer and one meter into the lowest point where remediation occurred.

- Final Evaluation of Residual Radioactivity in Soil for Unpaved Non-excavated Areas or Excavated Areas not Requiring Backfill

  In open land areas where sub-surface soil has been impacted by site operations, sub-surface soil samples will be obtained by use of direct push probe, rotary or percussive drilling, or other similar methods. Sub-surface samples will typically be obtained at each surface soil location. The FSS for impacted sub-surface soil will consist of:

  a. A surface sample to 15 cm;

  b. A composite sample from 15 cm to 1.5 m (Root stratum); and,

  c. If the SOF in the sample obtained from the Root stratum exceeds 0.5, a composite sample from 1.5 m to an appropriate depth (Deep stratum).

14.4.4.1.6.3 Paved Areas

Paved surfaces that remain at the site following decommissioning activities will require surveys for residual radioactivity that may be present on the exposed surface, and the collection of soil

samples from beneath the paved surface. The survey design of parking lots, roads and other paved areas will be based on soil survey unit sizes since they are outdoor areas where the exposure scenario is most similar to direct radiation from surface soil. Scan and total surface contamination measurement surveys are made as determined by the survey unit design. Paved areas may be separate survey units or they may be incorporated into other, larger open land survey units.

Where indications are that impacted soil could have been mixed by grade work prior to paving, the FSS design will define a reasonable depth of disturbed soil for evaluation based on an understanding of the construction, and examination of the soil cores. If sub-surface contamination is possible under paved or other covered areas, sub-surface volumetric samples will be collected using core bores as appropriate. These core bores can be obtained through use of split-spoon sampling, direct push probe, or larger drill rigs utilizing rotary or percussive drilling techniques.

Sub-surface samples beneath paved or concrete areas, or soil areas where remediation has not occurred, will typically be obtained at the location and frequency appropriate for unpaved areas, and will consist of the following:

1) A surface sample to a depth of 15 cm from the soil immediately beneath the asphalt or concrete (a bulk material sample from the asphalt or concrete is not necessary as this material is covered by the scan survey and total surface contamination measurements);

2) A composite sample from 15 cm to 1.5 m (Root stratum), and

3) If the SOF in the sample obtained from the Root stratum exceeds 0.5, then an additional composite sample will be taken from 1.5 m to an appropriate depth determined by the remediation and operational history of the survey unit.

14.4.4.1.6.4 <u>Groundwater</u>

Assessments of any residual radioactivity in groundwater at the site will be via groundwater monitoring wells. The monitoring wells installed at the site will monitor groundwater at both deep and shallow depths. Section 14.5 describes the groundwater monitoring to be conducted.

If there are positive results, above background, from samples collected in the sand/gravel or bedrock aquifers, then the corresponding dose will be calculated using the Dose to Source Ratios (DSRs) listed in DP Chapter 5.0, Table 5-14. Initially, the contribution to dose from the groundwater sample showing the highest individual aquifer sample result will be added to the dose attributable to the survey unit with the highest dose (calculated in accordance with Section 14.4.5.6.1) to ensure that the total dose remains below 25 mrem/yr. This contribution to dose is expected to be insignificant when compared to soil, however if this initial approach is

determined to be unduly conservative, then Westinghouse may choose to perform additional hydrogeological investigations. The investigations will be used to determine the extent of the groundwater contamination and a more realistic estimate of the groundwater source term for the purpose of performing the dose estimate as opposed to applying an individual maximum value. The NRC will be provided a report describing the method used to assess the groundwater source term if the maximum individual result is not deemed appropriate.

14.4.4.1.6.5 Sediments And Surface Water

Sediments will be assessed by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments, as appropriate. Such samples will be collected using approved procedures based on accepted methods for sampling of this nature. Sample locations will be established using the methods described in Section 14.4.3. Scanning in such areas is not normally applicable (it may be possible to scan in the site Pond once water is drained as described in Chapter 8.0).

Sediment samples will be evaluated against the DCGLs for soil. This is considered appropriate given that the action that would result in the greatest radiological impact to future inhabitants of the site would be to dredge up the sediment and use it for farming. If the sediment is left in place, then use of the soil DCGLs is conservative since many of the pathways considered in developing the soil DCGLs (direct exposure, uptake by plants, etc.) would not apply.

Assessment of residual radioactivity levels in surface water drainage systems will be via sampling of sediments, total surface contamination measurements, or both, as appropriate, making measurements at traps and other appropriate access points where radioactivity levels should be representative or bound those on the interior surfaces.

14.4.4.1.6.6 Active Rail Line

While the boundary of conceptual survey unit LSA-11-02 (Figure 14-14) will encompass the active rail line, the active rail line will not be surveyed or sampled as justified below; surveys and sampling will be limited to the 20 foot section of ground between the southern edge of the active rail line and the southern boundary of this survey unit. The random sampling locations that fall on the active rail line during survey design will be relocated to the southern edge of the railroad bed.

This approach for survey and sampling in this newly-defined survey unit is reasonable given the history, nature, and safety considerations of the active rail line. First, the rail has been in existence prior to the initial construction of the facility, thus the potential for subsurface contamination is very small. Second, the use of the rail line over time has served to fracture and compact the rail bed, resulting in a relatively impermeable surface. This compaction results in drainage of any precipitation (and radioactivity that may have been deposited by air deposition) to the edges of the rail bed. This is the area where the relocated samples will be collected, and

thus these samples should actually be biased to the location of the greatest potential for contamination.

14.4.4.2 Survey Instrumentation

The data quality objectives process includes the selection of instrumentation appropriate for the type of measurement to be performed (i.e., total surface contamination measurement, scan or both), that are calibrated to respond to a radiation field under controlled circumstances; evaluated periodically for adequate performance to established quality standards; and sensitive enough to detect the radionuclide(s) of interest with a sufficient degree of confidence.

When possible, instrumentation selection will be made to identify the ROC at levels sufficiently below the DCGL. Detector selection will be based upon detection sensitivity, operating characteristics, and expected performance in the field. The instrumentation will, to the extent practicable, use data logging to automatically record measurements to minimize transcription errors. Commercially available portable and laboratory instruments and detectors will be used to perform the following basic survey measurements:

- Surface scanning;
- Direct surface contamination measurements;
- Gamma spectroscopy analysis of soil and other bulk materials;
- Alpha spectroscopy analysis of soil and other bulk materials; and,
- Liquid scintillation counting of soil and other bulk materials.

Specific implementing procedures control the issuance, use, and calibration of instrumentation. The instrumentation currently proposed for use in the FSS is listed in Table 14-14.

The specific DQOs for instruments are established early in the planning phase for FSS activities, implemented by standard operating procedures and executed in the survey plan. Further discussion of the DQOs for instruments is provided below.

14.4.4.2.1 Instrument Selection

The selection and proper use of appropriate instruments for both total surface contamination measurements and laboratory analyses is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The survey plan design must establish acceptable measurement techniques for scanning and direct measurements. The DQO process must include consideration as to the type of radiation, energy spectrum and spatial distribution of radioactivity as well as the

characteristics of the medium to be surveyed (e.g., painted, scabbled, and chemically decontaminated).

Radiation detection and measurement instrumentation will be selected based on the type and quantity of radiation to be measured. The target MDC for measurements obtained using field instruments will be 50 percent of the applicable $DCGL_W$. The target MDC for measurements obtained using laboratory instruments will be 10 percent of the applicable $DCGL_W$. Measurement results with associated MDC that exceed these values may be accepted as valid data after evaluation by health physics supervision. The evaluation will consider the actual MDC, the reported value for the measurement result, and the fraction of the DCGL identified in the sample.

Instrumentation other than those listed in Table 14-14, or alternate measurement techniques, may be utilized provided the acceptability of the alternate instruments or measurement techniques for use in the FSS will be justified in a technical basis evaluation document prior to use. This evaluation will include the following:

- Description of the conditions under which the method would be used;
- Description of the measurement method, instrumentation and criteria;
- Justification that the technique would provide the required sensitivity for the given survey unit classification; and,
- Demonstration that the instrument provides sufficient sensitivity for measurement.

14.4.4.2.2 Calibration And Maintenance

Instruments and detectors will be calibrated for the radiation types and energies of interest or to a conservative energy source. Calibration will be performed on-site using HDP procedures or off-site by an approved vendor. Instrument calibrations will be documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels will also be attached to all portable survey instruments. Prior to using any survey instrument, the current calibration will be verified and all operational checks will be performed.

Radioactive sources used for calibration will be traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector is used, suitable NIST-traceable sources will be used for calibration, and the software set up appropriately for the desired geometry.

14.4.4.2.3 Response Checks

Prior to use on-site, all project instrument calibrations will be verified and initial response data collected. These initial measurements will be used to establish performance standards (response ranges) in which the instruments will be tested against on a daily basis when in use. An acceptable response for field instrumentation is an instrument reading within ±20 percent of the established check source value. Laboratory instrumentation standards will be within ±3-sigma as documented on a control chart.

The DQO process determines the frequency of response checks, typically before issue and after an instrument has been used (typically at the end of the work day but in some cases this may be performed during an established break in activity, e.g., lunch). This additional check will expedite the identification of a potential problem before continued use in the field. Instrumentation will be response checked in accordance with HDP Site procedures. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that FSS data are discarded, replacement data will be collected at the original locations.

14.4.4.2.4 Total Weighted Efficiency

Because a mixture of contaminants is potentially present as residual contamination on building surfaces, a total weighted efficiency may be calculated based on the guidance in Section 8.4 and Section 10.1 of Reference 14-10. The total weighted efficiency would account for the various energies of alpha and beta emissions from the primary contaminants and short lived progeny as well as account for surface conditions using the guidance provided in ISO 7503-1, Evaluation of surface contamination -- Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters (Reference 14-11).

A weighted efficiency is calculated for each contaminant, including progeny, as the product of the 2π instrument efficiency for detection, surface (source) efficiency, radiation yield, and radioactivity fraction. The instrument efficiency is determined using a NIST-traceable calibration source with a radiation emission average energy less than or equal to that of the average energy of the ROC(s). The surface efficiency selection is based on the contaminant's alpha or beta energy, not that of the calibration source. The yield is the radioactive branching ratio. The radioactivity fraction is calculated using the methodology presented in Section 14.1. The total weighted efficiency is then simply the sum of the weighted efficiency from each contaminant. An example calculation is provided in Table 14-19 based on nominal instrument efficiencies for a Ludlum Model 43-68 gas flow proportional counter. In the example, the total weighted efficiency was calculated to be 0.16 for alpha-plus-beta radioactivity when measuring Uranium enriched in U-235 to 4.5 weight percent (0.10 and 0.06 for alpha and beta radioactivity,

respectively). Details regarding the radioactivity fractions used in this example are provided in Section 14.1.

The gross $DCGL_W$, in counts per minute per 100 square centimeters (cpm/100 cm^2), is the conversion of the gross radioactivity $DCGL_W$, in dpm/100 cm^2, by using the total weighted efficiency and applying the probe correction factor. FSS measurements will be compared to this value.

14.4.4.2.5 Static MDC for Building and Structural Surfaces

For static (direct) surface measurements, with conventional detectors, such as those listed in Table 14-14, the MDC is calculated as follows:

(14-29)

$$\text{MDC}\,(\text{dpm}/100\text{cm}^2) = \frac{3 + 3.29\sqrt{(R_b)(T_s)\left(1 + \frac{T_s}{T_b}\right)}}{\frac{A}{100cm^2}(\varepsilon_t)(T_s)}$$

where:

A	=	probe area (cm^2)
ε_t	=	total weighted efficiency (c/d; 4π), is the product of the individual radionuclide weighted efficiencies. The weighted efficiency is the product of the 2π instrument efficiency (ε_i), surface (source) efficiency (ε_s), radiation yield, and radioactivity fraction.
R_b	=	background count rate (cpm)
T_b	=	background count time (minutes)
T_s	=	sample or measurement count time (minutes)
3	=	derived constant based on Type I and Type II errors of 0.05 (NUREG-1507, Sect 3.1)
3.29	=	derived constant based on the 95 percent confidence level (NUREG-1507, Sect 3.1)
100	=	conversion factor (detector area (cm^2) to 100 cm^2)

The static MDC was estimated for a detector having an area of 126 cm^2 and a nominal background count rate of 300 cpm. The total weighted efficiency was calculated to be 0.16 based on nominal instrument efficiencies for 4.5 percent U-235 enrichment. The estimated static MDC for building surfaces is calculated to be 415 dpm/100 cm^2.

(14-30)

$$MDC = \frac{3 + 3.29 \times \sqrt{300 \times 1 \times \left(1 + \frac{1}{1}\right)}}{0.16 \times 1 \times \left(\frac{126}{100}\right)} = 415\,\text{dpm/100cm}^2$$

14.4.4.2.6 HPGe Spectrometer Analysis

Gamma spectrometer systems will be calibrated to the various soil sample geometries that will be analyzed such as the 250 or 500 mL Marinelli container for soil, a 1 L Marinelli for water, and Petri dishes for small samples such as concrete dust and scale. The systems will be calibrated using NIST-traceable mixed gamma sources or intrinsic calibration routines. The counting system will have software-calculated MDC values that are less than or equal to the $DCGL_W$ for the analyte, with a range of 10-50 percent of the $DCGL_W$ being preferable. The MDCs as provided by the operational software will be similar to the following equation:

(14-31)

$$MDC\,(pCi/g) = \frac{3 + 4.65\sqrt{B}}{(K)(W)(t)}$$

where:

B	=	Number of background counts during the count interval *t*;
K	=	Proportionality constant that relates the detector response to the radioactivity level in a sample for a given set of measurement conditions;
W	=	Sample weight (dry grams); and,
t	=	Count time (minutes).
3	=	derived constant based on Type I and Type II errors of 0.05 (NUREG-1507, Sect 3.1)
4.65	=	derived constant based on the 95 percent confidence level (NUREG-1507, Sect 3.1)

The effect of analyzing a sample for multiple radionuclides should also be considered in meeting the sensitivity requirements and goals stated above. To ensure adequate sensitivity, Equation 4-3 of Reference 14-6, the unity rule equation, will be used. For this calculation, the MDC will be divided by the $DCGL_W$ for each radionuclide and the results for each radionuclide summed to calculate the SOF. In the case of U-234, the radionuclide activity concentration is estimated using the U-238:U-235 ratio rather than being inferred by the measurement of one radionuclide. To ensure adequate sensitivity, an estimated U-234 MDC will be calculated in order to include a term in the SOF calculation. The calculated SOF must be less than or equal to one, with a preferred value between 0.1 and 0.5.

The HRGS will be energy calibrated to properly identify the energy of detected gamma total absorption peaks (TAP). Each detector is calibrated using a NIST-traceable multi-energy gamma source. The specific source will include several gamma energies that span the range expected at the site. Coincident summing issues normally associated with the use of Eu-152 will not affect the energy calibration and will be overcome by keeping a minimum distance between the source and detector (i.e., at least a few inches away). The energy calibration will be performed separately for each detector.

ISOCS (*In Situ* Object Counting System) software is used to determine the efficiency for each measurement configuration. ISOCS allows a specific configuration to be modeled to determine the measurement efficiency for the configuration. Detailed parameters are specified using ISOCS software including detector position/distance relative to container, dimensions of material volume within container, specification of the material type and density in the container, and shielding materials between detector and waste (i.e., container and assay trailer walls).

Calibration curves for each container, geometry, and material are generated using the ISOCS software for each individual detector and for the summed detector response. The summed detector response (i.e., summation of all six detector spectrums after energy calibration shifts have been performed to align individual detector responses) provides better sensitivity of the overall or average radioactivity measured in the container.

When ISOCS is used in conjunction with NDA-2000, the range of container types, material types, and densities to be encountered may have calibration curves generated and linked to each combination in advance of measurements. This ensures an efficient day-to-day operation by allowing the system operator the ability to select the applicable container, geometry and material before starting each measurement to ensure the appropriate calibration curve is applied to the result.

All soil measurements will report radioactivity concentrations for U-235 and U-238 (inferred from the Pa-234 or Th-234 TAP). Thorium-232 radioactivity concentration (inferred from the Ac-228 TAP) will be reported for use in areas distinguishable from background or for sample results greater than the BTV of 1.7 pCi/g (see Appendix A of Reference 14-3). Results for the insignificant radionuclide Am-241 will also be reviewed to allow the identification of anomalous

results (see Section 14.1). Finally, gamma spectroscopy results for each sample will be reviewed for other gamma-emitting radionuclides present.

14.4.4.2.7 Scan MDCs

As described in MARSSIM, it is necessary to determine the scan sensitivity for field instrumentation utilized during the FSS. This will determine the effectiveness of the surface scans in the ability to determine whether an area meets the criteria for release and will also be a factor in determining the number of samples and measurements that will be required to demonstrate compliance.

Scan speeds will be established to the maximum extent practical to detect contamination at or below the release criteria for both open land soil and building and structural surface contamination surveys. In order to determine the scan sensitivity, it is first necessary to determine the Minimum Detectable Count Rate (MDCR) above background for the field instrumentation. This will be determined using the guidance in MARSSIM and NUREG-1507 with the following equations:

$$s_i = d'\sqrt{b_i} \tag{14-32}$$

$$MDCR = s_i\left(\frac{60}{i}\right) \tag{14-33}$$

where:

s_i	=	Minimum detectable source counts per counting interval;
d'	=	Index of sensitivity (Table 6.5 of MARSSIM);
b_i	=	Background counts per observation interval; and,
i	=	Observation interval (seconds).

For the purposes of the FSS, the index of sensitivity (d') value will be set to 1.38 as recommended in MARSSIM for a true positive proportion of 95 percent and a false positive proportion of 60 percent.

The observation interval, i, will be considered to be the amount of time for the detector to pass completely over the field of view or an area of concern such as a defined hot spot with a specified diameter. For building and structural surfaces, the observation interval is typically 1 second for scanning speeds that are 1 detector width per second. For open land areas for which

the detector has a wide view, this can be determined using MicroShield® modeling to assess the field of view of the instrument. Using this modeling program, it is estimated that at a scanning distance of about 6 inches, a field instrument such as the 44-10 NaI gamma scintillator would have a 75 percent response from a 5-foot diameter lens of contaminated soil in relation to an infinite slab source. This equates to an observation interval of approximately 5 seconds for a scanning speed of 1 foot per second (0.3048 meters per second). For conservatism, an observation interval of 1 second is typically used. Once the MDCR is determined for the field instrumentation, the scanning MDC will be calculated for building and structural surfaces and open land areas.

14.4.4.2.8 Building And Structural Surface Scan MDCs

Following the guidance in MARSSIM and NUREG-1507, the scan MDC for building and structural surfaces will be determined by using the following equation:

$$Scan\ MDC\ (dpm/100cm^2) = \frac{MDCR}{\sqrt{p}\,(\varepsilon_t)\left(\frac{A}{100\ cm^2}\right)} \qquad \textbf{(14-34)}$$

where:

$MDCR$	=	Minimum detectable count rate (cpm), calculated using Equation 14-29;
p	=	Surveyor efficiency (50 percent recommended by MARSSIM);
ε_t	=	Total efficiency; and,
A	=	Probe area (cm^2)
100	=	conversion factor (detector area (cm^2) to 100 cm^2)

For detectors with a large probe area, e.g., Ludlum 43-37, the term A / 100 cm^2 in Equation 14-34 may be omitted per technical discussion provided in Reference 14-10, Section 9.3.3.2 for Equation 9.14.

In the case of the scan measurements, the observation interval will be the time the probe is over a specific source of radioactivity. This time depends upon the scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. As previously mentioned, the scan speed is typically one probe width per second so the observation interval classically is and will be defined as 1 second.

The scan MDC was estimated for a 126 cm^2 gas proportional detector with a thin Mylar window (0.8 mg/cm^2). The surveyor efficiency (p) will be 0.5, as recommended by MARSSIM and NUREG-1507. The probe area is 126 cm^2 with a nominal background count rate of 300 cpm for poured concrete. The total weighted efficiency was estimated to be 0.16 based on nominal instrument efficiencies for 4.5 percent U-235 enrichment. The estimated scan MDC for building and structural surfaces is calculated to be 1,299 dpm/100 cm^2 and is illustrated below.

(14-35)

$$Scan\ MDC = \frac{1.38 \times \sqrt{300 \times \frac{1}{60}} \times \frac{60}{1}}{\sqrt{0.5} \times 0.16 \times \left(\frac{126}{100}\right)} = 1{,}299 \text{ dpm/100 cm}^2$$

14.4.4.2.9 Open Land Area Gamma Scan MDCs

Scan MDCs for various contaminants are listed in Table 6.4 of NUREG-1507. The radionuclides (contaminants) that will be measured include total Uranium, Am-241 and Th-232 + C. Note that while Am-241 was considered an insignificant radionuclide in Section 14.1, open land area gamma scans are likely to identify Am-241 if it is unexpectedly present. The calculation of the total Uranium scan MDCs is discussed in the sections below. Table 6.4 of NUREG-1507 lists scan MDCs of 1.8 pCi/g for Th-232 and 31.5 pCi/g for Am-241.

The scan MDC value (in pCi/g) for open land surface scans can be developed following the guidance in Section 6.8.2 of Reference 14-7 and Section 9.3.5 of Reference 14-10. This section of the DP follows the methodology in Reference 14-3 of postulating an elevated area, modeling the exposure rate using MicroShield® software, and then determining a scan MDC using manufacturer reported conversion factors for exposure rates to count rates. A scan MDC is calculated for each Uranium isotope and then the radioactivity fractions (provided in Table 14-5) are used to calculate a total Uranium scan MDC for a particular U-235 enrichment using Equation 9.15 of Reference 14-10. An example calculation is discussed below for a 2 in by 2 in NaI scintillation detector. Note that the calculations were only performed for the Surface CSM as it was the most limiting case.

a. Calculation of $MDCR_{surveyor}$

The $MDCR_{surveyor}$ for the detector was calculated using Equation 14-31, then dividing by the square root of the surveyor efficiency, using the following inputs:

- Background count rate of 10,000 cpm;
- Observation interval of 1 second;

- Index of sensitivity (d′) of 1.38; and,
- Surveyor efficiency of 0.5 for manually recorded data; for data obtained using GPS and subsequently post-processed using GIS software, the surveyor efficiency is not applicable and the MDC values are reduced by approximately 29 percent.

The $MDCR_{surveyor}$ was calculated to be 1,512 cpm and is illustrated below:

(14-36)

$$MDCR_{surveyor} = \frac{1.38\sqrt{10{,}000 \times \frac{1}{60}} \times \left(\frac{60}{1}\right)}{\sqrt{0.5}} = 1{,}512\ cpm$$

b. MicroShield® Modeling

A model of a postulated small elevated area was created in MicroShield® software v6.21. The model was setup with the following inputs and options consistent with the information provided on Page 6-21 of NUREG-1507:

- Cylinder Volume – End Shields;
- Height of 15 cm and radius of 28 cm;
- Dose Point #1 at x=0 cm, y=25 cm, and z=0 cm;
- Source material of concrete with a density of 1.6 grams per cubic centimeter (g/cm3);
- Air gap with density of 0.00122 g/cm3;
- All source activities equal to 8E-6 microCuries per cubic centimeter (µCi/cm3) per Equation 6-19 of Reference 14-7, which is equivalent to 5 pCi/g; and,
- Source input grouping method of standard indices.

Three models were created for the individual Uranium radionuclides of U-234, U-235, and U-238 and associated short-lived progeny, as shown below:

- U-234 only;
- U-235 and Th-231; and,
- U-238, Th-234 and Protactinium-234m (Pa-234m)

Ignoring gamma energies less than or equal to 15 keV, the total exposure rate at Dose Point #1 was 9.57E-5, 3.12E-1, and 3.39E-2 microRoentgen per hour (µR/h) for U-234, U-235 (with progeny), and U-238 (with progeny). Additionally, MicroShield® software provided the exposure rate for a number of gamma energies associated with each input source term.

c. Calculation of the MDER

The MicroShield® software results were independently tabulated by grouped gamma energies and exposure rates for each Uranium isotope. Table 6.3 of NUREG-1507 provides normalized detector count rate versus exposure rate calculations based on the manufacturer's detector response to Cs-137. The exposure rate for each gamma energy group was then multiplied by the count rate versus exposure rate to determine the weighted count rate versus exposure rate for each energy and the results were summed. The minimum detectable exposure rate (MDER) was then calculated by dividing the $MDCR_{surveyor}$ by the total weighted count rate versus exposure rate per Equation 6-21 of NUREG-1507. The results are presented in Table 14-20. The MDER calculations are shown below.

$$MDER_{U-234} = \frac{1{,}512\ cpm}{10{,}699\ cpm\ per\ \mu R/h} = 0.14\ \mu R/h \quad \textbf{(14-37)}$$

$$MDER_{U-235} = \frac{1{,}512\ cpm}{4{,}991\ cpm\ per\ \mu R/h} = 0.30\ \mu R/h \quad \textbf{(14-38)}$$

$$MDER_{U-238} = \frac{1{,}512\ cpm}{3{,}554\ cpm\ per\ \mu R/h} = 0.43\ \mu R/h \quad \textbf{(14-39)}$$

d. Calculation Individual Scan MDCs

The scan MDC for each Uranium isotope is calculated using Equation 6-22 of NUREG-1507 using the results provided above. The results are shown below. Note that the value of 5 pCi/g equates to the modeled source concentration.

