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Subject: DECOMMISSIONING PLAN CHAPTER 14 REVISION 1.1 FOR THE HEMATITE DECOMMISSIONING PROJECT (License No. SNM-00033, Docket No. 070-00036)

References:
1) NRC Letter (McConnell) to Westinghouse (Hackmann), dated October 13, 2011, U.S. Nuclear Regulatory Commission Approval of: (1) Westinghouse Hematite Decommissioning Plan, (2) Revised License Application, (3) Exemption from the Requirements of 10 CFR 70.24 and 70.22(a), and Issuance of Hematite License Amendment 57.
2) Westinghouse (Rood) Letter HEM-11-162 to NRC (Document Control Desk) dated December 19, 2011, "Decommissioning Plan Revision 1.0 for the Hematite Decommissioning Project"

In Reference 1, the U.S. Nuclear Regulatory Commission (NRC) approved the Westinghouse Electric Company LLC (Westinghouse) Decommissioning Plan (DP) and supplemental information which contained Westinghouse's responses to the NRC's Requests for Additional Information (RAIs). Reference 2 submitted DP Revision 1.0 to the NRC to incorporate the supplemental information into the DP itself, and to provide clarifying technical changes. NRC's initial review identified inconsistencies between the RAIs and Reference 2. Westinghouse requested that no further NRC effort be expended on Reference 2. By this letter, Westinghouse formally retracts Reference 2.

In addition, Westinghouse and NRC determined that only DP Chapter 14 needed to be revised to incorporate responses to RAIs. Accordingly, Enclosure 1 contains *HDP Decommissioning Plan Chapter 14 Revision 1.1*. Enclosure 2, *HDP Decommissioning Plan Revision 1.1 with Track Changes and Source Document Citations for Changes*, is a version of DP Chapter 14 Revision 1.1 that shows changes from Revision 0, with comment boxes identifying the source document for the change or identifying typographical corrections. To support some of the changes, Enclosure 3, *Non-RAI DP Technical Changes Reviewed with NRC*, is provided to place on the docket a matrix of DP changes that were reviewed with NRC but were not part of responses to RAIs.

Westinghouse has exercised its authority per Section 1.8 of the License Application Request approved by Reference 1 to make limited modifications to the Decommissioning Plan. DP Section 14.6, *FSS Reporting*, has been modified to change the arrangement of the content of final status survey reporting, but without changing the content itself. Enclosure 4, *Decommissioning Plan Section 14.6 Changes Made in Accordance with Section 1.8 of the License Application Request*, contains the modified text to Section 14.6. These modifications are provided for information and are not reflected in Revision 1.1 since Westinghouse committed that Revision 1.1 would only include modifications based on RAIs.

Please contact Dennis Richardson at 314-810-3376, should you have questions or need additional information.

Respectfully,

Robert D. Copp

Robert D. Copp
Director, Hematite Decommissioning Project

Enclosures:
1) HDP Decommissioning Plan Revision 1.1
2) HDP Decommissioning Plan Revision 1.1 with Track Changes and Source Document Citations for Changes
3) Non-RAI DP Technical Changes Reviewed with NRC
4) Decommissioning Plan Section 14.6 Changes Made in Accordance with Section 1.8 of the License Application Request

cc: J. J. Hayes, NRC/FSME/DWMEP/DURLD/MD
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ENCLOSURE 1

HDP Decommissioning Plan Chapter 14
Revision 1.1

Westinghouse Electric Company LLC
Hematite Decommissioning Project

Docket No. 070-00036

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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)

Microshield®[1]	computer software program
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

[1] MicroShield® is a trademark of Grove Software, Inc., registered in the U.S. and other countries.

ACRONYMS AND ABBREVIATIONS
(Continued)

TEDE	Total Effective Dose Equivalent
Th	Thorium
U	Uranium
UF_6	Uranium Hexafluoride
V&V	Verification And Validation
WMW	Wilcoxon Mann Whitney
WRS	Wilcoxon Rank Sum

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.

Volumetric DCGLs have not been developed for buildings that are expected to remain at the time of license termination based on no evidence of volumetric contamination from process knowledge and analysis to date. Should volumetrically contaminated material be identified, it is anticipated that it will be removed and shipped for disposal prior to final status survey. However, if the material will remain, appropriate DCGLs will be developed and submitted to NRC for approval.

The criteria used to determine whether volumetric contamination exists are: (a) scan or static survey measurements identify surface contamination exceeding the DGCL and scarifying the surface fails to reduce the contamination level; or (b) scan surveys biased to the locations of cracks or seams in concrete surfaces identify elevated activity that is not attributed to the radiological condition of the surface (e.g., a discrete particle that can be removed using a vacuum cleaner or an exposed surface within the crack that can be accessed for decontamination using hand tools). Conditions other than those described above will require more intrusive methods to evaluation the radiological condition such as breaking and removing concrete or obtaining core samples of concrete and underlying soil.

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)

$$DCGL_{W-Adjusted} = DCGL_W \times \left(\frac{25\,mrem/yr - 0.13\,mrem/yr}{25\,mrem/yr} \right) = DCGL_W \times 0.99$$

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.

$$\frac{1}{\frac{8.27E\text{-}1}{20{,}000}+\frac{3.72E\text{-}2}{19{,}000}+\frac{1.27E\text{-}1}{21{,}000}+\frac{2.83E\text{-}3}{1.3E7}+\frac{3.21E\text{-}3}{1{,}200}+\frac{5.57E\text{-}5}{2{,}700}+\frac{2.03E\text{-}6}{3{,}500}+\frac{2.68E\text{-}3}{3{,}400}} = 18{,}925 \text{ dpm}/100 \text{ cm}^2 \qquad (14\text{-}2)$$

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)

$$DCGL_{U-235,\mathrm{mod}} = \frac{1}{\left(\frac{1}{102.3}\right) + \left(\frac{9.24}{151.0}\right)} = 14.1 \text{ pCi/g}$$

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)

$$DCGL_{w,TotU}\,(pCi/g) = \frac{1}{\dfrac{f_{U\text{-}234}}{DCGL_{w,U\text{-}234}} + \dfrac{f_{U\text{-}235}}{DCGL_{w,U\text{-}235}} + \dfrac{f_{U\text{-}238}}{DCGL_{w,U\text{-}238}}}$$

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 4-10 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 modified U-235 DCGL values shown in Tables 14-9 and

14-10 (columns titled "Inferred Tc-99) are prohibited from use 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.

Note: Equations 14-9, 14-10, 14-13 and 14-14 were deleted in Revision 1.1.

(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)

$$\sigma_{SOF} = \sqrt{\left(\frac{\sigma_{U\text{-}234}}{DCGL_{w,U\text{-}234}}\right)^2 + \left(\frac{\sigma_{U\text{-}235}}{DCGL_{w,U\text{-}235}}\right)^2 + \left(\frac{\sigma_{U\text{-}238}}{DCGL_{w,U\text{-}238}}\right)^2 + \left(\frac{\sigma_{Tc-99}}{DCGL_{w,Tc-99}}\right)^2 + \left(\frac{\sigma_{Th-232}}{DCGL_{w,Th-232}}\right)^2 + \left(\frac{\sigma_{Ra-226}}{DCGL_{w,Ra-226}}\right)^2}$$

14.1.5.1.2 Sample – Sample Start Depth >1.5

For samples obtained at a depth > 1.5 m, the SOF will be calculated from the radioactivity concentrations (in pCi/g) of U-234, U-235, U-238, 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 Equation 14-11. The weighted sigma value is calculated using Equation 14-12.

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 1.1. 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 14-12a. Table 14-12a 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 AFs that will apply to soil below 1.5 m will be based on the Excavation DCGL if solely below 1.5 m (Table 14-12c); or will be based on the Uniform if a portion of the soil being evaluated is above 1.5 m (Table 14-12a). (Note that Table 14-12b represents an intermediate step in developing Table 14-12c).

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 $DCGL_{EMC}$:

(14-16)

$$DCGL_{EMC} = AF \times DCGL_W$$

The following equation was used to define the calculation of the $DCGL_{EMC,\ TotU}$ using Equation 14-8 and Equation 14-16:

(14-17)

$$DCGL_{EMC,TotU} = \frac{1}{\left(\dfrac{f_{U\text{-}234}}{AF_{U\text{-}234} \times DCGL_{w,U\text{-}234}} + \dfrac{f_{U\text{-}235}}{AF_{U\text{-}235} \times DCGL_{w,U\text{-}235}} + \dfrac{f_{U\text{-}238}}{AF_{U\text{-}238} \times DCGL_{w,U\text{-}238}} \right)}$$

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 14-12a or c;

$AF_{U\text{-}235}$ = AF for U-235 from Table 14-12a or c;

$AF_{U\text{-}238}$ = AF for U-238 from Table 14-12a or c;

$DCGL_{W,\ U\text{-}234}$ = 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 14-4 or Table 14-10 for all strata (pCi/g) – Table 14-4 applies to FSS since the "Inferred Tc-99" columns in Table 14-10 are prohibited from FSS use per Section 14.1.4.3.5; 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 14-12a.

(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.

Laboratories contracted to HDP are initially and periodically evaluated by Qualified Lead Auditors and Technical Specialists. The evaluations of laboratory QA/QC programs include: onsite audits for initial evaluation and on a triennial basis; as well as an annual Supplier Audit Evaluation to identify major changes to their quality program. Independent third party certifications of these laboratories, such as NELAP, NVLAP, and ISO 9001:2000 are also considered during their evaluation. Maintenance of applicable accreditations is imposed as a quality requirement on the purchase order.

Following receipt of laboratory data, HDP staff perform a data review to assess the validity of the data for use in the final status survey. This review includes an evaluation of the data to ensure that all of the data quality objectives (DQOs) have been met.

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 and total uranium from non-impacted areas were indistinguishable from the background data. The data, results, and ProUCL outputs for the Wilcoxon-Mann-Whitney and Quantile tests are provided in Table 14-25 and Figure 14-23.

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 Survey Methodologies During Removal Of Surface Contaminated Soil

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:

Approach 1:

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.

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 building surfaces (small office); 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.

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). 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). 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.

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.

(14-20)

$$\Delta = DCGL_w - LBGR$$

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.

(14-21)

$$\text{Relative Shift} = \frac{\Delta}{\sigma}$$

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 14-12a or c 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 a or c 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:

(14-27)

$$L = \sqrt{\frac{A}{0.866N}} \quad \text{for a triangular grid, or}$$

(14-28)

$$L = \sqrt{\frac{A}{N}} \quad \text{for a square grid}$$

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. Composite samples will be collected as follows:

- 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.

Soil Sample ID EP-13-30-SL

- To resolve a question regarding the potential vertical and lateral extent of Tc-99 around sample ID EP-13-30-SL, Westinghouse will obtain additional samples to determine the Tc-99 concentration in soil in the vicinity of this sample following the excavation of the evaporation pond area. If the soil represented by this sample is not excavated during remediation of the evaporation ponds, then subsurface samples within the excavation will be collected at depth within the vicinity of the original sampling location to determine whether Tc-99 contamination is present.

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.

The final status survey report will include the investigation sample results.

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 Cracks/Crevices, Wall-Floor Interfaces And Small Holes

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. The Final Status Survey will include biased measurements/sampling of cracks and interfaces between floors and walls. If volumetric contamination were present, core samples of the concrete would be obtained for laboratory analysis. 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. For piping that HDP has decided will remain in place after site closure, the final status survey method will be submitted for NRC review and approval, with approval received prior to implementation of final surveys of piping.

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.

To verify that buried piping that will remain after Site closure has not contaminated surrounding soil, HDP will utilize biased core bore samples through building slabs to evaluate soils adjacent to buried piping against appropriate DCGLs. Factors for determining biased location decisions will include location of pipe joints, low points, and any survey or video evidence available from the buried piping.

Also, the location of the decommissioning water treatment system within Building 230 will be included in the Final Status Survey, and the survey design will consider the potential for migration to sub-grade soil in the event that a leak should occur and the secondary containment does not effectively contain the spill.

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 (1.5 m 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. The criteria for terminating excavation are:

- Removal of buried debris/wastes;
- Removal of spent limestone;
- In-process surveys and sampling activities indicate the applicable DCGLs have been met, and compliance is confirmed by a successful final status survey (DP Chapter 14), including the additional sampling activities described in DP Section 14.4.3.4; and
- In process sampling activities indicate the applicable chemical RGs have been met. (Note: The excavation may terminate without achieving the chemical RGs for volatile organics if the excavation reaches the groundwater table as defined by the saturated zone.).

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).

- <u>Final Evaluation of Residual Radioactivity in Soil Following Backfill</u>

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.

- <u>Final Evaluation of Residual Radioactivity in Soil for Unpaved Non-excavated Areas or Excavated Areas not Requiring Backfill</u>

  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, a composite sample from 1.5 m to an appropriate depth (Deep stratum).

14.4.4.1.6.4 Groundwater

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.

(14-28a)

$$f_{GW} = \frac{Dose_{GW}}{25}$$

$$Dose_{GW} = MaxC_{GW}^{U-234} DSR_{GW}^{U-234} + MaxC_{GW}^{U-235} DSR_{GW}^{U-235} + MaxC_{GW}^{U-238} DSR_{GW}^{U-238} + MaxC_{GW}^{Tc-99} DSR_{GW}^{Tc-99}$$

Where: $MaxC_{GW}$ = the largest positive result, above background, in Sandy/Gravel, Jefferson City/Cotter or Roubidoux HSUs for that radionuclide (pCi/L). For the uranium isotopes, the sample's Total Uranium must exceed 8.6 pCi/L before any of that sample's isotopic results are considered "above background."

DSR_{GW} = the dose to source ratio for that radionuclide from DP Table 5-14, Groundwater DSRs (mrem/yr per pCi/L)

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. Final status survey procedures and associated training lesson plans specific to the performance of final status surveys will include calculation of a weighted efficiency as detailed in Chapter 14 of the DP. Training to these procedures will be administered to technicians prior to the implementation of final status survey, and subsequent changes to these procedures will be reviewed with technicians prior to implementation.

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)

$$\mathrm{MDC}\,(\mathrm{dpm/100cm^2}) = \frac{3 + 3.29\sqrt{(R_b)(T_s)\left(1 + \frac{T_s}{T_b}\right)}}{\frac{A}{100cm^2}(\varepsilon_t)(T_s)}$$

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)

$$MDC = \frac{3 + 3.29 \times \sqrt{300 \times 1 \times \left(1 + \frac{1}{1}\right)}}{0.16 \times 1 \times \left(\frac{126}{100}\right)} = 415\ \mathrm{dpm/100\,cm^2}$$

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 is best represented by the following equation:

(14-31)

$$MDC\ (pCi/g) = \frac{2.71 + 4.65\sqrt{B}}{(K)(W)(t)}$$

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)
	t	=	Count time (minutes)
	2.71	=	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 (inferred from the Ac-228 TAP), and Ra-226 (inferred from Pb-214/Bi-214). 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:

(14-32)

$$s_i = d'\sqrt{b_i}$$

(14-33)

$$MDCR = s_i\left(\frac{60}{i}\right)$$

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:

(14-34)

$$Scan\ MDC = \frac{MDCR}{\sqrt{p}\,(\varepsilon_t)\left(\frac{A}{100\ cm^2}\right)}$$

where: *MDCR* = minimum detectable count rate (cpm)

ε_t = Total efficiency (c/d),

p = Surveyor efficiency (unitless, typically assumed to be 0.5);

A = detector 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®, 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® 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® provided the exposure rate for a number of gamma energies associated with each input source term.

c. *Calculation of the MDER*

The MicroShield® 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.

