Westinghouse Non-Proprietary Class 3

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Our ref: HEM-17-9
Date: February 13, 2017

Subject: Westinghouse Hematite Decommissioning Project - Request for NRC Review of Final Status Survey Final Report Volume 3, Chapter 1, Land Survey Areas (LSA) Overview, Revision 3 (License No. SNM-00033, Docket No. 070-00036)

The purpose of this letter is to provide for the U.S. Nuclear Regulatory Commission (NRC) review of the FSS overview document Final Status Survey Final Report Volume 3, Chapter 1, Land Survey Areas (LSA) Overview, Revision 3.

Attachment 1 contains Final Status Survey Final Report Volume 3, Chapter 1, Revision 3. Attachment 2 contains a track change versions of Final Status Survey Final Report Volume 3, Chapter 1, Revision 3 for ease of review.

Please contact me at 314-810-3353, should you have questions or need additional information.

Sincerely,

Kenneth E. Pallagi
Licensing Manager,
Hematite Decommissioning Project

Attachment: 1) Final Status Survey Final Report Volume 3, Chapter 1, Land Survey Areas (LSA) Overview, Revision 3 (HDP-RPT-FSS-203)

cc: J. W. Smetanka, Westinghouse
M. R. Meyer, NRC/DUWP/MDB
J. A. Smith, NRC/DUWP/MDB

NMSS20

Attachment 1

Final Status Survey Final Report Volume 3, Chapter 1

Land Survey Areas (LSA) Overview, Revision 3

Westinghouse Electric Company LLC, Hematite Decommissioning Project

Docket No. 070-00036

Westinghouse

Final Status Survey Report

Hematite Decommissioning Project

Final Status Survey Final Report Volume 3, Chapter 1

TITLE: **Land Survey Areas (LSA) Overview**

REVISION: **3**

EFFECTIVE DATE: FEB 13 2017

Approvals:

Author: Kenneth E. Pallagi — Date: 02-13-2017

Owner/Manager: W. Clark Evers — Date: 2/13/17

HDP-RPT-FSS-203

REVISION LOG	
Revision No. Effect. Date	**Revision**
0 01/27/2016	Revision 0 is the initial issuance of the Land Survey Areas Overview.
1 05/17/2016	The NRC provided via email a Pre-Audit Submittal Table for Final Status Survey Final Report Volume 3, Chapter 1 on March 8, 2016, which contained comments generated by the review. During subsequent recurring weekly publicly noticed teleconferences, Westinghouse and the NRC discussed the path forward and resolution of the NRC comments for Final Status Survey Final Report Volume 3, Chapter 1. This revision implements the resolution of the comments.
2 10/27/2016	The NRC provided comments on the review of Final Status Survey Final Report Volume 3, Chapter 1, Revision 1, during recurring weekly publicly noticed teleconferences. Westinghouse and the NRC discussed the path forward and resolution of the NRC comments for Final Status Survey Final Report Volume 3, Chapter 1. This revision implements the resolution of the comments. The revision includes discussion of 1) GPS vs GWS Coverage, 2) clarification of section 3.1.2, Three Stratum DCGLs, and 3) addition of Appendix D.
3 See Cover Page	During the NRC review of Survey Area Release Record the NRC, during recurring weekly publicly noticed teleconferences, provided feedback in regards to the application of the WRS Test when applied to the Three Stratum approach. Westinghouse and the NRC discussed the path forward and resolution of the NRC comments. This revision implements the resolution of the comments.

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LIST OF ACRONYMS AND SYMBOLS

AEC	Atomic Energy Commission
AF	Area Factor
ALARA	As Low As Reasonably Achievable
AMSL	above mean sea level
bgs	below ground surface
CFR	Code of Federal Regulations
cm	centimeter(s)
cpm	count(s) per minute
CSM	Conceptual Site Model
DCGL	Derived Concentration Guideline Level
$DCGL_{EMC}$	DCGL for small area of elevated activity
$DCGL_{W}$	DCGL for average concentrations over a survey unit, used with statistical tests. ("W" suffix denotes "Wilcoxon")
DGPS	Differential Global Positioning System
DP	Hematite Decommissioning Plan
EMC	Elevated Measurement Comparison
EPA	U.S. Environmental Protection Agency
ft	foot (feet)
FSS	Final Status Survey
FSSP	Final Status Survey Plan
FSSFR	Final Status Survey Final Report
gcpm	gross count(s) per minute
GPS	Global Positioning System
GWS	Gamma Walkover Survey
HDP	Hematite Decommissioning Project
HP	Health Physics
HRCR	Hematite Radiological Characterization Report
HRGS	High Resolution Gamma Spectroscopy
HSA	Historical Site Assessment
I & C	Isolation and Control
IAL	Investigation Action Level
LSA	Land Survey Area
m	meter(s)
m^2	square meter(s)
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCL	Maximum Concentration Limit
MDC	Minimum Detectable Concentration
mrem	Milliroentgen Equivalent Man
NAD	North American Datum
NaI	Sodium Iodide
ncpm	net count(s) per minute
NCS	Nuclear Criticality Safety

NRC	U.S. Nuclear Regulatory Commission
pCi/g	picocurie(s) per gram
QC	Quality Control
Ra	Radium
RASS	Remedial Action Support Survey
RG	Remediation Goal (specific to chemical contaminants)
RML	Reuse Material Screening Action Level
ROC	Radionuclides of Concern
RSO	Radiation Safety Officer
SEA	Surrogate Evaluation Area
SNM	Special Nuclear Material
SOF	Sum of Fractions
SSC	Structures, Systems and Components
SU	Survey Unit
Tc	Technetium
TEDE	Total Effective Dose Equivalent
Th	Thorium
U	Uranium
UF_6	Uranium Hexafluoride
WRS	Wilcoxon Rank Sum
yr	year

1.0 INTRODUCTION

The objective of the Final Status Survey (FSS) is to demonstrate that the dose from residual radioactivity at the Hematite Decommissioning Project (HDP) Site does not exceed the annual dose criterion for license termination for unrestricted use as specified in U.S. Nuclear Regulatory Commission (NRC) Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination," and that the levels of residual radioactivity are As Low As Reasonably Achievable (ALARA). To demonstrate that this objective is achieved, an FSS will be performed on all impacted open land areas that are to remain at the time of license termination. The principal requirement is that the dose to future site occupants will be shown to be less than 25 millirem/year.

The conduct of remedial activities in land survey areas (LSAs) is described in the HDP Decommissioning Plan (DP) DO-08-004, *Hematite Decommissioning Plan* [Westinghouse 2009] (Reference 8.1) and associated documents as approved on October 13, 2011, by NRC letter (Reference 8.2) with the issuance of License SNM-33 Amendment 57 (Reference 8.3) and the DP Safety Evaluation Report (Reference 8.4). The goal of the decommissioning project is to release the facility for unrestricted use in compliance with the requirements of 10 CFR Part 20.1402. The FSS process described in the DP and associated documents, and discussed in this section, adheres to the guidance provided in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Reference 8.5) for planning, conducting, and evaluating surface soil final status radiological surveys. In addition to MARSSIM, 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 8.6);
- NUREG 1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions (Reference 8.7); and,
- NUREG 1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys (Reference 8.8).

The conduct of FSS activities for LSAs was carried out through the implementation of the following HDP procedures and their current Revision number:

- HDP-PO-FSS-700, Final Status Survey Program
- HDP-PR-FSS-701, Final Status Survey Plan Development
- HDP-PR-FSS-703, Final Status Survey Quality Control
- HDP-PR-FSS-711, Final Status Surveys and Sampling of Soil and Sediment
- HDP-PR-FSS-720, Final Status Survey Data Integrity and Database Management
- HDP-PR-FSS-721, Final Status Survey Data Evaluation
- HDP-PR-FSS-722, Final Status Survey Reporting

The current site LSA figure is provided for reference in Appendix A. A procedure revision history for the above procedures and other procedures discussed in this overview document is provided in Appendix B.

1.1 FSSFR Volume 3, Chapter 1, Revision 1

This revision (Revision 1) to FSSFR Volume 3, Chapter 1 has been issued to address and respond to pre-audit submittal comments provided by the NRC in regards to FSSFR Volume 3, Chapter 1, Revision 0. The NRC comments are provided in the document titled Pre-Audit Submittal Table for Final Status Survey Report Volume 3, Chapter 1 – Land Survey Areas (LSA) Overview {ML16068A239}. Westinghouse's response to the comments and the corresponding revisions made within this document accompanies the submittal of FSSFR Volume 3, Chapter 1, Revision 1, as submitted to the NRC in Westinghouse letter HEM-16-50.

2.0 REMEDIATION ACTIVITIES

Prior to the implementation of an FSS for any LSA, it was necessary to remove all structures, concrete slabs, buried waste and piping, vegetation, and/or any other obstructions, to the extent practicable, to allow the remaining soil to be assessed for radiological contamination and disposed offsite as appropriate. To achieve this objective, the scope of decommissioning activities conducted included the following:

- Installation of additional site infrastructure, including: temporary utilities, security equipment, rail spur and loading pad, soil treatment facility, water treatment system, equipment and soil staging areas, and temporary haul roads;
- Decontamination of structures, systems, and equipment intended to remain at the time of license termination;
- Performing radiological surveys of buried and embedded piping to assess nuclear criticality safety requirements, designing remediation "cut-plans", and developing waste disposition strategies;
- Demolishing and packaging for off-site disposal, site buildings and infrastructure not designated for unrestricted release;
- Excavation of soil, buried waste, and concrete foundations within impacted areas while segregating soil that was acceptable for re-use as backfill; and,
- Packaging and coordinating transportation for disposal of radioactive, hazardous and mixed waste.

These activities took over ten years to complete and included the removal and offsite disposal of the former Process Buildings and their contents, documented onsite burials, undocumented onsite burials, subterranean piping, a sanitary wastewater treatment plant, paved areas, concrete slabs, and other miscellaneous site structures. In some cases remediation, excavation, and surveys were all accomplished concurrently. In other cases, such as with the removal of the Process Buildings and the underlying concrete slabs, the radiological assessment of the underlying soil could not begin until the buildings and slabs were removed. Other areas onsite, such as the Site Pond, created its own set of challenges as incoming water had to be diverted and the pond dried before it could be fully evaluated and remediated.

Due to the significant diversity of tasks and activities that were required to be addressed before FSS of LSAs could be conducted, an overview is provided discussing each of these areas. This includes the Documented Burials, Undocumented Burials, Process Buildings, Vaults, Evaporation Ponds, Natural Gas Pipe Line, Red Room Roof Burial and Barns Areas, Sanitary Wastewater Treatment Plant, Site Pond/Site Creek Area, Tc-99 Area, Class 2 and 3 Survey Areas, Waste Disposal, and Backfill Operations.

2.1 Documented Burials

On-site burial was used as a disposal method for contaminated materials and wastes at Hematite from 1965, until 1970, in accordance with regulatory requirements and specific license authorizations. Detailed logbooks of these waste burials documented that there were 40 unlined pits located east of the site buildings as shown on Figure 2-1. These documented Burial Pits (collectively referred to as the Burial Pit Area) were used to dispose of waste materials generated by the fuel fabrication processes. These on-site burials were created under the governance of AEC regulations contained in 10 CFR 20.304 (1964, Reference 8.9). These regulations described the spacing of the pits, the thickness of the cover, and the quantity of radioactive material that could be buried in each pit. The nominal dimensions of each Burial Pit were 20 feet (ft) wide by 40 ft long by 12 ft deep and the regulations provided that these were supposed to include an approximate cover depth of 4 ft.

The site owner at the time, United Nuclear Corporation (and later Gulf United Nuclear Corporation) maintained detailed logs of waste burials occurring between July, 1965, and November, 1970. Each entry contains a date, a description of the waste buried, the weight of the Uranium measured or estimated for that waste, and a cumulative total of the Uranium buried in that particular pit. The weight of the contaminated item measured or estimated was determined to the nominal value of 1 gram which likely resulted in an over-estimate of the actual amount. Some entries also list the percent enrichment for the Uranium. The Burial Pit logs show a wide variety of wastes being buried in the pits; the majority of the listed waste is non-SNM waste, such as contaminated trash, drums, pails, bottles, rags, etc. Additional waste materials listed include Uranium process metals of various enrichments, metal wastes, liquid and solid chemical wastes, and HEPA filters.

The on-site burial of radioactive waste materials was terminated in November, 1970, as a result of an AEC violation issued to the Hematite facility for failure to adhere to revised AEC regulations concerning the quantity of material which could be buried onsite. An AEC Inspection Wrap-up Meeting memo (Reference 8.10), stated that a revision of 10 CFR 20 was enacted in June, 1970, that reduced burial limits for enriched Uranium. The licensee at the time had continued burials based upon the limits prior to June, 1970, resulting in the above AEC violation. It is noted that the Burial Pit logbook records, employee interviews, and the operational Uranium recovery process used during this time period consistently show efforts to maximize recovery and utilization of Uranium material whenever possible. Based on these records, Westinghouse believed that there was little likelihood the Documented Burial Pits contained significant quantities of recoverable SNM.