(14-40)

$$Scan\ MDC_{U-234} = (5\ pCi/g) \times \frac{0.14\ \mu R/h}{9.57E\text{-}5\ \mu R/h} = 7{,}383\ pCi/g$$

(14-41)

$$Scan\ MDC_{U-235} = (5\ pCi/g) \times \frac{0.30\ \mu R/h}{3.12E\text{-}1\ \mu R/h} = 4.9\ pCi/g$$

(14-42)

$$Scan\ MDC_{U-238} = (5\ pCi/g) \times \frac{0.43\ \mu R/h}{3.39E\text{-}2\ \mu R/h} = 62.8\ pCi/g$$

e. Calculation of Total Uranium Scan MDC

After establishing the individual scan MDCs, the total Uranium scan MDC can be calculated using the relative fractions of the individual Uranium isotopes using Equation 9.15 of Reference 14-10. Using the radioactivity fractions (provided in Table 14-5) for 20 percent U-235 enrichment, the total Uranium scan MDC is calculated below. Note that the actual calculation of 99.0 pCi/g shown below did not use any rounded values during the series of calculations and thus the equation shown below is for illustration only.

(14-43)

$$Scan\ MDC_{Total\ Uranium} = \frac{1}{\frac{0.9251}{7{,}383\ pCi/g} + \frac{0.0462}{4.9\ pCi/g} + \frac{0.0287}{62.8\ pCi/g}} = 99.0\ pCi/g$$

To demonstrate an example of the methodology presented in this section, the total Uranium scan MDC for a wide range of U-235 enrichments was calculated and compared to the total Uranium $DCGL_W$. This analysis was completed using a 2 in by 2 in NaI scintillation detector. The results are illustrated in Figure 14-18, Figure 14-19 and Figure 14-20 for the Plant Soil SEA, Tc-99 SEA, and Burial Pit SEA, respectively. Note that the Surface stratum $DCGL_W$ results correspond to Figure 14-4, Figure 14-5 and Figure 14-6 for the listed site areas.

The calculated total Uranium scan MDCs are generally consistent with those presented in Table 6.4 of NUREG-1507 in that they increase with U-235 enrichment. For high enrichments, the calculated total Uranium scan MDCs are greater than the scan MDC values presented in Table 6.4 of NUREG-1507.

From review of the figures, it can be seen that the total Uranium scan MDC exceeds the total Uranium $DCGL_W$. The implication is that if a total Uranium scan MDC is applied that exceeds the total Uranium $DCGL_W$, an analysis will be performed to determine if the instrumentation has adequate sensitivity to identify elevated areas bounded within the systematic sample locations. If the sensitivity is not adequate, an increase in the same size will be required.

14.4.4.2.10 Mapping Of Scan Data

The scan MDC for open land areas may be reduced further by using the field instrumentation coupled with a GPS unit by enabling the scan data to be logged, downloaded, and mapped. By logging and mapping the data, it enables the scan data to be reviewed in its entirety as a data set in correlation with survey unit characteristics such as paved areas and surface soil vs. subsurface soil, etc. By being able to statistically review the data by color coding and adjusting ranges of data values, patterns and areas of concern can be identified more readily than during real time scanning by the survey technician. Additionally, by using the GPS system, it is more readily available to relocate specific areas for further investigation, survey, and sampling as necessary. This technology eliminates the need to account for the surveyor efficiency, thereby reducing the scan MDC by approximately 29 percent.

14.4.4.2.11 MDC Summary

The specific MDCs for the instruments and techniques for final status surveys discussed in this DP will be used unless site-specific conditions warrant re-evaluation prior to post-remediation and/or FSS activities. Table 14-14 provides typical MDC values for the anticipated instruments to be used for FSS activities.

14.4.4.3 Surveillance Following Final Status Surveys

Isolation and control measures will be implemented through approved HDP Site procedures and will remain in force throughout final status survey activities and until there is no risk of contamination from decommissioning or the survey area has been released from the license. In the event that isolation and control measures established for a given survey unit are compromised, evaluations will be performed and documented to confirm that no radioactive material was introduced into the area that would affect the results of the FSS.

These evaluations will be controlled and documented in accordance with approved HDP Site procedures.

14.4.4.3.1 Surveillance Of Buildings And Structures

Routine surveys of removable surface contamination will be performed on buildings or structures in which a FSS has been completed until the time the building or structure is released from the site license. These routine operational health physics surveys will be used to verify that the as-

left radiological conditions in the area have not changed. These routine surveys will typically include survey locations on the floor and lower walls, and areas of ingress, egress, and storage. Locations will be selected on a judgmental basis, based on technician experience and conditions present in the survey area at the time of the evaluation. Location choices will be designed to detect the migration of removable surface contamination from decommissioning activities taking place in adjacent areas and in nearby areas that could cause a potential change in conditions.

If the area is suspect following the routine surveillance survey, then corrective measures will be taken, up to and including, a repeat of the FSS for the affected survey unit(s). Additionally, for any area that has completed FSS activities, any soil, sediment, or equipment relocated to that area will require a demonstration that the material being introduced does not result in resident radioactivity that is in excess of any FSS release criteria.

If a building or structure has been released from the site license, then soil, sediment or equipment will be prohibited from being stored or relocated to that building or structure.

14.4.4.3.2 Surveillance Of Open Land Areas

If the area is suspect following the evaluation (e.g., surface water transport of potentially contaminated sediment, soil pile that was not present during FSS), an investigation survey will be performed to confirm the FSS surveys validity. This investigation survey will involve judgment sampling of the suspect areas. If the results of the investigation survey indicates that contamination is statistically different than the initial FSS results (>2 standard deviations from the mean), then the investigation survey will be increased to include a larger physical area than the initial investigation survey. If the final results of the investigation survey are statistically different than the FSS survey results, then a full FSS survey of the affected areas will be performed.

Additionally, for any area that has completed FSS activities, any soil, sediment, or equipment relocated to that area will require a demonstration that the material being introduced does not result in resident radioactivity that is statistically different than that identified in the FSS.

If an open land area has been released from the site license, then soil, sediment or equipment will be prohibited from being stored or relocated to that open land area.

14.4.5 FINAL STATUS SURVEY DATA ASSESSMENT

The Data Quality Assessment (DQA) process, being implemented at the HDP Site, is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process will include a review of the DQOs and survey plan design, will include a review of preliminary data, will use appropriate statistical testing when applicable

(statistical testing is not always required, e.g., when all sample or measurement results are less than the $DCGL_W$), will verify the assumptions of the statistical tests, and will draw conclusions from the data.

Once the FSS data are collected, the data for each survey unit will be assessed and evaluated to ensure that it is adequate to support the release of the survey unit. Simple assessment methods such as comparing the survey data mean result to the appropriate $DCGL_W$ will be performed first. The SOF will be calculated for soil data to ensure a value less than unity to demonstrate compliance with the TEDE criterion, since several radioisotopes are measured. The specific non-parametric statistical evaluations will then be applied to the final data set as necessary including the EMC test and the verification of the initial data set assumptions. Once the assessment and evaluation is complete, any conclusions will be made as to whether the survey unit actually meets the site release criteria or whether additional actions will be required.

14.4.5.1 Review Of DQOs And Survey Plan Design

Prior to evaluating the data collected from a survey unit against the release criterion, the data are first confirmed to have been acquired in accordance with all applicable procedures and QA/QC requirements.

The DQO outputs will be reviewed to ensure that they are still applicable. The data collection documentation will be reviewed for consistency with the DQOs, such as ensuring the appropriate number of measurements or samples were obtained at the correct locations and that they were analyzed with measurement systems with appropriate sensitivity. The checklists provided in Section 5 of NUREG-1507, or similar, will be used in the review. Any discrepancies between the data quality or the data collection process and the applicable requirements will be resolved and documented prior to proceeding with data analysis. Data assessment will be performed by trained personnel using approved site procedures.

14.4.5.2 Preliminary Data Review

The first step in the data review process is to convert all of the survey results to DCGL units. Basic statistical quantities are then calculated for the sample data set (e.g., mean, standard deviation, and median). An initial assessment of the sample and measurement results will be used to quickly determine whether the survey unit passes or fails the release criterion or whether one of the specified non-parametric statistical analyses must be performed. This initial assessment is summarized in the evaluation matrices as provided in Table 14-21 and Table 14-22 for the WRS and Sign tests, respectively.

Individual measurements and sample concentrations will be compared to DCGL levels for evidence of small areas of elevated radioactivity or results that are statistical outliers relative to the rest of the measurements. Graphical analyses of survey data that depict the spatial correlation of the measurements are especially useful for such assessments and will be used to

the extent practical. At a minimum, a graphical review should consist of a posting plot and a histogram. Additional data review methodologies may be used and are detailed in Section 8.2.2 of MARSSIM.

Interpreting the results from a survey is most straightforward when all measurements are higher or lower than the $DCGL_w$. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the $DCGL_W$.

The statistical evaluations that will be performed will test the null hypothesis (H_o) that the residual radioactivity within the survey unit exceeds the $DCGL_W$. There must be sufficient survey data at or below the $DCGL_W$ to statistically reject the null hypothesis and conclude the survey unit meets the site release criteria. These statistical analyses may be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or electronic spreadsheets and/or databases.

14.4.5.3 Wilcoxon Rank Sum Test

The WRS Test is a non-parametric statistical evaluation typically used when the ROC is present in the background. In addition, this test is valid only when "less than" measurement results do not exceed 40 percent of the data set. Note that the use of "less than" values will be avoided whenever practical. In order to apply the WRS Test, a reference background area was established and reference measurements or samples collected. For the site, the WRS Test will be applied to the soil surveys using the guidance in Section 8.4 of MARSSIM. The WRS Test will be conducted as described below:

1. Each survey unit measurement, x_i, will be listed, typically consisting of only the systematic sampling and measurement locations to avoid bias in the statistical evaluation. The SOF will be calculated as necessary.

2. The background reference area measurements will be adjusted by adding the $DCGL_W$ to each background reference area measurement, y_i. When applying the unity rule, each contaminant of concern that is present in the background will be divided by the corresponding contaminant specified $DCGL_W$. These fractions will be totaled and added to 1 for the application of the unity rule.

3. The number of adjusted background reference area measurements, m, and the number of survey unit measurements, n, will be summed to obtain the total number of measurements for the combined data set, N, ($N = m + n$).

4. Survey unit measurements and adjusted background reference measurements will be pooled and ranked in order of increasing value from 1 to *N*. If several measurements have the same value, they will be assigned the average rank for that group of measurements.

5. If there are t "less than" values, they are all given the average of the ranks from one (1) to t, which is equal to (t+1)/2. Also, if there is more than one detection limit, all results below the largest detection limit will be treated as "less than" values.

6. The ranks for the adjusted background reference area measurements will be summed to obtain the critical value, W_r.

7. The critical value, W_r, will then be compared to the critical values provided in Table I.4 of MARSSIM, or equivalent. If the critical value, W_r, is greater than the value in the reference table, the null hypothesis can be rejected and the survey unit meets the release criterion. If the critical value, W_r, is less than or equal to the value in the reference table, the survey unit fails to meet the criterion.

In the event that the WRS Test fails, the survey unit will be re-evaluated to determine whether additional remediation will be required or the FSS re-designed to collect more data (i.e., a higher frequency of measurements and samples).

14.4.5.4 Sign Test

The Sign Test is a non-parametric statistical evaluation typically used in situations when evaluating sample analyses where the ROC are not present in background, they are present at acceptably low fractions as compared to the $DCGL_W$, or for gross radioactivity measurements for structural surfaces. For the site, the Sign Test will be applied to the building and structural surface surveys using the guidance in Section 8.3 of MARSSIM. The Sign Test will be conducted as described below:

1. Each survey unit measurement, x_i, will be listed. This will consist of the gross radioactivity measurement for building surfaces corrected for material specific background contributions. This will consist only of the systematic sampling and measurement locations to avoid bias in the statistical evaluation.

2. Each measurement, x_i, will be subtracted from the $DCGL_W$.

3. Differences where the value is exactly zero will be discarded and the number of measurements, *n*, reduced by the number of such zero measurements.

4. The number of positive differences will then be totaled. Measurements that are less than the release criteria provide evidence that the survey unit meets the site release criterion. The resulting total will be the test statistic S^+, or critical value.

5. The critical value, S^+, will then be compared to the critical value as provided in Table I.3 of Reference 14-6, for the total number of measurements taken, n, and the corresponding decision error α, which will be set at 0.05. Provided the critical value, S^+, is greater than the value as given in the reference table, the null hypothesis can be rejected and the survey unit meets the release criterion. If the critical value, S^+, is less than or equal to the value, the survey unit fails to meet the release criterion.

In the event that the Sign Test fails, the survey unit will be re-evaluated to determine whether additional remediation will be required or the FSS re-designed to collect more data (i.e., a higher frequency of measurements and samples).

14.4.5.5 Excavation Depth Considerations On Data Assessment

When the DQO process is modified as described in Section 14.4.3, a minor modification to the data assessment is also required. When the SOF is calculated for each sample location, using Equation 14-9, the $DCGL_W$ used depends on the elevation that the sample was collected, i.e., Root stratum vs. Deep stratum. The calculated SOF value is then used in the WRS test as described in Section 14.4.5.

However, when making the final determination of the dose consequence of the survey unit and when applying the unity rule across multiple CSMs, the average SOF needs to be weighted. The weighted average SOF is calculated using the following equation:

(14-44)

$$Average\ SOF_{Weighted} = f_{RZ} \sum_{i=1}^{n} \left(\frac{\overline{C}_{i,RZ}}{D_{i,RZ}} \right) + f_{DZ} \sum_{i=1}^{n} \left(\frac{\overline{C}_{i,DZ}}{D_{i,DZ}} \right)$$

where:

f_{RZ}	=	Fraction of survey unit area at the Root stratum depth;
n	=	Number of measured ROCs;
$\overline{C}_{i,RZ}$	=	Average concentration of *i*th measured ROCs in the Root stratum layer;
$D_{i,RZ}$	=	Root stratum $DCGL_W$ for the *i*th measured ROCs;
f_{DZ}	=	Fraction of survey unit area at the Deep stratum depth;
$\overline{C}_{i,DZ}$	=	Average concentration of *i*th measured ROCs in the Deep stratum layer; and,
$D_{i,DZ}$	=	Excavation $DCGL_W$ for the *i*th measured ROC.

(Note that the sum of f_{RZ} and f_{DZ} will equal one.)

A reasonable estimation of the area fractions, f_{RZ} and f_{DZ}, can be made by independently dividing the number of systematic locations in each depth layer by the total number of systematic locations. For example, if there are 15 systematic sampling locations and 10 are located at the depth of the Root Zone, then f_{RZ} will be equal to 10 / 15 = 0.67. If 14 of the locations are at the depth of the Deep Zone, then f_{DZ} will be equal to 14 / 15 = 0.93.

14.4.5.6 Elevated Measurement Comparison Evaluation

The EMC will be applied to Class 1 survey units only when an elevated area is identified by surface scans and/or biased and systematic samples or measurements. The EMC provides assurance that areas of elevated radioactivity receive the proper attention and that any area having the potential for significant dose contribution is identified. Locations identified by surface scans or sample analyses which exceed the $DCGL_W$ are subject to additional surveys to determine compliance with the elevated measurement criteria. Based upon the size of the elevated measurement area, the corresponding AF will be determined from Table 14-11 and Table 14-12 for building and structural surfaces and soil, respectively, using linear or exponential interpolation as necessary.

The EMC will be applied by summing the contributing dose fractions of the survey unit through the unity equation. This will be performed by determining the fraction of dose contributed by

the average radioactivity across the survey unit and by adding the additional dose contribution from each individual elevated area following the guidance as provided in Section 8.5.1 and Section 8.5.2 of MARSSIM.

14.4.5.6.1 Average Radioactivity Fraction

The average radioactivity within the survey unit will be determined from the systematic sampling and measurement results, excluding all biased measurements and any measurements within an elevated area. This is to ensure the proper statistical testing of the survey data without skewing the results of the evaluation. Any samples taken within an elevated area, including systematic and biased samples used to evaluate the average radioactivity within the elevated area, will be excluded from the survey unit average. Additionally, biased sampling results less than the $DCGL_W$ will typically be excluded as these were not randomly selected; however, these measurements may be included, with caution.

(14-45)

$$f_{Avg} = \sum_{j=1}^{x} \frac{\delta_j}{DCGL_{w_j}}$$

where:

f_{Avg}	=	Dose contribution from the average survey unit radioactivity;
x	=	Number of measured contaminants;
δ_j	=	Survey unit average radioactivity (pCi/g) of contaminant *j*; and,
$DCGL_{w\,j}$	=	Derived Concentration Guideline Level of contaminant *j*.

14.4.5.6.2 Elevated Area Fraction

The additional dose fraction or contribution from each elevated area will be determined by calculating the average radioactivity within the elevated area, subtracting the average radioactivity of the survey unit, and then dividing by the corresponding $DCGL_{EMC}$ which is the product of the $DCGL_W$ and the AF that applies to the size of the elevated area. The average survey unit radioactivity is subtracted as the dose contribution is already accounted for based upon the average radioactivity contribution to the dose as calculated above. The additional dose contribution from the elevated area(s) is/are a result of any elevated radioactivity in excess of the survey unit average.

(14-46)

$$f_{EMC} = \sum_{j=1}^{x} \sum_{i=1}^{y} \frac{(\tau_{i,j} - \delta_j)}{AF_{i,j} \times DCGL_{w_j}}$$

where:

f_{EMC}	=	Dose contribution from elevated area(s);
x	=	Number of measured contaminants;
y	=	Number of elevated areas;
$\tau_{i,j}$	=	Average radioactivity of contaminant *j* in elevated area *i*;
δ_j	=	Survey unit average radioactivity for contaminant *j*;
$AF_{i,j}$	=	AF for contaminant *j* based upon the size of elevated area *i*; and,
$DCGL_{w\,j}$	=	Derived Concentration Guideline Level of contaminant *j*.

14.4.5.6.3 Sum-Of-Fractions

Once all the dose contributions are determined, the SOF is applied using the results of Equation 14-34 and Equation 14-35 as follows.

$$f_{Avg} + f_{EMC} + f_{GROUNDWATER} \leq 1 \quad \textbf{(14-47)}$$

Provided the SOF is less than or equal to unity (1), the survey unit will pass the EMC. If the test fails, additional remediation will be performed as necessary to address the elevated areas. If the other statistical tests pass with the exception of the EMC test, remediation may be performed within these isolated area(s) only and the immediate area(s) re-surveyed without having to resurvey the entire survey unit as discussed in Section 8.5.3 of MARSSIM.

14.4.5.7 Data Conclusions

The results of the statistical testing, including the application of the EMC, allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release criterion through the rejection of the null hypothesis. The data provide statistically significant evidence that the level of residual radioactivity within the survey unit does not exceed the release criteria. The decision to release the survey unit will then be made with sufficient confidence and without any further analyses.

The second conclusion that can be made is that the survey unit fails to meet the release criteria. The data may not be conclusive in showing that the residual radioactivity is less than the release criteria. As a result, the data will be analyzed further to determine the reason for failure. Potential reasons may include:

- The average residual radioactivity exceeds the $DCGL_W$;
- The average residual radioactivity is less than the $DCGL_W$; however, the survey unit fails the EMC test;
- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release, (i.e., an adequate number of measurements was not performed); or,
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

"Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. The power of the statistical test is a function of the number of measurements made and the standard deviation of the measurement data. Quantitatively, the power is $1 - \beta$, where β is the Type II error rate (the probability of accepting the null hypothesis when it is actually false). A retrospective power analysis can be used in the event that a survey unit is found not to meet the release criterion to determine if this is indeed due to excess residual radioactivity or if it is due to an inadequate sample size. A retrospective power analysis may be performed using the methods as described in Section I.9 and Section I.10 of MARSSIM.

If the retrospective power analysis indicates insufficient power, then an assessment will be performed to determine whether the observed median concentration and/or observed standard deviation are significantly different from the estimated values used during the DQO process. The assessment may identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions may include failing the unit and starting the DQO process over, remediating some or all of the survey unit and starting the DQO process over and adjusting the LBGR to increase sample size. For example, the assessment determines that the median residual concentration in the survey unit exceeds the $DCGL_W$ or is higher than was estimated and planned for during the DQO process. A likely cause of action might be to fail the unit or remediate and resurvey using a new sample design. As another example, the assessment determines that additional samples are necessary to provide sufficient power. One course of action might be to determine the number of additional samples and collect them at random locations. Note, this method may increase the Type I error, therefore agreement with the regulator will be necessary prior to implementation. Another action would be to resample the survey unit with a new (and appropriate) number of samples and/or a new survey design.

There may be cases where the decision was made during the DQO process by the planning team to accept lower power. For instance, during the DQO process the calculated relative shift was found to be less than 1. The planning team adjusts the LBGR, evaluates the impact on power and accepts the lower power. In this case, the DQA process would require the planning team to compare the prospective power analysis with the retrospective power analysis and determine whether the lower power is still justified and the DQOs satisfied.

14.5 POST- REMEDIATION GROUNDWATER SAMPLING AND ANALYSIS

This section describes groundwater sampling that will be conducted during and following the completion of soil remediation and represents the FSS for groundwater. The goal of the sampling is to identify adverse affects on water quality as a result of excavation, and to verify the absence of any significant amount of residual radioactivity in the groundwater that could be a part of a credible exposure scenario. It is expected that remediation will result in a reduction in the radioactivity levels in water within the overburden soil, and will not have an adverse impact on groundwater from a dose perspective.

Monitoring well data was collected between 2004 and 2008 indicate that radioactivity in water is primarily limited to the overburden soil in source areas including the Burial Pits, evaporation ponds, and soil beneath buildings. Radioactivity in the bedrock groundwater underlying the site is generally within the range of background with the potential exception of Tc-99 concentrations at very low, insignificant levels (Reference 14-4).

The post-remediation sampling and analysis strategy will focus on monitoring vertical seepage in the clayey overburden and lateral migration in the sandy/gravelly deposits at the base of the clayey overburden. The approach is based on the site-specific hydrogeology, the pre-remediation groundwater contaminant distribution, and potential radionuclide transport pathway data as detailed in Chapter 3.0.

14.5.1 Locations Of Monitoring Wells

Following remediation, ground water will be monitored to assure that removal of the source term in the soil and burial areas is effective in protecting groundwater sources in the Sand/Gravel, Jefferson City-Cotter, and Roubidoux Hydrostratigraphic Unit (HSUs). This post-remediation monitoring will be performed at wells identified in Table 14-23 and Figure 14-21.

In general, the monitoring wells shown on Figure 14-21 are located down gradient (i.e., southeast) of the related source area with a goal of intercepting contamination released from the source areas. The primary post-remediation well network is composed of 12 monitoring wells screened in the Sand/Gravel HSU. Analysis will be performed quarterly on each of the wells to evaluate trends in sample results as discussed in Chapter 11.0 of this DP.

Primary wells GW-DD, GW-EE, GW-FF, and GW-GG are positioned down gradient (southeast) of the burial pits to assess ground water quality following removal of contaminated soil/materials from this area. A positive detection (defined as a concentration exceeding the MDA + Error [2σ] for Tc-99 or total uranium exceeding its BTV for groundwater) from a primary well sample will indicate potential down gradient migration. If this occurs, secondary wells GW-BB, GW-II, and GW-W, which are positioned further down gradient of the burial pits, will be monitored.

Primary wells GW-D, GW-S, GW-T, and GW-Z are positioned down gradient (southeast) of the process buildings to assess groundwater quality following building demolition and removal of contaminated soil from this area. A positive detection from a primary well sample will indicate potential down gradient migration. If this occurs, secondary wells GW-V and GW-W, which are positioned further down gradient of the process buildings, will be monitored

Primary wells GW-CC, GW-U, and GW-X are positioned down gradient (southeast) of the evaporation ponds and former leach field to assess groundwater quality following removal of contaminated soil from these areas. A positive detection from a primary well sample will indicate potential down gradient migration. If this occurs, secondary wells GW-HH, GW-Y, and GW-V, which are positioned further down gradient of the evaporation ponds and former leach field, will be monitored.

Primary well GW-AA is positioned down gradient (southeast) of the red room roof burial area and cistern/burn pit to assess groundwater quality following removal of contaminated soil and materials from these areas. A positive detection from a primary well sample will indicate potential down gradient migration. A positive detection in this well will prompt the installation of a secondary well further down gradient.

Three new monitoring wells (BR-13-JC, BR-14-JC, and BR-15-JC) will be installed in the Jefferson City-Cotter HSU down gradient of burial pit and Tc-99 source areas. The exact location of the sources areas will be determined during remediation. The wells are anticipated to be placed at locations to the south and east of the burial pit and ring storage area. These wells will be located closer to the central tract than currently monitored wells and are located in areas that, if contaminant migration occurs, will identify the degradation of the water within the post remediation monitoring timeframe.

Post remediation monitoring of the Jefferson City-Cotter HSU in the vicinity of the former process building will be through the installation of three new monitoring wells (BR-16-JC, BR-17-JC, and BR-18-JC) within the source and down gradient of the areas beneath the former process buildings where the highest levels of contamination were removed. These wells will be used to evaluate the potential for contaminant migration from the overburden into the shallow bedrock.

Post remediation monitoring of the Jefferson City-Cotter HSU in the vicinity of the former evaporation ponds will be through the installation of a new monitoring well (BR-19-JC) at a location down gradient of the primary (deep) evaporation pond.

Post remediation monitoring of the Roubidoux HSU will be conducted using the current sentry wells designated as BR-03-RB, BR-04-RB, BR-08-RB, and BR-10-RB.

Post-remediation monitoring wells will be sampled quarterly after the completion of remediation until license termination. The data collected will be used to confirm that the sum of the annual

dose from groundwater for all the radionuclides does not exceed the EPA Maximum Contaminant Level (MCL) of 4 millirem per year. Separately, the sum of the dose from all residual sources remaining after remediation, including soil and groundwater pathways, will be confirmed to result in an annual dose that does not exceed 25 millirem/year.

14.5.2 Frequency Of Post-Remediation Monitoring

Following completion of soil remediation, groundwater monitoring will begin at a quarterly frequency. An evaluation of the groundwater results will be performed to determine if the concentrations are stable, or are showing an increasing or decreasing trend. Monitoring will be discontinued when it can be determined that the radioactivity concentrations do not pose an unacceptable potential for dose utilizing the methods described in Chapter 5.0.