(14-37)

$$MDER_{U-234} = \frac{1{,}512\ cpm}{10{,}699\ cpm\ per\ \mu R/h} = 0.14\ \mu R/h$$

(14-38)

$$MDER_{U-235} = \frac{1{,}512\ cpm}{4{,}991\ cpm\ per\ \mu R/h} = 0.30\ \mu R/h$$

(14-39)

$$MDER_{U-238} = \frac{1{,}512\ cpm}{3{,}554\ cpm\ per\ \mu R/h} = 0.43\ \mu R/h$$

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)

$$Scan\ MDC_{Total\ Uranium} = \frac{1}{\dfrac{0.9251}{7{,}383\ pCi/g} + \dfrac{0.0462}{4.9\ pCi/g} + \dfrac{0.0287}{62.8\ pCi/g}} = 99.0\ pCi/g$$

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. 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. 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 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 14-11, 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: Item 14.4.4.2.9e from "Non-RAI DP Technical Changes Review with NRC" Attachment 2 to HEM-12-X.

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 a or c 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)

$$f_{EMC} = \sum_{j=1}^{x} \sum_{i=1}^{y} \frac{\left(\tau_{i,j} - \delta_j\right)}{AF_{i,j} \times DCGL_{w_j}}$$

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, including the contribution due to groundwater per Section 14.4.4.1.6.4, the SOF is applied using the results of Equation 14-34 and Equation 14-35 as follows.

(14-47)

$$\mathrm{f_{Avg} + f_{EMC} + f_{GROUNDWATER}} \leq 1$$

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 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 for radionuclides will focus on monitoring lateral migration in the Sand/Gravel Hydrostratigraphic Unit (HSU) at the base of the silty clay overburden and vertical seepage to the Jefferson City-Cotter and Roubidoux HSUs. 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. A Mann-Kendall analysis will be performed quarterly on each of the wells to evaluate trends in sample results.

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 (concentration above the MDA + Error) 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 (concentration above the MDA + Error) 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 (concentration above the MDA + Error) 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 (concentration above the MDA + Error) 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.

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. For example, assuming the Burial Pit remediation is completed during the second quarter of 2013, the post remediation monitoring wells for that area will be installed and developed during the third quarter of 2013, and sampled for laboratory analysis in the fourth quarter of 2013, even if remediation work is on-going elsewhere on-site.

Post-remediation monitoring wells will be sampled quarterly after the completion of remediation until license termination. The comparators for determining suitability for unrestricted use and license termination are the results of sequential quarterly sampling that show that the contribution to dose from the sum of all licensed radionuclides do 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 (See Section 14.4.4.1.6.4 above).

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.

14.6 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;
- 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, comparison of findings with the appropriate DCGL, and dose contribution from groundwater;

- 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).

- Appendix C: Post-remediation groundwater monitoring results and evaluation.

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[d]	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.
[d] The Deep Stratum DCGLs in this table shall not be used. As an ALARA measure, the Excavation DCGLs will be applied to soil at all depths below 1.5 m.

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[d]	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.
[c] "+ C" = plus the entire decay chain (progeny) in secular equilibrium.
[d] The Deep Stratum DCGLs in this table shall not be used. As an ALARA measure, the Excavation DCGLs in this table will be applied to soil at all depths below 1.5 m.

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 modified 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	DCGLW (pCi/g) By Conceptual Site Model									
	Surface Soil		Root Stratum		Deep Volumetric [a]		Uniform [b]		Excavation [a]	
	Measure Tc-99	Infer Tc-99 [d]	Measure Tc-99	Infer Tc-99 [d]	Measure Tc-99	Infer Tc-99 [d]	Measure Tc-99	Infer Tc-99 [d]	Measure Tc-99	Infer Tc-99 [d]
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.

d These values modified to infer Tc-99 are prohibited from use to demonstrate compliance with the final status survey dose criteria.

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 14-12a

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
Np-237 + D	1.0	1.1	1.1	1.1	2.6	4.5	7.1	11.0	23.4	52.4
Pu-239/240	1.0	1.1	1.1	1.1	3.6	9.5	23.5	43.0	65.5	83.4
Am-241	1.0	1.0	1.1	1.1	2.9	5.6	9.4	13.9	25.4	42.4
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	1.1	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
Np-237 + D	1.0	1.0	1.0	1.0	3.4	9.9	30.7	57.2	132.0	298.4
Pu-239/240	1.0	1.0	1.0	1.0	3.4	9.8	29.1	68.4	137.7	207.4
Am-241	1.0	1.0	1.0	1.0	3.1	7.8	17.4	31.0	62.2	109.8
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
Np-237 + D	1.0	1.7	4.7	9.7	31.0	84.0	221.3	425.7	981.7	2,218
Pu-239/240	1.0	1.0	1.0	1.0	3.4	9.8	29.1	68.4	137.7	207.3
Am-241	1.0	1.0	1.0	1.0	3.1	7.8	17.4	31.0	62.1	109.7

Table 14-12b

Calculated Area Factors Based On Excavation Scenario Constraints 1 And 2

Radionuclide	**Area Factor Based on Contiguous Elevated Area after Excavation (size of elevated area shown in m²)***					
	148	**100**	**30**	**10**	**3.0**	**1.0**
U-234	1.0	4.0	12	19	35	65
U-235 + D	1.0	1.3	2	2	4	7
U-238 + D	1.0	1.9	3	4	7	13
Tc-99	1.0	4.2	14	42	140	410
Th-232 + C	1.0	1.9	3	4	7	14
Ra-226 + C	1.0	2.3	4	5	10	20
Np-237 + D	1.0	3.6	9	17	37	79
Pu-239/240	1.0	4.1	13	32	71	117
Am-241	1.0	3.6	9	17	32	58
	Area Factor Based on Elevated Area being Uniformly Mixed after Excavation					
Any	1.0	2.0	6.7	20	67	200

*Note - An adjustment factor of 1.5/0.9 was applied during modeling for geometrical transformation between the excavation (200 m2 x 3 m) and modeled (700 m2 x 0.9 m) geometry.

Table 14-12c
Effective Area Factor For Use With Excavation DCGLs

Radionuclide	**Size of elevated area shown in m²**					
	148	**100**	**30**	**10**	**3**	**1**
U-234	1.0	<u>2.0</u>	<u>6.7</u>	19	35	65
U-235 + D	1.0	1.3	2	2	4	7
U-238 + D	1.0	1.9	3	4	7	13
Tc-99	1.0	<u>2.0</u>	<u>6.7</u>	<u>20</u>	<u>67</u>	<u>200</u>
Th-232 + C	1.0	1.9	3	4	7	14
Ra-226 + C	1.0	<u>2.0</u>	4	5	10	20
Np-237 + D	1.0	<u>2.0</u>	<u>6.7</u>	17	37	79
Pu-239/240	1.0	<u>2.0</u>	<u>6.7</u>	<u>20</u>	<u>67</u>	117
Am-241	1.0	<u>2.0</u>	<u>6.7</u>	17	32	58

Underlined values were constrained based on uniform mixing after excavation (200/area).

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	14-14
		02	Section 2	2	N/A	7,328	14-14
		03	Section 3	1	N/A	1,984	14-14
		04	Section 4	1	N/A	1,996	14-14
		05	Section 5	1	N/A	1,997	14-14
		06	Section 6	1	N/A	1,997	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	
LSA-12	**Lay Down Area**	07	Section 7	1	N/A	1,974	14-14
Piping Survey Areas							
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 National Nuclear Data Center, Brookhaven National Laboratory <nndc.bnl.gov/chart/>.
[b] Nominal 2π efficiency value for a 126 cm^2 gas flow proportional detector with a 0.8 mg/cm^2 window in the $\alpha + \beta$ 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®)	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			
15	1.02E-03	NA	NA
50	4.19E-05	11,800	5,170.59
100	5.38E-05	9,840	5,528.25
-	-	Total:	10,699
U-234 MDER (μR/h, Eq. 6-21 Of Reference 14-7):			**0.14**
U-235 Minimum Detectable Exposure Rate			
15	9.87E-03	NA	NA
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:	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®)	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			
15	1.83E-03	NA	NA
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:	3,554
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
PW-02	Bedrock	Secondary	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.

Table 14-25

ProUCL Statistical Assessment Input Data Set

Th-232 GS BKG	Th-232 GS NI	TotalU AS BKG	TotalU AS NI
0.53	-0.0859	1.1488	0.43
0.532	-0.0767	1.2636	0.75
0.68	-0.0743	1.289	1.00
0.758	-0.0536	1.3131	1.03
0.767	-0.0347	1.326	1.16
0.774	-0.00504	1.3679	1.47
0.802	0.0184	1.4172	1.51
0.806	0.0204	1.4375	1.51
0.814	0.024	1.4738	1.63
0.82	0.05	1.487	1.80
0.834	0.05	1.5101	1.80
0.877	0.0546	1.52	1.88
0.931	0.0776	1.5517	1.92
0.978	0.08	1.5812	1.96
0.997	0.103	1.6062	2.26
1	0.109	1.6364	2.60
1.01	0.117	1.7519	-
1.04	0.12	1.7582	-
1.14	0.121	1.7774	-
1.17	0.134	1.7904	-
1.17	0.14	1.8036	-
1.17	0.14	1.8345	-
1.18	0.142	1.8603	-
1.19	0.185	1.8724	-
1.38	0.205	1.8864	-
1.38	0.208	1.8919	-
1.43	0.216	1.9327	-
1.43	0.218	1.9407	-
1.46	0.226	1.968	-
1.49	0.253	1.9712	-
1.55	0.257	1.9908	-
1.83	0.258	1.993	-
-	0.261	-	-
-	0.304	-	-
-	0.304	-	-
-	0.308	-	-
-	0.32	-	-

Th-232 GS BKG	Th-232 GS NI	TotalU AS BKG	TotalU AS NI
-	0.327	-	-
-	0.331	-	-
-	0.342	-	-
-	0.343	-	-
-	0.345	-	-
-	0.35	-	-
-	0.358	-	-
-	0.363	-	-
-	0.374	-	-
-	0.376	-	-
-	0.377	-	-
-	0.377	-	-
-	0.384	-	-
-	0.399	-	-
-	0.401	-	-
-	0.404	-	-
-	0.42	-	-
-	0.425	-	-
-	0.425	-	-
-	0.432	-	-
-	0.443	-	-
-	0.478	-	-
-	0.48	-	-
-	0.509	-	-
-	0.526	-	-
-	0.533	-	-
-	0.537	-	-
-	0.54	-	-
-	0.546	-	-
-	0.573	-	-
-	0.574	-	-
-	0.579	-	-
-	0.58	-	-
-	0.584	-	-
-	0.595	-	-
-	0.6	-	-
-	0.607	-	-

Table 14-25 (continued)

ProUCL Statistical Assessment Input Data Set

Th-232 GS BKG	Th-232 GS NI	TotalU AS BKG	TotalU AS NI
-	0.632	-	-
-	0.632	-	-
-	0.638	-	-
-	0.64	-	-
-	0.647	-	-
-	0.65	-	-
-	0.651	-	-
-	0.66	-	-
-	0.682	-	-
-	0.682	-	-
-	0.69	-	-
-	0.692	-	-
-	0.694	-	-
-	0.708	-	-
-	0.711	-	-
-	0.719	-	-
-	0.726	-	-
-	0.735	-	-
-	0.74	-	-
-	0.742	-	-
-	0.743	-	-
-	0.75	-	-
-	0.771	-	-
-	0.774	-	-
-	0.784	-	-
-	0.791	-	-
-	0.793	-	-
-	0.807	-	-
-	0.816	-	-
-	0.818	-	-
-	0.82	-	-
-	0.824	-	-
-	0.828	-	-
-	0.83	-	-
-	0.846	-	-
-	0.846	-	-
-	0.852	-	-
-	0.854	-	-
-	0.854	-	-
-	0.859	-	-
-	0.86	-	-
-	0.862	-	-
-	0.865	-	-
-	0.865	-	-
-	0.875	-	-
-	0.878	-	-
-	0.883	-	-
-	0.89	-	-
-	0.906	-	-
-	0.91	-	-
-	0.913	-	-
-	0.915	-	-
-	0.919	-	-
-	0.94	-	-
-	0.95	-	-
-	0.951	-	-
-	0.959	-	-
-	0.96	-	-
-	0.962	-	-
-	0.97	-	-
-	0.974	-	-
-	0.987	-	-
-	0.99	-	-
-	0.99	-	-
-	0.992	-	-
-	0.993	-	-
-	0.996	-	-
-	1	-	-
-	1	-	-
-	1.01	-	-
-	1.02	-	-
-	1.03	-	-
-	1.04	-	-
-		-	-

ProUCL Statistical Assessment Input Data Set

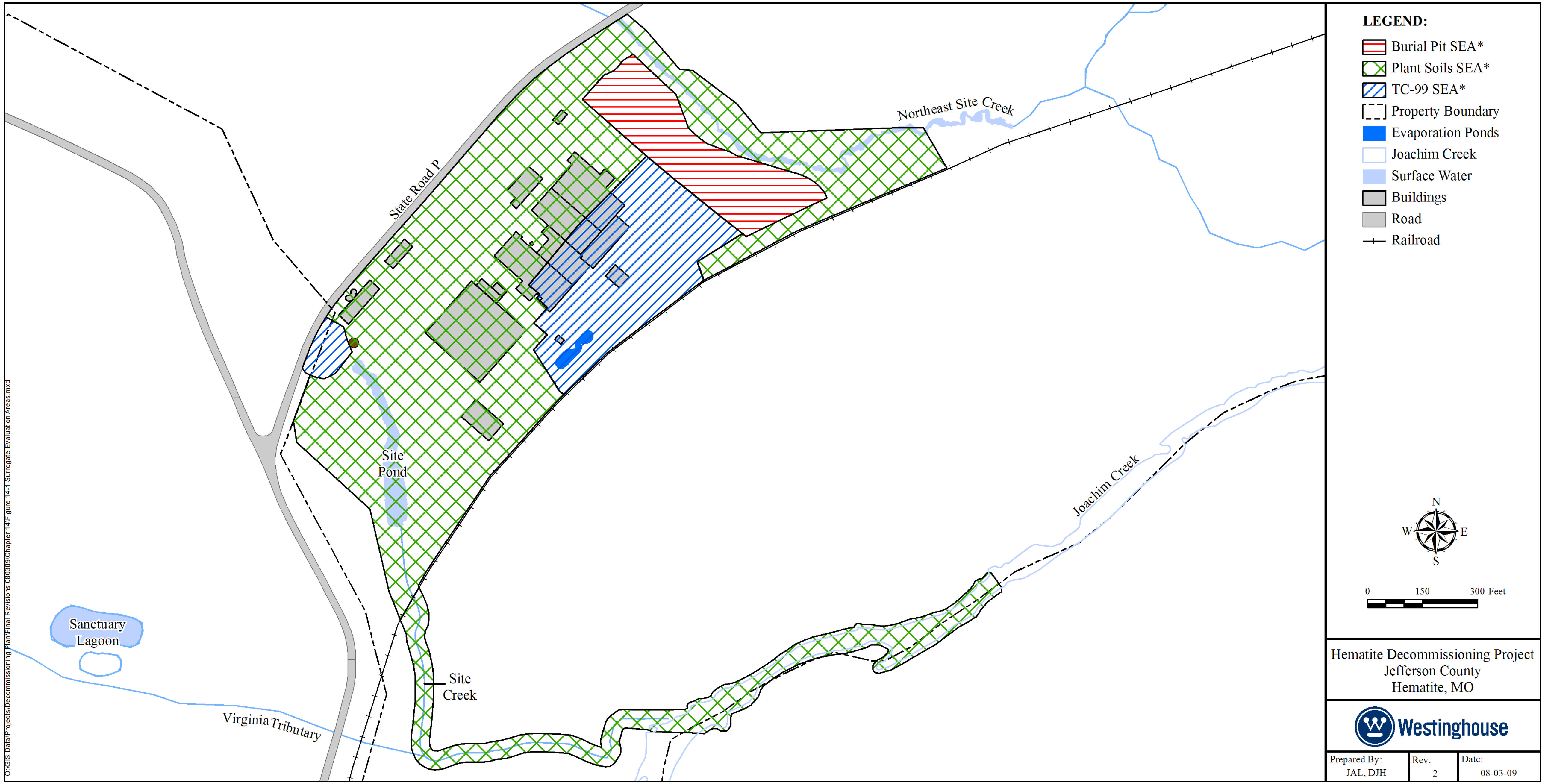
Th-232 GS BKG	Th-232 GS NI	TotalU AS BKG	TotalU AS NI
-	1.06	-	-
-	1.07	-	-
-	1.07	-	-
-	1.07	-	-
-	1.07	-	-
-	1.09	-	-
-	1.1	-	-
-	1.12	-	-
-	1.12	-	-
-	1.13	-	-
-	1.16	-	-
-	1.16	-	-
-	1.17	-	-
-	1.18	-	-
-	1.2	-	-
-	1.21	-	-
-	1.23	-	-
-	1.23	-	-
-	1.28	-	-
-	1.28	-	-