HDP developed consistent generic screening and handling approaches in preparation for the excavation and removal of buried wastes, contaminated soils, and sub-surface structures (e.g., concrete slabs, buried piping) in areas where it was determined fissile materials had a reasonable possibility to exist based on characterization data and historical knowledge. These approaches were analyzed from a Nuclear Criticality Safety (NCS) perspective in NCS Assessments (NCSAs) specific to buried waste exhumation, contaminated soil remediation, and sub-surface structure decommissioning. Although not an element of the FSS, screening for fissile materials and the required NCS controls were important due to the inherent safety significance of its performance during the remediation process. Screening typically involved duplicate performance of radiological surveys using sodium iodide scintillation detectors, and defined appropriate volumes of material to ensure that NCS limits were not exceeded. The objective of the *in-situ* radiological surveys was to identify any item or region of soil/waste with a fissile concentration exceeding 1 gram U-235 in any contiguous 10 liter volume. This provided a high degree of assurance that any items with elevated (i.e., nontrivial) levels of U-235 contamination would be identified. The in-situ radiological surveys were to be complemented by visual inspection of the survey area with the aim of identifying:

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

2) Items that resembled intact containers;

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

4) Metallic items.

While carrying out the fissile material screening and handling process the secondary remedial action objectives to identify hazardous materials (e.g., Volatile Organic Compounds or VOCs) and verify radioactivity concentrations of soil for potential use as backfill were also completed. In areas where fissile materials were suspected to exist based on historical knowledge, excavation continued until both visible and radiological evidence indicated that suspect materials had been removed. Once fissile material screening determined that fissile materials were not present in the remediation area in excess of the NCS Exempt Material Limit, NCS controls were curtailed. By making this the initial goal, remaining remediation, Health Physics, and Final Status Survey activities could proceed unencumbered by NCS controls. Identified items that exceeded NCS limits were segregated into designated Field Containers, which were placed in transport containers such as Collared Drums (CDs). Individual designated containers were then handled and stored in accordance with NCSA requirements.

In addition to the documented AEC authorized onsite burial pits, undocumented on-site burials were also conducted for disposal of general trash and items that may have been slightly contaminated. Prior to commencing excavation it was estimated that 20-25 of these non-AEC burials could exist for which there were no records. No written information was available that indicated the specific nature of the waste material buried in the undocumented burials. To provide clarity, the term "documented burial pit" refers to the AEC authorized burials that were identified in the site log books. The term "undocumented burials" refers to all other burials. The distinction is the "undocumented burials" were *not pits*, but shallow burials used to dispose site trash and debris. Both the documented burial pits and the undocumented burials were located within the larger area that is formally designated as the Burial Pit Area.

2.1.1 Remediation and Excavation

Excavation and removal of the Burial Pit Area soil was initially planned to begin at the northwest corner and continue towards the east and south. The soil excavation was to be performed in multiple burial pits concurrently ensuring sufficient space for heavy equipment to operate safely, while maximizing the handling of material available for re-use as backfill and minimizing cross-contamination. However, early in the excavation of the Burial Pit Area, Westinghouse recognized and made the decision that it was more efficient, provided a more comprehensive remediation process, and was more cost effective to excavate the entire area as opposed to trying to excavate each individual burial pit. This also had the added benefit of eliminating any need to correlate the size and content of the individual burial pits since not only were the pits being excavated, but so was any soil in between the pits.

The majority of materials buried in the Burial Pit Area were anticipated to be contaminated soil and trash, some laden with VOCs, floor tiles, glass wool, and laboratory glassware. Minor components of the buried waste volume were anticipated to include: acid-insoluble residue; filters; metallic debris; and, metallic oxides. However, because of the potential that fissile quantities of material could be found in the documented burial pits, excavations were performed in accordance with the limitations of the nuclear criticality safety assessment(s). In general, the order of techniques to be employed in removing soil and debris from the Burial Pit Areas was:

- Evaluate soil using in-situ gamma walkover surveys (GWS), VOC monitoring (with a Photo-Ionization detector), and visual inspection of the exposed surface, repeated for each newly exposed surface following the removal of each lift;
- Excavate and remove soil in nominal 6-inch lifts when under NCS controls, otherwise in 1-foot lifts;
- Excavate and segregate surface and subsurface soil based on: visual inspection; radiological and chemical survey/screening; supplemental sampling and analysis; the appropriate DCGLs; chemical Remediation Goals (RGs); and, the NCS Exempt Material Limit for potentially fissile material;
- Stockpile excavated soil at a safe distance adjacent to the excavation, or load into a haul truck for transfer to a Waste Consolidation Area (WCA) for further visual inspection;
- Using heavy equipment, excavate objects encountered in the soil; however, if deemed appropriate based upon GWS or NCS evaluation more precise methods and equipment could be used to excavate an object (e.g., hand-shoveling, small bucket excavator); and,
- Employ sloping and benching during the excavation process, as required, and continue until visible wastes are removed and in-process surveys and soil sampling meet specified acceptance criteria.

In March, 2012, excavation and remediation of the Burial Pit Area soil began in both the northwest and southeast corners of the Burial Pit Area simultaneously. Excavation and removal of soil was performed concurrently, which optimized the efficiency in handling of equipment and movement of material. The burial pit waste excavation activities concluded in December, 2014.

Initially the removal of the overburden began in 1 foot lifts as specified in the HDP DP. The intent of the 1 foot lifts was to ensure potentially fissile quantities of material could be identified for appropriate handling, and to enable waste to be identified and removed so a maximum volume of soil could be saved for reuse. This process was achieved through a GWS and visual inspection; any lumps of material that potentially exceeded 40g U-235 total were manually separated out and underwent a more thorough evaluation utilizing close proximity radiological surveys to determine if further handling requirements were necessary to maintain NCS compliance. At this point any material that was identified ex-situ to be greater than 15g U-235 required segregation from the waste stream, and special handling for NCS Control purposes. The NCS evaluation performed at the time identified that material in-situ less than 40g U-235 could go unidentified with a 12 inch lift, but any of the material identified ex-situ greater than 15g U-235 still required segregation. It had been determined that these were adequate ex-situ survey controls to provide a wide margin of Criticality Safety, and also demonstrate compliance with offsite waste disposal acceptance criteria. These activities were conducted under Procedure HDP-PR-HP-601, Remedial Action Support Surveys.

Within the first few months after excavation activities began, the NRC expressed concerns as to whether a NaI 2x2 detector could identify lumps of U-235 via in-situ scanning through a foot of soil, therefore potentially allowing material between 15g and 40g U-235 to be excavated prior to segregation. After further review and discussion with the NRC, HDP made the decision to perform subsequent NCS controlled excavations in 6 inch lifts. Soil/debris in areas not subject to NCS controls was still excavated in 12 inch lifts. This change was initiated in August, 2012, approximately four months after site excavation activities began and resulted in a revision to Procedure HDP-PR-HP-601.

The general locations of the former burial pits were known based on reviews of historical records, such as site aerial photographs taken during the time period the burials occurred, and the field observations of depressions that were visually discernible in the Burial Pit Area ground surface. The localization of the pits was further helped based on visual clues, and physical work activities performed during the excavation activities. Specifically, as the overburden soil was removed it was easy to visually identify the location of a burial pit based on a change in soil color. Even the undocumented burials could be easily identified by a change in soil color in spite of the fact that their size and shape was not as well defined as the documented pits. See Figures 2-2 and 2-3. Additionally, the equipment operators conducting the excavation could distinguish when they were digging in a burial pit based on the difference in the hardness of the soil. Workers could even detect the difference in the soil hardness when walking over a burial pit, which tended to be soft and spongy. Adding to the visual and soil hardness cues, the burial pit was also radiologically identifiable based on a GWS once reaching the contaminated layer. Figure 2-4 shows a radiation technician conducting surveys with a NaI 2x2 detector during burial pit remediation activities.

Overall the process of locating, assessing, and removing debris from the burial pits was very labor intensive. For example, on several occasions hundreds of small plastic vials were found. In each case, every vial had to be separately surveyed by two independent technicians using different instrumentation. While the majority of materials located within the burial pits were as anticipated, 215 radium contaminated filter plates made of steel, cast iron, and plastic, about 3 feet by 3 feet in size, were unexpectedly found. See Figures 2-5 and 2-6. It was determined that these were brought to the Hematite site from an offsite entity and did not originate from any onsite process or operation. These plates required a significant amount of time and effort to be decontaminated prior to disposal which resulted in the single largest contributor to worker dose (603 mrem max individual annual exposure) since the DP was approved in 2011. Figures 2-7 and 2-8 also show several examples of some of the waste removed from the burial pits.

As excavation and remediation of the Burial Pit Area progressed, it became apparent that most of the buried debris was located in the north and south ends and typically in closely aligned pits, while the middle area had minimal debris and contamination. As sloping and benching practices were employed, and due to the close nature of the pits, large areas ended up being remediated as opposed to individual standalone pits. This had the advantage of providing additional assurance that any radiological contaminate that could have migrated laterally from a pit was also likely remediated. Also as expected, the burial pits were generally of similar depth. This was expected because of the equipment that would have in all likelihood been used to dig the pits. A normal sized backhoe would have been the expected heavy equipment employed, which has a typical maximum reach of about 10 feet (plus or minus) when digging a trench. In addition, the depth of the pit was constrained by the regulatory requirement of 12 feet deep with 4 feet of cover. This knowledge helped when excavating a burial pit as it provided an informal bound of 16 feet for the depth of the pit, as well as providing a solid basis for not expecting two burials to be placed one on top of another.

Figures 2-9 through 2-18 provide a chronological aerial view of the Burial Pit Area excavation progress from March, 2012, when the removal of overburden soil had commenced through September, 2015, during backfilling of the area.

2.1.2 Ceasing Excavation

As excavation progressed in the Burial Pit Area, five activities came into play that determined the extent of remediation in a given survey unit (SU). These were: 1) ongoing remedial action support surveys (RASS), 2) conducting core bores to support release from criticality controls, 3) performing a final RASS, 4) sampling for VOC concentration in soils to determine if the chemical cleanup Remediation Goal (RG) had been met, and 5) conducting final status surveys (FSS). The RASS was conducted to: guide remediation activities, determine when an area or survey unit had been adequately prepared for FSS, and provide updated estimates of the parameters to be used for planning the FSS. During soil excavation, the RASS would serve to assess the potential concentration and amount of U-235 for comparison to the NCS Exempt Material Limit. In areas subject to NCS controls, a GWS with visual inspection was required to be performed independently by two different HP Technicians using two separate instruments prior to excavation of each layer of soil. In conjunction with the GWS, remediation areas were visually inspected prior to exhumation of any material. Once excavation of a SU reached a point where there was no longer any indication of burial pit waste, and the RASS indicated the soil met the NCS Exempt Limits, core borings were conducted. The core bores were required to be dug 7 feet deep below the original pre-excavation surface, with the additional constraint that it must also extend at least 3 feet below the excavated surface. This meant that any excavation greater than 4 feet deep would require a core bore of 3 feet below the excavated depth. While the purpose of the core bores was only to verify that NCS controls could be reduced, it also provided an additional opportunity to support the remediation objectives by allowing for a visual inspection to ensure that there was no longer an indication of a burial pit below the excavated elevation (surface). Core bores were performed using an auger of anywhere from 3 to 8 inches in diameter either manually or as an attachment to a skid steer. See Figure 2-18a.

These core bores were systematically placed based on a maximum 20 foot grid, eventually covering the entire Burial Pit Area. Core cuttings and the core bore hole were surveyed to provide radiological data to determine whether NCS controls could be suspended for that SU. The data was evaluated against pre-determined NCS Limits. The core cuttings were evaluated against a value of 47k net counts per minute (ncpm), and the core bore holes were evaluated against a value of 63k ncpm. These values were derived for a lump of uranium under two and six inches of soil respectively. The values differ because the cuttings were spread out in a layer while the bore hole core measurement was taken from the wall of the bore hole. The methodologies and results are detailed in NSA-TR-10-12, *Calibration Analysis for ^{235}U Response from Burial Pit Waste Materials at the Hematite Facility*. If the core bore data was less than the above criteria and no waste material indicative of a burial pit was found, it was used as a decision point that assumed the bottom of the burial pit had been reached and criticality controls were no longer necessary. Even though additional radiological remediation may still have been required for the soil surrounding a burial pit, by eliminating the NCS controls, work could progress at a faster pace and with fewer resources. (As a note, core bore data was not used in the development of the FSS Plan. It was only used in the assessment of the need for NCS controls.)