14.5.3 Sampling Method

Groundwater sampling will be conducted following site procedures using a low-flow technique that provides representative samples while reducing investigation derived waste. For each well, unfiltered and filtered groundwater samples will be collected and analyzed; turbidity will be measured in the field on unfiltered groundwater. Samples will be analyzed for gross alpha, gross beta, isotopic Uranium, and Tc-99. Comparison of radionuclide activities in paired filtered and unfiltered samples will be used to determine whether radionuclide migration, if any, is occurring through clay/colloidal transport.

14.5.4 Groundwater Dose Assessment

If there are positive results, above background, from samples collected in the sand/gravel or bedrock aquifers, then the corresponding dose will be calculated using the Dose to Source Ratios listed in DP Chapter 5.0, Table 5-14. Initially, the contribution to dose from the groundwater sample showing the highest individual aquifer sample result will be added to the dose attributable to the survey unit with the highest dose (calculated in accordance with Section 14.4.5.6.1) to ensure that the total dose remains below 25 mrem/yr. This contribution to dose is expected to be insignificant when compared to soil, however if this initial approach is determined to be unduly conservative, then additional hydrogeological investigations may be performed. The investigations will be used to determine the extent of the groundwater contamination and a more realistic estimate of the groundwater source term for the purpose of performing the dose estimate as opposed to applying an individual maximum dose value.

14.6 FINAL STATUS SURVEY REPORTING

Documentation of the FSS will transpire in two types of reports and will be consistent with Section 8.6 of NUREG-1575. An FSS Survey Unit Release Record will be prepared to provide a complete record of the as-left radiological status of an individual survey unit, relative to the specified release criteria. Survey Unit Release Records may be made available to the NRC for inspection. An FSS Final Report, which is a written report that is provided to the NRC for its review, will be prepared to provide a summary of the survey results and the overall conclusions which demonstrate that the site, or portions of the site, meets the radiological criteria for unrestricted use.

14.6.1 FSS Survey Unit Release Records

An FSS Survey Unit Release Record will be prepared upon completion of the final status survey for a specific survey unit. Sufficient data and information will be provided in the release record to enable an independent re-creation and evaluation at some future time. The format and content of the FSS Survey Unit Release Record will be as follows:

- Survey Unit Description, including unit size, descriptive maps, plots or photographs, including reference coordinates and historic changes in description;
- Classification Basis, including significant historical site assessment and characterization data used to establish the final classification as well as a statement on the impact groundwater had on the final classification;
- Data Quality Objectives stating the primary objective of the survey, and a brief description of the DQO process;
- Survey Design describing the design process, including methods used to determine the number of samples or measurements required based on statistical design, number of biased or judgmental samples or measurements required, method of sample or measurement locating, and a table providing a synopsis of the survey design;
- Survey Implementation describing survey methods and instrumentation used, accessibility restrictions to sample or measurement location, number of actual samples or measurements taken, documentation activities, Quality Control samples or measurements, and scan data collected in tabular format;
- Survey Results including types of analyses performed, types of statistical tests performed, statement of pass or failure of the statistical test(s);

- Quality Control results to include discussion of split samples and/or QC replicate measurements;
- Investigations and Results;
- Remediation activities, both historic and resulting from the final status survey;
- Changes from the FSS survey design including, but not limited to field changes, and reasons for survey unit reclassification (and the reasons for the initial misclassification);
- Data Quality Assessment;
- Anomalies occurring during the survey or in the sample results;
- Conclusion as to whether or not the survey unit satisfied the specified release criteria, a discussion of ALARA evaluations performed, and whether or not sufficient power was achieved; and,
- Attachments and enclosures to include supporting maps, diagrams, and sample statistical data.

14.6.2 FSS Final Reports

The ultimate product of the Data Life Cycle is an FSS Final Report which will be, to the extent practical, a stand-alone document with minimal information incorporated by reference. To facilitate the data management process, as well as overall project management, FSS Final Reports will usually incorporate multiple FSS Survey Unit Release Records. To minimize the incorporation of redundant historical assessment and other FSS program information, and to facilitate potential partial site releases from the current license, FSS Final Reports may be prepared and submitted in a phased approach. The format and content of the FSS Final Report is as follows:

- Introduction, including a discussion on the phased approach for submittals;
- FSS Program Overview to include sub-sections on survey planning, survey design, survey implementation, survey data assessment, and Quality Assurance and Quality Control measures;
- Site Information to include sub-sections on site description, survey area/unit description (specific to current phase submittal), summary of historical radiological data, conditions at the time of survey, identification of potential contaminants, and radiological release criteria;

- Final Status Survey Protocol to include sub-sections on Data Quality Objectives, survey unit designation and classification, background determination, FSS plans, survey design, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), survey methodology, quality control surveys, and a discussion of any changes that were made in the FSS from what was proposed in this DP;

- Survey Findings to include sub-sections on survey data conversion, survey data verification and validation, evaluation of number of sample/measurement locations, and comparison of findings with the appropriate DCGL;

- Appendix A: FSS Program and Implementing Procedures (initial phased submittal – subsequent submittals contain only revisions or additions to program and/or implementing procedures); and,

- Appendix B: FSS Technical Basis Documents (initial phased submittal – subsequent submittals contain only revisions or additions to FSS technical basis documents).

14.7 REFERENCES FOR CHAPTER 14.0

14-1 Code of Federal Regulations, Title 10, Part 20.1402, "Standards for Protection Against Radiation−Radiological Criteria for Unrestricted Use."

14-2 Westinghouse Electric Company Document No. DO-08-005, "Historical Site Assessment."

14-3 Westinghouse Electric Company Document No. DO-08-003, "Hematite Radiological Characterization Report."

14-4 Westinghouse Electric Company Document No. DO-08-008, "Derivation of Surrogates and Scaling Factors for Hard-To-Detect Radionuclides," Revision 0.

14-5 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria," Volume 2, Revision 1.

14-6 U.S. Nuclear Regulatory Commission, NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," Revision 1, August 2000, with June 2001 updates.

14-7 U.S. Nuclear Regulatory Commission, NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," 1998.

14-8 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated Decommissioning Guidance, Decommissioning Process for Materials Licensees," Volume 1, Revision 2.

14-9 U.S. Nuclear Regulatory Commission, NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," Revision 1, June 1998.

14-10 Abelquist, E.W., "Decommissioning Health Physics: A Handbook for MARSSIM Users," Institute of Physics Publishing, 2001.

14-11 International Organization for Standardization, ISO 7503-1, "Evaluation of surface contamination -- Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters," 1998.

Table 14-1

Site-Specific Building And Structural Surface DCGLs

Radionuclide	Occupancy $DCGL_W$ (dpm/100 cm^2) [a] By Conceptual Site Model	
	Small Office	Large Warehouse
U-234	20,000	49,000
U-235 + D [b]	19,000	37,000
U-238 + D [b]	21,000	49,000
Tc-99	13,000,000	13,000,000
Th-232 + C [c]	1,200	2,200
Np-237 + D [b]	2,700	4,000
Pu-239/240	3,500	5,300
Am-241	3,400	5,100

[a] The reported building DCGLs are in gross radioactivity limits rounded down (truncated) to two significant figures.
[b] "+ D" = plus short-lived decay products.
[c] "+ C" = plus the entire decay chain (progeny) in secular equilibrium.

Table 14-2

Site-Specific Soil DCGLs

Radionuclide	$DCGL_W$ (pCi/g) [a] By Conceptual Site Model				
	Surface Stratum	Root Stratum	Deep Stratum	Uniform Stratum	Excavation Scenario
U-234	545.4	252.7	3,099	209.6	935.6
U-235 + D [b]	109.7	68.7	3,254	55.3	223.2
U-238 + D [b]	319.2	196.6	3,247	181	591
Tc-99	162	32.3	105,800	26.9	79.4
Th-232 + C [c]	5	2.1	9,952	2.1	5.6
Ra-226 + C [c]	5.4	2.3	13,974	2	5.8

[a] The reported soil limits are the activities for the parent radionuclide as specified.
[b] "+ D" = plus short-lived decay products.
[c] "+ C" = plus the entire decay chain (progeny) in secular equilibrium.

Table 14-3

Buried Pipe Gross Activity DCGLs

Buried Pipe Diameter (inches)	Gross Activity DCGL (dpm/100cm^2)[a]
2	81,086
4	162,172
6	243,258
8	324,344
10	405,430
12	486,516
14	567,602
16	648,689
18	729,775
20	810,861
22	891,947
24	973,033
26	1,054,119
28	1,135,205
30	1,216,291
32	1,297,377
34	1,378,463
36	1,459,549
38	1,540,635
40	1,621,721
48	1,946,066

[a] The Gross Activity DCGL is based on the Root DCGLs for soil and the Activity Fractions from building drain samples.

Table 14-4

Adjusted Site-Specific Soil DCGLs

Radionuclide	$DCGL_W$ (pCi/g) [a] By Conceptual Site Model				
	Shallow Stratum	**Root Stratum**	**Deep Stratum**	**Uniform Stratum**	**Excavation Scenario**
U-234	**508.5**	**235.6**	**2890**	**195.4**	**872.4**
U-235 + D [b]	**102.3**	**64.1**	**3034**	**51.6**	**208.1**
U-238 + D [b]	**297.6**	**183.3**	**3028**	**168.8**	**551.1**
Tc-99	**151.0**	**30.1**	**98649**	**25.1**	**74.0**
Th-232 + C [c]	**4.7**	**2.0**	**9279**	**2.0**	**5.2**
Ra-226 + C [c]	**5.0**	**2.1**	**13029**	**1.9**	**5.4**

[a] The reported soil limits are the activities for the parent radionuclide as specified and were calculated using Equation 14-1 to account for the dose contribution from insignificant radionuclides (see Section 14.1.3.2).
[b] "+ D" = plus short-lived decay products. Values of U-235 DCGLs adjusted for Tc-99 are prohibited from use to demonstrate compliance with the final status survey dose criteria.
[c] "+ C" = plus the entire decay chain (progeny) in secular equilibrium.

Table 14-5

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction [a]	U-235 Activity Fraction [a]	U-238 Activity Fraction [a]	U-238:U-235 Ratio [a]	U-234:U-235 Ratio [a]
0.1	0.2285	0.0049	0.7666	155.37	46.31
0.2	0.2864	0.0091	0.7045	77.61	31.55
0.3	0.3358	0.0126	0.6516	51.69	26.64
0.4	0.3785	0.0156	0.6059	38.73	24.19
0.5	0.4157	0.0183	0.5660	30.95	22.73
0.6	0.4484	0.0206	0.5310	25.77	21.76
0.7	0.4775	0.0227	0.4999	22.06	21.07
0.72	0.4829	0.0230	0.4941	21.44	20.96
0.8	0.5034	0.0245	0.4721	19.28	20.56
0.9	0.5267	0.0261	0.4472	17.12	20.17
1.0	0.5477	0.0276	0.4247	15.40	19.85
1.1	0.5668	0.0289	0.4043	13.98	19.60
1.2	0.5842	0.0301	0.3857	12.80	19.39
1.3	0.6001	0.0312	0.3687	11.81	19.22
1.4	0.6147	0.0322	0.3530	10.95	19.07
1.5	0.6282	0.0332	0.3386	10.21	18.95
1.6	0.6407	0.0340	0.3253	9.56	18.84
1.7	0.6523	0.0348	0.3129	8.99	18.75
1.8	0.6631	0.0355	0.3014	8.48	18.67
1.9	0.6731	0.0362	0.2907	8.03	18.59
2.0	0.6825	0.0368	0.2806	7.62	18.53
2.1	0.6913	0.0374	0.2712	7.25	18.48
2.2	0.6996	0.0380	0.2624	6.91	18.43
2.3	0.7074	0.0385	0.2541	6.61	18.39
2.4	0.7147	0.0390	0.2463	6.32	18.35
2.5	0.7216	0.0394	0.2390	6.06	18.32
2.6	0.7282	0.0398	0.2320	5.83	18.29
2.7	0.7344	0.0402	0.2254	5.60	18.26
2.8	0.7403	0.0406	0.2191	5.40	18.24
2.9	0.7459	0.0409	0.2132	5.21	18.22
3.0	0.7512	0.0413	0.2075	5.03	18.20
3.1	0.7562	0.0416	0.2022	4.86	18.18
3.2	0.7611	0.0419	0.1971	4.70	18.17
3.3	0.7657	0.0422	0.1922	4.56	18.15

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
3.4	0.7701	0.0424	0.1875	4.42	18.14
3.5	0.7743	0.0427	0.1830	4.29	18.14
3.6	0.7783	0.0429	0.1788	4.16	18.13
3.7	0.7822	0.0432	0.1747	4.05	18.12
3.8	0.7859	0.0434	0.1708	3.94	18.12
3.9	0.7894	0.0436	0.1670	3.83	18.11
4.0	0.7928	0.0438	0.1634	3.73	18.11
4.1	0.7961	0.0440	0.1599	3.64	18.11
4.2	0.7993	0.0441	0.1566	3.55	18.10
4.3	0.8023	0.0443	0.1534	3.46	18.10
4.4	0.8053	0.0445	0.1503	3.38	18.10
4.5	0.8081	0.0446	0.1473	3.30	18.10
4.6	0.8108	0.0448	0.1444	3.22	18.11
4.7	0.8135	0.0449	0.1416	3.15	18.11
4.8	0.8160	0.0451	0.1389	3.08	18.11
4.9	0.8185	0.0452	0.1363	3.02	18.11
5.0	0.8209	0.0453	0.1338	2.95	18.12
5.1	0.8232	0.0454	0.1314	2.89	18.12
5.2	0.8254	0.0455	0.1291	2.83	18.13
5.3	0.8276	0.0456	0.1268	2.78	18.13
5.4	0.8297	0.0457	0.1246	2.72	18.14
5.5	0.8317	0.0458	0.1225	2.67	18.14
5.6	0.8337	0.0459	0.1204	2.62	18.15
5.7	0.8356	0.0460	0.1184	2.57	18.16
5.8	0.8375	0.0461	0.1164	2.53	18.16
5.9	0.8393	0.0462	0.1145	2.48	18.17
6.0	0.8410	0.0463	0.1127	2.44	18.18
6.1	0.8427	0.0463	0.1109	2.39	18.18
6.2	0.8444	0.0464	0.1092	2.35	18.19
6.3	0.8460	0.0465	0.1075	2.31	18.20
6.4	0.8476	0.0466	0.1058	2.27	18.21
6.5	0.8492	0.0466	0.1042	2.24	18.22
6.6	0.8506	0.0467	0.1027	2.20	18.23
6.7	0.8521	0.0467	0.1012	2.16	18.24

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
6.8	0.8535	0.0468	0.0997	2.13	18.24
6.9	0.8549	0.0468	0.0982	2.10	18.25
7.0	0.8563	0.0469	0.0968	2.07	18.26
7.1	0.8576	0.0469	0.0955	2.03	18.27
7.2	0.8589	0.0470	0.0941	2.00	18.28
7.3	0.8602	0.0470	0.0928	1.97	18.29
7.4	0.8614	0.0471	0.0915	1.95	18.30
7.5	0.8626	0.0471	0.0903	1.92	18.31
7.6	0.8638	0.0471	0.0891	1.89	18.32
7.7	0.8649	0.0472	0.0879	1.86	18.34
7.8	0.8661	0.0472	0.0867	1.84	18.35
7.9	0.8672	0.0472	0.0856	1.81	18.36
8.0	0.8682	0.0473	0.0845	1.79	18.37
8.1	0.8693	0.0473	0.0834	1.76	18.38
8.2	0.8703	0.0473	0.0824	1.74	18.39
8.3	0.8713	0.0474	0.0813	1.72	18.40
8.4	0.8723	0.0474	0.0803	1.70	18.41
8.5	0.8733	0.0474	0.0793	1.67	18.42
8.6	0.8742	0.0474	0.0783	1.65	18.44
8.7	0.8752	0.0474	0.0774	1.63	18.45
8.8	0.8761	0.0475	0.0764	1.61	18.46
8.9	0.8770	0.0475	0.0755	1.59	18.47
9.0	0.8779	0.0475	0.0746	1.57	18.48
9.1	0.8787	0.0475	0.0738	1.55	18.50
9.2	0.8796	0.0475	0.0729	1.53	18.51
9.3	0.8804	0.0475	0.0721	1.52	18.52
9.4	0.8812	0.0475	0.0712	1.50	18.53
9.5	0.8820	0.0476	0.0704	1.48	18.55
9.6	0.8828	0.0476	0.0696	1.46	18.56
9.7	0.8836	0.0476	0.0688	1.45	18.57
9.8	0.8843	0.0476	0.0681	1.43	18.58
9.9	0.8851	0.0476	0.0673	1.41	18.60
10.0	0.8858	0.0476	0.0666	1.40	18.61
10.5	0.8893	0.0476	0.0631	1.32	18.67

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
11.0	0.8925	0.0476	0.0599	1.26	18.74
11.5	0.8954	0.0476	0.0569	1.20	18.81
12.0	0.8982	0.0476	0.0542	1.14	18.87
12.5	0.9007	0.0475	0.0517	1.09	18.94
13.0	0.9031	0.0475	0.0494	1.04	19.01
13.5	0.9053	0.0474	0.0472	1.00	19.08
14.0	0.9074	0.0474	0.0452	0.95	19.15
14.5	0.9094	0.0473	0.0433	0.92	19.23
15.0	0.9112	0.0472	0.0416	0.88	19.30
15.5	0.9130	0.0471	0.0399	0.85	19.37
16.0	0.9146	0.0470	0.0384	0.82	19.44
16.5	0.9162	0.0469	0.0369	0.79	19.51
17.0	0.9176	0.0468	0.0355	0.76	19.59
17.5	0.9190	0.0467	0.0342	0.73	19.66
18.0	0.9204	0.0466	0.0330	0.71	19.74
18.5	0.9216	0.0465	0.0318	0.68	19.81
19.0	0.9229	0.0464	0.0307	0.66	19.88
19.5	0.9240	0.0463	0.0297	0.64	19.96
20.0	0.9251	0.0462	0.0287	0.62	20.03
20.5	0.9262	0.0461	0.0277	0.60	20.11
21.0	0.9272	0.0459	0.0268	0.58	20.18
21.5	0.9282	0.0458	0.0260	0.57	20.26
22.0	0.9292	0.0457	0.0251	0.55	20.34
22.5	0.9301	0.0456	0.0244	0.53	20.41
23.0	0.9309	0.0454	0.0236	0.52	20.49
23.5	0.9318	0.0453	0.0229	0.51	20.56
24.0	0.9326	0.0452	0.0222	0.49	20.64
24.5	0.9334	0.0451	0.0215	0.48	20.72
25.0	0.9342	0.0449	0.0209	0.47	20.79
25.5	0.9349	0.0448	0.0203	0.45	20.87
26.0	0.9356	0.0447	0.0197	0.44	20.94
26.5	0.9363	0.0445	0.0192	0.43	21.02
27.0	0.9370	0.0444	0.0186	0.42	21.10
27.5	0.9376	0.0443	0.0181	0.41	21.17

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
28.0	0.9382	0.0442	0.0176	0.40	21.25
28.5	0.9389	0.0440	0.0171	0.39	21.33
29.0	0.9394	0.0439	0.0167	0.38	21.40
29.5	0.9400	0.0438	0.0162	0.37	21.48
30.0	0.9406	0.0436	0.0158	0.36	21.56
30.5	0.9411	0.0435	0.0154	0.35	21.64
31.0	0.9417	0.0434	0.0150	0.35	21.71
31.5	0.9422	0.0432	0.0146	0.34	21.79
32.0	0.9427	0.0431	0.0142	0.33	21.87
32.5	0.9432	0.0430	0.0138	0.32	21.94
33.0	0.9437	0.0429	0.0135	0.31	22.02
33.5	0.9441	0.0427	0.0131	0.31	22.10
34.0	0.9446	0.0426	0.0128	0.30	22.18
34.5	0.9450	0.0425	0.0125	0.29	22.25
35.0	0.9455	0.0423	0.0122	0.29	22.33
35.5	0.9459	0.0422	0.0119	0.28	22.41
36.0	0.9463	0.0421	0.0116	0.28	22.49
36.5	0.9467	0.0420	0.0113	0.27	22.56
37.0	0.9471	0.0418	0.0110	0.26	22.64
37.5	0.9475	0.0417	0.0108	0.26	22.72
38.0	0.9479	0.0416	0.0105	0.25	22.80
38.5	0.9483	0.0415	0.0102	0.25	22.87
39.0	0.9487	0.0413	0.0100	0.24	22.95
39.5	0.9490	0.0412	0.0098	0.24	23.03
40.0	0.9494	0.0411	0.0095	0.23	23.11
40.5	0.9497	0.0410	0.0093	0.23	23.18
41.0	0.9501	0.0408	0.0091	0.22	23.26
41.5	0.9504	0.0407	0.0089	0.22	23.34
42.0	0.9507	0.0406	0.0087	0.21	23.42
42.5	0.9511	0.0405	0.0085	0.21	23.50
43.0	0.9514	0.0404	0.0083	0.20	23.57
43.5	0.9517	0.0402	0.0081	0.20	23.65
44.0	0.9520	0.0401	0.0079	0.20	23.73
44.5	0.9523	0.0400	0.0077	0.19	23.81

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
45.0	0.9526	0.0399	0.0075	0.19	23.89
45.5	0.9529	0.0398	0.0074	0.19	23.96
46.0	0.9532	0.0396	0.0072	0.18	24.04
46.5	0.9534	0.0395	0.0070	0.18	24.12
47.0	0.9537	0.0394	0.0069	0.17	24.20
47.5	0.9540	0.0393	0.0067	0.17	24.28
48.0	0.9543	0.0392	0.0066	0.17	24.35
48.5	0.9545	0.0391	0.0064	0.16	24.43
49.0	0.9548	0.0390	0.0063	0.16	24.51
49.5	0.9550	0.0388	0.0061	0.16	24.59
50.0	0.9553	0.0387	0.0060	0.15	24.67
50.5	0.9555	0.0386	0.0058	0.15	24.74
51.0	0.9558	0.0385	0.0057	0.15	24.82
51.5	0.9560	0.0384	0.0056	0.15	24.90
52.0	0.9563	0.0383	0.0054	0.14	24.98
52.5	0.9565	0.0382	0.0053	0.14	25.06
53.0	0.9567	0.0381	0.0052	0.14	25.13
53.5	0.9570	0.0380	0.0051	0.13	25.21
54.0	0.9572	0.0378	0.0050	0.13	25.29
54.5	0.9574	0.0377	0.0048	0.13	25.37
55.0	0.9576	0.0376	0.0047	0.13	25.45
55.5	0.9578	0.0375	0.0046	0.12	25.53
56.0	0.9581	0.0374	0.0045	0.12	25.60
56.5	0.9583	0.0373	0.0044	0.12	25.68
57.0	0.9585	0.0372	0.0043	0.12	25.76
57.5	0.9587	0.0371	0.0042	0.11	25.84
58.0	0.9589	0.0370	0.0041	0.11	25.92
58.5	0.9591	0.0369	0.0040	0.11	25.99
59.0	0.9593	0.0368	0.0039	0.11	26.07
59.5	0.9595	0.0367	0.0038	0.10	26.15
60.0	0.9597	0.0366	0.0037	0.10	26.23
60.5	0.9599	0.0365	0.0037	0.10	26.31
61.0	0.9600	0.0364	0.0036	0.10	26.39
61.5	0.9602	0.0363	0.0035	0.10	26.46

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
62.0	0.9604	0.0362	0.0034	0.09	26.54
62.5	0.9606	0.0361	0.0033	0.09	26.62
63.0	0.9608	0.0360	0.0032	0.09	26.70
63.5	0.9610	0.0359	0.0032	0.09	26.78
64.0	0.9611	0.0358	0.0031	0.09	26.86
64.5	0.9613	0.0357	0.0030	0.08	26.93
65.0	0.9615	0.0356	0.0029	0.08	27.01
65.5	0.9616	0.0355	0.0029	0.08	27.09
66.0	0.9618	0.0354	0.0028	0.08	27.17
66.5	0.9620	0.0353	0.0027	0.08	27.25
67.0	0.9621	0.0352	0.0026	0.08	27.33
67.5	0.9623	0.0351	0.0026	0.07	27.40
68.0	0.9625	0.0350	0.0025	0.07	27.48
68.5	0.9626	0.0349	0.0024	0.07	27.56
69.0	0.9628	0.0348	0.0024	0.07	27.64
69.5	0.9629	0.0347	0.0023	0.07	27.72
70.0	0.9631	0.0346	0.0023	0.07	27.80
70.5	0.9632	0.0346	0.0022	0.06	27.87
71.0	0.9634	0.0345	0.0021	0.06	27.95
71.5	0.9635	0.0344	0.0021	0.06	28.03
72.0	0.9637	0.0343	0.0020	0.06	28.11
72.5	0.9638	0.0342	0.0020	0.06	28.19
73.0	0.9640	0.0341	0.0019	0.06	28.27
73.5	0.9641	0.0340	0.0019	0.05	28.34
74.0	0.9643	0.0339	0.0018	0.05	28.42
74.5	0.9644	0.0338	0.0017	0.05	28.50
75.0	0.9646	0.0338	0.0017	0.05	28.58
75.5	0.9647	0.0337	0.0016	0.05	28.66
76.0	0.9648	0.0336	0.0016	0.05	28.74
76.5	0.9650	0.0335	0.0015	0.05	28.81
77.0	0.9651	0.0334	0.0015	0.04	28.89
77.5	0.9652	0.0333	0.0015	0.04	28.97
78.0	0.9654	0.0332	0.0014	0.04	29.05
78.5	0.9655	0.0331	0.0014	0.04	29.13

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
79.0	0.9656	0.0331	0.0013	0.04	29.21
79.5	0.9658	0.0330	0.0013	0.04	29.29
80.0	0.9659	0.0329	0.0012	0.04	29.36
80.5	0.9660	0.0328	0.0012	0.04	29.44
81.0	0.9661	0.0327	0.0011	0.03	29.52
81.5	0.9663	0.0326	0.0011	0.03	29.60
82.0	0.9664	0.0326	0.0011	0.03	29.68
82.5	0.9665	0.0325	0.0010	0.03	29.76
83.0	0.9666	0.0324	0.0010	0.03	29.83
83.5	0.9667	0.0323	0.0009	0.03	29.91
84.0	0.9669	0.0322	0.0009	0.03	29.99
84.5	0.9670	0.0322	0.0009	0.03	30.07
85.0	0.9671	0.0321	0.0008	0.03	30.15
85.5	0.9672	0.0320	0.0008	0.02	30.23
86.0	0.9673	0.0319	0.0008	0.02	30.30
86.5	0.9674	0.0318	0.0007	0.02	30.38
87.0	0.9676	0.0318	0.0007	0.02	30.46
87.5	0.9677	0.0317	0.0007	0.02	30.54
88.0	0.9678	0.0316	0.0006	0.02	30.62
88.5	0.9679	0.0315	0.0006	0.02	30.70
89.0	0.9680	0.0315	0.0006	0.02	30.78
89.5	0.9681	0.0314	0.0005	0.02	30.85
90.0	0.9682	0.0313	0.0005	0.02	30.93
90.5	0.9683	0.0312	0.0005	0.01	31.01
91.0	0.9684	0.0311	0.0004	0.01	31.09
91.5	0.9685	0.0311	0.0004	0.01	31.17
92.0	0.9686	0.0310	0.0004	0.01	31.25
92.5	0.9687	0.0309	0.0003	0.01	31.33
93.0	0.9688	0.0309	0.0003	0.01	31.40
93.5	0.9689	0.0308	0.0003	0.01	31.48
94.0	0.9690	0.0307	0.0003	0.01	31.56
94.5	0.9691	0.0306	0.0002	0.01	31.64
95.0	0.9692	0.0306	0.0002	0.01	31.72
95.5	0.9693	0.0305	0.0002	0.01	31.80

Table 14-5 (continued)

Radioactivity And Isotopic Ratios Relative To Enrichment

Enrichment (%)	U-234 Activity Fraction	U-235 Activity Fraction	U-238 Activity Fraction	U-238:U-235 Ratio	U-234:U-235 Ratio
96.0	0.9694	0.0304	0.0001	0.00	31.87
96.5	0.9695	0.0303	0.0001	0.00	31.95
97.0	0.9696	0.0303	0.0001	0.00	32.03
97.5	0.9697	0.0302	0.0001	0.00	32.11
98.0	0.9698	0.0301	0.0000	0.00	32.19
98.5	0.9699	0.0301	0.0000	0.00	32.27
100.0	0.9702	0.0298	0.0000	0.00	32.50

[a] Though calculations were performed for "enrichments" less than 0.7 percent, those calculated values are subject to significant error due to limitations of the original empirically-derived formulas. Additional calculations should be performed if the weight percent of U-235 is less than 0.7 percent.