Th-232 GS BKG	Th-232 GS NI	TotalU AS BKG	TotalU AS NI
-	1.3	-	-
-	1.3	-	-
-	1.33	-	-
-	1.33	-	-
-	1.33	-	-
-	1.34	-	-
-	1.35	-	-
-	1.37	-	-
-	1.37	-	-
-	1.4	-	-
-	1.41	-	-
-	1.42	-	-
-	1.45	-	-
-	1.46	-	-
-	1.47	-	-
-	1.49	-	-
-	1.5	-	-
-	1.62	-	-

All data in pCi/g
GS = gamma spectroscopy
AS = alpha spectroscopy
BKG = background
NI = non-impacted

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

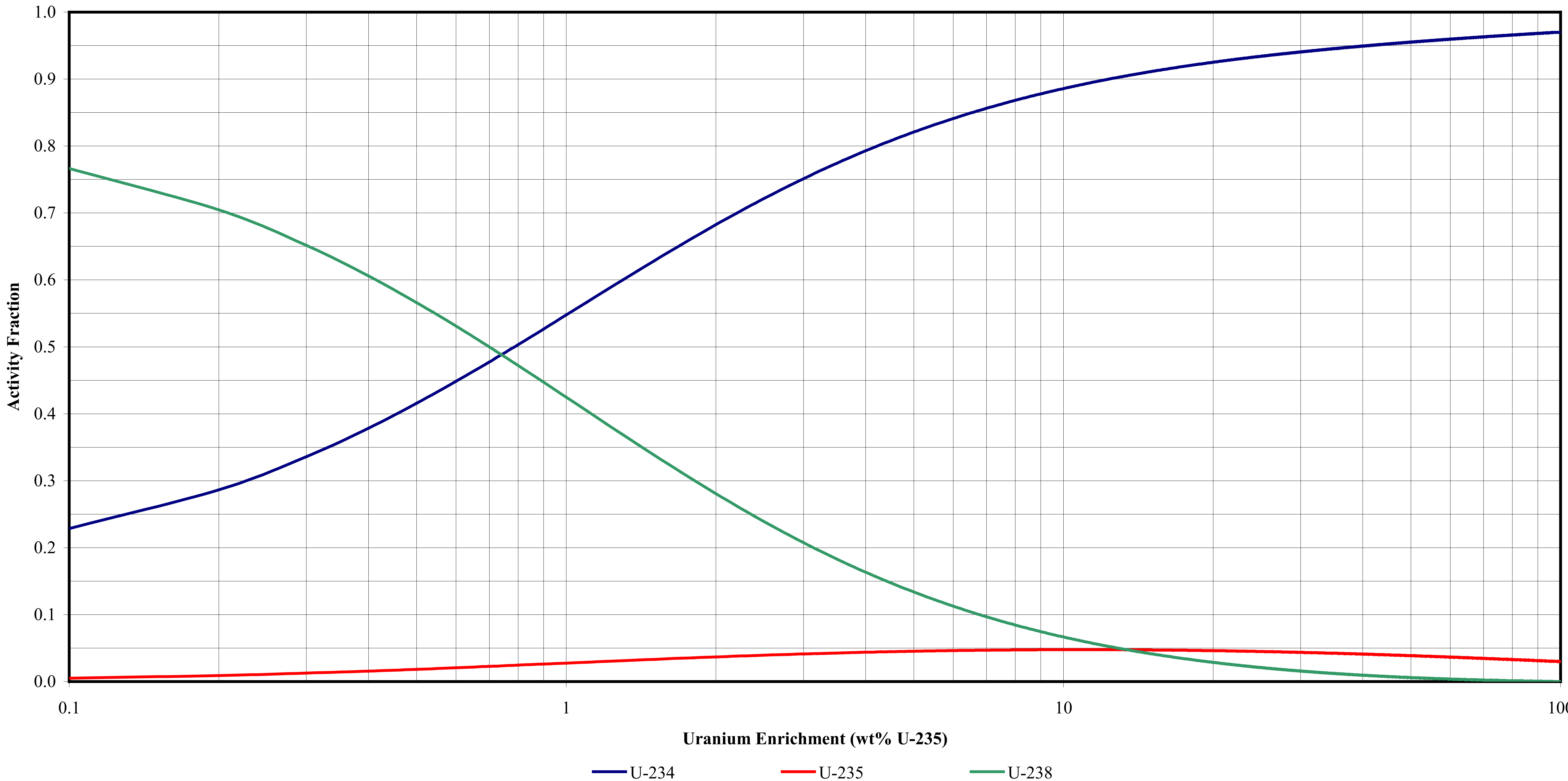


Figure 14-3