Following removal of NCS controls, remediation would often continue to remove any remaining radiological contamination in the soil surrounding the burial pit debris field. This work would continue in conjunction with RASS until it was determined the DCGLs for that SU were be met. At this time, radiological remediation was complete unless additional remediation was required based on subsequent FSS activities. However, excavation in many cases continued after radiological concerns were addressed due to the need to remove VOCs in the soil. In a number of cases, VOC remediation resulted in the removal of soil down to the phreatic surface. Once all radiological and VOC remediation activities were completed, a final RASS survey was performed to ensure the area met required DCGLs and to obtain data for FSS design. The FSS was then performed to demonstrate the SU met the NRC unrestricted release requirements based on a GWS and soil sampling results.

2.1.3 Retrospective Review

As the remediation of the Burial Pits neared completion, two issues were identified for examination as a retrospective review to provide assurance that all contaminates related to the burial pits were adequately removed. The first issue involved the potential for a radioactive contaminate to migrate through the soil and collect in an area below where a burial pit or radioactivity was known to have existed. In this case, because an area of elevated contamination could pose a long term risk factor, it was deemed sensible to examine the potential that an area like this could still exist in spite of the extensive remediation activities that had been performed. The second issue was whether another burial "pit" could potentially exist below where the current excavation activities ceased.

To address the first issue, Westinghouse evaluated the potential for the existing radionuclides identified to migrate through the soil. As discussed in Volume 1, Chapter 1 of the FSSFR, the primary nuclides of concern at Hematite were U-234, U-235, U-238, Ra-226, Th-232, and Tc-99. Of these radionuclides, only the Tc-99 had the potential to move downward through the soil and collect within a timeframe consistent with existence of the Hematite facility. The other nuclides are generally non-soluble and more readily bind to soil particles which retards their movement. Because Tc-99 is less likely to bind to soil and is more mobile it will travel more readily with water. A detailed discussion of Tc-99 and limestone as its source is provided in Section 2.10.

A review was conducted of the Burial Pit Area surface and sub-surface soil sampling data collected between 2004, and 2008, and summarized in the Hematite Radiological Characterization Report. The review indicated that 94 surface soil samples (at 0.0-0.5 feet) were analyzed for Tc-99. Fifty-five of the 94 samples had Tc-99 activity above the Minimum Detectable Concentration (MDC) with a maximum result of 68.3 picocurie per gram (pCi/g) and an average of 2.6 pCi/g. All of these areas were remediated when the overburden was excavated.

There were 89 soil samples within the root zone (0.5-5.0 feet) that were analyzed for Tc-99. One sample had a value of 33.8 pCi/g, and this area was remediated. Twenty-one of the 89 samples had Tc-99 activity above the MDC with a maximum result of 14.8 pCi/g and an average of 1.0 pCi/g. None of the results were in excess of the Tc-99 Uniform DCGL of 25.1 pCi/g or the Root Stratum DCGL value of 30.1 pCi/g. Although below the DCGL, many of these areas were still remediated in conjunction with excavation activities.

There were 144 soil samples within the deep zone (> 5.0 feet) that were analyzed for Tc-99. Twenty-seven of the 144 samples had Tc-99 activity above the MDC with a maximum result of 38 pCi/g and an average result of 0.75 pCi/g. None of the results were in excess of the Tc-99 Excavation DCGL value of 74.0 pCi/g. Regardless, many of the areas where these samples were collected were also remediated. Also reviewed were 106 soil sample results of Tc-99 in the deep zone from 16 feet down to a maximum of 35 feet. The highest value recorded was 9.34 pCi/g, which indicated the limestone had contributed very little Tc-99 at depth in the Burial Pit Area. While Tc-99 can migrate readily in soil the data shows that it was not an issue in the Burial Pit Area based on characterization results obtained between 2004, and 2008.

There was also a concern that previous hybrid groundwater monitoring wells could have provided a conduit for Tc-99 to migrate downward. A hybrid well was a well that had been installed with a screen that extended from the overburden clay to the sand-gravel aquifer, which could allow transport between layers. Seven hybrid wells (BP-17, BP-20A, BP-21, NB-61, WS-25, WS-27, WS-29) were installed in the Burial Pit Area in 2004. From each of these seven wells, Tc-99 water sample results were evaluated to determine if any well analytical results for Tc-99 exceeded a threshold value of the MDC plus any measurement error. Only well BP-17 exceeded this threshold value, with a value of 43.2 pCi/L Tc-99 on 9/15/2008. (The EPA drinking water standard for Tc-99 is about 900 pCi/L.) This well was located in LSA 10-12, in the area where the limestone was buried so the higher level of Tc-99 in this well was not unexpected.

In 2011, Westinghouse committed to investigate the potential for a preferential pathway of Tc-99 and uranium along a monitoring hybrid well, and to determine whether contaminated soil existed in proximity to a hybrid monitoring well (HEM-11-56, May 5, 2011, Attachment 1). When hybrid wells were abandoned they would 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 would continue until reaching refusal (typically bedrock) or a depth of 35 feet. The soil cuttings that were removed during the boring process would be surveyed for indications of elevated radioactivity as a qualitative measure and sampled by laboratory analysis.

In 2006, hybrid well NB-61, which was located in LSA 10-11 and was close to the natural gas pipeline was abandoned. Since this pre-dated HEM-11-56, cuttings from this well were not collected for analysis. In 2012, the remaining six hybrid wells (BP-17, BP-20A, BP-21, WS-25, WS-27, WS-29) plus three groundwater monitoring wells in the Burial Pit Area were abandoned. When the wells were abandoned they were over drilled as required. Soil cuttings that were removed during the boring process were surveyed for indications of elevated radioactivity as a qualitative measure and sampled for laboratory analysis. For the six hybrid wells and three groundwater monitoring wells abandoned in 2012, radiological samples were collected in the cuttings of each 5 foot interval to the bottom of the boring which was in the range of 28-35 feet below ground surface. A total of 62 samples were collected. The maximum Tc-99 concentration identified was 21.1 pCi/g from the 5-10 foot interval of the BP-17 well cuttings. This area was subsequently excavated. In 2013, to better characterize the area around BP-17 and assess the potential extent of Tc-99 contamination four core borings were conducted surrounding BP-17. The highest reading was in the boring east of BP-17, which showed a maximum Tc-99 value of 59.6 pCi/g at 28-30 feet.

Subsequent to the initial four core borings, another six investigation borings were conducted in 2013, in the area where BP-17 was located. A total of 86 soil samples were collected from the ground surface to bedrock from the ten borings conducted. None of the sample results from the four initial cores exceeded the applicable Tc-99 Excavation DCGL of 74 pCi/g. The maximum Tc-99 value identified was 59.6 pCi/g at 28-30 ft on the boring collected just east of BP-17. The remaining samples collected in the other three borings contained a maximum value of 4.25 pCi/g. The maximum values of Tc-99 identified from the additional six investigation borings was 33.1 pCi/g at 16-20 feet.

Surficial deposits of spent limestone were located in the central portion of the Burial Pit Area along the eastern slope. The spent limestone was remediated and the sent to US Ecology for disposal. The overall depth of the limestone was approximately 2 – 4 feet below original grade.

In summary, based on sample results, it has been determined that there is very little probability that Tc-99 could exist at levels that exceed the Excavation DCGL at a depth below where excavation activities ceased because there was very little Tc-99 that existed prior to excavation and remediation activities in the majority of the burial pit area. The Tc-99 that was identified was predominately in the areas where limestone had been placed, which was expected since the limestone was the path by which the Tc-99 was introduced into the waste stream. In addition, the areas where the Tc-99 was identified were ultimately excavated.

To address the second issue of whether another burial pit could potentially exist below where the excavation activities ceased, a number of factors were considered. In some locations it is clearly demonstrated that another burial pit could not exist because the excavation went to the phreatic surface or sample results from core borings that went to bedrock showed there was no additional radiological contamination, while in other cases it is demonstrated through a combination of a number of factors.

The first things considered were the nature of the burial pits themselves. The documented licensed burial pits were fairly easy to identify by location based on historical records, examination of site aerial photographs taken during the time period the burials took place, and depressions visually discernable in the ground surface. In none of the historical records, photographs, or anecdotal evidence, including input from former Hematite workers, was it indicated, observed, or implied that burial pits were stacked one on top of another.

In regards to the contents of the burial pits, the characterization studies performed prior to the development of the DP had identified numerous radionuclides of concern (ROC) that were used to define the site DCGLs, which included radium as a ROC. While there was no documented historical process or other licensed activity onsite that would explain the presence of radium in the burial pits, it had none-the-less been identified. It was therefore not unexpected and it was anticipated that radium would be encountered during the remediation of the Burial Pit Area.

As could be expected, based upon the documented history of site operations and the information contained in the logbooks, the number and size of the radium contaminated filter plates found during the Burial Pit Area remediation was not anticipated. It is understandable then to assume that previous owners of the facility may have allowed the radium contaminated filter plates generated at another location to be buried in the Burial Pit Area. No other unanticipated waste was identified during remediation. The decision to remediate the Burial Pit Area in its entirety rather than locating individual burial pits for subsequent remediation proved to be a prudent and invaluable decision. The process provided an extremely high degree of assurance that all radioactive wastes were identified and removed from the Burial Pit Area, regardless if the specific type of waste or radionuclide was expected or not.

Ultimately, the excavation remediation activities bore out what was expected in regards to burial pit configuration, size, location, and the radionuclides of concern based on all the above sources of information. No burials (documented or undocumented) of any kind were found in any other location on the site, or one burial pit on top of another burial pit anywhere within the excavated

Burial Pit Area during excavation remediation work.

The second factor to consider is the visual nature of the burial pits. At the start of the Burial Pit Area excavation the overburden layer of soil was removed. Once this overburden layer was gone, the location of a burial pit was visually observable based on the discernable change in soil color. They could also be identified by the equipment operators conducting the excavation by the difference in the hardness of the soil, and even the workers could detect the difference in the soil hardness when walking over the burial pit. In essence, the burial pits were very easy to visually identify. Also of significance is the burial pit could be radiologically identified, even before it was fully uncovered. In some instances the actual buried debris was encountered a few feet below the observable change in soil color/texture, but was still identifiable using a 2 x 2 NaI detector. Once the bottom of the burial pit was reached it was evident by the change in color of the soil, the hardness difference between the softer burial pit material and the hard native soil, a disappearance of debris, and, a sharp decline in radiological readings. Figure 2-18b is an example of the evident change in color of the soil as excavation is nearing the bottom of a burial pit. It is important to be aware that had another burial existed below an already excavated burial pit, the soil between the "lower" and "upper" pit would have to have been disturbed, which would have been visually evident, and this was never identified to occur. Overall, the visually observable change in the soil conditions provided a very good confirmation that there was no buried waste located further down.

As excavation/remediation was conducted, the depth at which the remediation was necessary to remove radiological debris varied throughout the burial area. Radiological surveys were conducted continually as the burial pits were remediated. Prior to NCS controls being suspended, core bores were taken to verify the burial pit had ended as evidenced by visual inspection, the radiation readings from the bore holes, and the bore hole cuttings. These core bores were systematically placed based on a 20 foot grid over the entire burial pit area. A technical basis document (HDP-TBD-NC-205, *Assessment of the Adequacy of Lateral Subsurface Soil Sampling in the Burial Pit Area*) was developed to statistically verify the 20 foot spacing was acceptable to ensure that an undiscovered burial pit should be identified with a high level of confidence. Figure 2-19 provides a map showing the locations of all the core bores that were conducted for criticality control purposes. While this process was not conducted for the purpose of demonstrating a SU was ready for FSS, the fact that over 600 bore holes were drilled did provide additional assurance that no other burial pit existed below the depth of an already remediated pit.

Separate from the radiological remediation activities, excavation was also necessary to comply with State of Missouri regulatory requirements for VOCs. To meet the RGs for VOCs, a large portion of the Burial Pit Area was remediated beyond what was required to be remediated for radiological contamination. In some cases, this resulted in the entire core boring previously conducted for criticality concerns to be dug out as the excavation activities continued. In several locations, soil was excavated down to the phreatic zone. After all these activities, not one instance was identified where a second burial pit was located below an existing burial pit, or where radiological remediation activities had ceased.