Table 14-6

Building And Structural Surface Radioactivity Fractions

Radionuclide	Radioactivity Fraction [a]
U-234	8.27E-01
U-235 + D [b]	3.72E-02
U-238 + D [b]	1.27E-01
Tc-99	2.83E-03
Th-232 + C [c]	3.21E-03
Np-237 + D [b]	5.57E-05
Pu-239/240	2.03E-06
Am-241	2.68E-03
Sum For All Radionuclides :	**1.0**
Sum For Uranium Only :	**9.91E-01**

[a] Values are taken from Table 4-2 of DP Chapter 4.0.
[b] "+ D" = plus short-lived decay products.
[c] "+ C" = plus the entire decay chain (progeny) in secular equilibrium.

Table 14-7

Building And Structural Surfaces Gross Radioactivity $DCGL_W$ For Small Office

Radionuclide	$DCGL_W$ (dpm/100 cm^2)	Radioactivity Fractions Based on Characterization Data [a]
U-234	20,000	8.27E-01
U-235 + D	19,000	3.72E-02
U-238 + D	21,000	1.27E-01
Tc-99	13,000,000	2.83E-03
Th-232 + C	1,200	3.21E-03
Np-237 + D	2,700	5.57E-05
Pu-239/240	3,500	2.03E-06
Am-241	3,400	2.68E-03
	Totals:	**1.0**
	Gross Activity $DCGL_W$ (dpm/100 cm^2) [b] :	**18,925**

[a] Values are taken from Table 4-2 of DP Chapter 4.0.
[b] Calculated using Equation 4-4 of MARSSIM and rounded down (truncated) to two significant figures.

Table 14-8

Distribution Ratios For U-235 To Infer Tc-99

Site Area	Distribution Ratio Per Surrogate Evaluation Area (SEA)[a, b]		
	Surface Soil	Root Stratum Soil	Deep Stratum Soil
Plant Soil SEA	9.24	9.63	5.94
Tc-99 SEA	46.11	20.47	21.84
Burial Pit SEA	5.91	3.83	4.76

[a] Mean Tc-99:U-235 Ratio plus 1.645 x Standard Deviation of the Mean
[b] Taken from Table 4-2 of Reference 14-4

Table 14-9

Modified U-235 Soil $DCGL_W$ Values Accounting For Tc-99

Site Area	Modified U-235 $DCGL_W$ [a] (pCi/g) By Conceptual Site Model				
	Shallow Stratum	Root Stratum	Deep Stratum	Uniform Stratum	Excavation Scenario
Plant Soil SEA	14.1	3.0	2565	2.5	11.8
Tc-99 SEA	3.2	1.4	1815	1.2	3.3
Burial Pit SEA	20.4	7.0	2647	5.8	14.5

[a] Calculated using Equation 4-1 of MARSSIM. Values of U-235 DCGLs adjusted for Tc-99 are prohibited from use to demonstrate compliance with the final status survey dose criteria.

Table 14-10

Adjusted And Modified Soil $DCGL_W$ Values For Survey Design and Remedial Action Support

Radionuclide	$DCGL_W$ (pCi/g) By Conceptual Site Model									
	Surface Soil		Root Stratum		Deep Volumetric [a]		Uniform [b]		Excavation [a]	
	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99
Plant Soil SEA										
Total Uranium [c]	394.3	191.7	202.4	52.8	2917	2895	170.2	44.1	706.3	202.8
U-234	508.5	508.5	235.6	235.6	2890	2890	195.4	195.4	872.4	872.4
U-235	102.3	14.1	64.1	3.0	3034	2565	51.6	2.5	208.1	11.8
U-238	297.6	297.6	183.3	183.3	3028	3028	168.8	168.8	551.1	551.1
Tc-99	151.0	N/A	30.1	N/A	98649	N/A	25.1	N/A	74.0	N/A
Th-232 + C	4.7	4.7	2.0	2.0	9279	9279	2.0	2.0	5.2	5.2
Ra-226 + C	5.0	5.0	2.1	2.1	13029	13029	1.9	1.9	5.4	5.4

a The distribution ratio for Deep Stratum soil was used to calculate the $DCGL_W$ for Total Uranium and U-235 when inferring Tc-99

b The distribution ratio for Root Stratum soil was used to calculate the $DCGL_W$ for Total Uranium and U-235 when inferring Tc-99

c Total Uranium $DCGL_W$ values were calculated using Equation 4-4 of MARSSIM, adjusted $DCGL_W$ values from Table 14-4, modified U-235 $DCGL_W$ values from Table 14-9, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4% in soil.

Table 14-10 (continued)

Adjusted And Modified Soil $DCGL_W$ Values For Survey Design and Remedial Action Support

Radionuclide	$DCGL_W$ (pCi/g) By Conceptual Site Model									
	Surface Soil		Root Stratum		Deep Volumetric [a]		Uniform [b]		Excavation [a]	
	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99
Tc-99 SEA										
Total Uranium [c]	394.3	62.9	202.4	28.8	2917	2837	170.2	24.0	706.3	69.7
U-234	508.5	508.5	235.6	235.6	2890	2890	195.4	195.4	872.4	872.4
U-235	102.3	3.2	64.1	1.4	3034	1815	51.6	1.2	208.1	3.3
U-238	297.6	297.6	183.3	183.3	3028	3028	168.8	168.8	551.1	551.1
Tc-99	151.0	N/A	30.1	N/A	98649	N/A	25.1	N/A	74.0	N/A
Th-232 + C	4.7	4.7	2.0	2.0	9279	9279	2.0	2.0	5.2	5.2
Ra-226 + C	5.0	5.0	2.1	2.1	13029	13029	1.9	1.9	5.4	5.4

a The distribution ratio for Deep Stratum soil was used to calculate the $DCGL_W$ for Total Uranium and U-235 when inferring Tc-99

b The distribution ratio for Root Stratum soil was used to calculate the $DCGL_W$ for Total Uranium and U-235 when inferring Tc-99

c Total Uranium $DCGL_W$ values were calculated using Equation 4-4 of MARSSIM, adjusted $DCGL_W$ values from Table 14-4, modified U-235 $DCGL_W$ values from Table 14-9, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4%.

Table 14-10 (continued)

Adjusted And Modified Soil $DCGL_W$ Values For Survey Design and Remedial Action Support

Radionuclide	$DCGL_W$ (pCi/g) By Conceptual Site Model									
	Surface Soil		Root Stratum		Deep Volumetric [a]		Uniform [b]		Excavation [a]	
	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99
Burial Pit SEA										
Total Uranium [c]	394.3	235.3	202.4	95.1	2917	2899	170.2	79.6	706.3	236.3
U-234	508.5	508.5	235.6	235.6	2890	2890	195.4	195.4	872.4	872.4
U-235	102.3	20.4	64.1	7.0	3034	2647	51.6	5.8	208.1	14.5
U-238	297.6	297.6	183.3	183.3	3028	3028	168.8	168.8	551.1	551.1
Tc-99	151.0	N/A	30.1	N/A	98649	N/A	25.1	N/A	74.0	N/A
Th-232 + C	4.7	4.7	2.0	2.0	9279	9279	2.0	2.0	5.2	5.2
Ra-226 + C	5.0	5.0	2.1	2.1	13029	13029	1.9	1.9	5.4	5.4

a The distribution ratio for Deep Stratum soil was used to calculate the $DCGL_W$ for Total Uranium and U-235 when inferring Tc-99

b The distribution ratio for Root Stratum soil was used to calculate the $DCGL_W$ for Total Uranium and U-235 when inferring Tc-99

c Total Uranium $DCGL_W$ values were calculated using Equation 4-4 of MARSSIM, adjusted $DCGL_W$ values from Table 14-4, modified U-235 $DCGL_W$ values from Table 14-9, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4%.

Table 14-11

Area Factors For Building Surfaces (Building Occupancy)

Radionuclide	Elevated Measurement Area (m^2)		
	6.5	**4**	**1**
U-234	1.0	1.6	6.5
U-235 + D	1.0	1.6	6.4
U-238 + D	1.0	1.6	6.5
Tc-99	1.0	1.6	6.4
Th-232 + C	1.0	1.6	6.4
Np-237 + D	1.0	1.6	6.5
Pu-239/ Pu-240	1.0	1.6	6.5
Am-241	1.0	1.6	6.5

+ D = plus short-lived decay products.
+ C = plus the entire decay chain (progeny) in secular equilibrium.

Table 14-12

Area Factors For Soil Contamination

Radionuclide	Elevated Measurement Area (m^2)									
	153,375	**10,000**	**3,000**	**1,000**	**300**	**100**	**30**	**10**	**3**	**1**
Surface Soil										
U-234	1.0	1.5	2.2	2.6	7.8	19.3	41.7	67.3	96.0	119.5
U-235 + D	1.0	1.1	1.2	1.2	1.3	1.5	1.8	2.6	5.4	12.1
U-238 + D	1.0	1.2	1.5	1.6	2.2	2.6	3.4	4.9	10.2	22.3
Tc-99	1.0	1.0	1.0	1.0	3.4	10.3	34.2	102.2	338.5	1,009
Th-232 + C	1.0	1.0	1.1	1.1	1.4	1.7	2.3	3.5	7.3	16.9
Ra-226 + C	1.0	1.1	1.2	1.2	1.8	2.2	3.0	4.5	9.6	22.4
Root Soil										
U-234	1.0	1.2	1.3	1.4	4.1	9.4	19.2	33.0	67.9	130.4
U-235 + D	1.0	1.0	1.1	1.1	1.9	2.3	2.9	4.1	8.3	17.9
U-238 + D	1.0	1.1	1.3	1.3	2.5	3.6	5.0	7.2	14.8	31.5
Tc-99	1.0	1.0	1.0	1.0	3.4	10.3	34.3	103.0	343.3	1,029
Th-232 + C	1.0	1.0	1.0	1.0	2.1	3.0	4.2	6.0	12.8	28.4
Ra-226 + C	1.0	1.0	1.1	1.1	2.4	3.9	5.8	8.7	18.5	41.6
Uniform Soil										
U-234	1.0	1.2	1.3	1.3	4.0	9.3	19.6	34.3	70.5	132.8
U-235 + D	1.0	1.1	1.1	1.1	1.9	2.5	3.3	4.7	9.6	20.5
U-238 + D	1.0	1.1	1.3	1.3	2.5	3.6	5.0	7.2	14.9	31.6
Tc-99	1.0	1.0	1.0	1.0	3.4	10.3	34.3	102.9	342.7	1,027
Th-232 + C	1.0	1.0	1.0	1.0	2.1	3.0	4.2	6.1	12.9	28.9
Ra-226 + C	1.0	1.1	1.1	1.1	2.5	4.1	6.1	9.1	19.3	43.4

Excavation Scenario						
Radionuclide	**148**	**100**	**30**	**10**	**3**	**1**
U-234	1.0	2.0	6.7	19.2	35.1	64.5
U-235 + D	1.0	1.3	1.5	2.0	3.5	7.0
U-238 + D	1.0	1.9	2.7	3.7	6.6	13.2
Tc-99	1.0	2.0	6.7	20	67	200
Th-232 + C	1.0	1.9	2.7	3.8	7.0	14.3
Ra-226 + C	1.0	2.0	3.5	5.1	9.6	19.7

Table 14-13

Laboratory Analysis Methods And Sensitivities

Analyte	Medium	Method	Sensitivity Soil (pCi/g)	Sensitivity Water (pCi/l)	Description
Gross alpha	Water	EPA 900.0	NA	3	Gas Flow Proportional Counter
Gross beta	Water	EPA 900.0	NA	4	Gas Flow Proportional Counter
Ra-226	Soil and Water	EML GA-01-R MOD EPA 901.1	0.5	1	Gamma Spectrometry
Tc-99	Soil and Water	EPA-906.0 EML TC-02-RC MOD ASTM C-1387	2	5	Liquid Scintillation Counting
Th-232	Soil and Water	EML A-01-R MOD ASTM D-3972	0.1	0.2	Alpha Spectrometry
U-234	Soil and Water	EML A-01-R MOD ASTM D-3972	0.1	0.2	Alpha Spectrometry
U-235	Soil and Water	EML A-01-R MOD ASTM D-3972	0.1	0.2	Alpha Spectrometry
	Soil and Water	EML GA-01-R MOD EPA 901.1	0.5	5	Gamma Spectrometry
U-238	Soil and Water	EML A-01-R MOD ASTM D-3972	0.1	0.2	Alpha Spectrometry
	Soil and Water	EPA 901.1	1	5	Assume secular equilibrium with Th-234
Total U	Soil and Water	ASTM D-3972	0.1	0.2	Derived from alpha spectrometry data
Am-241	Soil and Water	EML A-01-R MOD ASTM D-3972	0.05	0.1	Alpha Spectrometry
Np-237	Soil and Water	EML A-01-R MOD ASTM D-3972	0.05	0.1	Alpha Spectrometry
Pu-239/240	Soil and Water	EML A-01-R MOD STM D-3972	0.05	0.1	Alpha Spectrometry

Table 14-14

Typical Field Instruments For Performing Final Status Surveys

Instrument/Detector Type	Radiation Detected	Scale/Range	Typical Background	Typical MDC 95 Percent Confidence Level	Usage
Scintillation (Ludlum 2224 or equivalent) rate meter/scalar with Ludlum 43-89	Alpha Beta	0 to 500,000 cpm	<10 cpm ~300 cpm	100 dpm/100 cm^2 (direct alpha) 700 dpm/100 cm^2 (direct beta) 1,500 dpm/100 cm^2 (scan)	RASS and FSS
Ludlum Model 2360/Ludlum 43-68 or equivalent Gas Flow Proportional	Alpha Beta	0 to 500,000 cpm	<10 cpm ~300 cpm	100 dpm/100 cm^2 (direct alpha) 400 dpm/100 cm^2 (direct alpha/beta) 1,100 dpm/100 cm^2 (scan)	RASS and FSS
Ludlum Model 2360/Ludlum 43-37 or equivalent Gas Flow Proportional Floor Monitor	Alpha Beta	0 to 500,000 cpm	<30 cpm ~1,200 cpm	5,500 dpm/100 cm^2 (scan)	RASS and FSS
Ludlum Model 2360/Ludlum HP-260 or equivalent, Geiger-Mueller (20 cm^2 Pancake)	Beta Gamma	0 to 500,000 cpm 720 cpm = 0.2 µR/h	100 cpm	2,100 dpm/100 cm^2 (direct) 8,000 dpm/100 cm^2 (scan)	General characterization and RASS
Ludlum Model 19 Micro-R meter or equivalent 1 in by 1 in NaI detector	Gamma	0 to 3,000 µR/h or 0 to 5,000 µR/h	5 to 8 µR/h	1 to 2 µR/h	General characterization and RASS
3 in by ½ in NaI scintillation detector digital scalar or equivalent	Gamma	0 to 500,000 cpm	2,500 cpm avg. shielded 7,000 cpm avg. unshielded	250 cpm 500 cpm	General characterization and RASS
Ludlum Model 2360/Ludlum 44-10 or equivalent 2 in by 2 in NaI scintillation detector	Gamma	0 to 500,000 cpm	10,000 cpm	84 pCi/g (3 percent enriched Uranium)[1] 99 pCi/g (20 percent enriched Uranium)[1] 122 pCi/g (50 percent enriched Uranium)[1] 140 pCi/g (75 percent enriched Uranium)[1]	RASS and FSS
Ludlum Model 2360 or equivalent/FIDLER NaI scintillation detector	Gamma	0 to 500,000 cpm	12,500 cpm	8 pCi/g (3 percent enriched Uranium)[1] 12 pCi/g (20 percent enriched Uranium)[1] 16 pCi/g (50 percent enriched Uranium)[1] 18 pCi/g (75 percent enriched Uranium)[1]	RASS and FSS
Pressurized ion chamber (Reuter-Stokes HPIC)	Gamma	0 to 10 R/h	5 to 8 µR/h	~0.1 µR/h Accuracy ± 5 percent at 10 µR/h	Environmental gamma exposure rate
Bicron AB-100 scintillation detector	Alpha Beta	0 to 500,000 cpm	<10 cpm ~750 cpm closed beta ~1,500 cpm open beta	70 dpm/100 cm^2 (direct) 850 dpm/100 cm^2 (direct) 3,900 dpm/100 cm^2 (scan)	General characterization and RASS

Table 14-14 (continued)

Typical Field Instruments For Performing Final Status Surveys

Instrument/Detector Type	Radiation Detected	Scale/Range	Typical Background	Typical MDC 95 Percent Confidence Level	Usage
Energy*Solutions* GARDIAN-III or equivalent Intermodal (dimensions 230" X 85" X 61") Box Counting System (6 - 40% HPGE detectors), 50k lbs of soil with a 10 minute count time.	Gamma	N/A	Varies based on geometry and configuration	<1 pCi/g (U-235) <12 pCi/g (U-238)	RASS and FSS
Tennelec Gas Flow Proportional	Alpha Beta	N/A	<2 cpm alpha <6 cpm beta	25 dpm alpha 30 dpm beta	General characterization, RASS and FSS

[1] **MDC values assume actions based on surveyor observations with a surveyor efficiency of 50%. If actions are based on post-processed data evaluation surveyor efficiency is not applicable and the MDC values are reduced by approximately 29%.**

Table 14-15

Survey Unit Size Limitations

Classification	Area Type	Suggest Maximum Area (m^2)
Class 1	Open Land	2,000
	Structures	100
Class 2	Open Land	10,000
	Structures	1,000
Class 3	Open Land	No Limit
	Structures	No Limit

Table 14-16

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Building Survey Areas							
BSA-01	**Building 110**	01	Sub-Surface Soil	3	N/A	506	14-18
		02	Exterior Surfaces Walls and Roof	3	N/A	895	14-18
		03	Interior Surfaces Floors, Walls and Ceilings	3	460	1749	14-18
BSA-02	**Building 230**	01	Sub-surface Soil	3	N/A	3642	14-19
		02	Exterior Surfaces Walls and Roof	3	N/A	5112	14-19
		03	Rod Load Area - Section 1 Floor and Lower Walls	1	92	230	14-19
		04	Rod Load Area – Section 2 Floor and Lower Walls	1	100	165	14-19

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Building Survey Areas							
BSA-02	**Building 230**	05	Rod Load Area – Section 3 Floor and Lower Walls	1	76	151	14-19
		06	Rod Load Area – Section 4 Floor and Lower Walls	1	73	202	14-19
		07	Rod Load Area – Section 5 Floor and Lower Walls	1	72	176	14-19
		08	Rod Load Area – All Sections Upper Walls and Ceiling	2	N/A	947	14-19
		09	Cushman Room Floor and Lower Walls	1	71	139	14-19
		10	Cushman Room Upper Walls and Ceiling	2	N/A	142	14-19
		11	Gadolinium Room Floor and Lower Walls	1	61	124	14-19

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Building Survey Areas							
BSA-02	**Building 230**	12	Gadolinium Room Upper Walls and Ceiling	2	N/A	124	14-19
		13	"U-Shaped" Area (N) – Section 6 Floor, Lower Walls and Stairs	2	939	1,429	14-19
		14	"U-Shaped" Area (S) – Section 7 Floor and Lower Walls	2	979	1,545	14-19
		15	"U-Shaped" Area – Section 8 Trench	1	36	45	14-19
		16	"U-Shaped" Area – Section 9 Floor	1	36	36	14-19
		17	"U-Shaped" Area – All Sections Upper Walls and Ceiling	3	N/A	4,263	14-19

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Building Survey Areas							
BSA-02	**Building 230**	18	Warehouse Area Floors, Walls, Ceilings and Stairs	3	1096	3,681	14-19
		19	2nd Floor Mezzanine Floor, Walls, Ceiling and Roof	3	767	2,005	14-19
		20	Ventilation Ducting	2	N/A	N/A	14-19
BSA-03	**Building 231**	01	Sub-Surface Soil	3	N/A	558	14-20
		02	Exterior Surfaces Walls and Roof	3	N/A	1,212	14-20
		03	Interior Surfaces Floors, Walls and Ceilings	2	558	1,770	14-20

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Open Land Survey Areas							
LSA-01	**South Site Waterways**	01	Joachim Creek	3	N/A	10,072	14-17
		02	Site Creek	2	N/A	2,324	14-17
LSA-02	**Site Pond**	01	Site Pond - North	1	N/A	1,792	14-17
		02	Site Pond - Central	1	N/A	1,736	14-17
		03	Site Pond - South	1	N/A	1,720	14-17
LSA-03	**West Open Land Area**	01	West Open Land Area	3	N/A	10,879	14-17
LSA-04	**Southwest Open Land Area**	01	Southwest Open Land Area	3	N/A	10,309	14-17

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Open Land Survey Areas							
LSA-05	**Barns and Cistern Open Land Area**	01	Cistern Burn Pit Area	1	N/A	1,708	14-17
		02	Barns Area	1	N/A	1,761	14-17
LSA-06	**North Open Land Area**	01	North Open Land Area	3	N/A	14,723	14-17
LSA-07	**North Central Open Land Area**	01	Primary Parking Lot	3	N/A	3,440	14-17
LSA-08	**Central Open Land Area**	01	Section 1	1	N/A	1,773	14-17
		02	Section 2	1	N/A	1,614	14-17
		03	Section 3	1	N/A	1,694	14-17

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Open Land Survey Areas							
LSA-08	**Central Open Land Area**	04	Section 4	1	N/A	1,717	14-17
		05	Section 5	1	N/A	1,714	14-17
		06	Section 6	1	N/A	1,900	14-17
		07	Section 7	1	N/A	1,916	14-17
		08	Section 8	1	N/A	1,895	14-17
		09	Section 9	1	N/A	1,885	14-17
		10	Section 10	1	N/A	1,948	14-17

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Open Land Survey Areas							
LSA-08	**Central Open Land Area**	11	Section 11	1	N/A	1,955	14-17
		12	Section 12	1	N/A	1,872	14-17
		13	Section 13	1	N/A	1,889	14-17
		14	Section 14	1	N/A	1,972	14-17
LSA-09	**Rail Spur Open Land Area**	01	East Rail Spur Area	2	N/A	2,599	14-17
		02	West Rail Spur Area	1	N/A	1,953	14-17
LSA-10	**Burial Pits Open Land Area**	01	Section 1	1	N/A	1,862	14-17

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Open Land Survey Areas							
LSA-10	**Burial Pits Open Land Area**	02	Section 2	1	N/A	1,951	14-17
		03	Section 3	1	N/A	1,939	14-17
		04	Section 4	1	N/A	1,937	14-17
		05	Section 5	1	N/A	1,959	14-17
		06	Section 6	1	N/A	1,954	14-17
		07	Section 7	1	N/A	1,946	14-17
LSA-11	**East Open/Southeast Open Land Area**	01	Section 1	3	N/A	24,715	14-14