Uranium Radioactivity Ratios

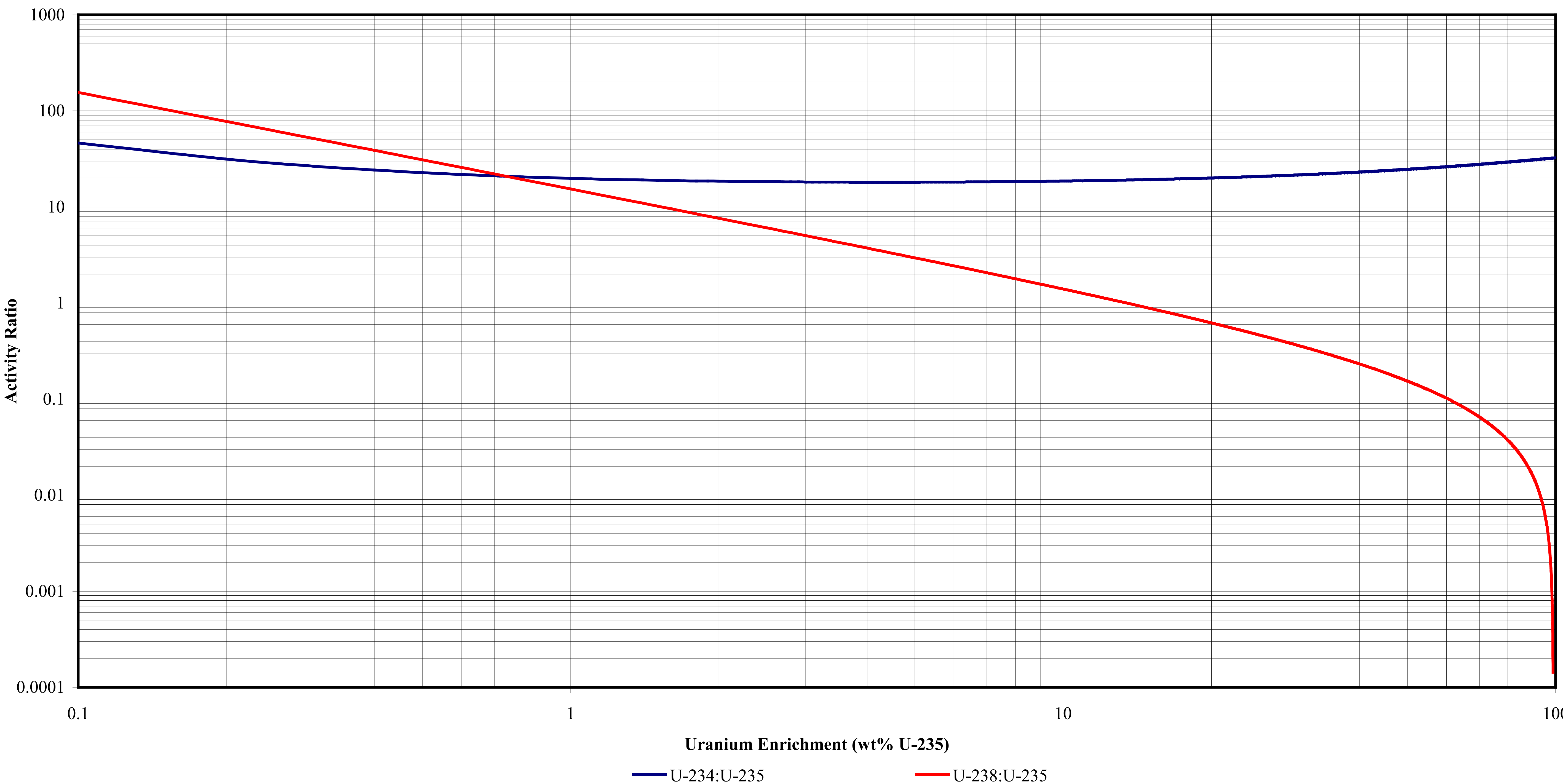


Figure 14-4

Sensitivity Analysis Of Total Uranium $DCGL_W$ For The Plant Soil SEA

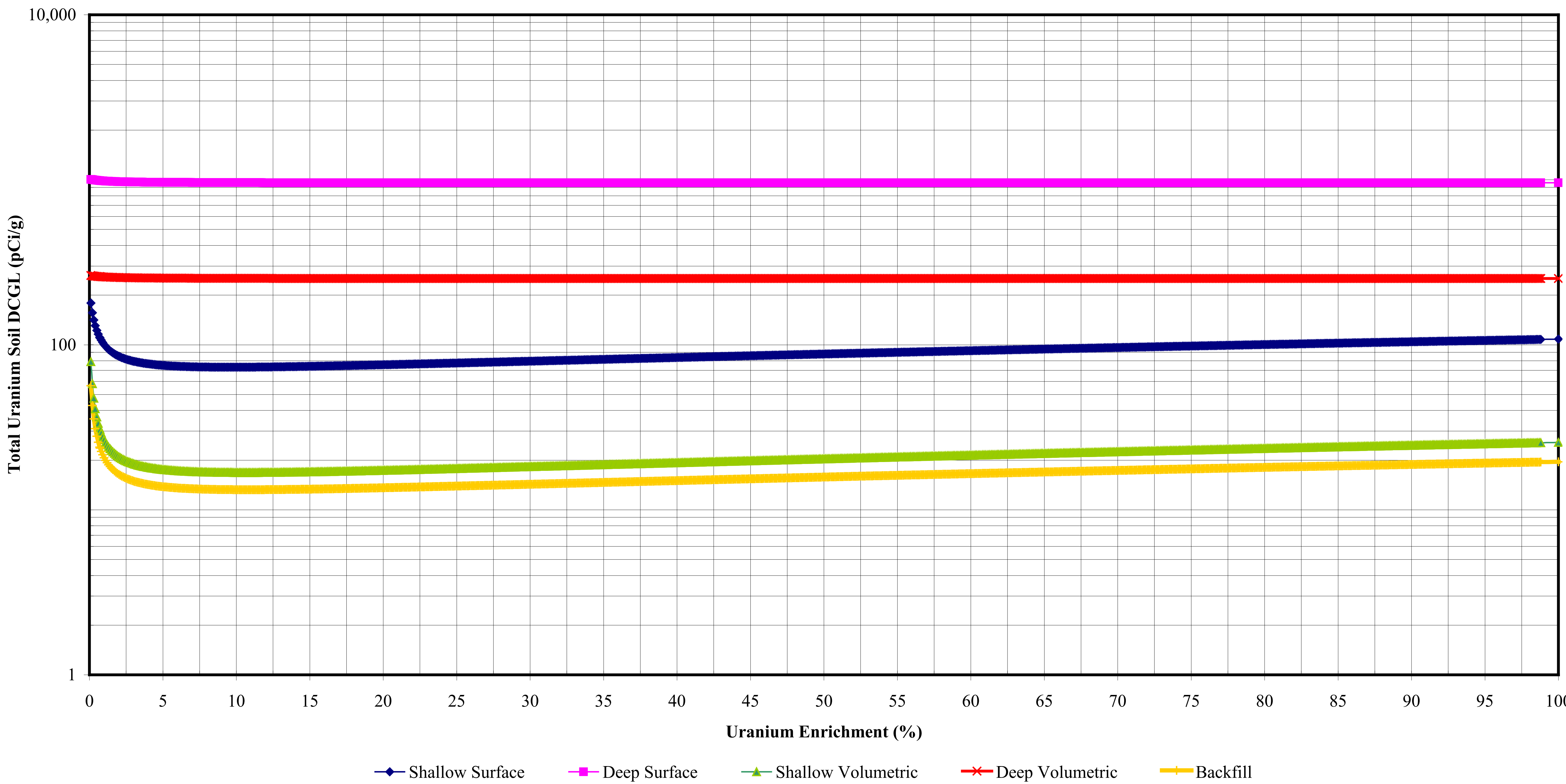


Figure 14-5

Sensitivity Analysis Of Total Uranium DCGL$_W$ For The Tc-99 SEA

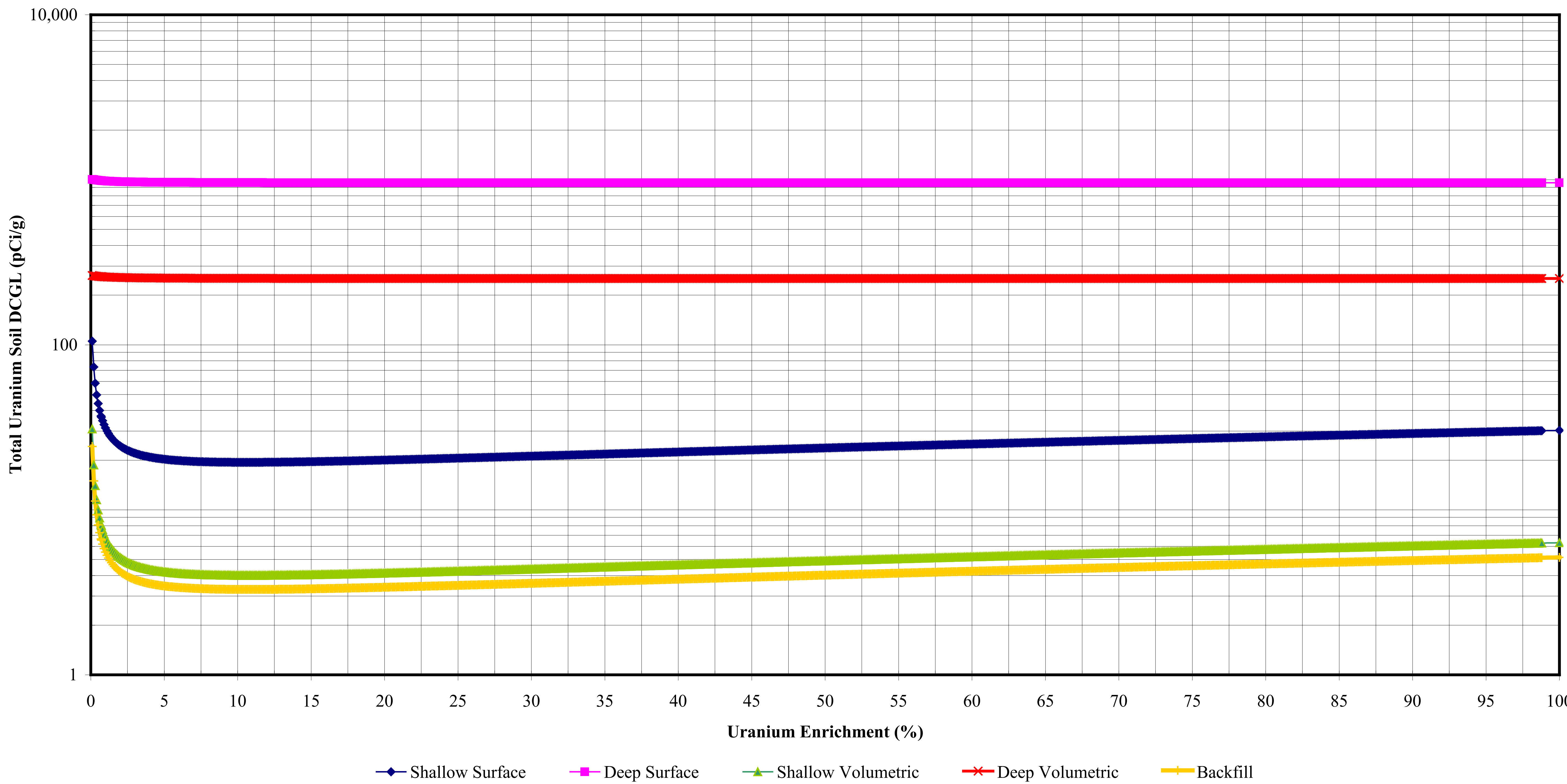


Figure 14-6

Sensitivity Analysis Of Total Uranium $DCGL_W$ For The Burial Pit SEA

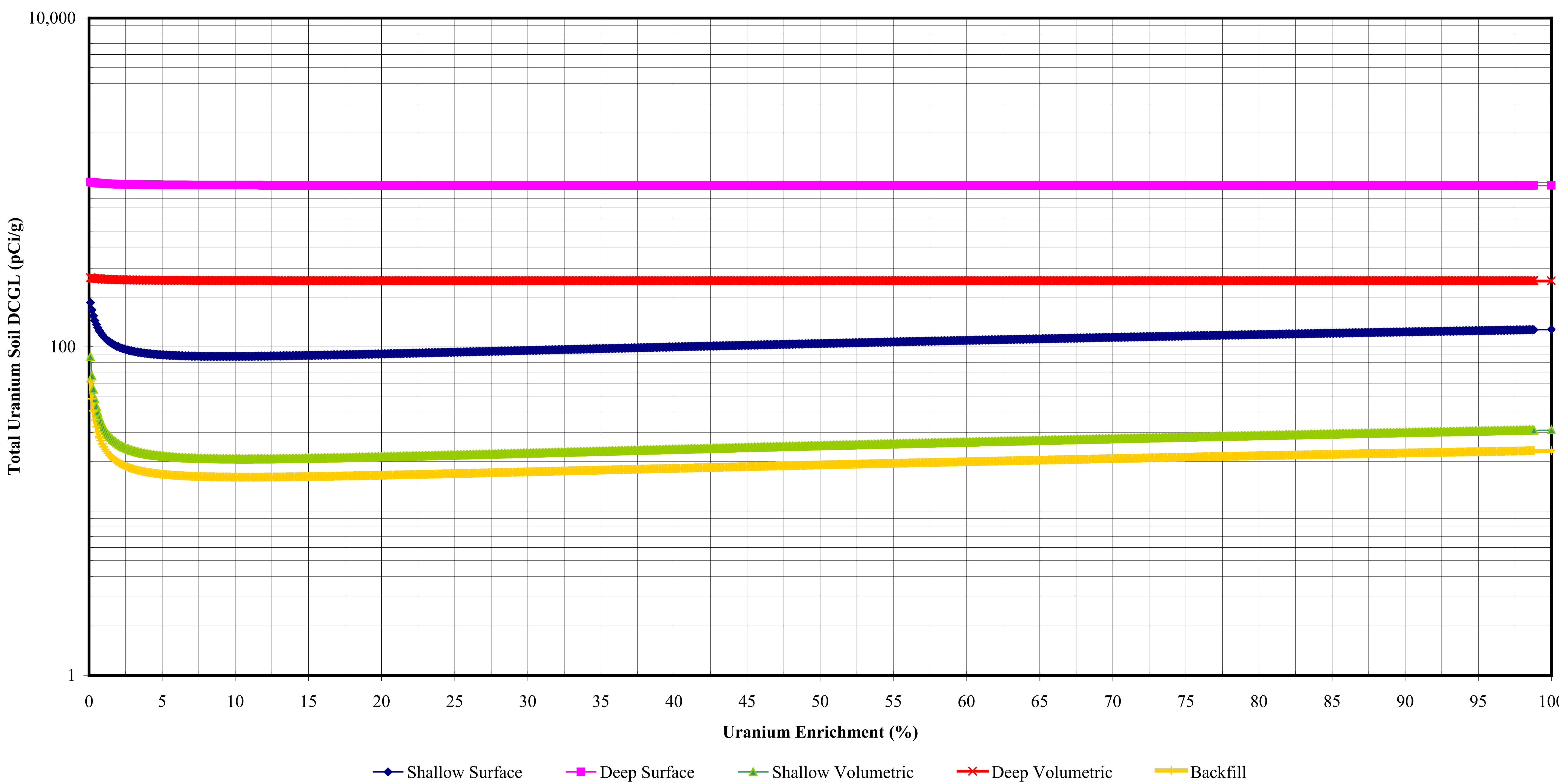


Figure 14-7