A review was conducted to examine sample results from soil borings carried out from 2003, to 2014, for site characterization purposes. Soil samples from these borings were analyzed using gamma spectroscopy for any radionuclides and separately assessed for the presence of Tc-99. These borings provided evidence of both buried waste and the lack of buried waste. Due to criticality concerns, many initial borings were intentionally placed to avoid the burial pits. In subsequent characterization efforts, borings were conducted in the burial pits as well. Overall this provided a good mix of data that was representative of the entire Burial Pit Area. Figure 2-20 provides a map showing the locations of characterization core borings in the Burial Pit Area. Based on the review, none of the sample results showed there was contamination at a depth that would be indicative of a second burial pit below an existing burial pit. The core boring data was also examined against final excavation elevations. While these results are summarized within the FSSFR Report for each individual SU, in summary, all areas of contamination identified through characterization surveys were remediated entirely or at a minimum, to levels below the applicable DCGL for that stratum and SU.

Lastly, following the completion of all remediation activities, and in accordance with the DP, final status surveys were conducted that involved a 100% walkover scan performed with a 2 x 2 NaI detector, and a prescribed number of systematic soil samples (and additional biased soil samples as necessary) collected for analysis. This was completed for each SU within the Burial Pit Area.

Figure 2-21 provides a map of the post remediation excavation depths in the Burial Pit Area. The depths shown are the feet of material excavated from the original surface elevation at that location. The maximum excavated depth was approximately 24 ½ feet.

Based on the above retrospective review, all the above factors in combination provide an extremely high level of assurance that there are no undiscovered burial pits or localized elevated areas of contamination that were not remediated. This is based on, 1) the ability to visibly observe the locations of the burial pits during remediation activities, 2) criticality related core borings that were performed, 3) additional excavation activities to remove VOCs, 4) not one instance was documented in the log books, stated by former plant workers, or identified during excavations throughout the burial area that a second burial pit was located below an existing burial pit, 5) characterization data based on borings in the Burial Pit Area demonstrates all known areas of radiological contamination were already below or were remediated below the applicable DCGLs, and 6) the acceptable analytical results of FSS soil sampling and 100% gamma walkover surveys.

2.2 Undocumented Burials

2.2.1 History

Interviews with former employees indicated that undocumented on-site burials may have occurred as early as 1958, or 1959. Available employee interview records indicate that three or four burials may have been performed each year, prior to 1965, for disposal of general trash and items that may have been slightly contaminated. Accordingly, it was estimated that 20-25 of these non-AEC 10 CFR 20.304 burials could exist for which there were no records. Burials prior to 1965 were not documented (logged), as they were not considered to contain significant quantities of SNM, and were not known to contain radioactive wastes. No information was available that indicated the specific nature of the waste material buried in the undocumented pits. Additionally, no evidence was found that indicated that burial of known uranium-bearing materials occurred prior to 1965.

These undocumented burials were also believed to have been in the same general area as the Documented Burial Pits, and/or were in the proximity of site buildings in the eastern portion of the Central Tract (see Figure 2-12). Also, no specific information was located that indicated the specific nature of the waste material buried in these undocumented burial pits. Additionally, no evidence was found to indicate that burial of known uranium bearing materials (i.e., levels greater than free-release criteria) occurred during this time period.

2.2.2 Remediation and Excavation

Remediation and excavation activities for the Undocumented Burials were a continuation of and conducted in the same manner as the Documented Burial Pits. Initially, the entire area where documented Burial Pits were expected to be found was excavated to a depth of at least 4 ft to identify any burials. This depth was based on the AEC requirement that at least 4 ft of cover be placed over any regulated burial pits. As this process proceeded, undocumented burials, mostly to the west of the documented burial pit area, were also being uncovered. This was consistent with interviews with former licensee employees. These undocumented burials were found to have a minimal amount of cover, often just one or two feet, and there was no consistency to the shape and depth of these burials.

Although there was no evidence that any of these undocumented burials would contain fissile quantities of material, excavation practices including NCS controls remained in use until such time as it could be demonstrated the controls could be lifted. Figure 2-22 depicts the original Documented Burial Pit Area and the outer boundary of the area defined by the extent of the locations of the undocumented burials that were eventually uncovered. No documented burials were located outside of the area where the burial pit logs indicated they would be, but some undocumented burials were located within the documented Burial Pit Area.

Debris remediated from the undocumented burials was generally less contaminated than debris from the Documented Burial Pit Area and consisted of mostly trash and construction debris. Radiation surveys of this material identified the same nuclides identified in the Documented Burial Pit Area but typically at lower concentrations. These materials were dispositioned as required following the same practices employed for the documented burials. From an operational standpoint the excavation and remediation of the undocumented burials was conducted in the same manner as the documented burials.

2.3 Process Building

2.3.1 History

The primary structure removed during site remediation/demolition activities was the Process Building, which collectively included Building 240, Building 253, Building 254, Building 255, Building 256, and Building 260. The Process Building housed equipment associated with the chemical conversion of Uranium into compounds, solutions, and metals, and for the fabrication of Uranium compounds into physical shapes. The former location of these buildings is shown in Figure 2-23. An aerial view of the Process Building prior to demolition can be seen in Figure 2-24.

In June, 2001, Hematite ceased principal activities and shut the facility down. Although License Amendment No. 52 was issued in June, 2006 (Reference 8.12), authorizing the dismantlement and demolition of buildings, Westinghouse utilized the period between ceasing licensed operations in 2001, and initiating building demolition in 2011, by conducting decontamination of the various Hematite buildings to facilitate demolition. This included the removal and disposal of all Process Building process equipment and components, and remaining product.

2.3.2 Demolition

Demolition sequence of the Process Building included the removal of the processing equipment, then building demolition, and then the removal of the foundations, floor slabs, and associated drains, with all debris being shipped off-site for disposal.

Demolition of the Process Buildings took place during May, and June, 2011. To minimize the potential spread of contamination during demolition, the entire interior surface of the Process Building was sprayed with a chemical control fixative. An additional measure included spraying water on demolition debris to reduce dust and other airborne particulates.

Between May, 2013, and November, 2015, the Process Building slab was removed. The mid-easterly portion of the slab was left in place to be used as a haul road for the movement of vehicles and heavy equipment around the site in support of decommissioning activities. The haul road was removed in early November, 2015. Removal of the Process Building slab (except Haul Road) then allowed for access to the underlying soil for remediation. Excavation and remediation activities on the soil under the slab commenced in May, 2013, and all remediation of the slabs including under the Haul Road was completed in January, 2016.

2.3.3 Remediation and Excavation

Based on historical records and interviews it was identified that non-native material was introduced under the existing Process Buildings near the existing northwestern side of Building 253. As part of preparing for the construction of Building 253, native soil was excavated due to the presence of gross alpha radioactivity exceeding 30 pCi/g and spent limestone was used as backfill. Because of concerns about undermining Building 240, the excavation was stopped before all soil exceeding this limit was removed. The average alpha concentration in soil on the surface at the conclusion of this excavation was 17 pCi/g. The maximum alpha concentration was 82 pCi/g. Per "Building 253 Construction Site Soil" (Reference 8.13), this historical excavation area reached a maximum depth of approximately six feet below the planned floor level. Based on this description, the maximum depth of the excavation was likely to have been approximately eight to ten feet. However, based on interviews with personnel that were present when Building 253 was constructed, the excavation depth may have extended to a depth of 10-15 feet.

Prior to placing backfill, the then owner, Combustion Engineering, requested that NRC allow spent limestone that was stored on-site to be used as backfill material within the excavation area under Building 253. Per NRC letter to ASEA Brown Boveri/Combustion Engineering, "Authorizes Backfill of Area of Building 253 Construction Site" (Reference 8.14), dated July 2, 1990, the NRC allowed spent limestone from two piles meeting a 30 pCi/g limit (alpha) to be used as fill below Building 253 with the understanding that the fill would have to be removed upon facility decommissioning. Per a Combustion Engineering letter to NRC, "Spent Limestone Results" (Reference 8.15), dated April 7, 1989, the two limestone piles had average gross alpha concentrations of 7 pCi/g and 8 pCi/g.

Due to the presence of residual radioactivity in soil that could not previously be removed, and due to the presence of spent limestone as backfill, these areas were required to be excavated to the extent that all limestone was removed. This issue was addressed in a letter to the NRC, Westinghouse (E. K. Hackmann) letter to NRC (Document Control Desk), HEM-11-56, dated May 5, 2011, "Evaluation of Technetium-99 Under the Process Buildings" (Reference 8.16).

Remediation of the soil under the slab was conducted with the same procedures and practices employed for all other open land areas. All limestone used as backfill to support the construction of Building 253 was removed and shipped as waste, as were any other soils/materials identified as waste requiring offsite disposal.

The Process Building area also contained subterranean process piping. All of this piping was removed and disposed as waste. While additional areas of contamination were identified in the soil during the removal of the Process Building slab and subterranean piping, these areas were due to spills originating from the former Process Building operations and not due to burials. Figure 2-38 is a photograph of the Process Building Area showing the excavation and remediation that occurred under the Process Building slab. Figure 2-39 is a photograph taken on November 7, 2015, showing the area after removal and remediation of the haul road along with the remaining subterranean piping.

FSSFR Volume 4, Chapter 1, Building Survey Areas Overview, provides further photographs and discussion regarding the history, demolition and disposal of the Process Building.

2.4 Storage Areas/Vaults

2.4.1 History

The two storage areas/vaults constructed at the Hematite facility were the South Storage Area and the West Storage Area.

The South Storage Area was a standalone building (Building 252, see Figure 2-23). Constructed in 1960, in response to an increased need for storage, the building was used for storing process materials as well as final product storage. In 1974, all equipment was removed from the South Storage Vault and this area was decontaminated. Subsequently, during the commercial nuclear era, this building was used for radioactive waste storage.

The West Storage Area was located in Building 235, which was also a standalone building. The original building was constructed in 1956, housed the West Storage Area, and was utilized as the outgoing storage building where final Uranium products were stored. During the commercial nuclear era this building stored source material. In 1992, when Building 230 was constructed, Building 230 was placed such that the two buildings shared a common wall.

2.4.2 Demolition

The West Storage Area was not demolished until March, 2015. The demolition of this vault was delayed in case sufficient fissile material was identified during excavation of the Burial Pit Area that required the vault to meet storage requirements. No quantities of fissile material were ever found that met this requirement. Once the Burial Pit Area excavation and remediation was completed, the West Storage Area was demolished.

Building 252 was demolished in May, 2001, during the demolition of the Process Building.

Prior to demolition both vaults were sprayed with a chemical control fixative to limit airborne dust, as well as being sprayed with water during demolition activities. Neither facility contained a drain or underground piping, nor was significant radiological contamination identified in the soil beneath the buildings following demolition and removal.

FSSFR Volume 4, Chapter 1, Building Survey Areas Overview, provides photographs and further discussion regarding the history, demolition, and disposal of the Storage Buildings.

2.4.3 Remediation

The soil underlying Building 252 was located in LSA 08-12 and was remediated during the remediation of that survey unit. The soil underlying Building 235 was located in LSA 08-17 and was remediated during the remediation of that survey unit.

2.5 Evaporation Pond Area

2.5.1 History

The Hematite facility had two Evaporation Ponds that were placed into operation between 1962, and 1964, that were used for on-site disposal of process filtrates, low-level contaminants, and high-enriched and low-enriched Uranium materials. Use of the evaporation ponds was discontinued in 1978. In 1979, 1985, and again in the 1990's, these ponds were partially remediated. The Ponds had been known to overflow during periods of high precipitation, thereby impacting the soils around the Evaporation Ponds. Employee interviews and recent experience confirmed overflows have occurred, which made it an expectation that remediation of this area would be required.

Figure 2-40 is from an aerial photograph taken in September, 2012, showing one of the Evaporation Ponds (with a tarp covering to minimize water intrusion) and the location of an adjacent 8 inch natural gas pipe line (red line). The second pond has been covered with soil, but is located adjacent to and below (towards the bottom of the photo) the visible pond. Figure 2-41 provides a second view that shows the extent of the second pond (covered over) below the first pond. A historical site review determined that the ponds were placed into operation after the Natural Gas Pipeline (NGP) was constructed. There was a concern that contamination from the Evaporation Ponds could have migrated under the NGP. Over most of its onsite traverse the depth of the NGP is approximately 4½-5 ft, but in the vicinity of the evaporation ponds the NGP is 4-7 ft in depth. The close proximity of a Union Pacific rail line is also seen in Figures 2-40 and 2-41.

The Evaporation Ponds consisted of a primary Pond and a larger, secondary/overflow Pond with a 1.5 foot berm around each Pond. The Ponds were originally lined with approximately 10 in. of rock (nominal diameter of 0.5 to 3 inches). The size of the primary Pond was approximately 30 ft by 40 ft, and the secondary Pond was 30 ft by 85 ft. While the Evaporation Ponds were designed and built to receive filtrates from the low enrichment processes, they were also used for the retention of both high and low enrichment recovery waste liquids. Historical documentation also indicates retention of other liquid waste solutions in the Evaporation Ponds. Examples of these waste liquids include acidic cleanup solutions, organic solvent solutions (perchloroethylene and trichloroethylene), oils, building sump contents, and mop water. The precipitates and solids were allowed to settle and the water evaporated naturally. As additional liquids were added to the primary Pond, the overflow flowed through a pipe into the secondary Pond.