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Open Land Survey Areas							
LSA-11	**East Open/Southeast Open Land Area**	02	Section 2	3	N/A	5,394	14-14
LSA-12	**Lay Down Area**	01	Section 1	2	N/A	7,308	14-17
		02	Section 2	2	N/A	7,328	14-17
		03	Section 3	1	N/A	1,984	14-17
		04	Section 4	1	N/A	1,996	14-17
		05	Section 5	1	N/A	1,997	14-17
		06	Section 6	1	N/A	1,997	14-17

Table 14-16 (continued)

Initial Survey Area/Unit Classification And Description List

Survey Area Code	Survey Area Description	Survey Unit Code	Survey Unit Description	Initial MARSSIM Class	Area (m^2)		Figure No.
					Floor Area	Total Area	
Piping Survey Areas							
LSA-12	**Lay Down Area**	07	Section 7	1	N/A	1,974	14-17
PSA-01	**Storm Drain System**	01	Storm Drain System	1	N/A	N/A	N/A
PSA-02	**Septic Treatment System**	01	Septic Treatment System	1	N/A	N/A	N/A
PSA-03	**Building Drain System**	01	Building 110 Floor Drains	1	N/A	N/A	N/A
		02	Building 230 Floor Drains	1	N/A	N/A	N/A
PSA-04	**Public Water System**	01	Public Water System	3	N/A	N/A	N/A
PSA-05	**Raw Water System**	01	Raw Water System	3	N/A	N/A	N/A

Table 14-17

Scan Coverage

Area Classification	Scan Coverage	Surface Activity Measurements Or Soil Samples
Class 1	100 percent	As determined by statistical tests; additional measurements/samples to account for small areas of elevated activity as necessary
Class 2	10 to 100 percent	As determined by statistical tests
Class 3	1 to 10 percent (Judgmental)	
Non Impacted	N/A	

Table 14-18

Investigation Levels

Survey Unit Classification	Flag Scanning Measurement Result When:	Flag Direct Measurement Or Sample Result When:
Class 1	> $DCGL_{EMC}$	> $DCGL_{EMC}$ or > $DCGL_W$ and > a statistical parameter-based value
Class 2	> $DCGL_W$ or > scan MDC	> $DCGL_W$
Class 3	> $DCGL_W$ or > scan MDC	> 50 percent of $DCGL_W$

Table 14-19

Total Weighted Efficiency Example Calculation

Radionuclide	Radiation/Maximum Energy (MeV) [a]	Instrument Efficiency [b]	Surface Efficiency [c]	Yield	Activity Fraction [d]	Weighted Efficiency
Am-241	Alpha/5.6	0.35	0.25	100%	2.682E-03	0.0002
Np-237	Alpha/5.0	0.35	0.25	100%	5.573E-05	0.0000
Pu-239	Alpha/5.2	0.35	0.25	100%	2.027E-06	0.0000
Tc-99	Beta/0.294	0.32	0.25	100%	2.829E-03	0.0002
Th-232	Alpha/4.1	0.35	0.25	100%	3.214E-03	0.0003
Ra-228 [e]	Beta/0.046	0.00	0.00	100%	3.214E-03	0.0000
Ac-228 [e]	Beta/2.13	0.40	0.50	100%	3.214E-03	0.0006
Th-228 [e]	Alpha/5.5	0.35	0.25	100%	3.214E-03	0.0003
Ra-224 [e]	Alpha/5.8	0.35	0.25	100%	3.214E-03	0.0003
U-234	Alpha/4.9	0.35	0.25	100%	8.010E-01	0.0701
U-235	Alpha/4.7	0.35	0.25	100%	4.424E-02	0.0039
Th-231 [f]	Beta/0.390	0.32	0.25	100%	4.424E-02	0.0035
U-238	Alpha/4.3	0.35	0.25	100%	1.460E-01	0.0128
Th-234 [f]	Beta/0.270	0.32	0.25	100%	1.460E-01	0.0117
Pa-234m [f]	Beta/2.20	0.40	0.50	100%	1.460E-01	0.0292
					Total Weighted Efficiency:	**0.13**

[a] Data from <nndc.bnl.gov/chart/>.
[b] Nominal 2πefficiency value for a 126 cm2 gas flow proportional detector with a 0.8 mg/cm2 window in the α + β mode.
[c] Based on guidance provided in ISO 7503-1 (Reference 14-11).
[d] From Table 14-7.
[e] Progeny from decay of Th-232. Assumes complete radon emanation.
[f] Progeny from decay of Uranium parent radionuclides.

Table 14-20

2 in by 2 in NaI Minimum Detectable Exposure Rate

Energy (keV) (From MicroShield® software)	Exposure Rate (μR/h)	cpm per μR/h (Table 6.3 Of Reference 14-7)	cpm per μR/h (weighted)
U-234 Minimum Detectable Exposure Rate			
50	4.19E-05	11,800	5,170.59
100	5.38E-05	9,840	5,528.25
Total:	**9.57E-05**	Total:	10,699
U-234 MDER (μR/h, Eq. 6-21 Of Reference 14-7):			**0.14**
U-235 Minimum Detectable Exposure Rate			
30	2.73E-03	5,160	45.1
60	2.40E-04	13,000	10.0
80	1.03E-02	12,000	396.0
100	1.40E-02	9,840	442.3
150	4.11E-02	6,040	794.3
200	2.44E-01	4,230	3,303.8
Total:	**3.12E-01**	Total:	4,991
U-235 MDER (μR/h, Eq. 6-21 Of Reference 14-7):			**0.30**

Table 14-20 (continued)

2 in by 2 in NaI Minimum Detectable Exposure Rate

Energy (keV) (From MicroShield® software)	Exposure Rate (μR/h)	cpm per μR/h (Table 6.3 Of Reference 14-7)	cpm per μR/h (weighted)
U-238 Minimum Detectable Exposure Rate			
60	1.97E-03	13,000	757.4
80	1.25E-04	12,000	44.2
100	8.11E-03	9,840	2357.8
800	3.54E-03	710	74.2
1,000	2.01E-02	540	320.7
Total:	**3.39E-02**	Total:	3,554
U-238 MDER (μR/h, Eq. 6-21 Of Reference 14-7):			**0.43**

Table 14-21

Data Evaluation When The WRS Test Is Used

Measurement Results	Conclusion
Difference between the maximum survey unit measurement and the minimum reference area measurement is less than the $DCGL_W$. (i.e., SOF as applied to the difference between the maximum survey measurement and minimum reference area measurement for the radionuclides of concern is less than one [1])	The survey unit meets the release criteria
Difference of the survey unit average and reference area average is greater than the $DCGL_W$. (i.e., SOF as applied to the difference between the average survey unit measurements and the reference area measurements for the radionuclides of concern is greater than one [1])	The survey unit fails, additional remediation required.
Difference between any survey unit measurement and any reference area measurement is greater than the $DCGL_W$; however, the difference of the survey unit average and the reference area average is less than $DCGL_W$ (i.e., SOF as applied to the difference between any survey unit measurement and any reference area measurement exceeds 1; however the SOF as applied to the difference between the average of the survey unit measurements and reference area measurements is less than one [1])	Conduct the WRS test and EMC

Table 14-22

Data Evaluation When The Sign Test Is Used

Measurement Results	Conclusion
All concentrations less than the $DCGL_W$. (i.e., SOF for each measurement location is less than one [1])	The survey unit meets the release criteria.
Average concentration greater than the $DCGL_W$ (i.e., SOF as applied to the average activity of each radionuclide of concern is greater than one [1])	The survey unit fails, additional remediation required.
Some measurements greater than the $DCGL_W$; however, the average is less than the $DCGL_W$ (i.e., sum of fraction for any individual measurement exceeds 1; however the SOF as applied to the average activity of each radionuclide of concern is less than one [1])	Conduct the Sign Test and EMC

Table 14-23

Proposed Post Remediation Groundwater Monitoring Wells

Well ID No.	HSU	Post-Remediation Protocol			Existing or Proposed
		Purpose	Parameters	Sample Frequency	
GW-AA	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-D	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-S	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-T	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-U	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-X	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-Z	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Existing
GW-CC	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Proposed
GW-DD	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Proposed
GW-EE	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Proposed
GW-FF	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Proposed
GW-GG	Sand/Gravel	Primary	Tc-99, Isotopic U	Quarterly	Proposed
GW-BB	Sand/Gravel	Secondary	Tc-99, Isotopic U	Quarterly	Existing
GW-V	Sand/Gravel	Secondary	Tc-99, Isotopic U	Quarterly	Existing
GW-W	Sand/Gravel	Secondary	Tc-99, Isotopic U	Quarterly	Existing
GW-Y	Sand/Gravel	Secondary	Tc-99, Isotopic U	Quarterly	Existing
GW-HH	Sand/Gravel	Secondary	Tc-99, Isotopic U	Quarterly	Proposed
GW-II	Sand/Gravel	Secondary	Tc-99, Isotopic U	Quarterly	Proposed
BR-13-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-14-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-15-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-16-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-17-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-18-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-19-JC	Jefferson City Cotter	Primary	Tc-99, Isotopic U	Quarterly	Proposed
BR-04-RB	Roubidoux	Primary	Tc-99, Isotopic U	Quarterly	Existing
BR-08-RB	Roubidoux	Primary	Tc-99, Isotopic U	Quarterly	Existing
BR-10-RB	Roubidoux	Primary	Tc-99, Isotopic U	Quarterly	Existing
BR-03-RB	Roubidoux	Primary	Tc-99, Isotopic U	Quarterly	Existing

Table 14-24

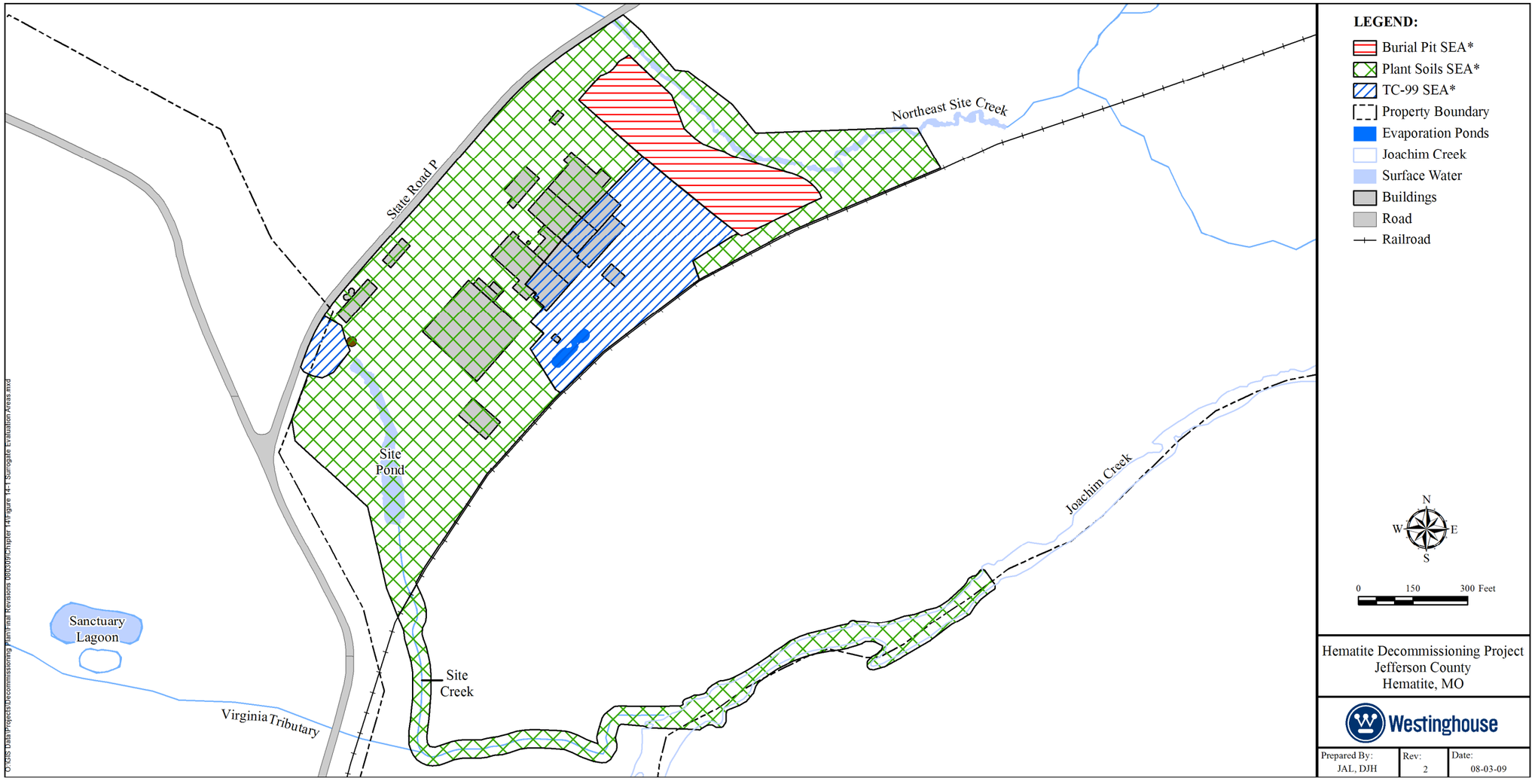
Final Status Survey Sampling and Survey Summary for the Various Scenarios

Final Status Survey Scenarios	Gamma Walkover Survey	Sampling Protocol at Each Systematic Station
Final Status Survey Performed Prior to Backfill*	100 percent of Surfaces	• Excavation surface is within the Root Stratum: A composite sample is collected through the remainder of the Root Stratum and a separate sample is collected from the top 15 cm of the Deep Stratum. • Excavation surface is within the Deep Stratum: A sample is collected in the top 15 cm of the exposed Deep Stratum.
Final Status Survey Performed Following Backfill*	100 percent of surfaces with the exception of areas filled with off-site borrow	• Coring or drilling to the lowest point where remediation occurred (ensures through the backfill) and then compositing a sample from a coring that extends one meter deeper than the lowest point where remediation occurred.
Final Status Survey for Paved/Unpaved Non-excavated Areas or Excavated Areas not Requiring Backfill	Minimum scan coverage is dependent on the classification of the Survey Unit	• A surface sample is collected from the top 15 cm. • A composite sample from 15 cm to 1.5 m is collected. • If the SOF in the sample obtained from the Root Stratum exceeds 0.5, a composite sample is collected from 1.5 m to an appropriate depth (Deep Stratum).

*Note: Peripheral areas of a Survey Unit that have not been excavated or areas not requiring backfill after excavation are surveyed using the "Final Status Survey for Paved/Unpaved Non-excavated Areas or Excavated Areas not Requiring Backfill" scenario.

Figure 14-1

Surrogate Evaluation Areas



*Surrogate Evaluation Area (SEA)

NOTE: With regard to Joachim Creek, the Historical Site Assessment (HSA) and radiological characterization results did not indicate the presence of residual radioactivity in excess of background levels, and thus Joachim Creek and the area immediately adjacent could be considered non-impacted. However, Tc-99 was detected in samples collected at locations just below the confluence of the Site Creek with the Virginia Tributary, and thus the Site Creek has been designated as an impacted area. Consistent with MARSSIM (Reference 14-6) regarding the use of impacted area buffer zones, a reasonably conservative and prudent approach has been taken by establishing an impacted (Class 3) buffer zone along a portion of the Joachim Creek. This buffer zone extends from the confluence of the Site Creek and the Joachim Creek eastward along the Joachim Creek to the location of the nearest radiological characterization sample collected on the Joachim Creek.