Elevated SOF Contour Plot – Surface Stratum

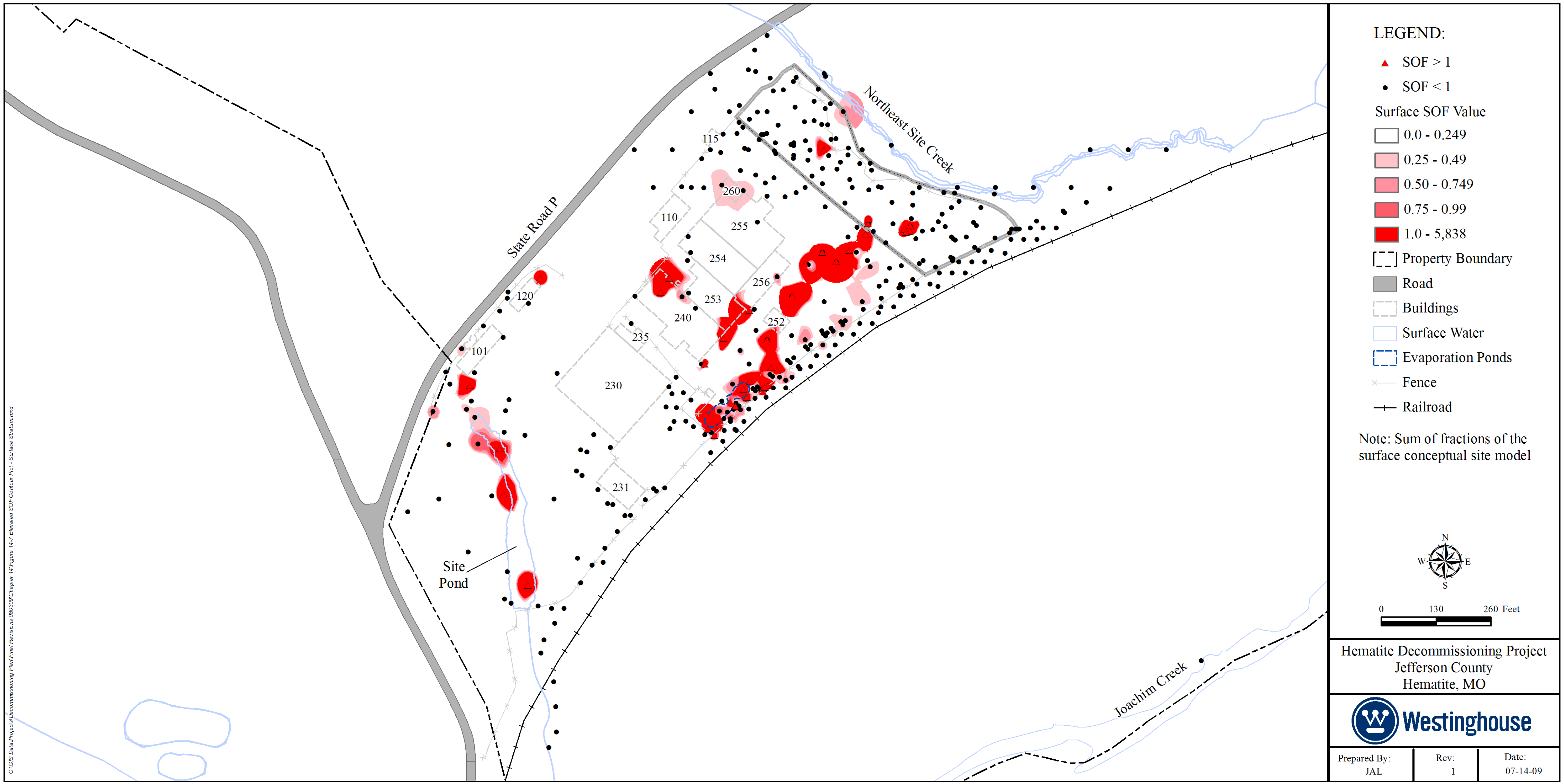


Figure 14-8

Elevated SOF Contour Plot – Root Stratum

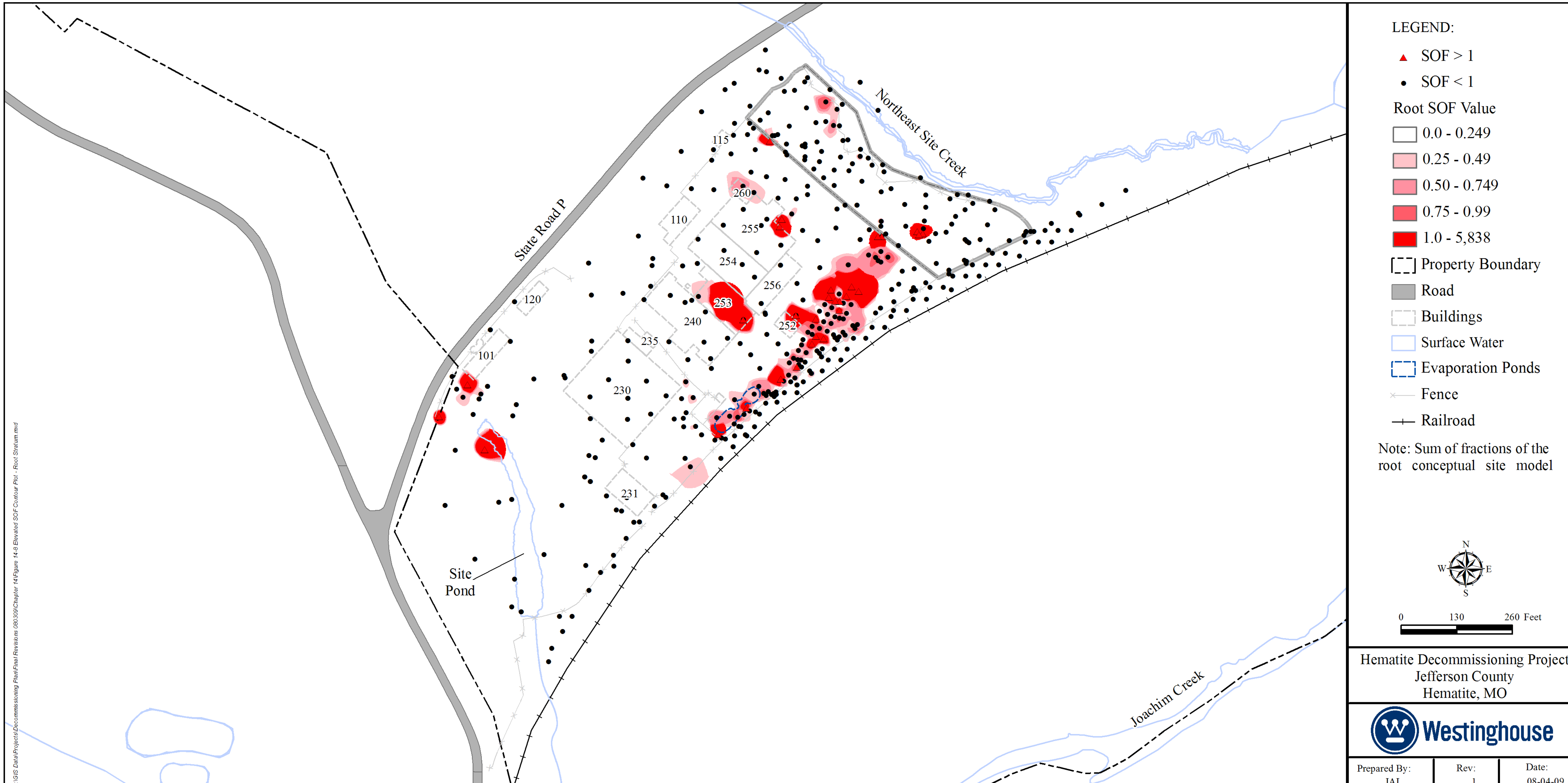


Figure 14-9

Elevated SOF Contour Plot – Deep Stratum

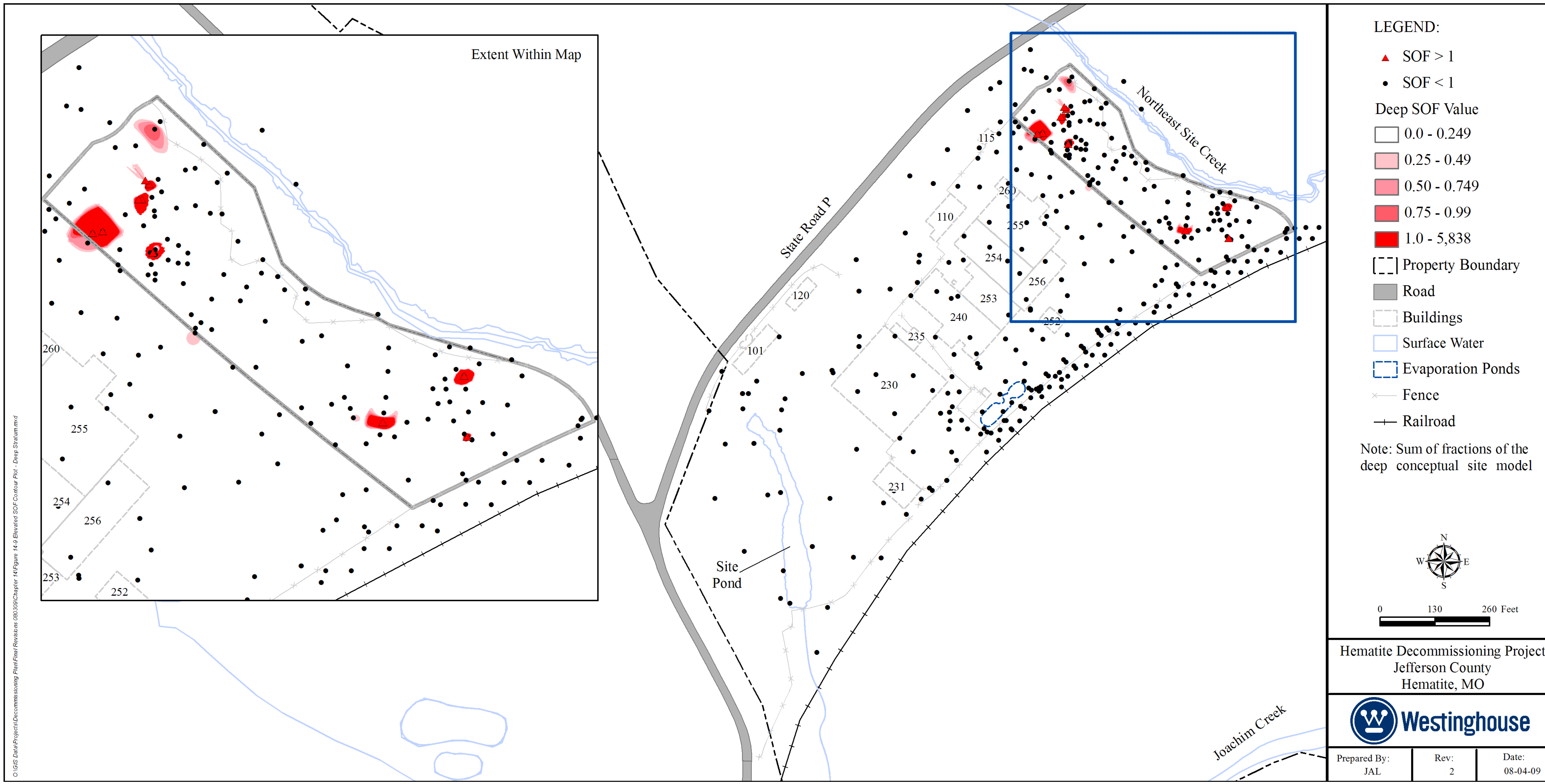


Figure 14-10

Elevated SOF Contour Plot – Total Strata

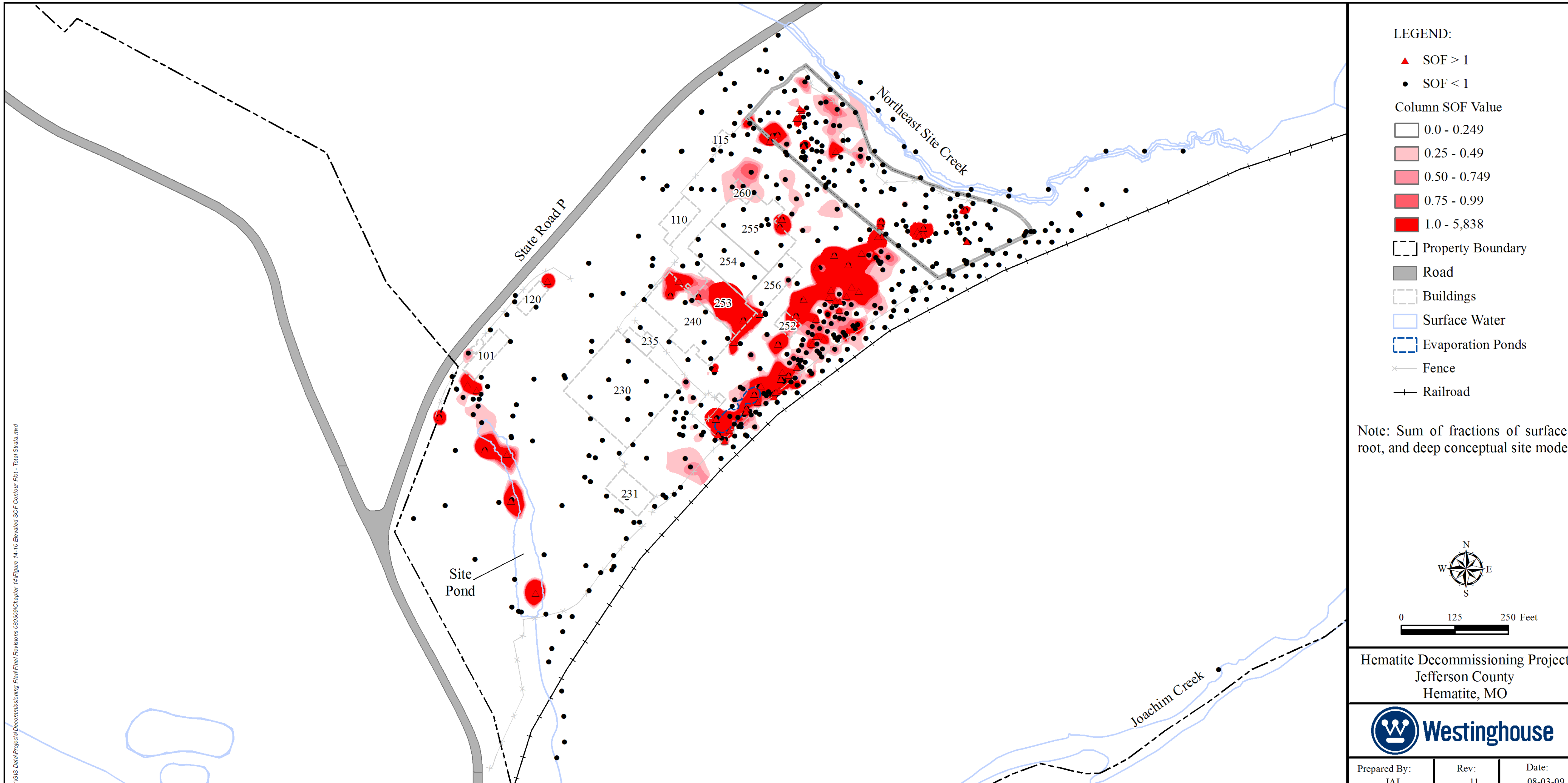


Figure 14-11

Impacted Area For FSS

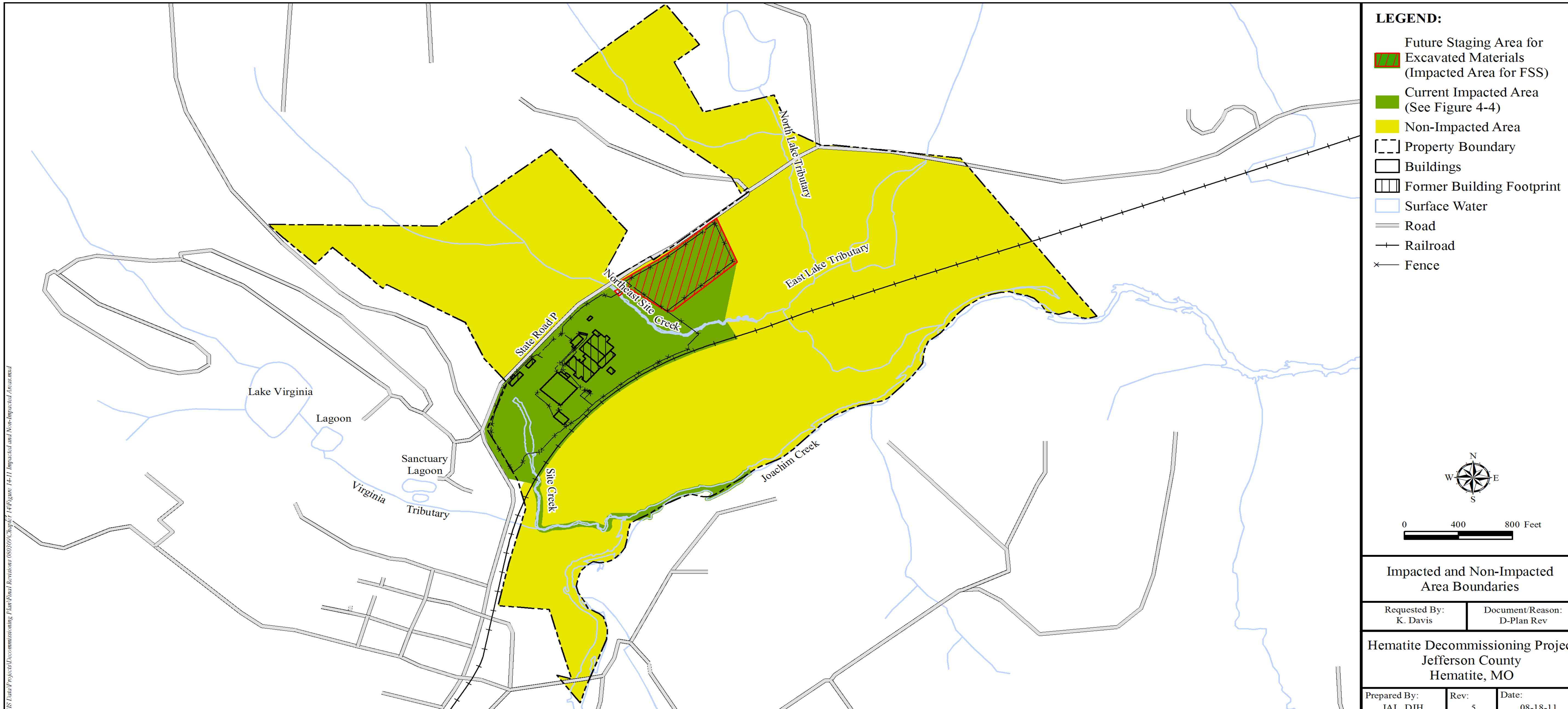


Figure 14-12