After CE purchased the Hematite facility in 1974, use of the Evaporation Ponds was curtailed to allow only the retention of spent potassium hydroxide scrubber solution from the Uranium dry recycle process and liquids from startup testing of the wet recovery process. Use of the Ponds was discontinued altogether in September, 1978. In 1979, 700 ft^3 of sludge was pumped from the primary Pond. The sludge was dried and shipped to a licensed burial facility between 1982, and 1984. Additional decommissioning efforts for the Evaporation Ponds were undertaken by CE in 1984, in response to NRC directives. As a result, in 1985, CE removed approximately 2,800 ft^3 of sludge, rock, and soil from the primary Evaporation Pond. Detailed sampling following the remediation effort determined the average total Uranium contamination of soil in the Evaporation Pond was below the 250 pCi/g total Uranium decontamination limit set by the NRC, however, spot contamination levels in excess of the limit remained. Approximately 1,200 ft^3 of soil and rock were also removed from the secondary Evaporation Pond in 1987. Subsequent soil/sediment samples collected from the Evaporation Ponds following these remediation efforts revealed an average concentration of Uranium in the Evaporation Ponds below the 250 pCi/g limit; however, individual sample results showed that soil/sediment contamination levels in excess of the limit remained.

On May 4, 1995, a decommissioning plan for the Evaporation Ponds was incorporated by amendment into the site license. Following additional characterization of the Evaporation Ponds, this decommissioning plan was revised based on more extensive characterization results (References 8.17 and 8.18). The Evaporation Ponds' decommissioning plan was implemented over the next four years and resulted in the removal of approximately 6,000 ft^3 of additional soil/sediment for disposal. Surveys and sampling of the Evaporation Pond area conducted in 1999 indicated an average concentration of 170 pCi/g U-235, with several samples yielding higher results, up to 745 pCi/g U-235. In addition, Uranium concentrations of approximately 100 pCi/g were detected at depths of 10 ft below ground surface (Reference 8.19). Remediation efforts associated with the Evaporation Ponds were suspended in 1999, to evaluate additional remediation techniques and options.

Because of the known levels of contamination in the area of the Evaporation Ponds, additional characterization was performed between 2004, and 2014. The objective of these characterizations was to fully understand the extent of the contamination to ensure that excavation and remediation could be conducted safely and effectively. A total of 136 samples were collected from the surface to depths of over 30 feet. Because the NGP line runs through the Evaporation Ponds, remediation activities in this area created a potential risk of damage to the 58 year old line. While this risk was low, the resulting consequences if the natural gas were ignited would be unacceptably high when considering the safety of workers and members of the public. The primary concern with excavating near the NGP was the need to provide a required slope (1:1) to any excavation, or use some type of side-wall shoring. The deeper the excavation the further out a slope would need to extend to ensure required worker safety. To achieve the same level of remediation of the Evaporation Ponds as conducted elsewhere on-site, the depth of the excavation would require sloping that could potentially undermine the NGP.

Based on the results of samples collected from 2004, to 2014, it was determined that in most cases, identified contamination was between the surface and a depth of 10 feet and could be remediated without significant concern for the NGP. However, several sample locations at depths of 20 to 30 feet showed Tc-99 in excess of the DCGL. The maximum value identified was 221 pCi/g, at a depth of 26 to 28 feet, in the southwest portion of the Evaporation Ponds (Sample location EP-2014-6). (Tc-99 was the only nuclide of concern that was identified in excess of the DCGLs at depth in the vicinity of the Evaporation Ponds.) Due to these sample locations' close proximity to the NGP, normal excavation practices were not possible to reach that depth while maintaining a 1:1 sidewall slope to ensure worker safety. For this reason an additional systematic sampling initiative was undertaken for the Evaporation Pond area in April, and May, 2015, which included collecting samples where the previously high Tc-99 samples had been identified at depth.

The initiative was comprised of 21 core borings in a systematic grid and nine additional core borings located where previous sample results had shown elevated Tc-99 contamination. (Figure 2-41a shows the locations of the core borings that were conducted during the initiative.) The borings were conducted to collect soil samples in four foot increments to a depth of 35 feet or until refusal (e.g., bedrock), which typically occurred at a depth close to 32 feet. The grid itself was laid over LSA 08-11, which had been previously adjusted to cover just the area of the

Evaporation Ponds. Sample EP-15-27 was collected in the same location as EP-2014-6, where the sample containing 221 pCi/g of Tc-99 was taken. At a depth of 16-20 feet, EP-15-27 showed 82.0 pCi/g of Tc-99. In the 20-24 and 24-28 foot sections the results showed 4.34 and 4.19 pCi/g respectively. The sections of soil above 16 feet were all between 2 and 3 pCi/g. The value of 82.0 pCi/g was the highest sample collected from the total 212 samples collected during the initiative. Core bore EP-15-07, which was drilled very close to EP-15-27 showed 26.0 pCi/g at 16-20 feet. The second highest reading collected was 50.50 pCi/g from EP-15-04, also at 16-20 feet. EP-15-04 was southwest of EP-15-07 and EP-15-27, and at the southwest end of the Evaporation Pond. Sample locations EP-15-01, -02, -05, -08, -11, -28, -29 and -30, all to the southwest, south and southeast of EP-15-04, -07 and -27 (the prevailing direction of groundwater flow) were less than or equal to 14.0 pCi/g

While this initiative was not the final FSS for LSA 08-11, it was performed with the intent that the data would be used in support of demonstrating that the dose from LSA 08-11 would be less than 25 mrem. Additional detail supporting the release of LSA 08-11 is provided in the FSSFR for LSA 08-11.

2.5.2 Remediation

With the information provided by the April, and May, 2015, core bore sample results the extent of excavation for the Evaporation Ponds was further defined as well as providing the necessary information to ensure safe remediation near the natural gas pipe line. Remediation was completed in the same manner as the other LSAs at HDP. Figure 2-42 shows the excavated Evaporation Ponds.

2.6 Natural Gas Pipeline Area

2.6.1 History

An 8 inch diameter high pressure natural gas transmission pipe line runs the length of the southeast (SE) border of the Hematite Site. The NGP was installed in 1956, and is a sole source of natural gas for local communities. In 2008, Westinghouse used air-knife equipment to unearth and positively identify the precise location and depth of the NGP at approximately 40 ft. intervals. Markers were installed at these locations to provide surface level visual indications of the buried NGP. Civil Survey techniques were used to record coordinates and the surface elevation and depth of the NGP at each interval. The location of the NGP is identified in Figure 2-43 by the depths documented in the diagram. The top value is ground surface and the bottom value is top of the NGP.

In Westinghouse letter HEM-11-96 (Reference 8.19), Westinghouse made the commitment to contact the Laclede Gas Company, owners of the NGP, in advance of excavation and/or sampling within five feet of the NGP to discuss any necessary precautions or controls. On November 6, 2014, HDP staff met with Laclede Gas Company representatives to get their input on safely working around the NGP. The Laclede Gas Company representatives considered that excavating down to the top of the NGP was acceptable. The first choice of the representatives for protecting the structural integrity of the NGP was to not excavate if at all possible closer than two feet of the sides and not under the NGP. If excavation was to be conducted within two feet of the sides of the NGP, then an appropriate 1:1 slope should be employed. If excavation could not be avoided within that structural support area for the NGP, then shoring of the NGP would be required at a minimum of every 20 ft.

Soil impacted by former Site operations was located near the NGP. Specifically, the Hematite facility had two former evaporation ponds that were used for the retention of process filtrates, low-level liquid wastes, and high- and low-enriched Uranium-containing materials. Figures 2-40 and 2-41 show the proximity of the Evaporation Ponds to the NGP. In this area radiological contamination was identified close to the NGP on the northwest side of the NGP as expected, due to the Evaporation Ponds, but also on the SE side between the NGP and the rail line. Contamination requiring remediation SE of the NGP was close enough to the surface to enable removal through excavation while maintaining a required 1:1 sloping ratio for sloping sidewalls.

Early in the project there was a concern in regards to how remediation could successfully be accomplished near the NGP, as characterization data identified Tc-99 contamination at depth where the Evaporation Ponds were located, adjacent to the NGP. To excavate to that depth and maintain a 1:1 sloping ratio would not have been feasible that close to the NGP, and might have required shoring or other means to enable remediation. However, subsequent core bore sampling determined that the extent of contamination in the Evaporation Pond area was much less than originally believed, which allowed excavation and remediation activities to proceed in a normal manner and without potential risk to the NGP.

2.6.2 Remediation

For removal of contamination close to the NGP, hand digging with shovels was conducted. The Laclede Gas Company representatives were onsite during portions of the excavation activities around the NGP. Only a few small areas of surface soil contamination were identified elsewhere along the NGP and were remediated. Figure 2-44 is a photograph of remediation activities at the Natural Gas Pipeline.

2.7 Barns Area and Red Room Roof Burial Area

2.7.1 History

The Barns Area consisted of two barns, one wood and one tile, and two concrete silos. Figure 2-26 shows the location of the barns in relation to the entire site (lower right corner). Figure 2-45 shows the barns prior to demolition. The Wood Barn (Building 120) existed on the property prior to purchase by Mallinckrodt. It had a dirt floor and was used to store both clean and contaminated equipment throughout the facility's operating period. The Tile Barn (Building 101) also existed on the property prior to purchase by Mallinckrodt and was a former dairy barn. It was designated as a temporary storage facility and was used to store both clean and radiologically contaminated equipment. The Tile Barn's main floor was concrete. The walls were constructed of hollow tile to an elevation about 12 feet off the floor. This building was also used to store emergency equipment during the commercial nuclear phase of operations. The silos were not used in conjunction with any licensed activities.

The Area 240-2, Red Room area of Building 240, was used for UF_6 conversion of highly-enriched Uranium. The Red Room roof was replaced in the late 1984-85 timeframe and the old roof was buried in an area south of the Tile Barn, otherwise known as the Red Room Roof Burial area. The Cistern Burn Pit area, southwest of Building 101 and adjacent to the Red Room Roof Burial area, was used to burn wood pallets that may have been contaminated. This general area was also known to have been used for temporary storage of scrap materials. In 1993, soil contamination was discovered in the Red Room Roof Burial/Cistern Burn Pit area during renovations to the Tile Barn.

All equipment, materials, and debris were removed from the Barns and disposed prior to demolition.

2.7.2 Demolition

Demolition of the Barns began in late March, 2011, and the Barns were the first buildings to be demolished as part of the building demolition project. All Barn building demolition debris was loaded onto trucks and shipped offsite for disposal. To minimize the potential for the spreading of contaminated dust, water was sprayed on the debris as demolition was conducted.

FSSFR Volume 4, Chapter 1, Building Survey Areas Overview, provides photographs and a further discussion regarding the history, demolition and disposal of the Barns.

2.7.3 Remediation

In December, 2012, the concrete slab comprising the floor of the Tile Barn was removed. Upon removal of the floor slab a previous foundation was discovered, indicating that the Tile Barn had replaced another barn/structure on the same location. In December, 2012, excavation and remediation of soils in the Barns Area was started. These areas were not subject to the same criticality controls as the burial pits because radiological characterization work performed had shown little to no fissile material present in these areas. Therefore, the survey and soil excavation process was different than that of the Burial Pit Area. Overall no widespread contamination of the soil in the Barns Area was identified. Localized hot spots and some fuel pellets were identified and remediated; otherwise the area was generally clean. Following soil remediation, a FSS was performed.

Subsequent to the FSS the NRC with support from ORAU conducted a confirmatory survey of the Barns Area. The NRC survey results were documented in NRC Inspection Reports 07000036/13001 {ML13154A125} and 07000036/2014001 {ML14084A566}. The reports stated, in part, that the plan and instructions for survey unit LSA-05-01 were in accordance with MARSSIM and the DP, and the results for walkover surveys, concrete surface scans and activity measurements, and radionuclide concentration in soil samples and smears were consistent with the license and below the DCGLs, with a few exceptions in LSA-05-01, in which three samples exceeded the DCGLs. The NRC Inspector who was on-site and also performing surveys during FSS of the area discussed these results with the HDP RSO and determined the HDP soil sample results were also elevated in those same areas. HDP had remediated to the extent possible in those specific areas per the direction of the Missouri Department of Transportation. LSA 05-01 contains a portion of the base of Missouri State Road P and any further excavation could have destabilized the roadway.