Figure 14-2

Uranium Radioactivity Fractions

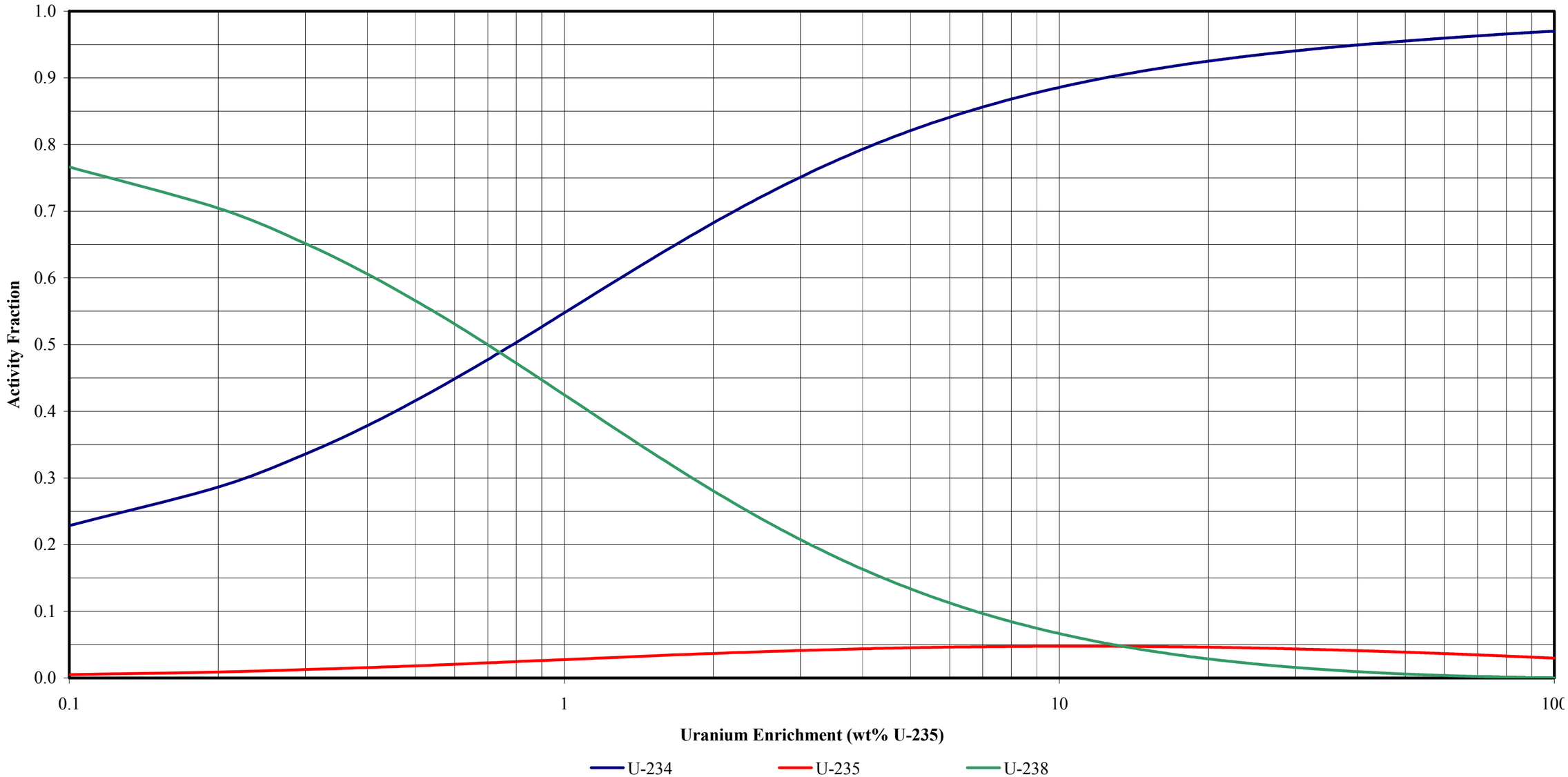


Figure 14-3

Uranium Radioactivity Ratios

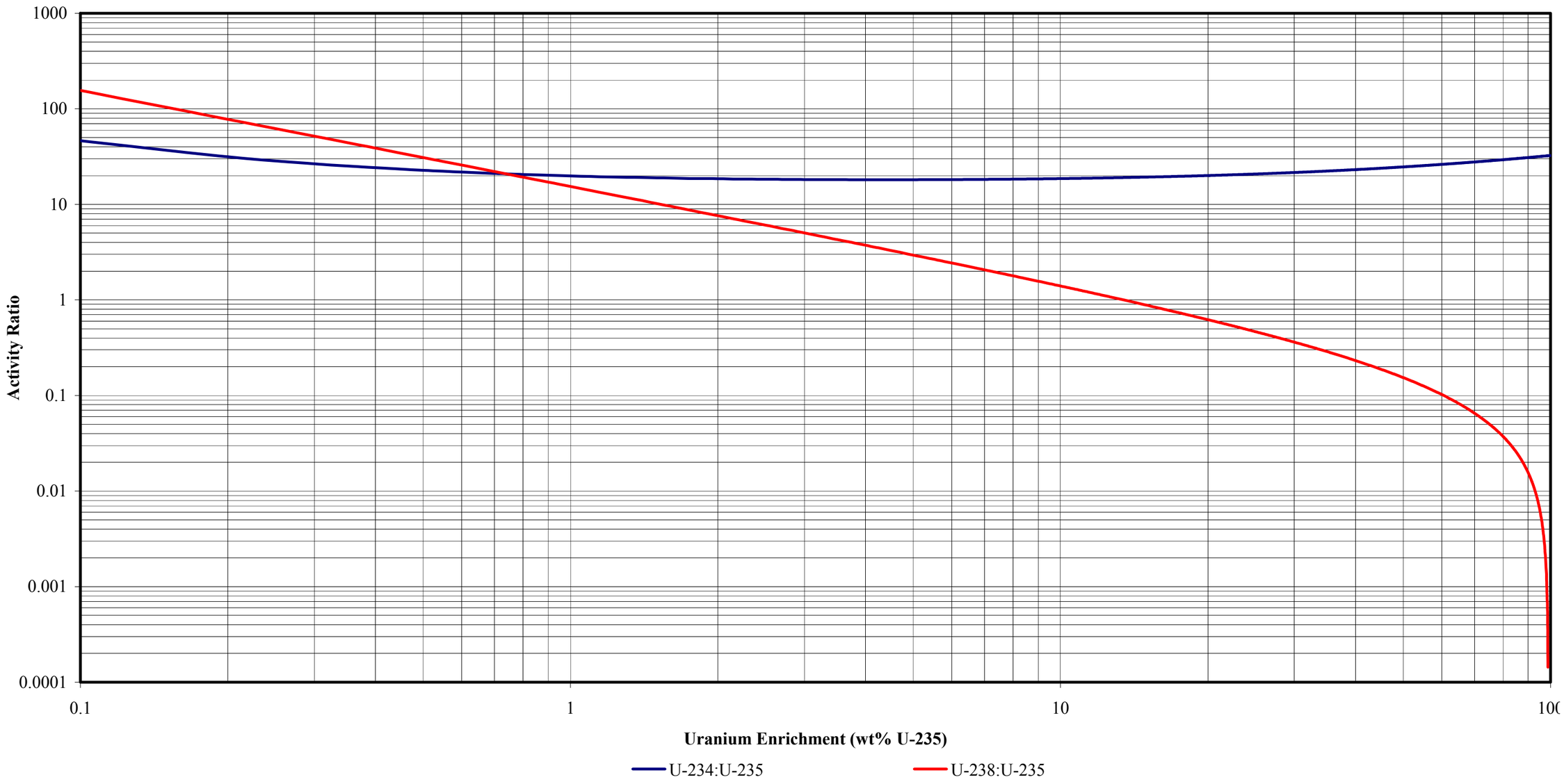


Figure 14-4

Sensitivity Analysis Of Total Uranium $DCGL_W$ For The Plant Soil SEA

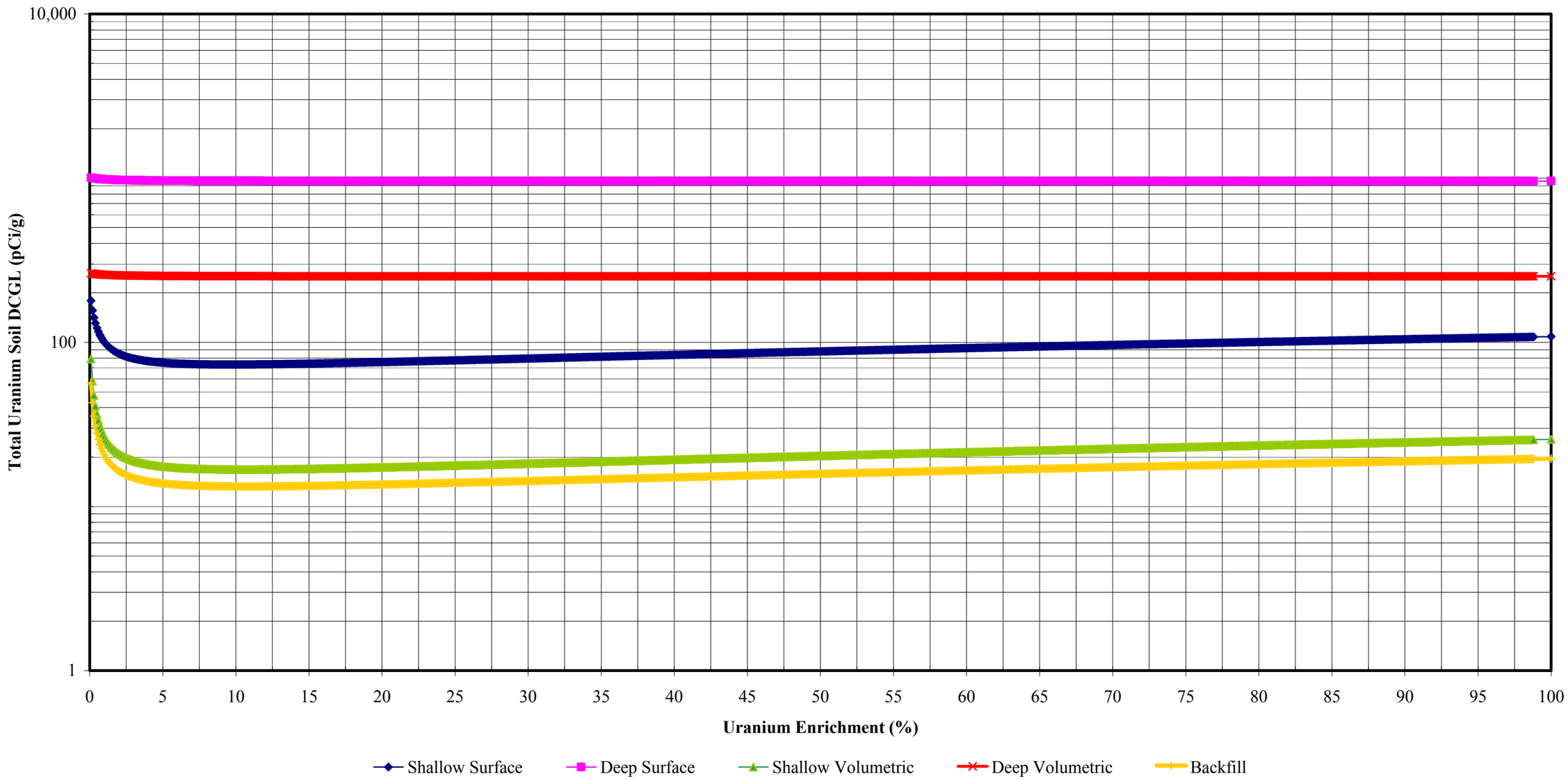


Figure 14-5

Sensitivity Analysis Of Total Uranium $DCGL_W$ For The Tc-99 SEA

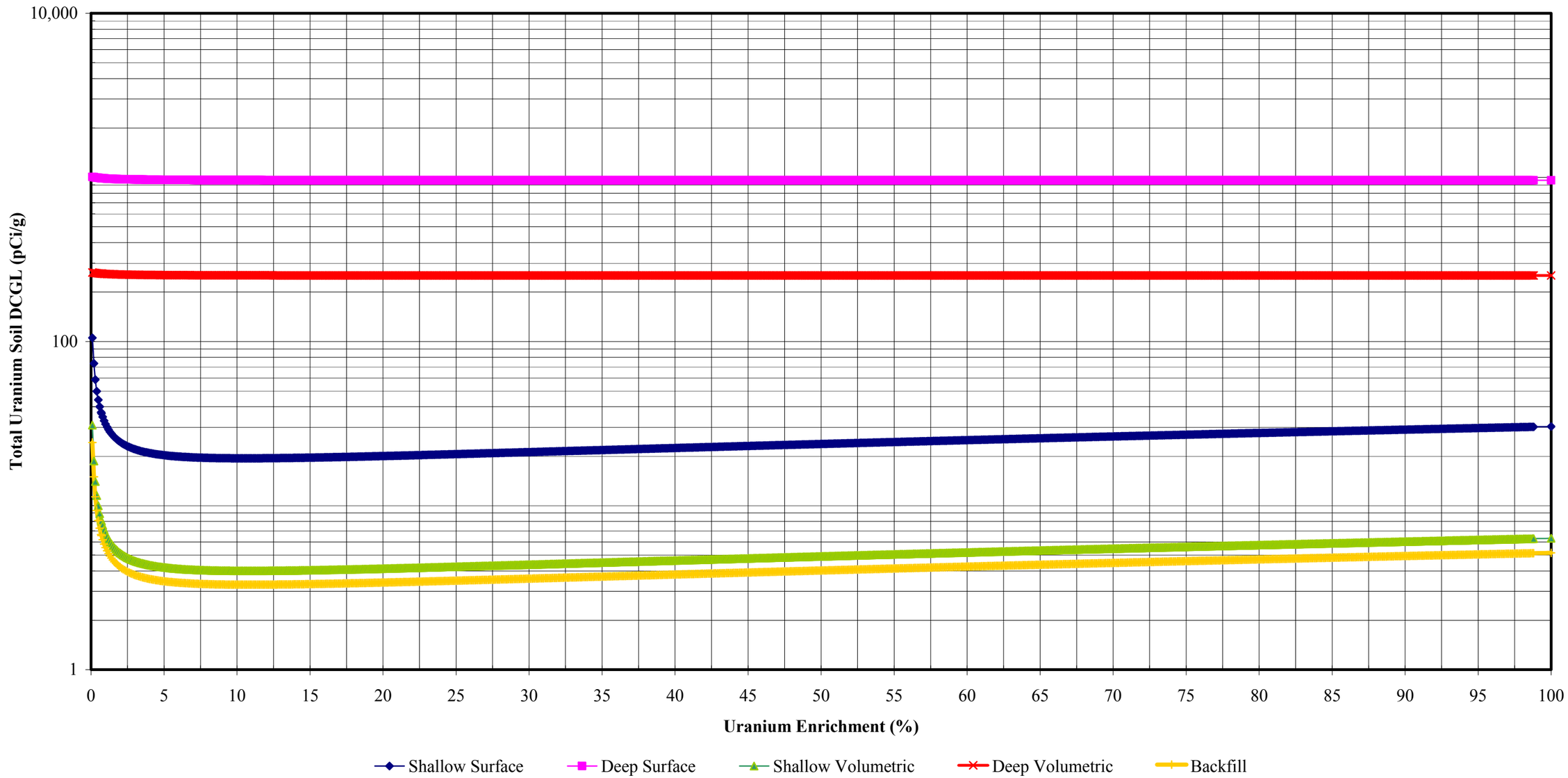


Figure 14-6

Sensitivity Analysis Of Total Uranium $DCGL_W$ For The Burial Pit SEA

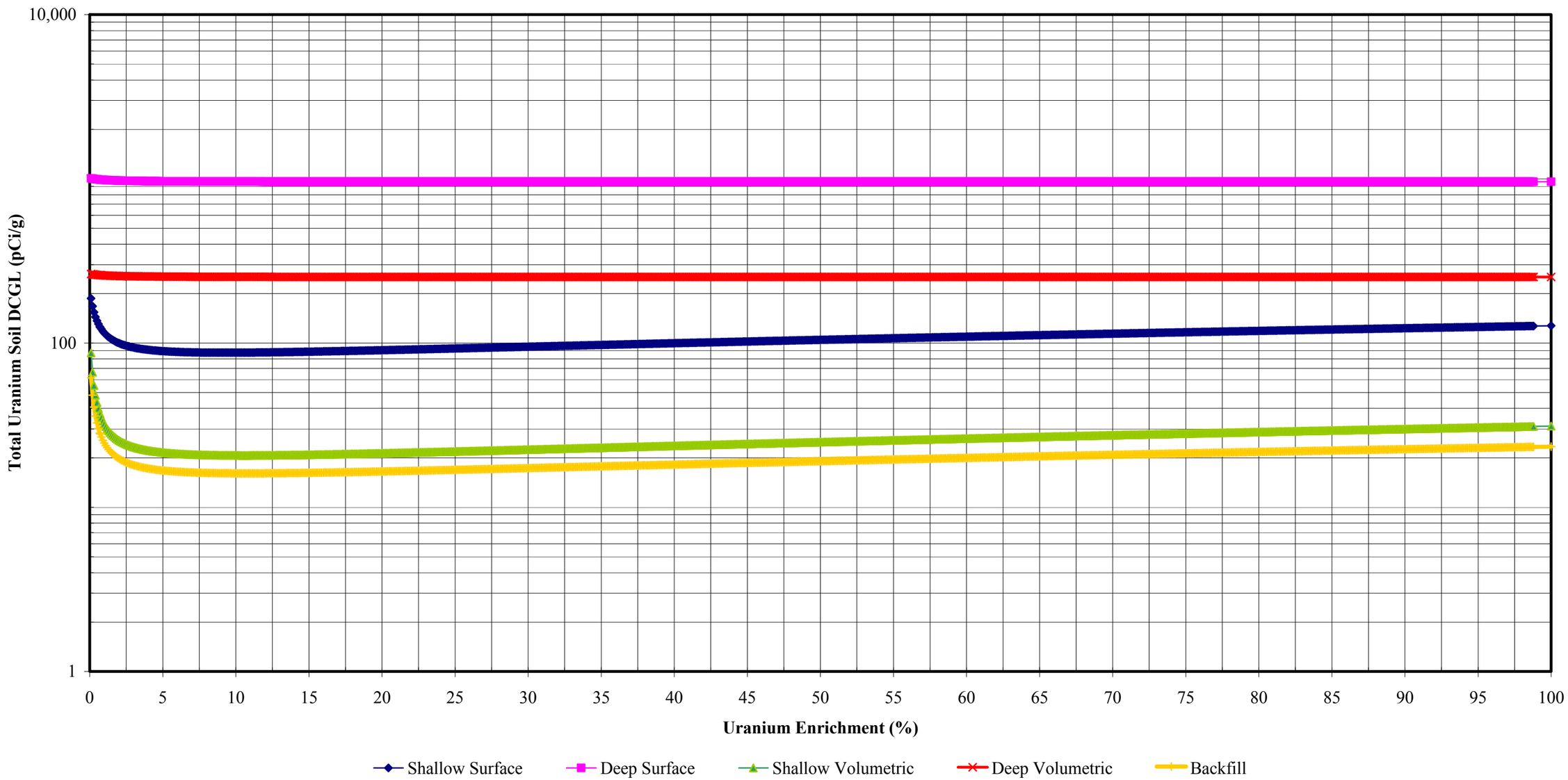


Figure 14-7

Elevated SOF Contour Plot – Surface Stratum

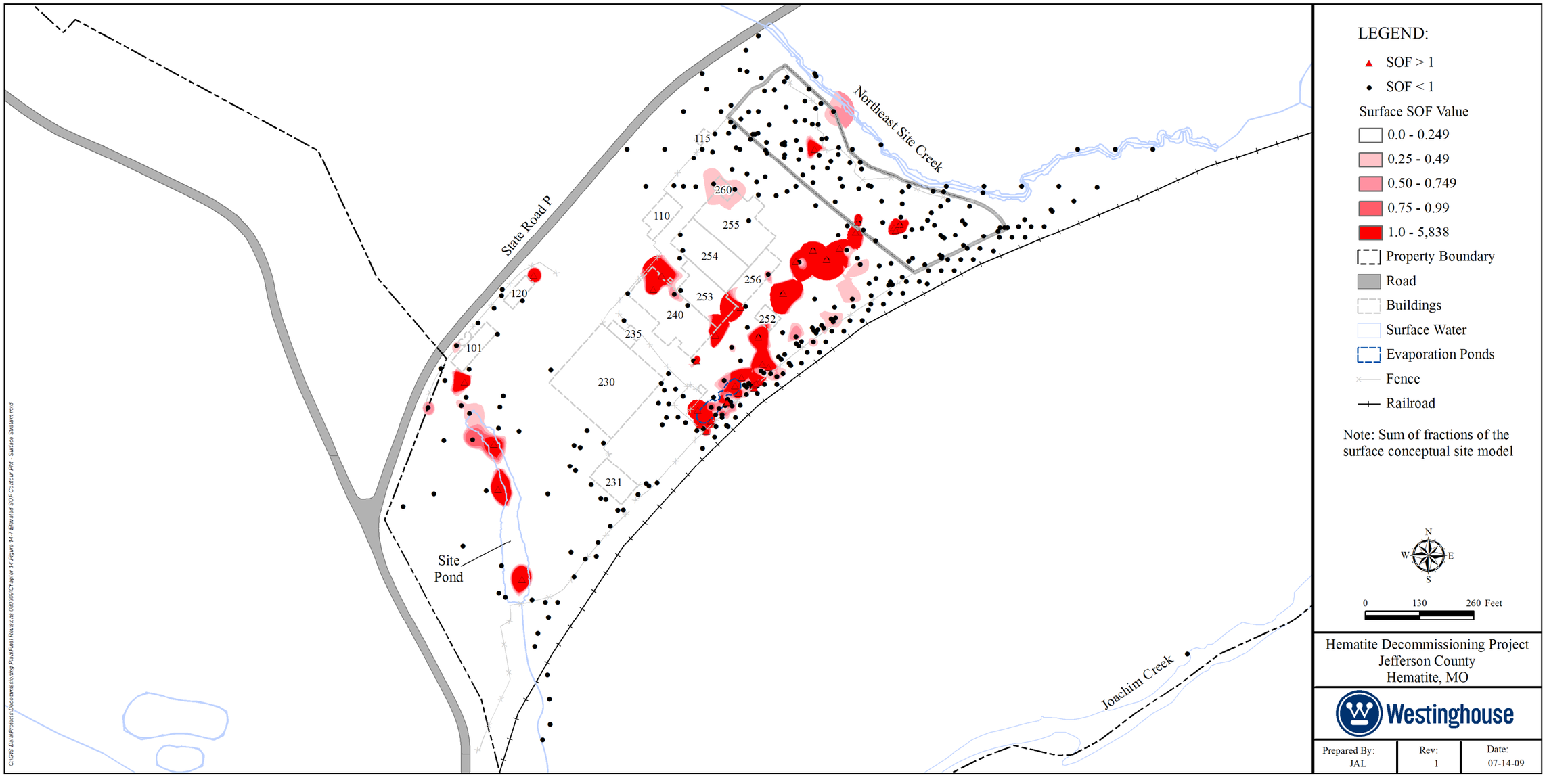


Figure 14-8

Elevated SOF Contour Plot – Root Stratum

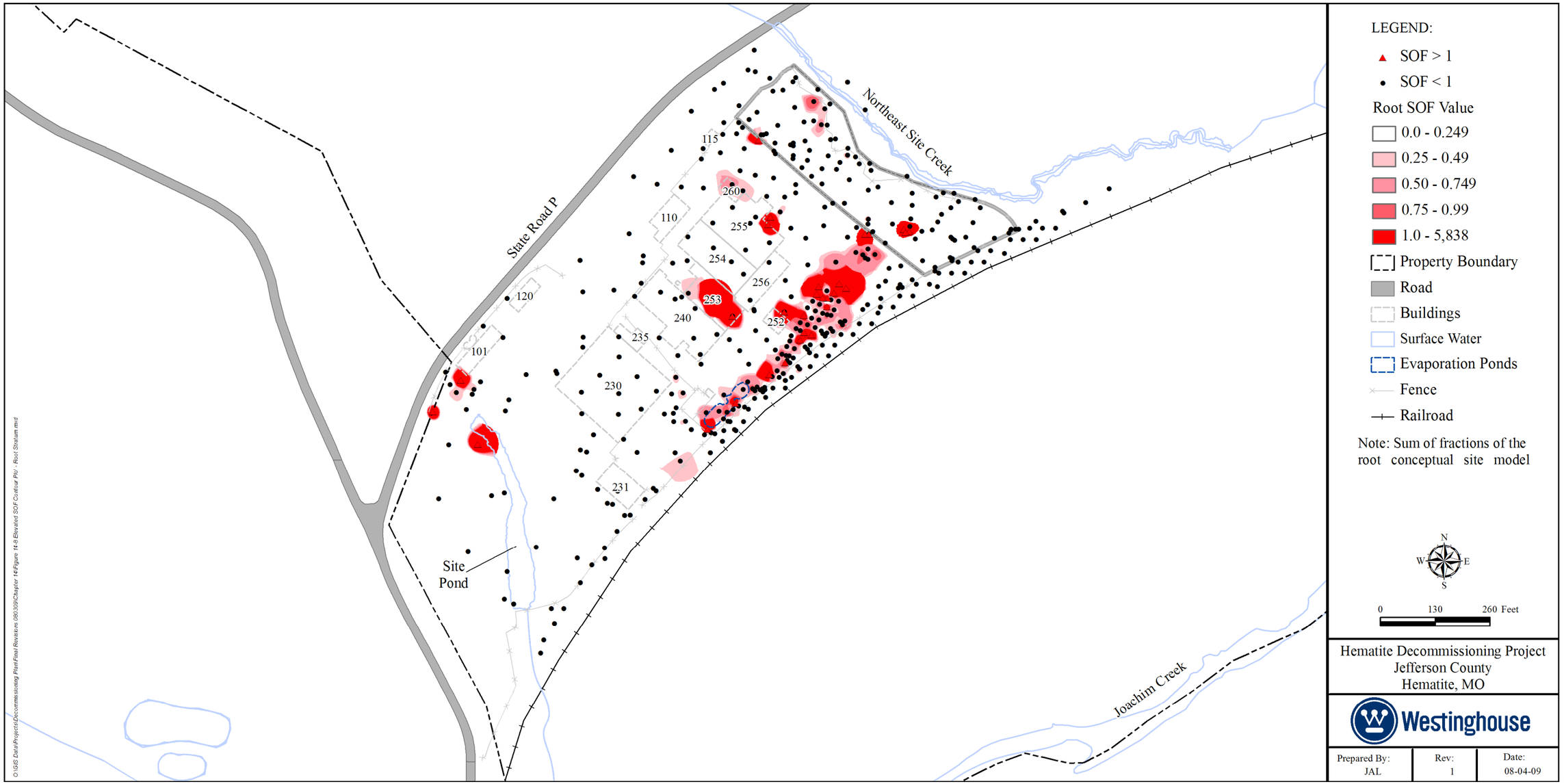


Figure 14-9