Initial Classification Of Impacted Soil Areas

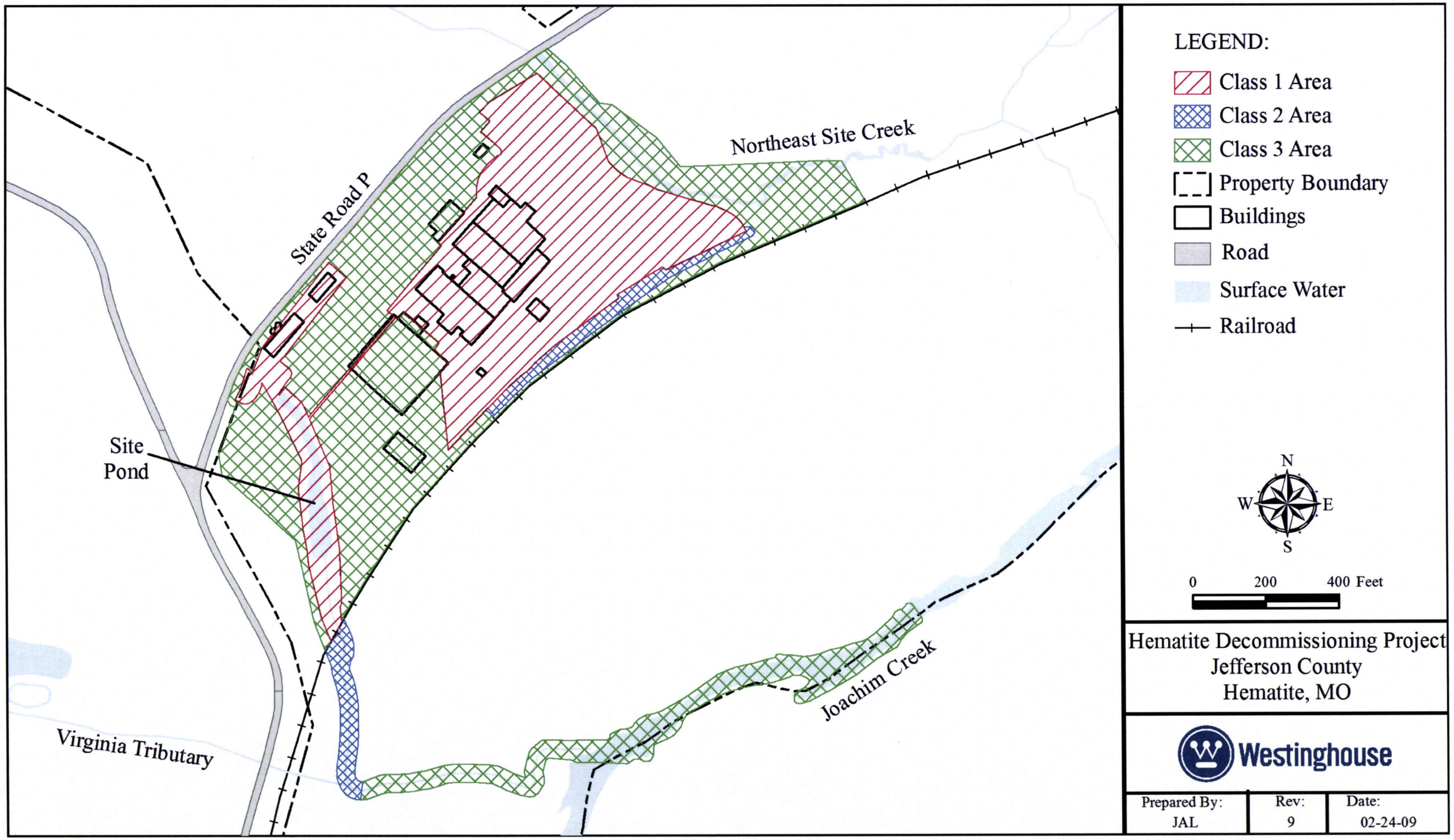


Figure 14-13

Site Survey Area Boundaries

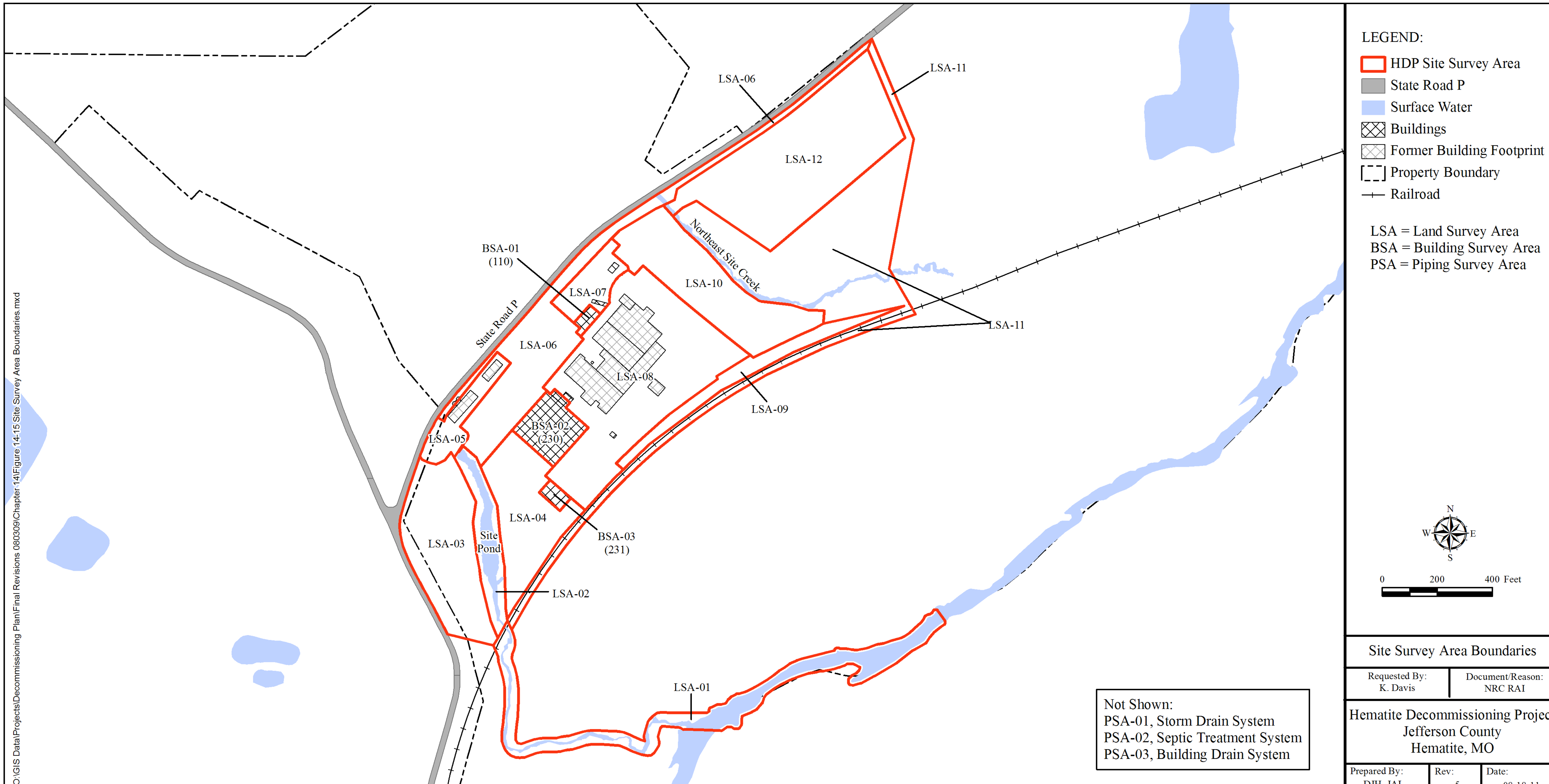


Figure 14-14

Conceptual Open Land Area Survey Units

Figure 14-15

Building 110 Survey Units

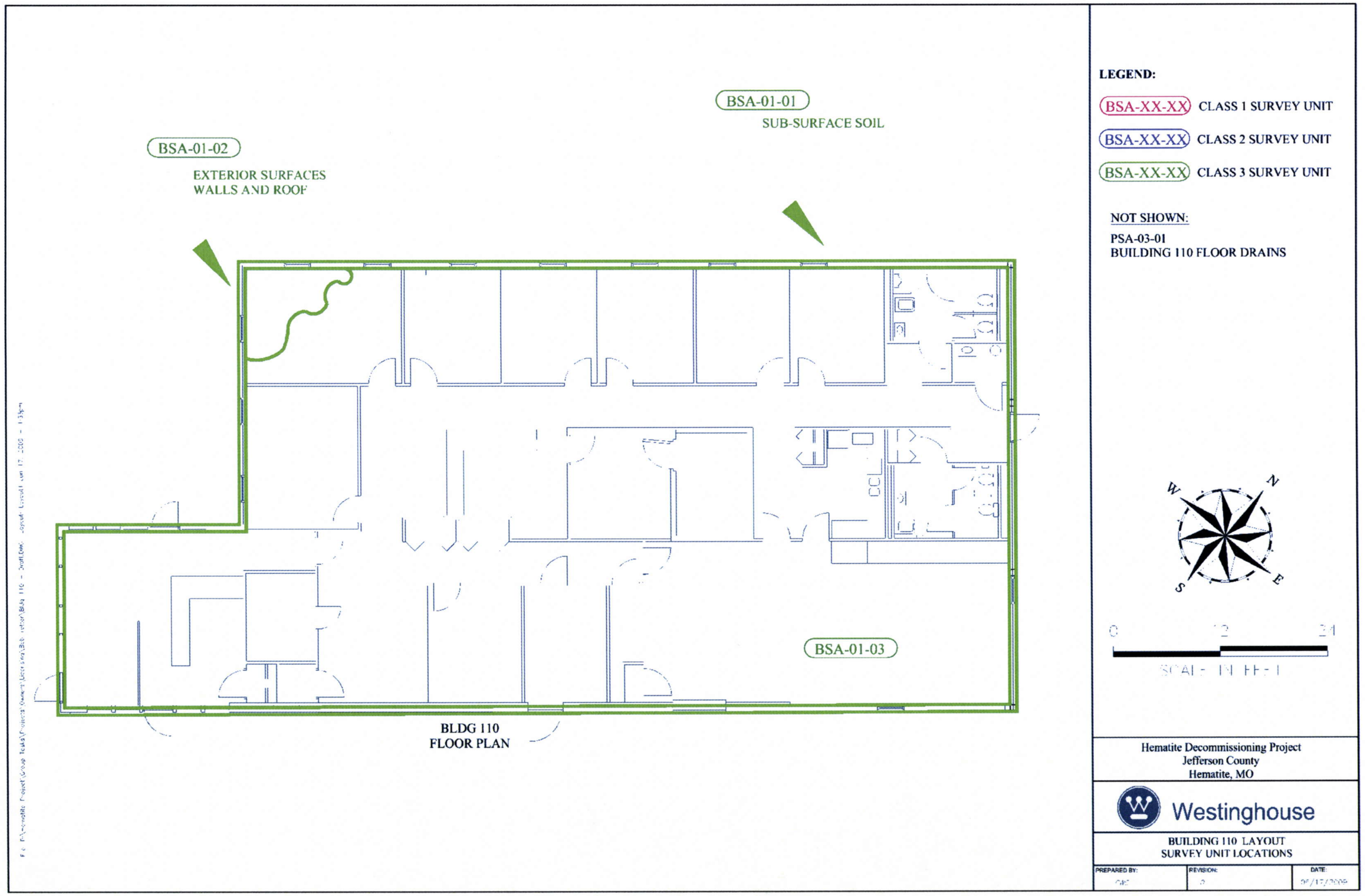


Figure 14-16

Building 230 Survey Units (Ground Floor)

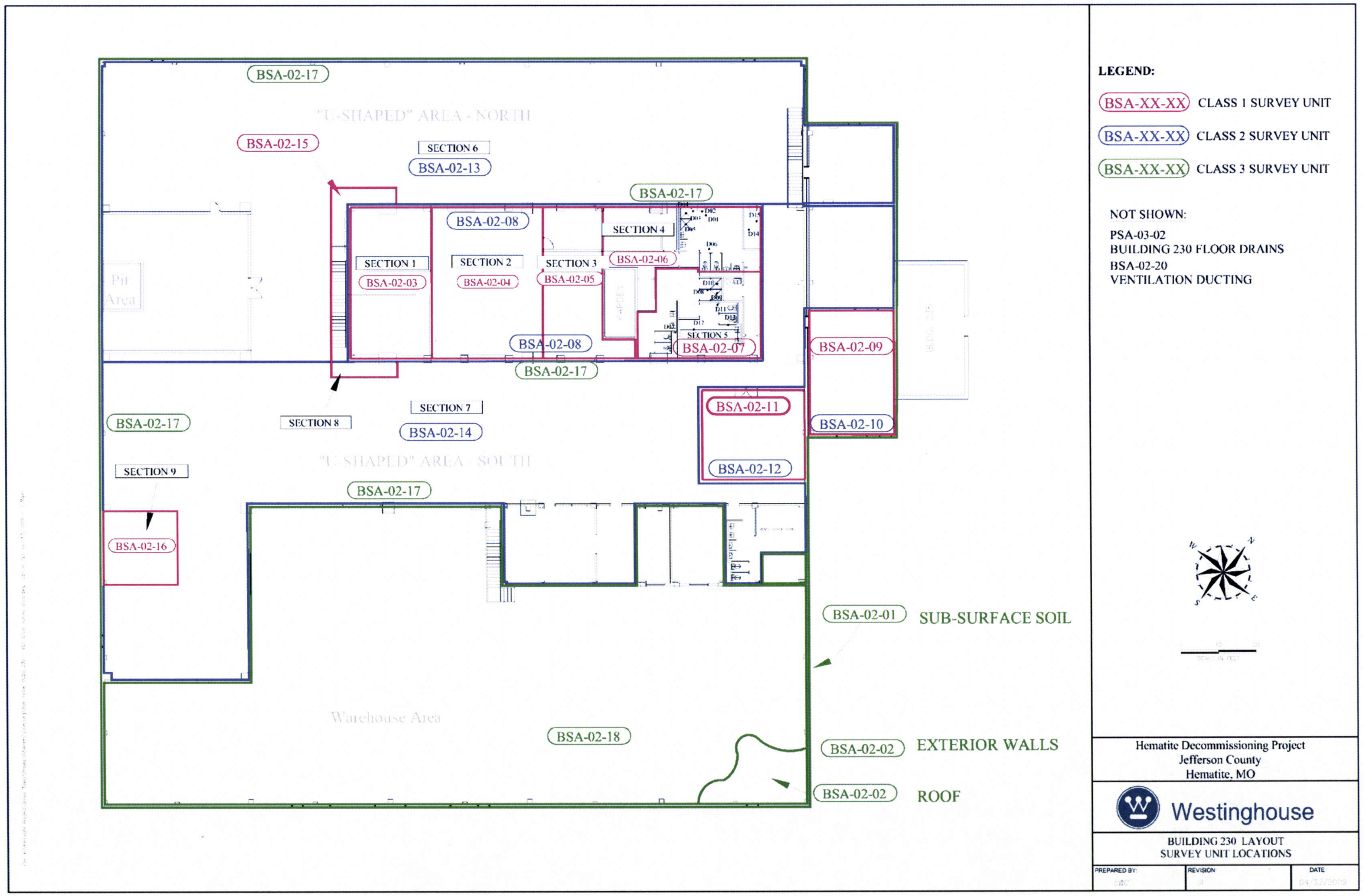


Figure 14-16 (continued)

Building 230 Survey Units (Mezzanine)