Based on the HDP sample results and knowledge that the NRC's samples results were consistent with the on-site survey results it was determined that an EMC evaluation would be performed for the survey unit that would demonstrate compliance with the release criteria. To facilitate remediation in the southeasterly section of LSA 05-01, while completing the work necessary to return stability back to the base of Missouri State Road P, the survey unit was separated into two individual Class 1 survey units. The section that contains the base of Missouri State Road P remained designated as LSA 05-01 and the section yet to be completed was designated LSA 05-04. To restore stability to the roadway LSA 05-01 was immediately backfilled after the completion of FSS. One layer of backfill soil came from Reuse Stockpile 2 (which will be discussed in FSSFR Volume 2, Chapter 2, regarding the stockpiled soils) and the rest of the backfill was brought in from offsite (and will be included in FSSFR Volume 2, Chapter 9, regarding the off-site borrow soils). Figure 2-52, which was taken during backfill of the Barns Areas shows the location of LSA 05-01, LSA 05-02, LSA 05-03 and LSA 05-04.

The Red Room Roof Burial/Cistern Burn Pit Area was remediated following the remediation of the Barns Area. Remediation proceeded as expected with no widespread contamination of the soil in the Red Room Roof Burial/Cistern Burn Pit Area identified. Figure 2-53 shows remediation of the Red Room Roof Burial Area.

2.8 Sanitary Wastewater Treatment Plant

2.8.1 History

The original Sewage Treatment System was designed such that drains inside buildings were directed to a buried holding (septic) tank connected to a leach field. Liquid wastes from personnel showers, mop water, and small spills in process areas were directed to various floor drains leading to the septic tank and leach field of the former system. Between 1977, and 1978, the septic tank and leach field were bypassed and abandoned in place. The modified Sewage Treatment System was connected to new wastewater treatment equipment located just northwest of the Evaporation Ponds and designated as the Sanitary Wastewater Treatment Plant (SWTP). This modified SWTP utilized the existing Sewage Treatment System discharge pipe and discharged to the Site Creek just below the dam. Degradation of the buried SWTP discharge pipe was identified in 2007, during remediation, when it was discovered that no flow existed at the discharge outfall effluent sampling point. Degradation of the effluent pipe had progressed to the point that the majority of the liquid effluent entering this line did not reach the discharge point at the Site Creek. Evidence of subsurface contamination indicated that liquids from the degraded pipe leaked through cracks or breaks in the pipe, resulting in effluent migrating into the surrounding soils. Since the effluent of the SWTP may have contained residual radioactivity (within approved regulatory release limits), remediation of this system and associated effluent piping and adjacent soil was addressed during decommissioning.

2.8.2 Remediation

Figure 2-54 shows the concrete slab covering the location of the abandoned septic tank. Below this slab was the septic tank with a separate prefabricated concrete cover that was removed and disposed. Due to the close proximity of the septic tank to the foundation of Building 230, it was only partially removed so the structural integrity of the building would not be compromised. All piping associated with the original septic tank and leach field was removed as part of remediation. Figure 2-55 shows the interior of the septic tank. The construction of the septic tank was such that should there have been any transport of a discrete quantity of SNM it would have settled to the bottom of the septic tank. The septic tank was remediated using conventional methods such as power washing and scabbling. Radiological surveys and sampling of the septic tank only identified low levels of fixed residual contamination with no removable contamination identified. No discrete or recoverable quantities of SNM were found during remediation of the septic tank. FSS surveys conducted on the portion of the septic tank that remains showed 17% of the DCGL for systematic samples and 50% to 90% of the DCGL for biased samples. All smears were less than the MDA. The radiological data for the septic tank (BSA 04-02) will be included in the survey area release record for LSA 08-17, the survey unit in which it resides.

Figure 2-56 shows the excavation of the leach field in the lower portion of the photograph. The discharge pipe from the leach field and the SWTP went southwest (towards the upper left in the photo) under the concrete slab, and to the south (left) of the smaller building (Building 231). The line continued to the Site Creek where it discharged just below the dam (Outfall 1). Between Building 231 and the Site Creek the discharge line passed under the NGP. Between the leach field/SWTP and the concrete slab, the discharge line was removed. Between the Site Creek and the NGP the discharge line was also removed. The remaining portion of the line was left intact. This section, which could be accessed via a manhole in the concrete slab as well as the ends, was power washed. The radiological data for the SWTP Discharge Line (SAN-1) will be included in the survey area release record for LSA 08-10, LSA 08-15 and LSA 04-04, the survey units in which it resides.

Figure 2-57 shows the SWTP Tanks prior to demolition and removal, located between Building 230 to the north (left) and the Evaporation Ponds to the south (right). Figure 2-58 shows the SWTP Tanks exposed after excavating around them prior to demolition. The original intent was to decontaminate the tanks and dispose of them generally intact at US Ecology Idaho. However, to adequately characterize and decontaminate the interior of the tanks would have required a difficult confined space entry that raised numerous safety concerns. Based on commitments to the NRC, the interior of the tanks required a systematic grid sampling of concrete samples of the walls to obtain sufficient radiological data to demonstrate the material would meet the waste acceptance criteria (WAC) at US Ecology. This would have been time consuming and the structural integrity of the walls was questionable. It was therefore determined it was cost effective to simply demolish the tanks in place and ship the rubble to EnergySolutions in Utah, which had waste acceptance criteria that would accommodate disposal at that location, although at a higher cost of disposal, but eliminating the need to enter the tanks. Figure 2-59 shows the excavation remaining after demolition and removal of the SWTP Tanks.

2.9 Site Pond/Site Creek Area

2.9.1 History

The Site Pond is fed by a natural spring located on the northwest portion of the site and generally contains water year around. The Site Pond receives storm water runoff from the plant area. The Site Creek is the effluent from below the dam of the Site Pond, and receives discharge from the Sanitary Wastewater Treatment Plant (SWTP) directly below the dam. It flows through a culvert beneath the railroad track, and joins the effluent from the Lake Virginia drainage basin; the combined stream discharges to Joachim Creek. The Site Creek normally flows year round. Figure 2-60 is a photograph of the Site Pond prior to remediation.

In 1995, it was identified that occasional upsets in the operation of the SWTP over a period of time had resulted in contamination collecting in the Site Creek sediments. The contamination sediment had settled between the dam and the point where the Site Creek passes beneath the railroad tracks. Prior to remediation, sediment samples showed total Uranium concentrations within the range of 40 pCi/g to 800 pCi/g.

A Water Treatment System (WTS) was maintained onsite during decommissioning to process potentially contaminated water that resulted from decommissioning operations, from precipitation that entered work areas, or from excavations that encountered ground water. Collected water was directed to the WTS for analysis, and treatment as required. A portion of the system was comprised of an ion exchange resin bed to remove metals, including uranium and Tc-99 that may have been in solution and not removed by the granulated activated carbon filters in the system. On August 16, 2014, the presence of resin beads was identified in Tank T-5 of the WTS, the final holding tank before water was discharged into the Site Pond (through Outfall #003a). In addition, a small amount of resin beads was identified in a water sample collected at Outfall #003a. Upon discovery, a Stop Work Order was issued and all water treatment and discharge activities were paused until the problem was located and the WTS component that had failed was taken out of service. Restoration of the system included resin recovery efforts up to the point of discharge into the Site Pond. HDP routinely updated the NRC on recovery of the system and resin recovery. An assessment of the WTS estimated that 1.627 cubic feet of resin was unrecoverable and determined to be in the Site Pond as it would not travel downstream into the Site Creek due to the presence of the Site Pond Dam. This volume of resin was evaluated to contain below the NCS limit of 0.1 g/L U-235. This volume of resin would be recovered during the remediation of the Site Pond and disposed, along with sediment.

NRC Inspection Report 07000036/2014005(DNMS) dated February 20, 2015, (Reference 8.21), stated in part that, "During the collection of water and sediment radiological sampling, neither the licensee nor the NRC identified resin beads downstream of the outfall separating the Site Pond from the Site Creek which feeds into Joachim Creek. The NRC Inspector completed walk-over surveys and collected soil/sediment and water samples of potentially impacted areas. Walk-over surveys/sample results did not identify abnormal radiation levels or radioactive concentrations. Consequently, the NRC determined that the failure of the resin-tank did not result in any significant release of resins, and associated licensed material that may have been collected on it, to be released from the site."

2.9.2 Remediation

Remediation of the Site Creek was accomplished by diverting the Creek and then removing the sediment with a backhoe to a depth of approximately 0.5 ft to 3 ft between the site dam and the railroad tracks. The removed material was dried and shipped to an offsite licensed disposal facility. Sediment was removed until the average remaining contamination was less than 30 pCi/g, with no single sample above 90 pCi/g. Remaining residual radioactivity after remediation of the Site Creek averaged 22 pCi/g, with a maximum concentration of 85 pCi/g. Samples taken at the confluence of the Site Creek and Joachim Creek indicated contamination had not extended to Joachim Creek.

Remediation of the Site Pond and surrounding area required diverting inflow to the Site Pond, followed by draining, excavating, and removing sediments and soil. A water-inflow bypass basin was constructed to divert the Site Spring and storm water discharge during remediation of the Site Pond. Figure 2-61 shows the construction of the bypass diversion. Once the inflow-bypass system was operating and diverting the inflow of the Site Spring and storm water discharge, the Site Pond was drained. Site Pond water was drawn down in accordance with site discharge permits, sampled, and processed as necessary for discharge under the Water Management Plan. Figure 2-62 shows the Site Pond near the completion of excavation and remediation.

Remediation of the concrete dam required limited decontamination (power washing and scabbling) to meet the appropriate DCGLs and RGs. The radiological data for the Site Dam (BSA 04-01) will be included in the survey area release record for LSA 02-03, the survey unit in which it resides.

2.9.3 Site Pond Post FSS Event

FSS had been completed in Site Pond survey unit LSA 02-01 in late July, 2015, with isolation and control of the survey unit being maintained in accordance with site procedures. In summary, on or about August 30, 2015, a significant un-forecasted rain event occurred and moved 15 radiologically contaminated items from LSA 05-04, into adjacent LSA 02-01. As remediation in LSA 05-04 had not yet been completed and is upstream of LSA 02-01, it was determined that the contaminated items originated in LSA 05-04. The contaminated items were removed from LSA 02-01 and the affected area resurveyed. All downstream LSAs were resurveyed to verify the absence of radiologically contaminated items in early September, 2015.

On November 27, 2015, the NRC issued a Notice of Violation in regards to the event (Reference 8.31). In the Notice of Violation was a request for Westinghouse to complete an extent of condition in regards to the possibility of other instances in which radiologically contaminated items could have been transferred into a survey unit in which FSS had been completed. Westinghouse letter HEM-15-131, (Reference 8.32) which was a response to the Notice of Violation, also included the extent of condition investigation results. The extent of condition assessment concluded that there were no radiologically contaminated items residing in any previously FSS completed survey units. The NRC reviewed the extent of condition and in letter dated January 19, 2016 (Reference 8.33), acknowledged receipt of the assessment, review of the assessment and that there were no further questions at that time. Additional information in regards to the event will be provided in the survey area release record for LSA 02-01.

2.10 Tc-99 Areas

Technetium-99 was identified as a radionuclide of concern at Hematite. A historical review of site operations determined that the only source of the Tc-99 was as a contaminant in Department of Energy supplied UF6 originating from reprocessed/recycled spent nuclear fuels and not part of site operations. In 1967, five dry scrubber columns were installed for removal of hydrogen fluoride from the off-gas associated with the conversion of uranium hexafluoride (UF6) to uranium oxide (UO2). These dry scrubber columns used limestone rock chips as the off-gas scrubber media. The limestone media was periodically replaced and the waste limestone stored outside or utilized as onsite fill material, including locations in the northeast portion of the Burial Pit Area. While the limestone was used as an off-gas scrubber media, it also had an affinity for Tc-99. During Hematite operations, the limestone scrubber media became contaminated with Tc-99. Therefore, the primary areas impacted by Tc-99 were those locations where the waste limestone was disposed following removal from the scrubbers.

During excavation activities the identification of Tc-99 contamination was challenging and slowed the overall excavation and remediation process since field instrumentation did not exist for the measurement of Tc-99. As a result, the identification and quantification of Tc-99 had to be based on a laboratory analysis of soil samples. The typical turnaround time from collection to receipt of laboratory results took seven days.

Because the limestone waste from the scrubbers was the source of the Tc-99, all the limestone was removed prior to completion of excavation, RASS, and FSS activities. Limestone was identified to have been stored in four areas onsite: in the Documented Burial Pit Area, as fill under the Process Building before the building was constructed, west of the Red Roof Barn Burial Area, and in a pile just northeast of the central tract area of the site. (See Figure 2-23)

2.10.1 Tc-99 in Documented Burial Pit Area

As discussed in Section 2.1 "Retrospective Review", a review was conducted of the Burial Pit Area surface and sub-surface soil sampling data summarized in the Hematite Radiological Characterization Report. The maximum Tc-99 sample result collected in the Burial Pit Area was 68.3 pCi/g collected within the surface stratum. None of the results were in excess of the Tc-99 Excavation DCGL value of 74.0 pCi/g, and regardless, many of the areas where these samples were collected were also remediated.