Elevated SOF Contour Plot – Deep Stratum

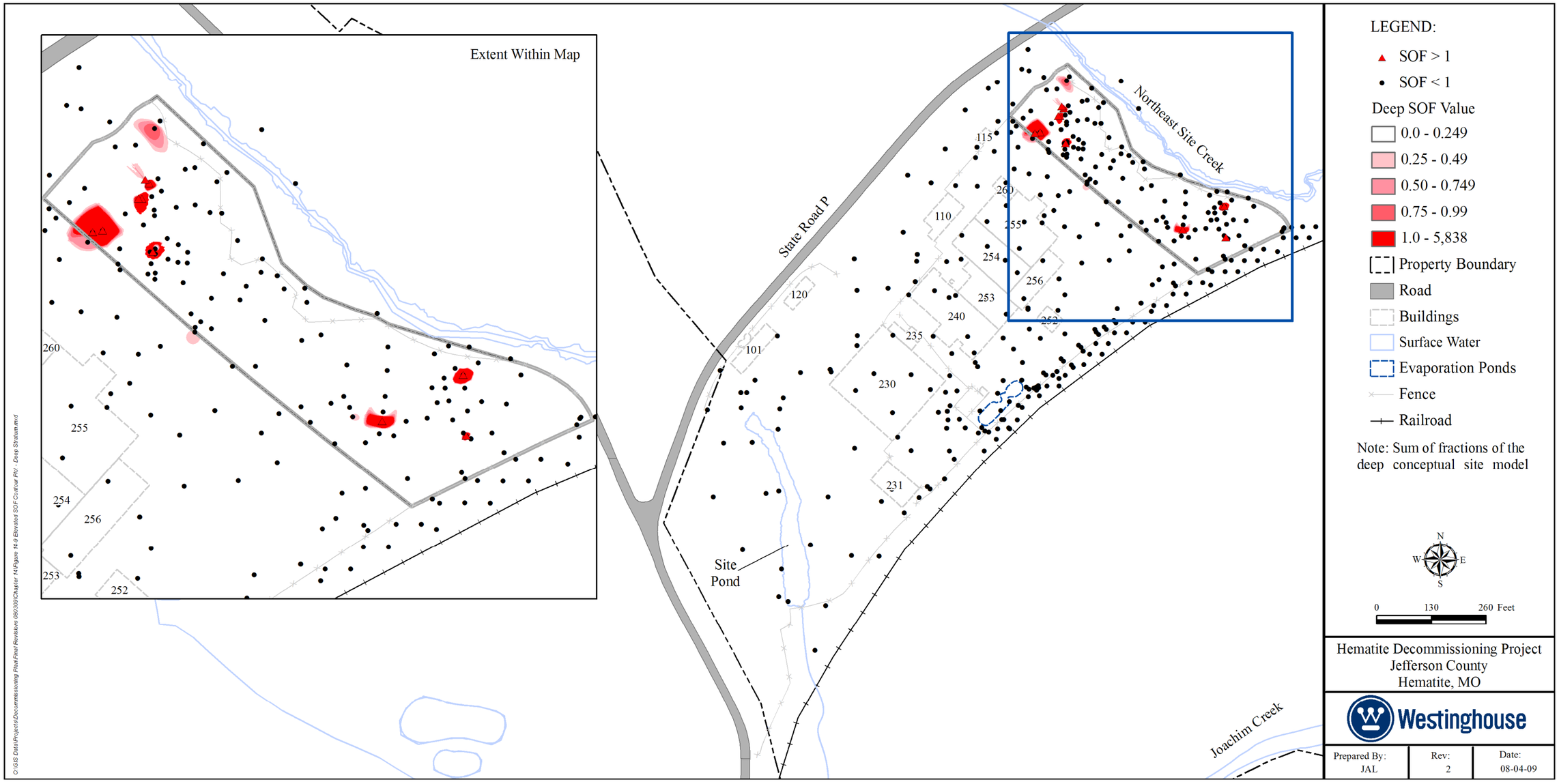


Figure 14-10

Elevated SOF Contour Plot – Total Strata

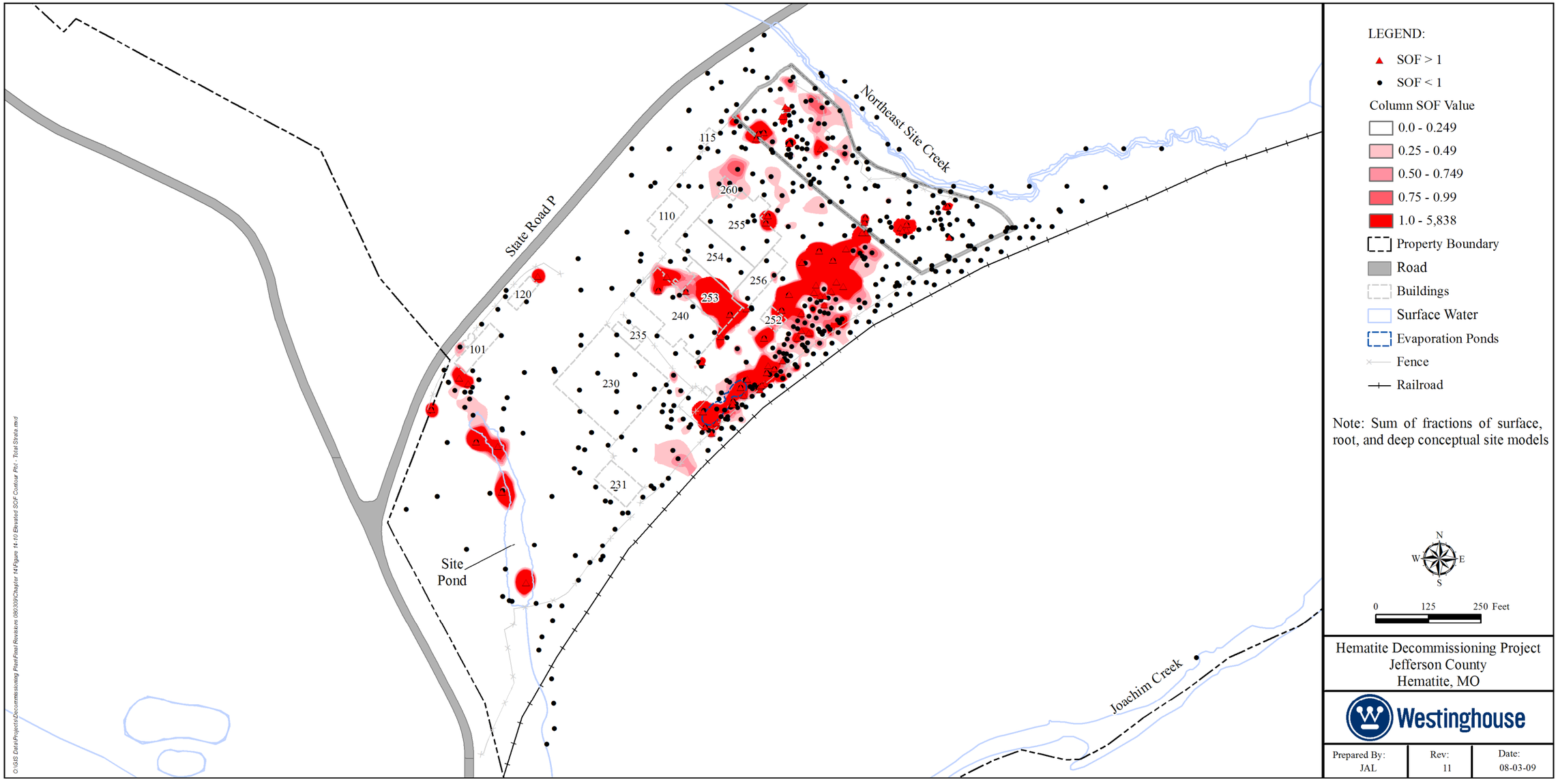


Figure 14-11

Impacted Area For FSS

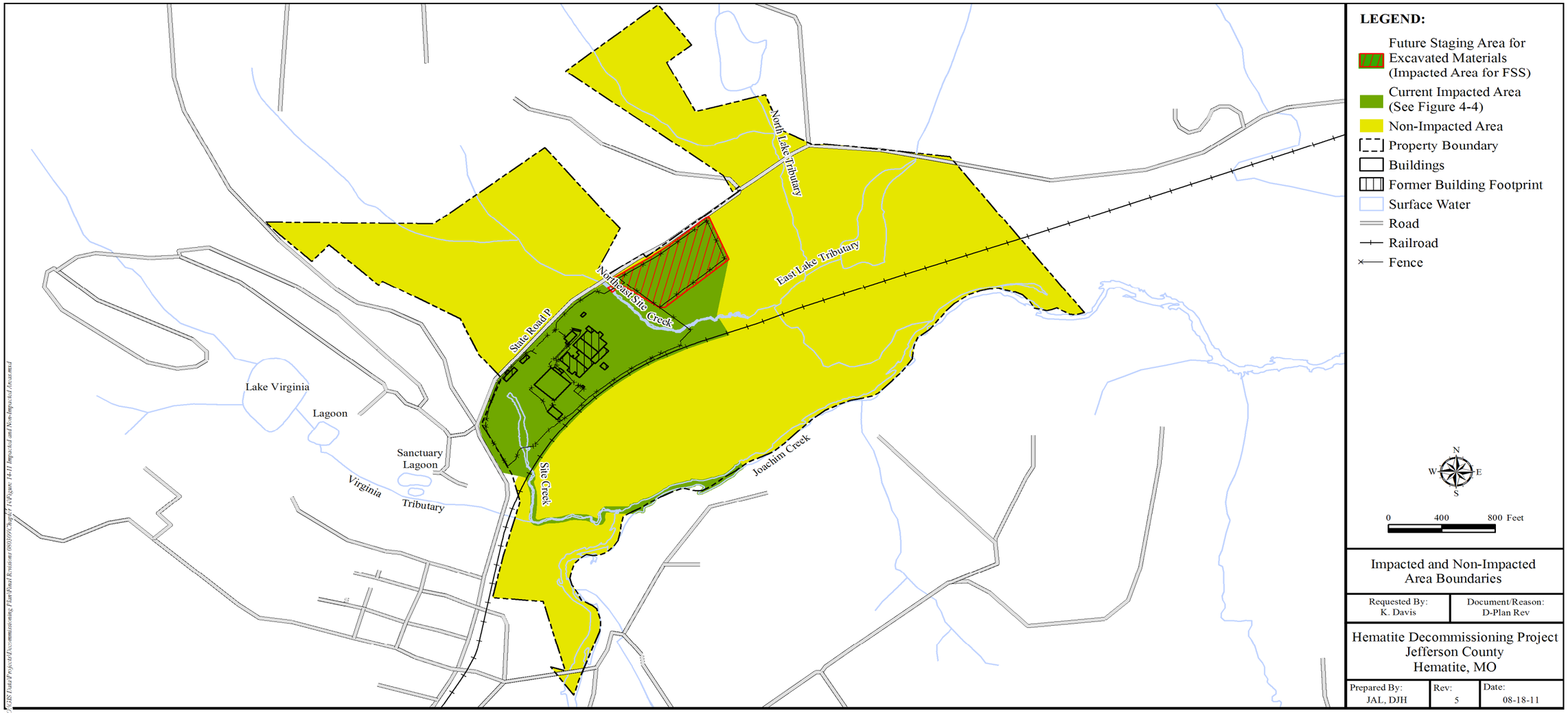


Figure 14-12

Initial Classification Of Impacted Soil Areas

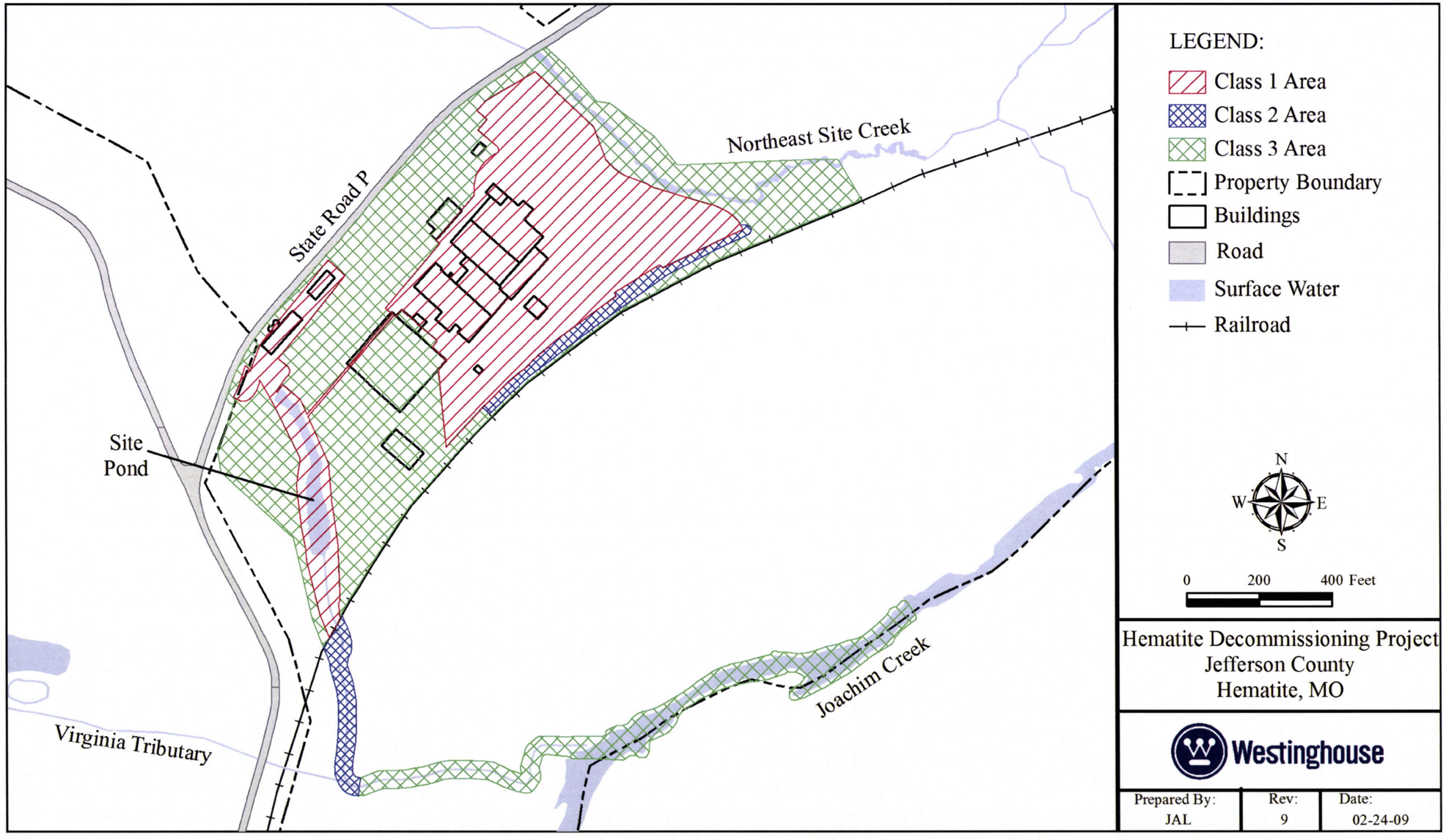


Figure 14-13

Site Survey Area Boundaries

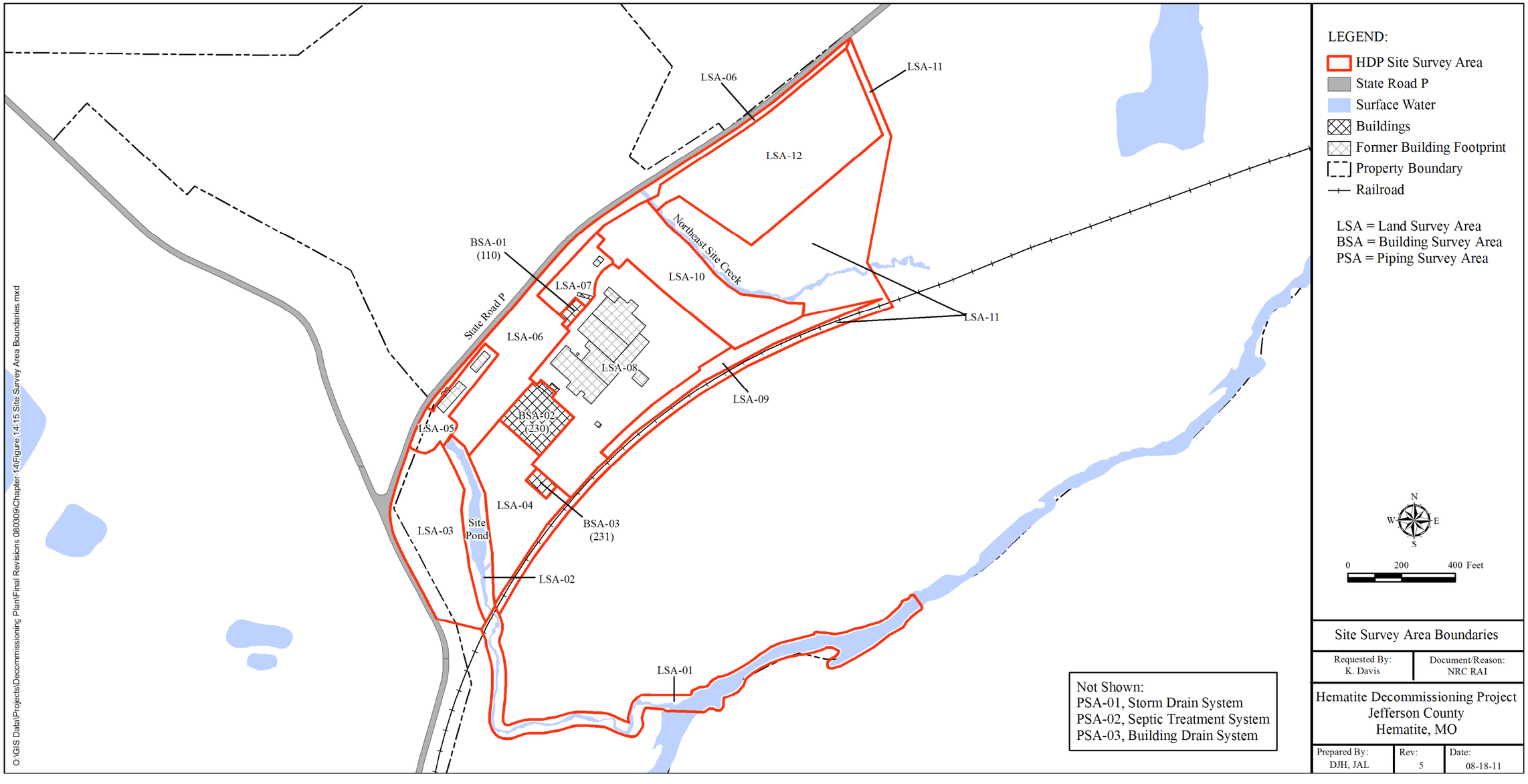


Figure 14-14

Conceptual Open Land Area Survey Units

Figure 14-15

Building 110 Survey Units

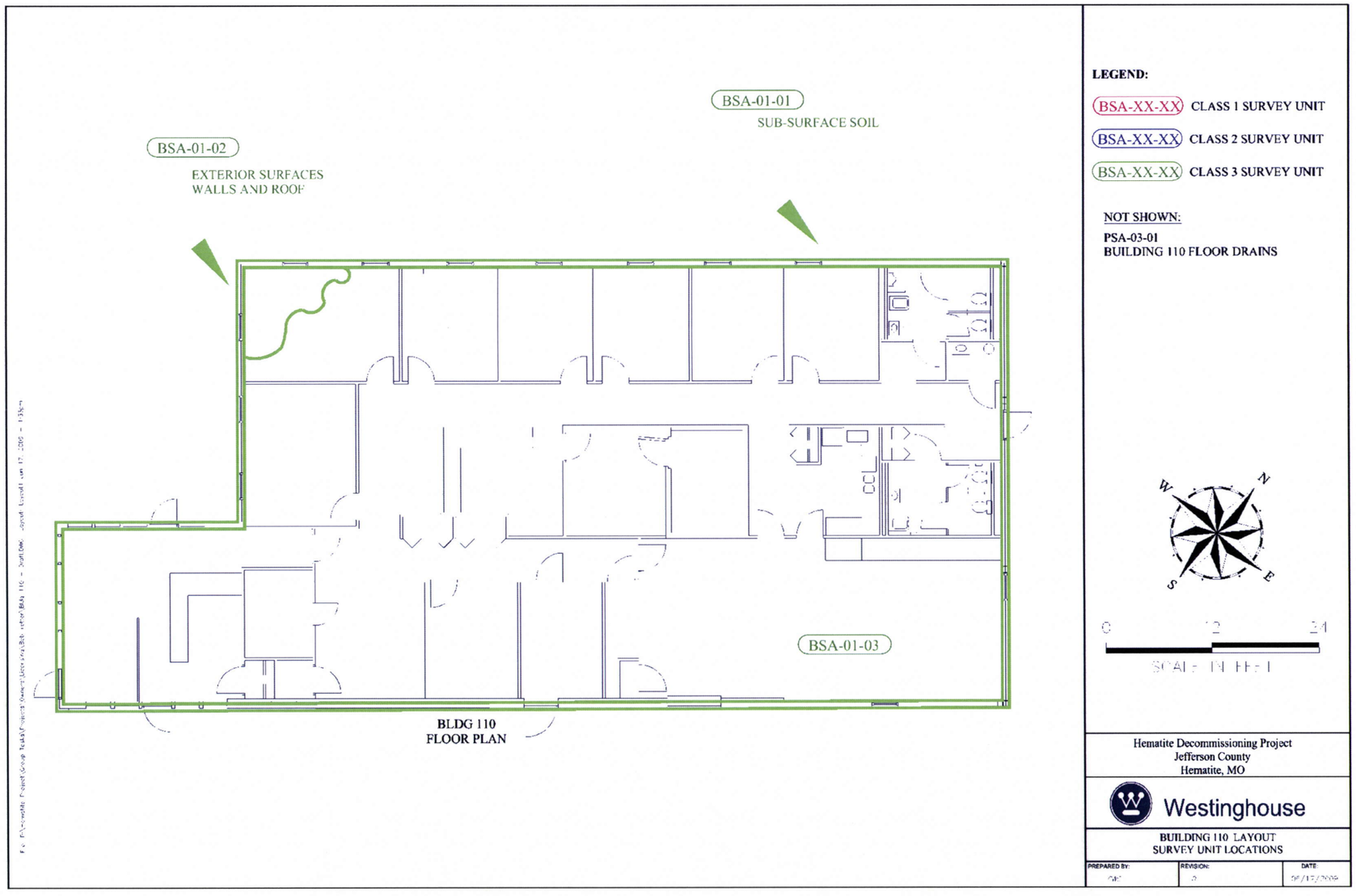


Figure 14-16

Building 230 Survey Units (Ground Floor)

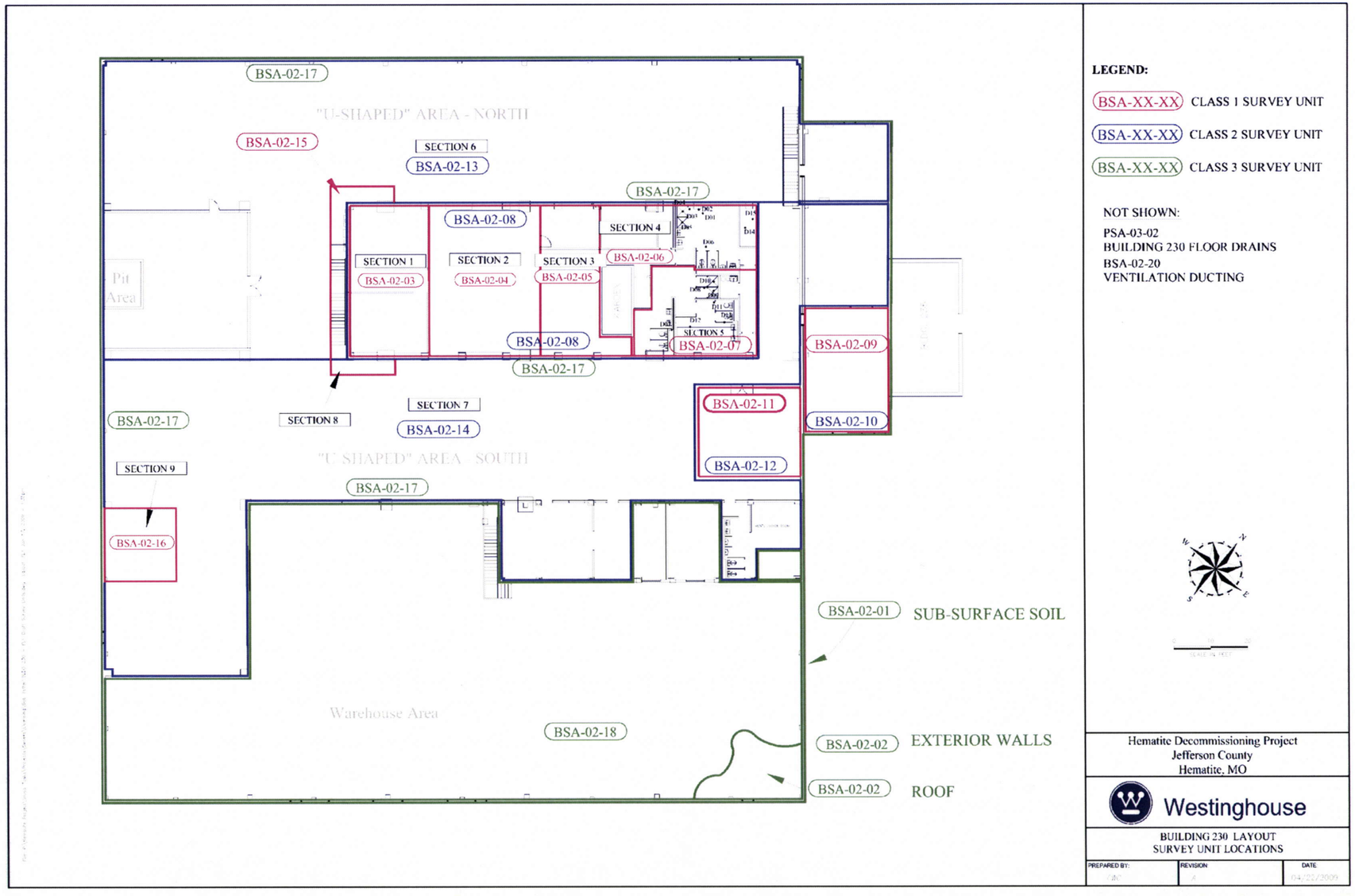


Figure 14-16 (continued)

Building 230 Survey Units (Mezzanine)