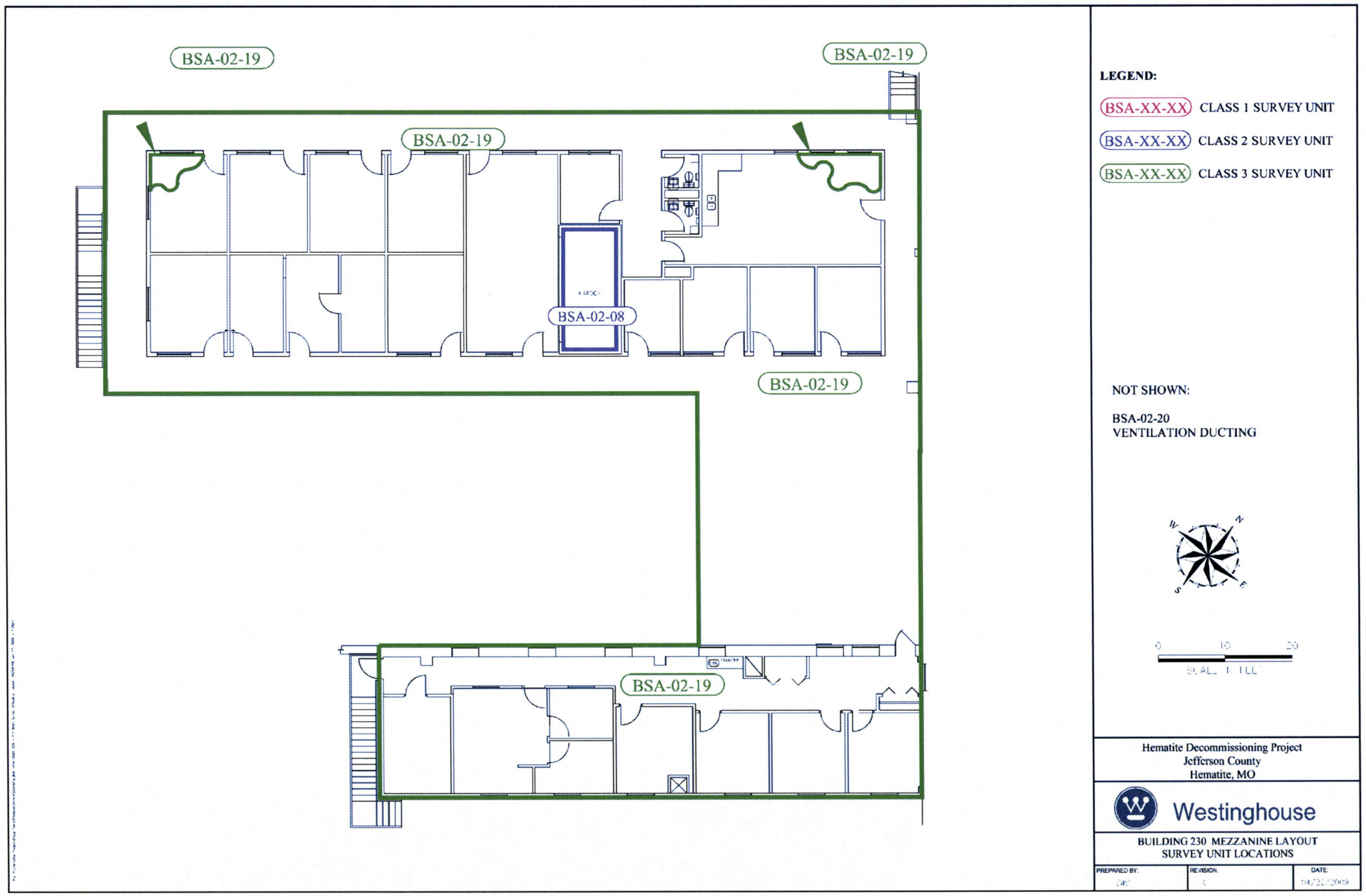


Figure 14-17

Building 231 Survey Units

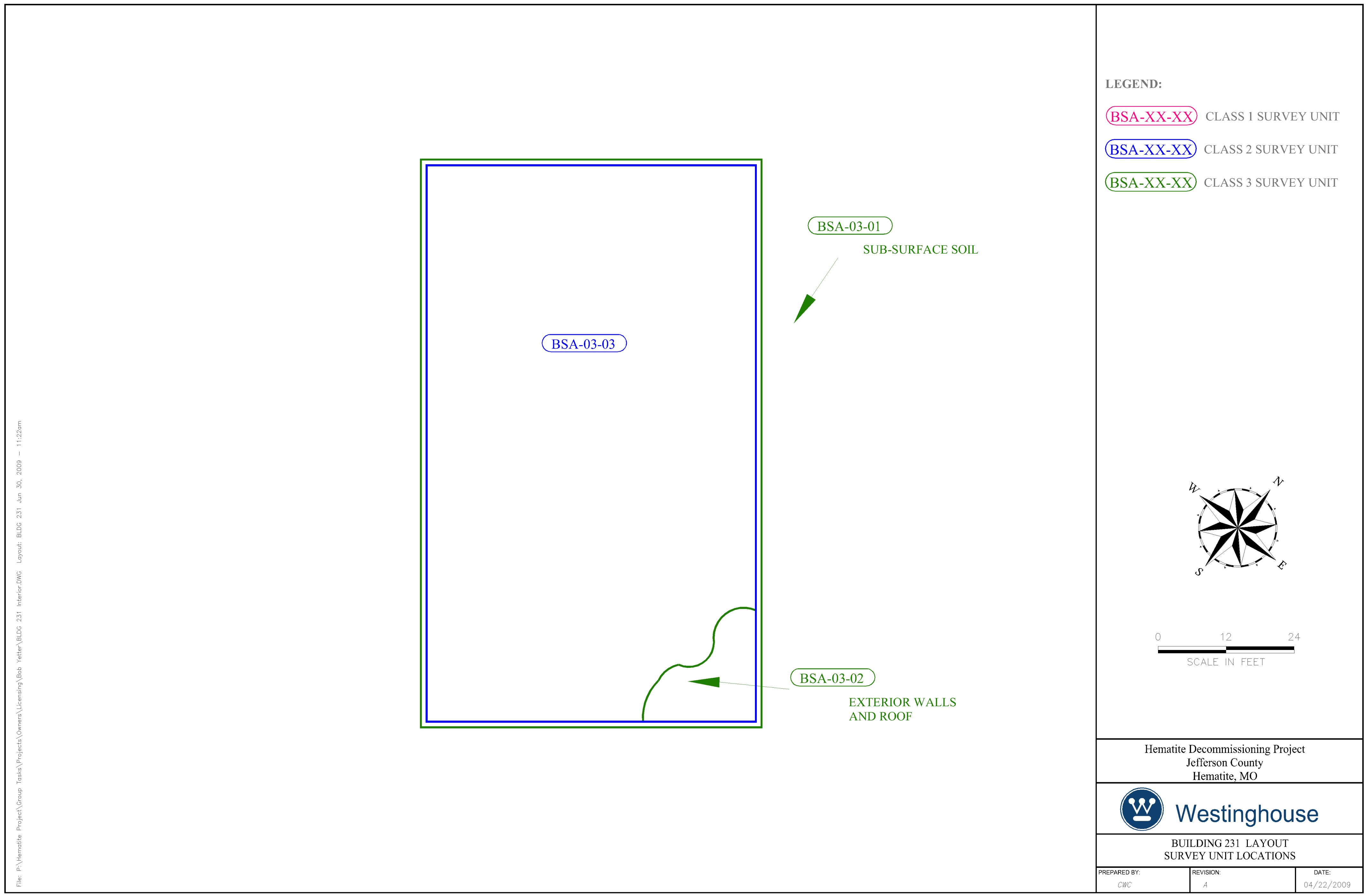


Figure 14-18

Plant Soil SEA Total Uranium Scan MDC vs. Total Uranium $DCGL_W$

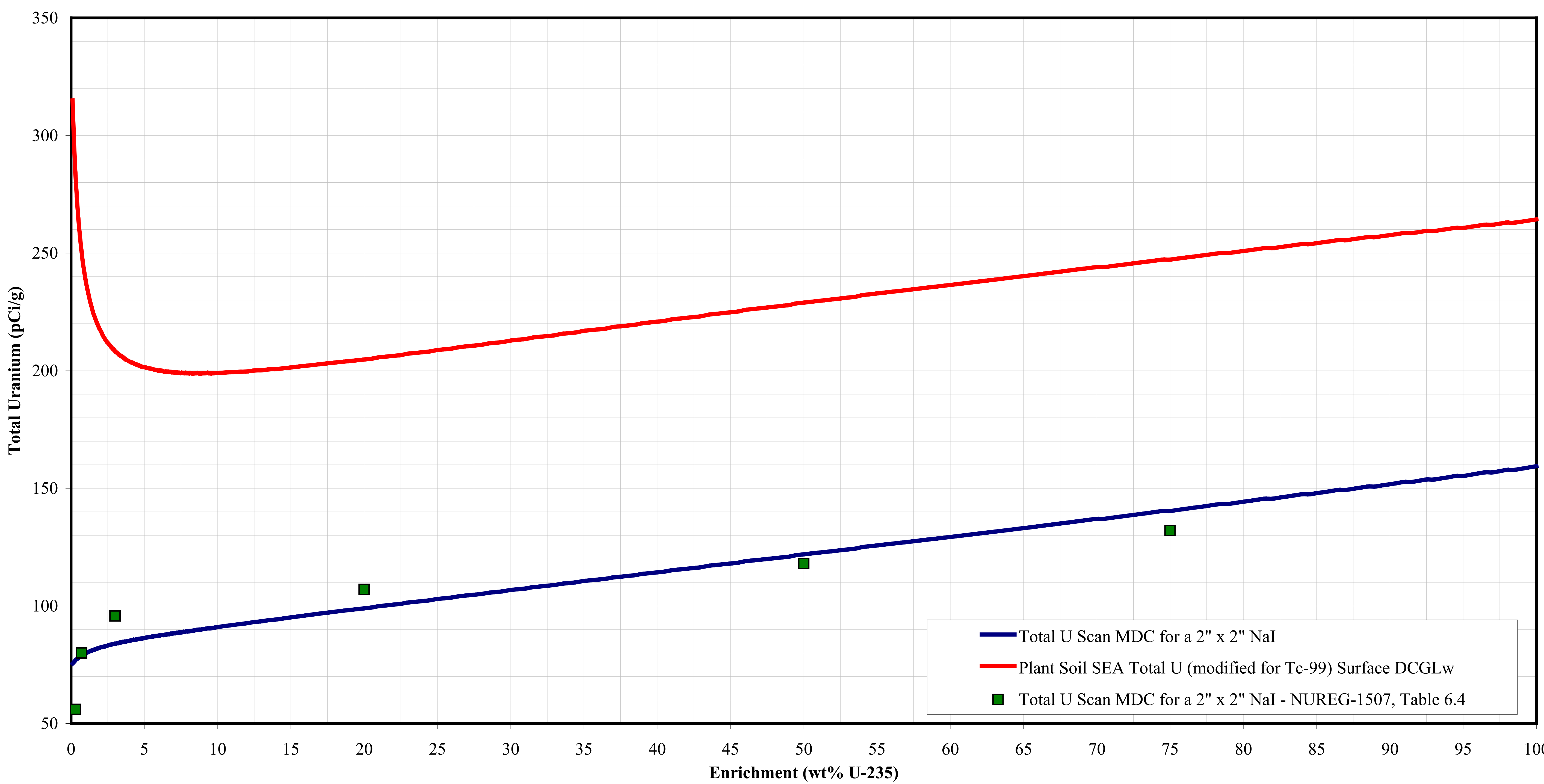


Figure 14-19

Tc-99 SEA Total Uranium Scan MDC vs. Total Uranium $DCGL_W$

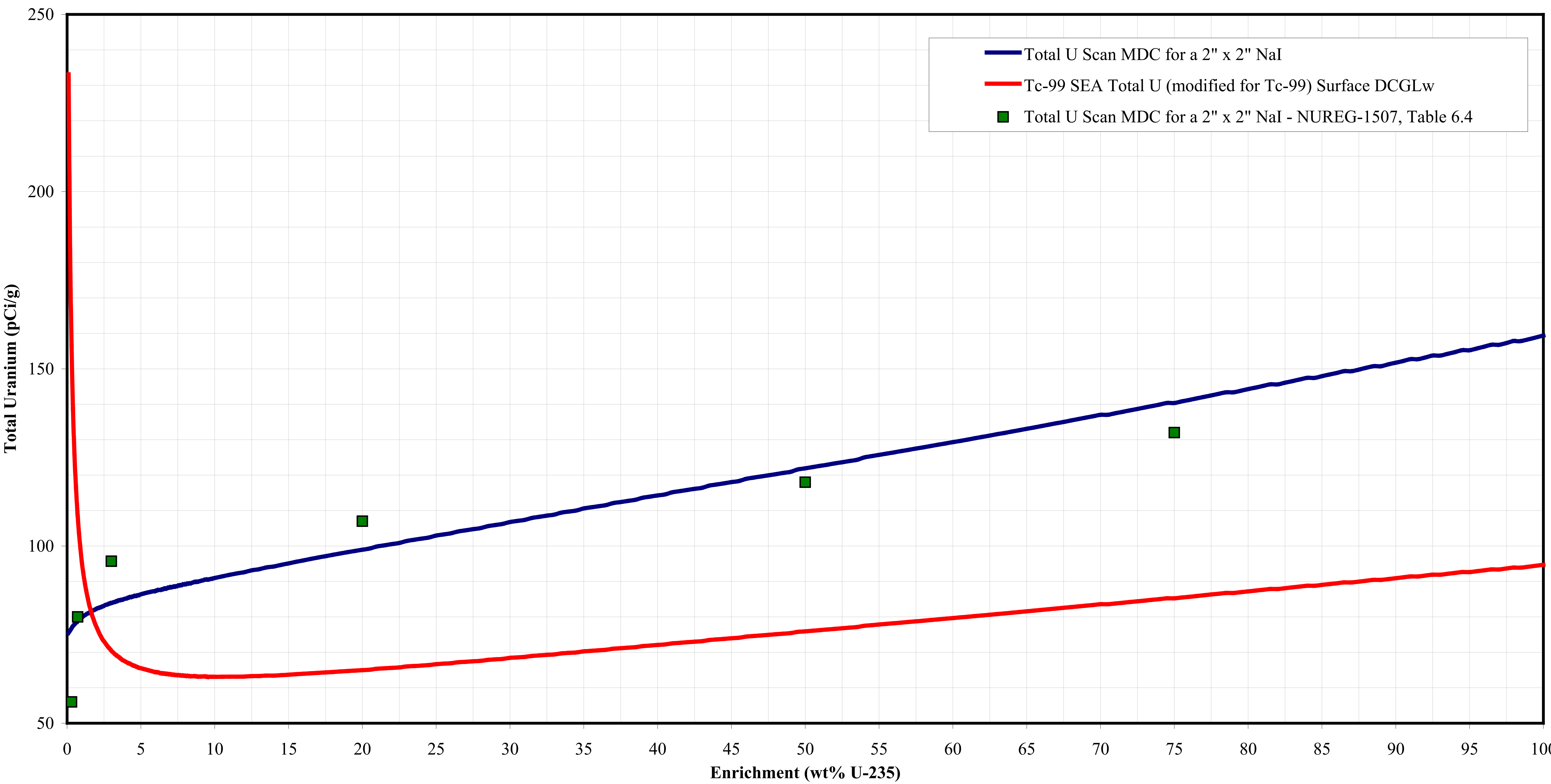


Figure 14-20

Burial Pit SEA Total Uranium Scan MDC vs. Total Uranium $DCGL_W$

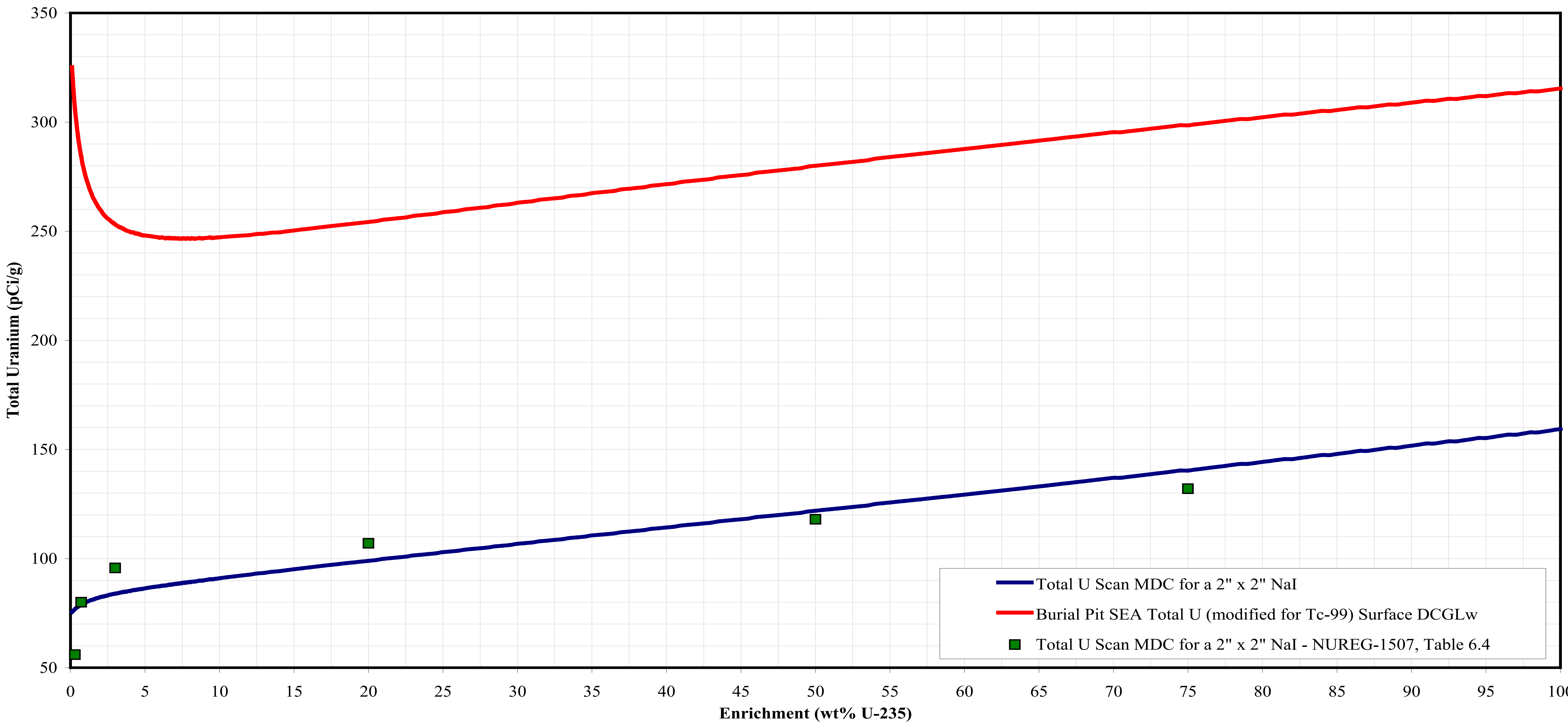
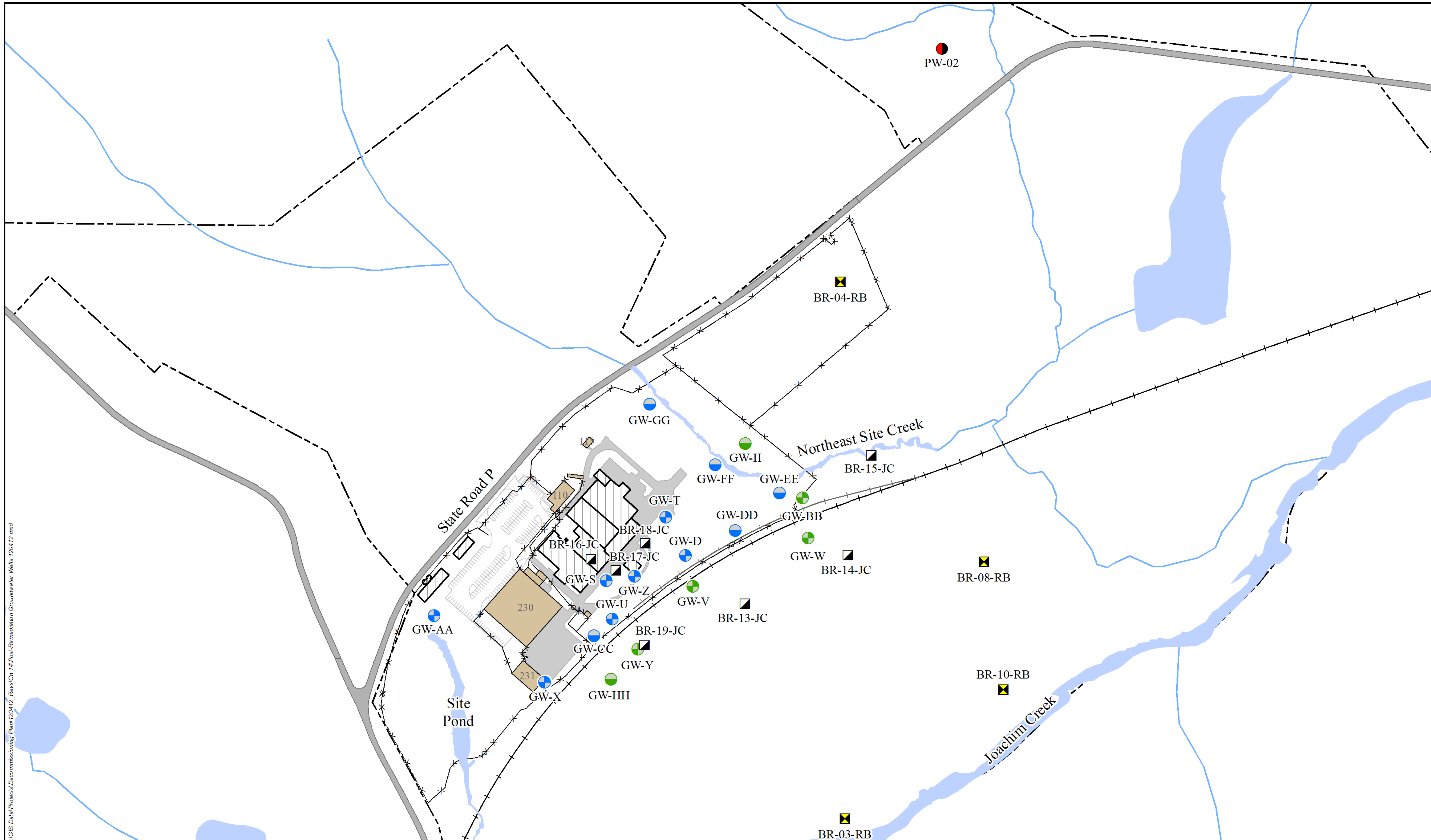