Between 2004, and 2008, various core boring characterizations were conducted throughout the Burial Pit Area. A review of this data identified 106 samples that were characterized as collected at a depth of 16 feet or greater and sampled for Tc-99. Out of these samples the highest value was only 9.34 pCi/g. The next highest was 4.99 pCi/g. From over 300 samples collected at all depths, the highest value of Tc-99 was 68.3 pCi/g. Only five samples exceeded the Uniform $DCGL_W$ of 25.1 pCi/g, and four of these were at the surface while the fifth was at 6.5 feet. In other words, there was no expectation that Tc-99 had leached off the limestone and migrated throughout the Burial Pit Area and into the groundwater.

In spite of the expectation that Tc-99 contamination was not a significant problem, thousands of soil samples were collected and analyzed for Tc-99 during the excavation and remediation of the Burial Pit Area. In looking at the "as left condition", a total of 246 FSS samples were collected in the Burial Pit Area and analyzed for Tc-99. Every sample result was less than their applicable $DCGL_W$, with an average Tc-99 activity of 0.33 pCi/g, and a maximum result of 19.1 pCi/g (collected in LSA 10-12).

In summary, based on sample results, it has been determined that there is very little probability that Tc-99 could exist at levels that exceed the excavation DCGL at a depth below where excavation activities ceased because there was very little Tc-99 that existed in the majority of the Burial Pit Area to begin with. The Tc-99 that was identified, as expected, was predominately in the areas where limestone had existed. In addition, all areas where spent limestone was encountered were ultimately excavated.

2.10.2 Tc-99 Used As Fill Under the Process Building Before the Building Was Constructed

Based on historical records and interviews, non-native material was introduced under the Process Building 253. As part of preparing for the construction of Building 253, native soil was excavated due to the presence of gross alpha radioactivity exceeding 30 pCi/g and spent limestone was used as backfill. Because of concerns about undermining Building 240, the excavation was stopped before all soil exceeding this limit was removed. The average alpha concentration in soil of the surface at the conclusion of this excavation was 17 pCi/g. The maximum alpha concentration was 82 pCi/g.

A May 14, 1990, letter to the NRC, *Building 253 Construction Site Soil* (Reference 8.23), includes a figure that provides the locations within the excavation that were selected for soil sampling and subsequent analysis for gross alpha activity concentration. Per this letter, the excavation area reached a maximum depth of approximately six feet below the planned floor level. Based on this description, the maximum depth of the excavation was likely to have been approximately eight to ten feet. However, later interviews with personnel that were present when Building 253 was constructed indicated that the excavation depth may have extended to a depth of 10-15 feet.

Prior to placing backfill, Combustion Engineering requested that NRC allow spent limestone that was stored on-site to be used as backfill material within the excavation area under Building 253. In an NRC letter to ASEA Brown Boveri/Combustion Engineering, Authorizes Backfill of Area of Bldg. 253 Construction Site, dated July 2, 1990, (Reference 8.24), the NRC authorized spent limestone from two piles meeting a 30 pCi/g limit (alpha) to be used as fill below Building 253 with the understanding that the fill may have to be removed upon facility decommissioning. Figure 2-63 shows the location of the limestone backfill. Per a Combustion Engineering letter to the NRC, Spent Limestone Results, (Reference 8.25), dated April 7, 1989, the two limestone piles had average gross alpha concentrations of 7 pCi/g and 8 pCi/g.

Due to the presence of residual radioactivity in soil that could not previously be removed, and due to the presence of spent limestone as backfill, these areas were excavated to remove the limestone and the underlying soil until the approved DCGLs were met.

2.10.3 Tc-99 West of the Red Roof Barn Burial Area

A deposit of spent limestone was located west of the Barns, near State Road P. This area was remediated and the limestone sent to US Ecology for disposal. The limestone encroached upon State Road P, and during portions of the remediation, one lane of State Road P had to be closed (see Section 2.7 discussion). The overall depth of the limestone was approximately 4 – 6 feet below original grade. The limestone was removed to the maximum extent practical; however the extent of excavation activities was limited due to Missouri Department of Transportation (MDOT) restrictions which were put in place to prevent potentially undermining the integrity of State Road P. Soil sampling was performed both vertically downward from the surface and horizontally into the bank of the roadway in order to accurately measure the "as left" condition of the area. This evaluation will be discussed in detail further in the LSA 05-01 Survey Area Release Record.

2.11 Class 1, Class 2 and Class 3 Survey Areas

To allow a more concentrated survey effort in the areas likely to be contaminated, impacted survey areas are subdivided into Class 1, Class 2, or Class 3 survey units (SU). At Hematite, survey units were contiguous areas with similar characteristics and contamination potential. Survey units were 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 was consistent with the exposure pathway modeling used to convert residual radioactivity into dose.

The suggested maximum survey unit sizes by classification are recommended by MARSSIM (Ref. 8.5). This Guidance has been taken into consideration when delineating survey units; and as appropriate, were 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 up to 2,200 m^2 versus the MARSSIM recommended size of 2,000 m^2, the area was considered only one survey unit.

Open Land survey units were designed to have compact shapes rather than highly irregular (gerrymandered) shapes unless unusual shapes were practical given appropriate site operational history or site topography. Flexibility was also maintained to modify survey units based changing site conditions and information. Although these boundaries were altered in several instances, the classification for the purpose of final status survey was never reduced from the original classification.

Survey results were converted to appropriate units of measure and compared to investigation levels to determine if the action levels for investigation had been exceeded. Measurements exceeding the investigation action levels were investigated. If confirmed within a Class 1 survey unit, the location of an elevated concentration was evaluated using an elevated measurement comparison, or the location was remediated and re-surveyed. However, in a few instances survey results were confirmed to exceed the investigation level within a Class 2 or Class 3 survey unit, which required reclassification and a re-survey to be performed consistent with the change in classification. Although this occurred infrequently, it did occur on several occasions. Those occurrences are discussed in the FSSFR chapters specific to those survey units.

Survey units were 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 DCGLw. 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 DCGLw. 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 DCGLw. 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 DCGLw, 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.

2.12 Waste Disposal

The HDP radioactive waste management program was designed to safely control the handling, packaging, transport, and disposal of solid wastes generated during decommissioning activities. Site activities were performed in accordance with the Waste Management and Transportation Plan (Reference 8.22). The WMTP is an integrated plan for management of radioactive and non-radioactive waste, and is designed to protect the personnel, public, and environment during the generation, handling, and transportation of waste. As waste was exposed during remediation activities, additional waste characterization and initial segregation steps were implemented. After removal of the waste, final characterization and segregation of waste into the appropriate waste type was completed. Based upon waste type, waste was either directly loaded into containers or stockpiled awaiting packaging, treatment, or transportation. To prevent the spread of contamination during storage, piles were maintained utilizing Best Management Practices (BMPs). Packaged waste was transported using approved, qualified, and permitted carriers.

HDP used both rail service and truck/trailer as means of transportation of radioactive waste. For both modes of transportation, verification was performed to ensure carriers were permitted to carry the load, and that appropriate security plans were in effect. Pre-transportation checklists from approved procedures were used to ensure compliance with applicable United States Department of Transportation (DOT) and NRC regulations.

The majority of the solid radioactive waste generated during decommissioning was associated with excavation activities. The two general types of solid radioactive waste generated were:

- Demolition debris such as concrete rubble, building materials, piping, conduit, and exhumed Burial Pit waste; and,
- Volumetrically contaminated material such as soil, sediment, charcoal, resin, and limestone.

Solid radioactive waste generated by the project was Class A waste.

The specific isotopes and activity associated with the solid radioactive waste was dependent on the location where the waste was generated:

- The Burial Pit area was contaminated primarily with uranium isotopes: U-234, U-235 and U-238, including the associated decay daughter products; and to a lesser extent Ra-226, and Th-232.
- Radiologically impacted soil areas included the Barn Area, Red Room Roof Burial Area, Site Pond, Site Creek, and Leach Field Areas, and were contaminated primarily with the uranium isotopes. A portion of these areas was also contaminated to a lesser extent with Tc-99.
- The area southeast of and under the processing buildings was contaminated primarily with uranium isotopes, their associated decay daughter products, and Tc-99.

Radioactive waste handling was accomplished primarily with mechanical equipment such as excavators, front-end loaders, and trucks. However, there were occasions when smaller equipment and/or hand shoveling was employed.

For areas where it was determined that there was a reasonable possibility of finding fissile materials based on characterization data and historical knowledge, HDP developed consistent generic screening and handling approaches. Relative to the removal of buried wastes, contaminated soils, and sub-surface structures (e.g., concrete slabs and buried piping), these generic screening and handling approaches were analyzed from a NCS perspective in NCSAs specific to buried waste exhumation and contaminated soil remediation, and sub-surface structure decommissioning. Screening for fissile materials during remediation was the initial goal. Screening typically involved duplicate and independent performance of radiological surveys, using sodium iodide scintillation detectors, and defined volumes of material to ensure that NCS limits had not been exceeded. The objective of the in-situ radiological surveys was to identify any item or region of soil/waste with a fissile concentration exceeding 1 gram U-235 in any contiguous 10 liter volume. The 1 gram U-235/10L threshold provided a high degree of assurance that any items with elevated (i.e., nontrivial) levels of U-235 contamination would be identified. The in-situ radiological surveys were complemented by visual inspection of the survey area with the aim of identifying: 1) Items with the potential to contain fissile material (e.g., a process filter); 2) Items that resemble intact containers; 3) Bulky objects with linear dimensions exceeding the permitted excavation '*cut depth*'; and, 4) Metallic items.

During fissile material screening, the secondary remedial action objectives of identifying hazardous materials (e.g., VOCs), and verifying radioactivity concentrations of potential backfill soils were also completed. Excavation continued in areas of suspected fissile materials to a depth where historical knowledge, and/or visible and radiological evidence indicated that suspect materials had been removed. Once fissile material screening determined that fissile materials were not present in the remediation area in excess of the NCS Exempt Material Limit, NCS controls were then curtailed. By making this the initial goal, continued excavation, Health Physics, and Final Status Survey activities could proceed unencumbered by NCS controls. Identified items that exceeded NCS limits were segregated into designated Field Containers, which were placed in transport containers such as Collared Drums (CDs). Individual designated containers were handled and stored in accordance with NCSAs.

To assess and manage low level radioactive waste (LLRW), field screening was performed to establish an initial material classification, followed by sampling and analysis to validate the field screening classification. The general sequence for excavation, removal, and handling of LLRW was as follows:

- Excavated LLRW was loaded directly into haul trucks for transfer to the WCA or stockpiled until a sufficient quantity was available for transport to the WCA for final visual inspection and assay by High Resolution Gamma Spectroscopy (HRGS); and,
- LLRW was sent to the WHA for stockpiling, loading, verification of compliance with WAC, and subsequent transportation to off-site disposal facilities.

Transportation for off-site disposal was generally by gondola cars; however, alternate conveyances which met the requirements of the Waste Management and Transportation Plan were also utilized. Solid radioactive waste was shipped for processing and/or disposal to an appropriately licensed or permitted facility. The following facilities were used for solid radioactive waste disposition: EnergySolutions, Inc., Oak Ridge, TN; EnergySolutions, Inc., Clive, UT; U S Ecology Idaho, Grandview, ID; and Waste Control Specialists, Andrews TX.

Regardless of the facility selected for processing and disposal, waste was prepared for transport in accordance with the receiving facility's WAC, facility license, or NRC approved exemption, and the DOT regulations.

2.13 Backfill Operations

Site restoration included backfilling excavated areas, compacting to a standard proctor, spreading topsoil, reseeding, and removing temporary features that impeded final site restoration. Excavated soil determined to be below the appropriate DCGLs, and meeting other regulatory requirements for re-use, was used as backfill material. However, additional off-site backfill material was imported from off-site source(s) to provide sufficient soil to meet site cover requirements for radiological and chemical constituents. Grading was performed to achieve pre-remediation contours to the maximum extent practical. Adjustments were made to the grade to mitigate the potential for surface water to pool over the remediated site. Reseeding of backfilled areas was performed with a MDNR approved seed mixture to limit the potential for erosion. Topsoil was placed above backfill material in areas to be seeded, and was cultivated and graded to ensure a smooth, uniform grade with positive drainage towards wetland areas. Winter rye seed and/or other MDNR approved cover were utilized.