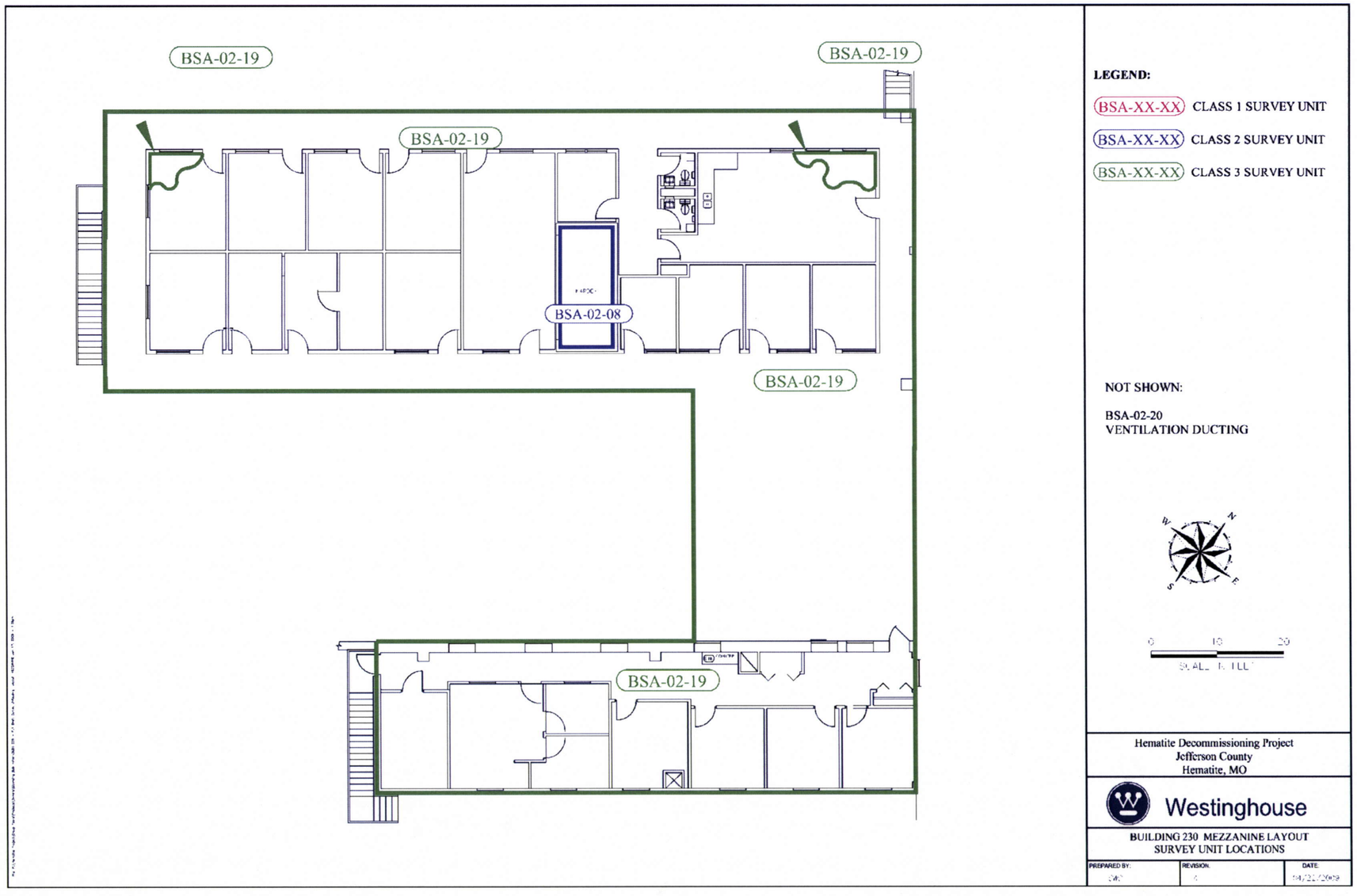


Figure 14-17

Building 231 Survey Units

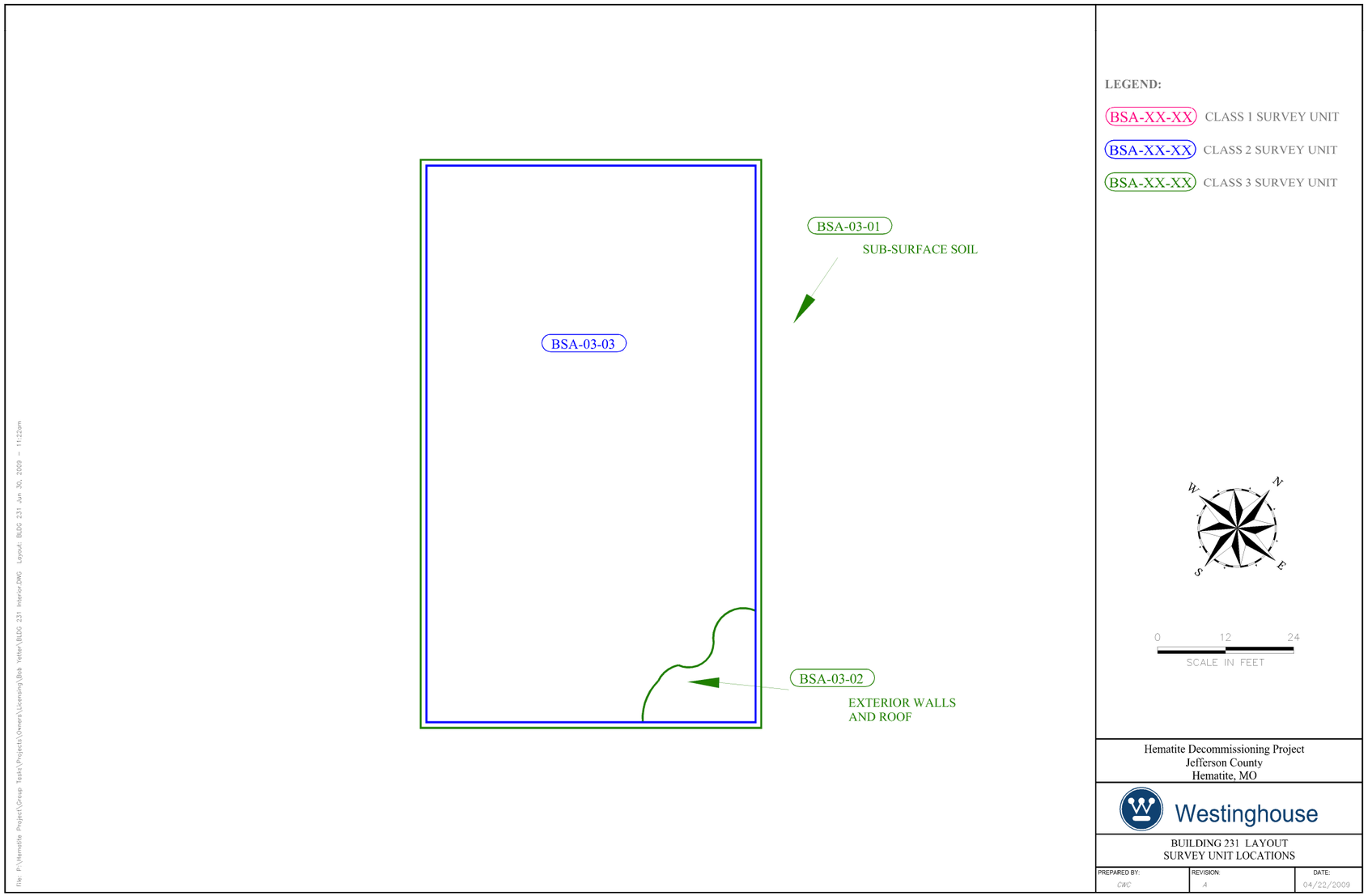


Figure 14-18

Plant Soil SEA Total Uranium Scan MDC vs. Total Uranium $DCGL_W$

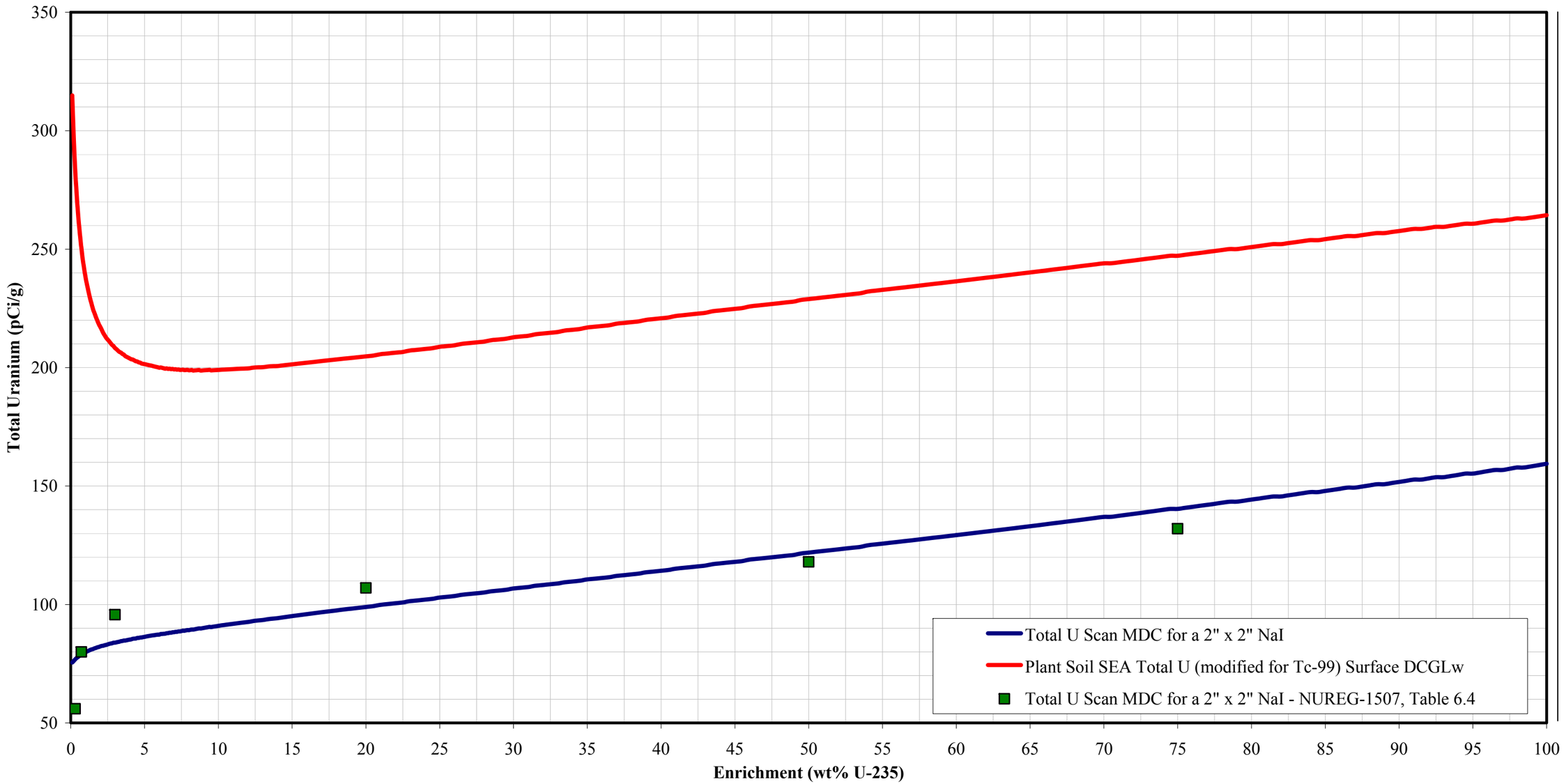


Figure 14-19

Tc-99 SEA Total Uranium Scan MDC vs. Total Uranium $DCGL_W$

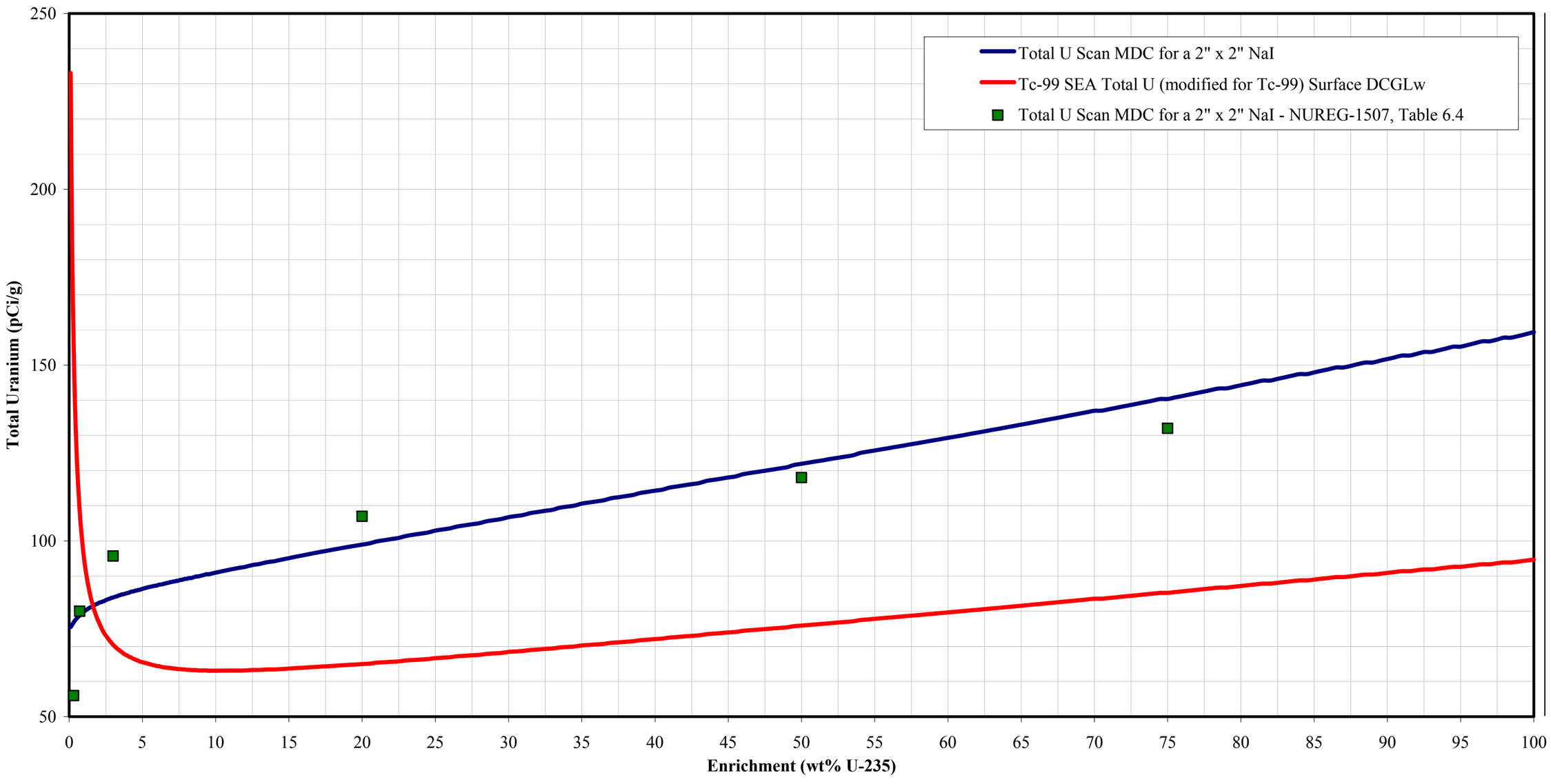


Figure 14-20

Burial Pit SEA Total Uranium Scan MDC vs. Total Uranium $DCGL_W$

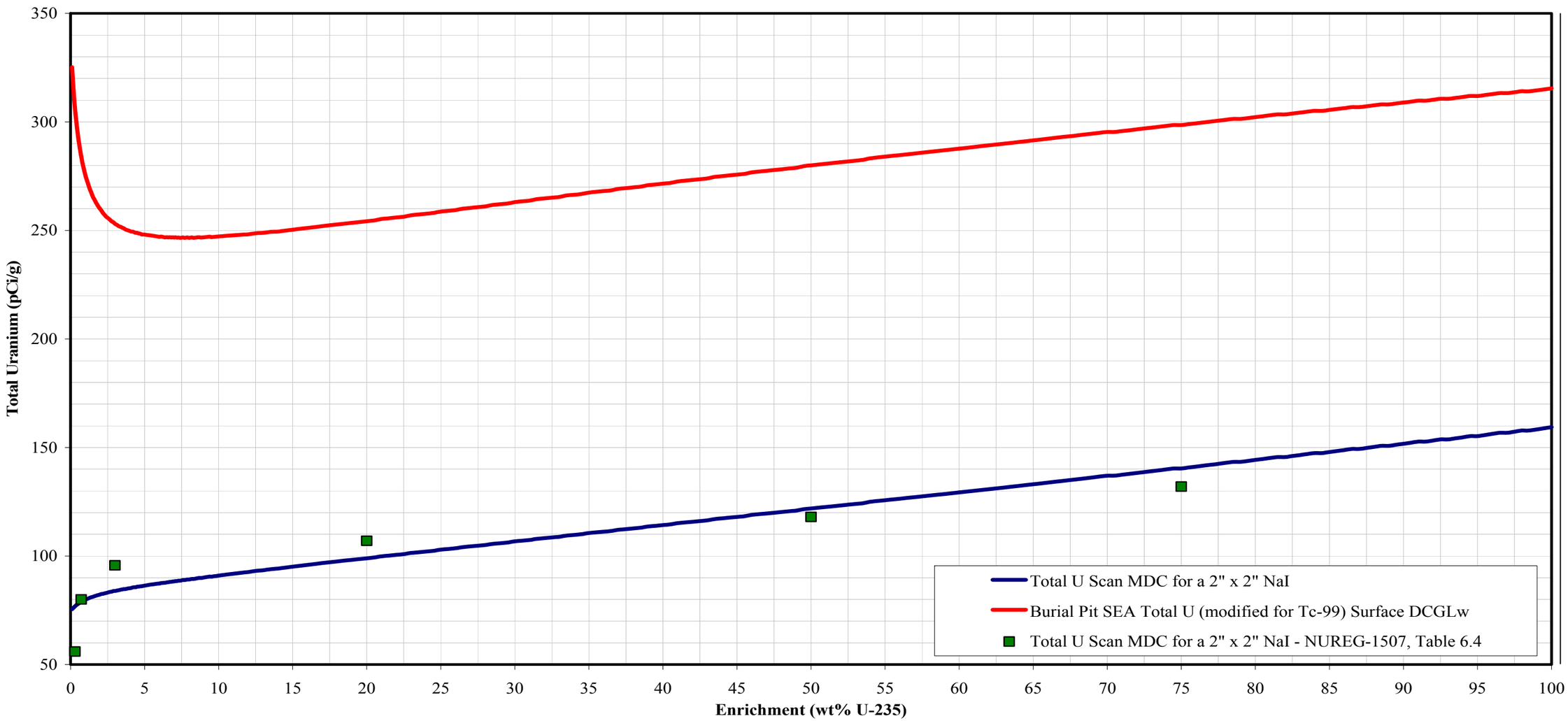


Figure 14-21

Post-Remediation Groundwater Monitoring Wells

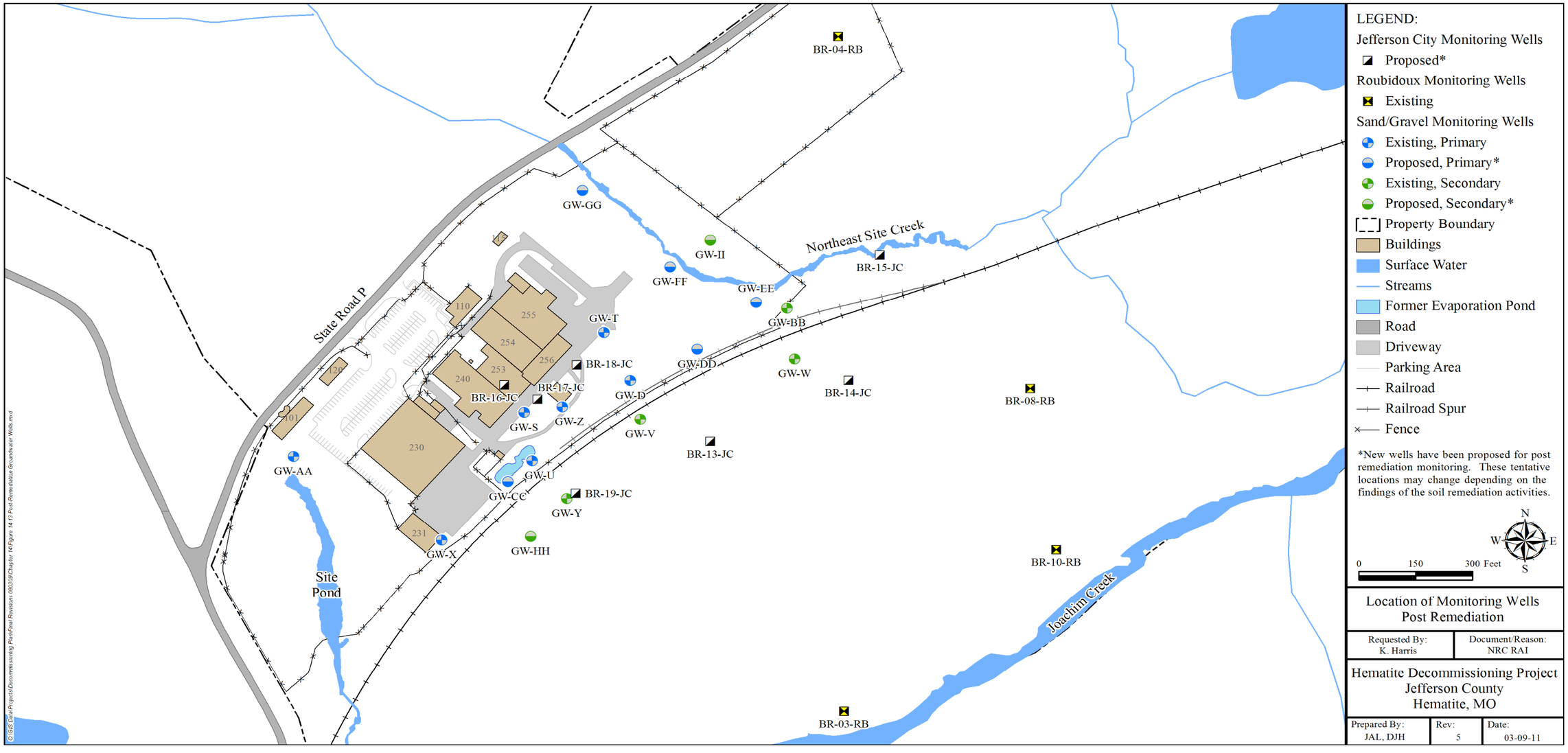


Figure 14-22

Conceptual Investigation Sample Stations Associated with the Process Buildings

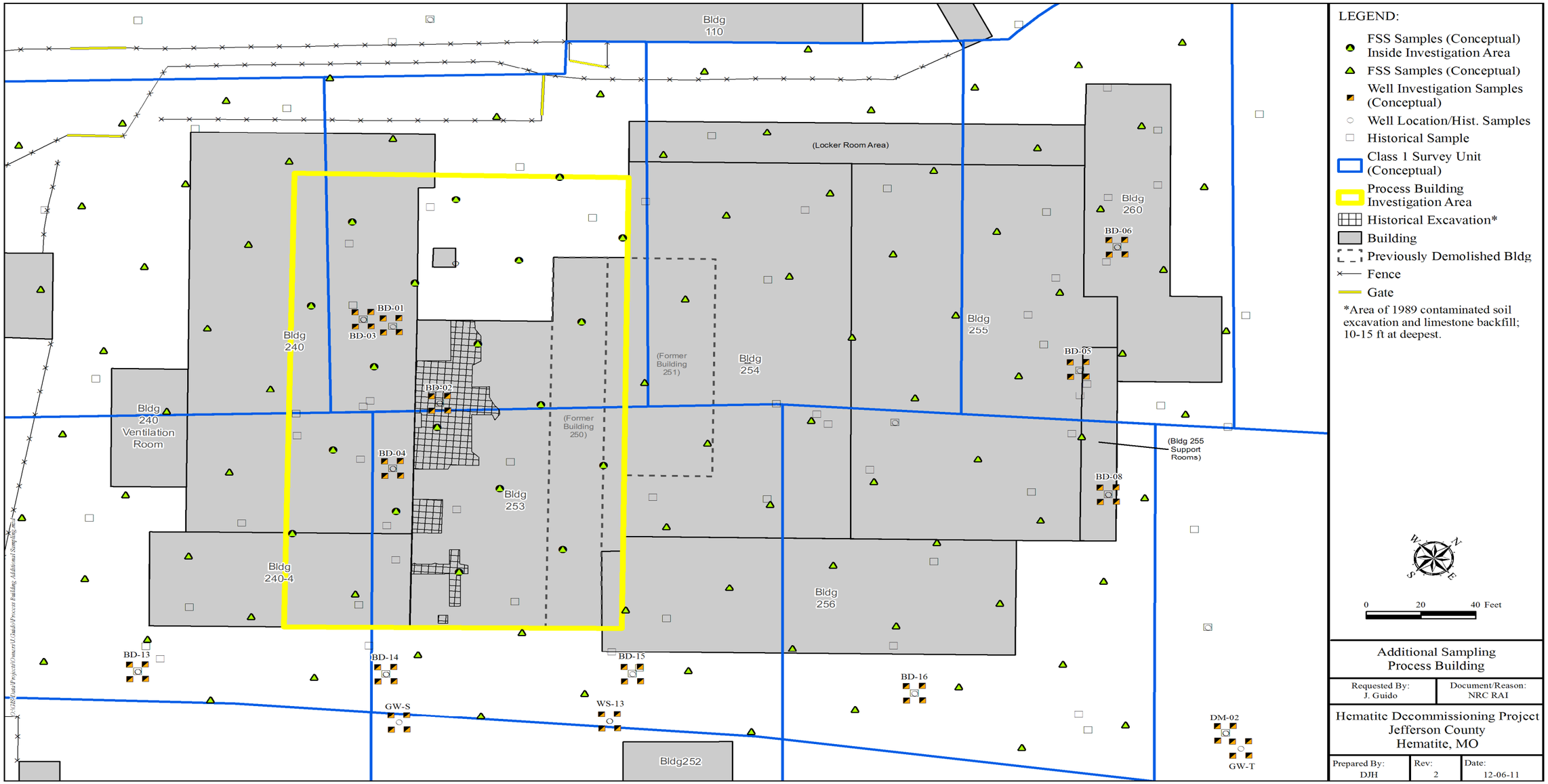


TABLE OF CONTENTS

ACRONYMS AND ABBREVIATIONS

CFR	Code Of Federal Regulations
NRC	U.S. Nuclear Regulatory Commission
SNM	Special Nuclear Material

15.0 DECOMMISSIONING COST ESTIMATE AND FUNDING PLAN

In accordance with Code of Federal Regulations (CFR) 10 CFR 70.38(g)(4)(v) (Reference 15-1) and consistent with NUREG-1757, Volume 3 (Reference 15-2), the Hematite Decommissioning Funding Plan (Reference 15-3) as incorporated into this Section provides:

1. An updated, detailed cost estimate for decommissioning;

2. A comparison of the estimated costs with present funds set aside for decommissioning; and

3. A plan for assuring the availability of adequate funds for completion of decommissioning.

As detailed further in the Hematite Decommissioning Funding Plan and summarized in this Section, the Hematite decommissioning cost estimate adequately reflects the costs to carry out required decommissioning activities prior to and through license termination.

15.1 DECOMMISSIONING COST ESTIMATE

This Section summarizes the basis for a site-specific cost estimating for the decommissioning of the Hematite Site. Consistent with current U.S. Nuclear Regulatory Commission (NRC) policy, the Hematite decommissioning cost estimate is intended to provide an estimate of "radiological decommissioning costs," which are those costs associated with radiological decommissioning necessary for termination of the NRC materials license and release of the site for unrestricted use, including costs associated with maintaining decommissioning financial assurance as required by 10 CFR 70.25(f). The Hematite Decommissioning Project (HDP) cost estimate does not include those costs associated with removal and disposal of nonradioactive structures and materials beyond that necessary for license termination.

15.1.1 COST ESTIMATING

The Hematite Decommissioning Funding Plan includes costs for labor, equipment and supplies, overhead and contractor profit, sampling and laboratory analysis, and miscellaneous expenses such as license fees, insurance, and taxes. A contingency factor of 25 percent is applied to the sum of the estimated radiological decommissioning costs (Reference 15-2). The cost estimate does not take credit for salvage value that might be realized from the sale of potential assets during or after decommissioning or reduced taxes that might result from payment of decommissioning costs or site control and maintenance costs.

The Hematite Decommissioning Funding Plan provides a summary of the estimated remaining decommissioning costs. The estimated costs include costs for the applicable major

decommissioning and site control and maintenance activities specified in Section A.3 of NUREG-1757, Volume 3, including: (a) planning and preparation, (b) decontamination and/or dismantling of facility components, (c) packaging, shipment, and disposal of radioactive wastes, (d) a final radiation survey, and (e) restoration of contaminated areas on site grounds. The cost of these activities are supplied in the following categories:

1. Planning and Preparation;
2. Decontamination and/or Dismantling of Radioactive Facility Components;
3. Building Demolition;
4. Restoration of Contaminated Areas on Facility Grounds;
5. Final Status Survey;
6. Site Stabilization and Long-Term Surveillance;
7. Packing Material Costs;
8. Shipping Costs;
9. Waste Disposal Costs;
10. Equipment/Supply Costs;
11. Laboratory Costs;
12. Miscellaneous Costs; and
13. Subcontractor Costs

15.1.2 COST ESTIMATE DESCRIPTION

The methodology used to develop the cost estimate is consistent with the approach presented in NUREG-1757, Volume 3. Specifically, the Hematite decommissioning cost estimate is developed based on documented and reasonable assumptions, and the estimated cost is sufficient to allow an independent third party to assume responsibility for decommissioning the facility. In addition, the cost estimate uses a unit cost factor method for estimating decommissioning costs. The unit cost factors are reasonable and consistent with NRC guidance, and incorporate site-specific considerations whenever practicable.

Quantities and volumes of material and waste to be removed during decommissioning have been estimated using facility records and the results of site characterization efforts. Unit cost factors were applied to the volumes and quantities to estimate the "activity dependent" costs. "Period dependent" costs were determined from a critical path schedule based on the removal activity duration.

The Hematite Decommissioning Funding Plan contains the three basic parts specified in NUREG-1757, Volume 3. Specifically, it contains: (1) a facility description; (2) the estimated decommissioning costs; and (3) key assumptions.

Westinghouse will periodically update the Hematite decommissioning cost estimate based on actual decommissioning progress. The cost estimate updates will reflect the most current knowledge with respect to staffing requirements and work/activity schedules, remaining scheduled decommissioning remediation efforts, and adjustments for current radioactive waste disposal volumes and rates. Pursuant to the requirements of 10 CFR 70.25(e), updates to the cost estimate and associated funding levels are performed at intervals not exceeding three years.

15.2 FINANCIAL ASSURANCE

Since Hematite decommissioning funds are obtained when needed (rather than by setting aside and accumulating funds for decommissioning), Westinghouse maintains financial assurance for radiological decommissioning costs via a Parent Company Guarantee as stipulated in 10 CFR 70.25(f)(2). Financial assurance for remaining radiological decommissioning activities is reviewed periodically to ensure that an adequate level of financial assurance is maintained (i.e., the Parent Company Guarantee is maintained in an amount at least equivalent to the estimated remaining decommissioning costs).

Pursuant to the requirements of 10 CFR 70.25(e), Attachment 1 of the Hematite Decommissioning Funding Plan contains a certification by Westinghouse, as licensee for Hematite License No. SNM-33, that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning. Also NRC (Hsia) letter to Westinghouse (Gutman), dated August 29, 2011, "Approval of Request to Use a Parent Company Guarantee as Financial Assurance for Decommissioning (TAC L33096)," (Reference 15-4) approves the Parent Company Guarantee per the requirements of 10 CFR 70.25(f)(2).

The Parent Company Guarantee is written consistent with NUREG 1757, Vol. 3, Appendix A, Section (A.10.4) and contains all the necessary terms and conditions identified in Checklist 10-B of NUREG 1757, Vol. 3, Appendix A. A standby trust fund has been established to receive funds from the Parent Company Guarantee. This standby trust fund satisfies the criteria described in Section A.17 and in Checklist 17–A of NUREG 1757, Vol. 3, Appendix A.

15.3 REFERENCES FOR CHAPTER 15.0

15-1 Code of Federal Regulations, Title 10, Part 70, "Domestic Licensing of Special Nuclear Material."

15-2 U.S. Nuclear Regulatory Commission, NUREG-1757, "Consolidated NMSS Decommissioning Guidance, Financial Assurance, Recordkeeping, and Timeliness." Volume 3, September 2003.

15-3 Westinghouse Electric Company Document No. DO-09-001, "Hematite Decommissioning Funding Plan," Revision 0.0.

15-4 NRC (Hsia) letter to Westinghouse (Gutman), dated August 29, 2011, "Approval of Request to Use a Parent Company Guarantee as Financial Assurance for Decommissioning (TAC L33096)".