Figure 14-21

Post-Remediation Groundwater Monitoring Wells



Location of Monitoring Wells Post Remediation	
Requested By: K. Davis	Document/Reason: DP Ch 14

Hematite Decommissioning Project
Jefferson County
Hematite, MO

Prepared By:	Rev:	Size:	Date:
JAL, DJH	0	11x17	01-09-13

Figure 14-22

Conceptual Investigation Sample Stations Associated with the Process Buildings

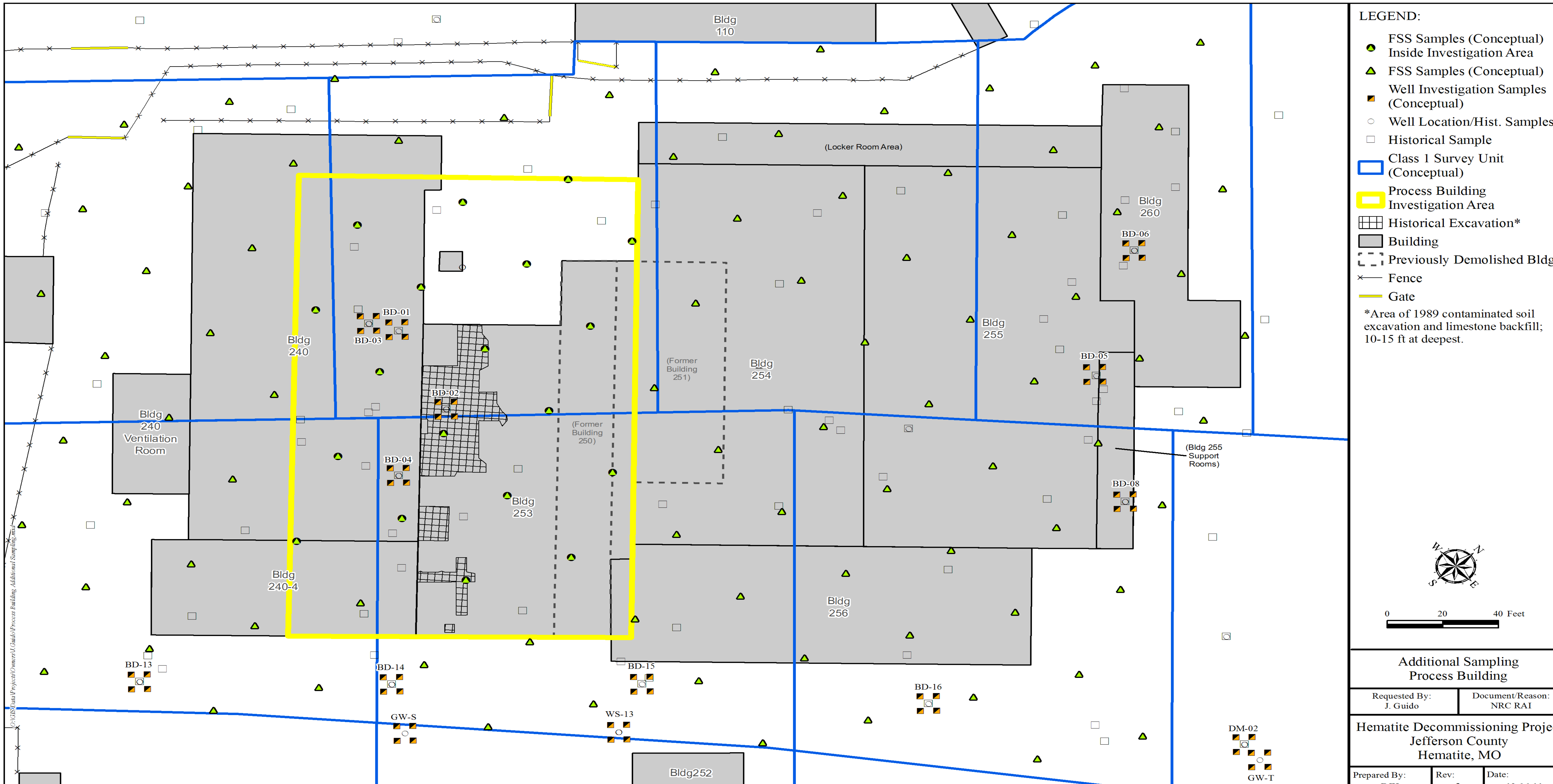


Figure 14-23

ProUCL Quantile and Mann-Whitney Test Results for Th-232 and Total Uranium

Th-232 Quantile Test ProUCL Results

Non-parametric Quantile Hypothesis Test for Full Dataset (No NDs)

User Selected Options

From File	ProUCL.wst
Full Precision	OFF
Confidence Coefficient	99%
Null Hypothesis	Site or AOC Concentration Less Than or Equal to Background Concentration (Form 1)
Alternative Hypothesis	Site or AOC Concentration Greater Than Background Concentration

Area of Concern Data: Th-232 GS NI
Background Data: Th-232 GS BKG

Raw Statistics

	Site	Background
Number of Valid Observations	187	32
Number of Distinct Observations	163	28
Minimum	-0.0859	0.53
Maximum	1.62	1.83
Mean	0.709	1.06
Median	0.74	1.005
SD	0.407	0.314
SE of Mean	0.0298	0.0555

Quantile Test

H0: Site Concentration <= Background Concentration (Form 1)

Approximate R Value (0.015)	15
Apprcximate K Value (0.015)	15
Number of Site Observations in 'R' Largest	9
Calculated Alpha	N/A

Conclusion with Alpha = 0.015

Do Not Reject H0, Perform Wilcoxon-Mann-Whitney Ranked Sum Test

Th-232 Mann-Whitney U Test ProUCL Results

Wilcoxon-Mann-Whitney Site vs Background Comparison Test for Full Data Sets without NDs

User Selected Options

From File	ProUCL.wst
Full Precision	OFF
Confidence Coefficient	99%
Substantial Difference	0.000
Selected Null Hypothesis	Site or AOC Mean/Median Less Than or Equal to Background Mean/Median (Form 1)
Alternative Hypothesis	Site or AOC Mean/Median Greater Than Background Mean/Median

Area of Concern Data: Th-232 GS NI
Background Data: Th-232 GS BKG

Raw Statistics

	Site	Background
Number of Valid Observations	187	32
Number of Distinct Observations	163	28
Minimum	-0.0859	0.53
Maximum	1.62	1.83
Mean	0.709	1.06
Median	0.74	1.005
SD	0.407	0.314
SE of Mean	0.0298	0.0555

Wilcoxon-Mann-Whitney (WMW) Test

H0: Mean/Median of Site or AOC <= Mean/Median of Background

Site Rank Sum W-Stat	19153
WMW Test U-Stat	-4.28
WMW Critical Value (0.010)	2.326
P-Value	1

Conclusion with Alpha = 0.01

Do Not Reject H0, Conclude Site <= Background

P-Value >= alpha (0.01)

Figure 14-23 (continued)

ProUCL Quantile and Mann-Whitney Test Results for Th-232 and Total Uranium

Total Uranium Quantile Test ProUCL Results

	Non-parametric Quantile Hypothesis Test for Full Dataset (No NDs)
User Selected Options	
From File	C:\ Documents and Settings\ guidojs\ Desktop\ totalU RevNI ProUCL IN.xls.wst
Full Precision	OFF
Confidence Coefficient	95%
Null Hypothesis	Site or AOC Concentration Less Than or Equal to Background Concentration (Form 1)
Alternative Hypothesis	Site or AOC Concentration Greater Than Background Concentration

Area of Concern Data: TotalURevNI

Background Data: TotalU AS BKG

Raw Statistics

	Site	Background
Number of Valid Observations	16	32
Number of Distinct Observations	16	32
Minimum	0.425	1.149
Maximum	2.596	1.993
Mean	1.543	1.655
Median	1.572	1.694
SD	0.566	0.252
SE of Mean	0.141	0.0445

Quantile Test

H0: Site Concentration <= Background Concentration (Form 1)

Approximate R Value (0.052)	10
Approximate K Value (0.052)	6
Number of Site Observations in 'R' Largest	4
Calculated Alpha	0.0537

Conclusion with Alpha = 0.052

Do Not Reject H0, Perform Wilcoxon-Mann-Whitney Ranked Sum Test

Total Uranium Mann-Whitney U Test ProUCL Results

	Wilcoxon-Mann-Whitney Site vs Background Comparison Test for Full Data Sets without NDs
User Selected Options	
From File	C:\ Documents and Settings\ guidojs\ Desktop\ totalU RevNI ProUCL IN.xls.wst
Full Precision	OFF
Confidence Coefficient	95%
Substantial Difference	0
Selected Null Hypothesis	Site or AOC Mean/ Median Less Than or Equal to Background Mean/ Median (Form 1)
Alternative Hypothesis	Site or AOC Mean/ Median Greater Than Background Mean/ Median

Area of Concern Data: TotalURevNI

Background Data: TotalU AS BKG

Raw Statistics

	Site	Background
Number of Valid Observations	16	32
Number of Distinct Observations	16	32
Minimum	0.425	1.149
Maximum	2.596	1.993
Mean	1.543	1.655
Median	1.572	1.694
SD	0.566	0.252
SE of Mean	0.141	0.0445

Wilcoxon-Mann-Whitney (WMW) Test

H0: Mean/ Median of Site or AOC <= Mean/ Median of Background

Site Rank Sum W-Stat	362
WMW Test U-Stat	-0.667
WMW Critical Value (0.050)	1.645
P-Value	0.748

Conclusion with Alpha = 0.05

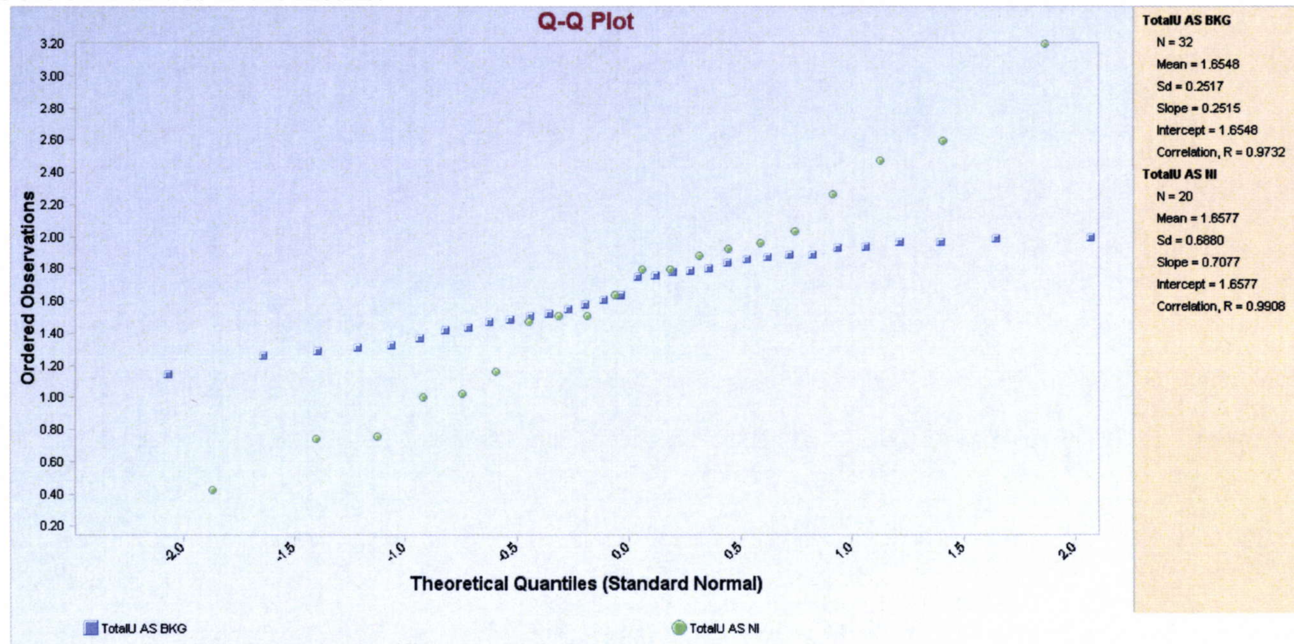
Do Not Reject H0, Conclude Site <= Background

P-Value >= alpha (0.05)

Figure 14-23 (continued)

ProUCL Quantile and Mann-Whitney Test Results for Th-232 and Total Uranium

Total Uranium Quantile Plot – Provides the entire background and non-impacted data distribution, ranging from the lowest to the highest. The vertical axis represents measures concentrations (pCi/g) and the horizontal axis represents percentiles of each distribution. This plot illustrates the similarities in the means.



Total Uranium Box Plot – Depicts the background and non-impacted data through five-number summaries: sample minimum (excluding outliers), lower quartile (25th percentile of the data), median (50th percentile of the data), upper quartile (75th percentile of the data), and sample maximum (excluding outliers). The boxes illustrate data that range from the lower quartile to the upper quartile. The box plots display differences between the two data populations without assuming an underlying statistical distribution (non-parametric). This plot illustrates the similarities in the medians.