Due to the significantly large volume of off-site borrow material required to backfill excavations, Westinghouse identified multiple sources of off-site borrow. In regards to off-site borrow material the DP states:

- DP Section 8.8 - "*Additional off-site backfill material will be imported from an approved off-site source(s), as needed, and tested to ensure it meets site cover requirements for radiological and chemical constituents.*"
- DP Section 14.4.4.1.6.2 - "*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.....*"

Based upon the above stated DP requirements, HDP performed a lengthy process of radiological testing, statistical analyses, and documentation of the off-site borrow material prior to NRC approval. That effort is documented in FSSFR Volume 2 Chapter 1 (HDP-RPT-FSS-100). Subsequently, in NRC (Norato) letter to Westinghouse (Fussell) dated October 8, 2015, *U.S. Nuclear Regulatory Commission Conclusions Associated with the Utilization of Off-site Borrow Material at the Westinghouse Hematite Site*, (Reference 8.34) the NRC provided approval of the off-site borrow referenced in HEM-15-39 (the Horine Road site) (Reference 8.35). The NRC also stated that "*The conclusions presented in this letter are also applicable to any other source of off-site borrow material. Westinghouse informed the NRC on September 24, 2015, that they have procured access to two other sources of off-site borrow. Westinghouse should consider the conclusions contained in this letter as they may apply to these two resources.*"

In conducting the radiological remediation of the HDP site, approximately 180,000 cubic yards of soil was excavated and shipped offsite for disposal or retained for reuse. To replace this soil, several locations near the HDP site were approved where offsite soil could be obtained and used to supplement onsite stockpiled soil that had been segregated for reuse. Each of these locations represented native Missouri soil from farm fields. To complete backfilling the excavated areas onsite, approximately 150,000 cubic yards of offsite borrow soil was brought to the site from these locations by truck and used as backfill.

2.13.1 Management of Soil Used as Backfill

There are two categories of soil used to backfill excavations and complete final grade contouring of the site. The primary type and source of soil used is Off-site Borrow as discussed above. The second type and source is reuse soil which is soil that has been determined to have met the unrestricted release criteria. FSSFR Volume 2, Chapter 1, Reuse Soil and Off-site Borrow Material Overview, provides a detailed discussion on the methodology utilized to determine that both off-site borrow material and reuse soil is acceptable for use as backfill.

As off-site borrow material has been determined to contain no radiological contamination there is no restriction on placement of this material.

Although the reuse soil generated on the site has been determined to be acceptable for unrestricted release, the fact that it does contain residual radioactive contamination necessitates managing the dose associated with each individual reuse stockpile in relation to the total dose for the survey unit. In addition, placement of the reuse soil within a survey unit and also in which stratum the soil is placed must be considered.

Placement of reuse soil and off-site borrow is managed by the use of Work Package HDP-WP-ENG-802, Backfill & Restoration. In summary, the process for placement of reuse soil is as follows; the Radiation Safety Officer (RSO) evaluates the FSS data and the expected dose contribution from all sources for all survey unit(s) to be backfilled; based upon the evaluation the RSO selects the appropriate survey unit and if required the stratum in which the reuse soil is to be placed ensuring the total dose for the survey unit(s) will meet the release criteria; the onsite backfill placement tracking forms are completed which provide the survey unit(s) and depth of excavation in which the reuse soil will be placed; and, in addition to the directions provided by the tracking form to ensure proper placement of the reuse soil, the placement of the reuse soil and the associated elevations are verified by topographical measurements by HDP Engineering.

3.0 RELEASE CRITERIA

In order to demonstrate that the site meets requirements for unrestricted site release, site-specific release criteria or Derived Concentration Guideline Levels (DCGLs) were developed using dose modeling. The DCGLs represented isotope-specific release criteria. Because multiple radionuclides were present at the same time in varying quantities, the dose contribution from each radionuclide was considered such that the total dose from all radionuclides did not exceed the dose based limit. Section 5.1 of FSSFR Volume 1, Chapter 1 provides a discussion of the radionuclides of significance at Hematite.

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.

Surface or volumetric concentrations that correspond to the maximum annual dose criterion are established for the average residual radioactivity in a survey unit and are called the $DCGL_W$. 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.

Conceptual Site Models (CSMs) were developed for soil and the surfaces of remaining buildings. The critical groups and exposure pathways were identified and described. Dose model parameters were selected and sensitivity analyses performed. DCGLs were then calculated for soil and building surfaces. The soil DCGLs are specific to a given CSM and will result in a Total Effective Dose Equivalent (TEDE) of 25 mrem/yr to the average member of the critical group for that CSM. For the building surface DCGLs, two room sizes were considered for the DCGL calculations, representing a small office and an open warehouse. The Small Office CSM resulted in the most limiting DCGLs. Considering the very low levels of residual surface contamination present in the buildings and the limited effort that would be required to reduce surface contamination to acceptable levels, the DCGLs based on the Small Office CSM were used for all building surfaces.

As open land area and building remediation activities were completed, FSSs were conducted to demonstrate that the dose from residual radioactivity at the HDP Site did not exceed the annual dose criterion for license termination for unrestricted use specified in 10 CFR 20.1402 (Reference 8.26), and that the levels of residual radioactivity were ALARA. FSS was performed on all impacted open land areas and structures, systems, and components (SSCs) that will remain at the time of license termination. 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 were to: select/verify survey unit classification; demonstrate that the potential dose from residual radioactivity was below the release criterion for each survey unit; and, demonstrate that the potential dose from small areas of elevated radioactivity were below the release criterion for each survey unit.

Documentation of the FSS provides a FSS Survey Area Release Record for each survey unit. A FSS Final Report (FSSFR Volume 7, Chapter 1) will be prepared that includes the survey area release records (Chapters), and provides a summary of the survey results and the overall conclusions that demonstrate the site meets the radiological criteria for unrestricted use.

3.1 LSA Release Criteria

The soil contamination, including surface and subsurface, was modeled using two different source term geometries; 1) a soil column with three distinct layers that represent different exposure pathways and depths, and 2) a soil column with uniform contamination over the entire depth of the Contaminated Zone. Therefore, the surface and subsurface contamination is represented by four different CSMs (RESRAD models). The first subsurface geometry assumes a soil column that is comprised of three stratums as follows:

Surface - surface soil to a depth of 15 cm below the ground surface;

Root - subsurface soil starting at 15 cm and extending to 1.5 m below the ground surface to include the entire root stratum; and

Deep - subsurface soil located below 1.5 m (i.e., below the root stratum) and extending to the bottom of the Contaminated Zone which was conservatively estimated to be 6.7 m below the ground surface.

The Surface stratum represents the typical surface contamination configuration, i.e., the top 15 cm of soil. The Root stratum represents soil in the root zone (15 cm to 1.5 meters) and accounts for the potential removal of soil due to erosion over a 1,000-year period. The root depth is assumed to be 0.9 m and potential erosion over 100 years is estimated to be 0.6 m. Using the combined 1.5 m depth ensures that the thickness of the root stratum will equal or exceed the 0.9 m root depth for an entire 1,000-year period. The Deep stratum represents soil below the root stratum starting at a depth of 1.5 meters and extending to the bottom of the Contaminated Zone, which was conservatively estimated to be 6.7 m deep.

The second subsurface geometry, Uniform, is comprised of one stratum of uniform soil contamination from the ground surface to the bottom of the Contaminated Zone (6.7 m).

DCGLs were calculated for each of the four CSMs discussed above including Shallow, Root, Deep and Uniform. The four CSMs were designed to address the various configurations that may be present during remediation and at the time of the FSS. DCGLs were also calculated for an excavation scenario CSM to evaluate the effect of changing the in-situ soil configuration after license termination.

The Hematite Radiological Characterization Report (HRCR) (Reference 8.30) discusses the characterization efforts from the several characterization survey campaigns and compiles the data into one report. The data were reviewed, separated into logical areas, and evaluated against the proposed soil and building surface DCGLs. The data sets were tabulated and basic statistics calculated. The data sets were evaluated to determine preliminary classification (Class 1, Class 2, or Class 3) for the impacted areas in accordance with guidance from MARSSIM (NUREG 1575). The soil data were also evaluated to estimate the lateral and vertical depth of the contamination.

Table 3-1 presents the site-specific DCGLs as developed for soil for all four CSMs.

3.1.1 Uniform DCGLs

The subsurface geometry for Uniform DCGLs is comprised of one stratum of uniform soil contamination from the ground surface to the bottom of the Contaminated Zone (6.7 m). Demonstration of compliance with the Uniform DCGL is simply a comparison of the DCGL to the average concentration of residual contamination regardless of the depth of the contamination.

DCGL values were initially derived for building surfaces in 2005 and for site soil in 2006. The modeling and resulting DCGL values for building surfaces and site soil were reviewed and revised in Chapter 5 of the HDP. The DCGLs were approved by the NRC with issuance of License Amendment 57 on October 13, 2011.

To help expedite the speed and efficiency of the FSSR process, Westinghouse decided that the Uniform DCGL would be used when evaluating the results of the LSA final status surveys unless otherwise approved by the RSO. It was believed that as a result of the overall quality of the soil remediation that was conducted the vast majority of the LSAs would meet the Uniform DCGL criteria. In the few instances where the three stratum approach would be necessary (surface, root, deep), the RSO would approve the deviation from using the Uniform DCGLs.

3.1.2 Three Stratum DCGLs

Compliance with the "three layer" geometry requires consideration of the Surface, Root, and Deep layers independently. After the original DP submittal and approval, Westinghouse agreed with the NRC that the Deep DCGLs should not be used as they were not protective of the intruder scenario, and the Excavation DCGLs were developed as a replacement. Only the Excavation DCGLs will be used when evaluating the Deep layer as part of the "three layer" approach. Because each of the three DCGLs (Surface, Root, Excavation) represent 25 mrem/yr from each layer independently, the unity rule was used to demonstrate compliance when contamination was present in more than one soil layer. In some cases, less than three layers were present, for example, when there was no contamination below the depth of 1.5 m. In this case, compliance was demonstrated using the unity rule for the Surface and Root DCGLs only.

Overall, the Uniform DCGLs were applied in 60 of the 69 LSAs while the "Three Stratum" DCGLs were applied in nine of the 69 LSAs. Of those nine LSAs, LSAs 09-02, 09-03, 10-12, 08-09, 08-11, 08-12 and 08-13 received offsite soil as backfill. As a result the Surface Stratum would have no dose contribution in those LSAs. In the remaining two LSAs, 08-01 and 08-02, reuse soil from Stockpile 4-7 was placed (only) in the Deep Stratum.

The following is a hypothetical example (see diagram below) of the "3 layer" approach which includes showing how dose from on-site reuse soil will be added into the survey unit that is being evaluated. This hypothetical example would assume that one stockpile was used as backfill within the survey unit, the stockpile was evaluated against the Uniform $DCGL_w$, and the stockpile exceeded the Tc-99 Uniform MIL therefore placement was restricted to the Deep stratum

To simplify the discussion this hypothetical SU is assumed to have no elevated areas that exceed a SOF of 1.0, so no EMC evaluation is necessary, and also assumed to have no remaining subterranean piping. Therefore the dose contributions for these are each set to zero.

As a conservative measure, and to greatly simplify the calculation, area weighting factors will not be used for on-site reuse soil. When an on-site reuse soil stockpile is used in a survey unit, the *entire dose* for that stockpile will be attributed to that survey unit, even if the entire stockpile is not used.

The attached diagram presents a cross sectional view of this hypothetical survey unit, and the following calculations sum the total potential dose for each layer. Please note that the drawing is not to scale, and is only intended as a visual aid for this hypothetical discussion. In the diagram, the Surface stratum has been entirely removed, and only the Root stratum and Deep stratum remain. Using the typical minimum of 8 systematic soil sample locations that would fall in a Class 1 Survey Unit, we assume that 2 of the 8 samples fall in the remaining Root stratum, and all 8 samples are also collected in the remaining Deep stratum.

1. Remaining Surface SOF is 0.0 (layer totally removed)

   Offsite borrow soil is used to backfill the surface stratum.

   Surface stratum dose contribution to SU is 0.00 SOF

2. Remaining Root stratum contribution is (2/8)x(0.20)=0.05

   Offsite borrow soil is used to backfill the Root stratum.

   Root stratum dose contribution to SU is 0.05 SOF

3. Remaining Deep stratum contribution to SU is (8/8) x (0.15) = 0.15

   The onsite reuse soil stockpile with a 0.15 Uniform SOF is used, all dose from the stockpile is conservatively assigned to SU, even if entire stockpile is not used. Use of the Uniform DCGL when restricting the stockpile to the Deep stratum only is also conservative.

   Deep stratum dose contribution to SU is 0.15 + 0.15 = 0.30 SOF

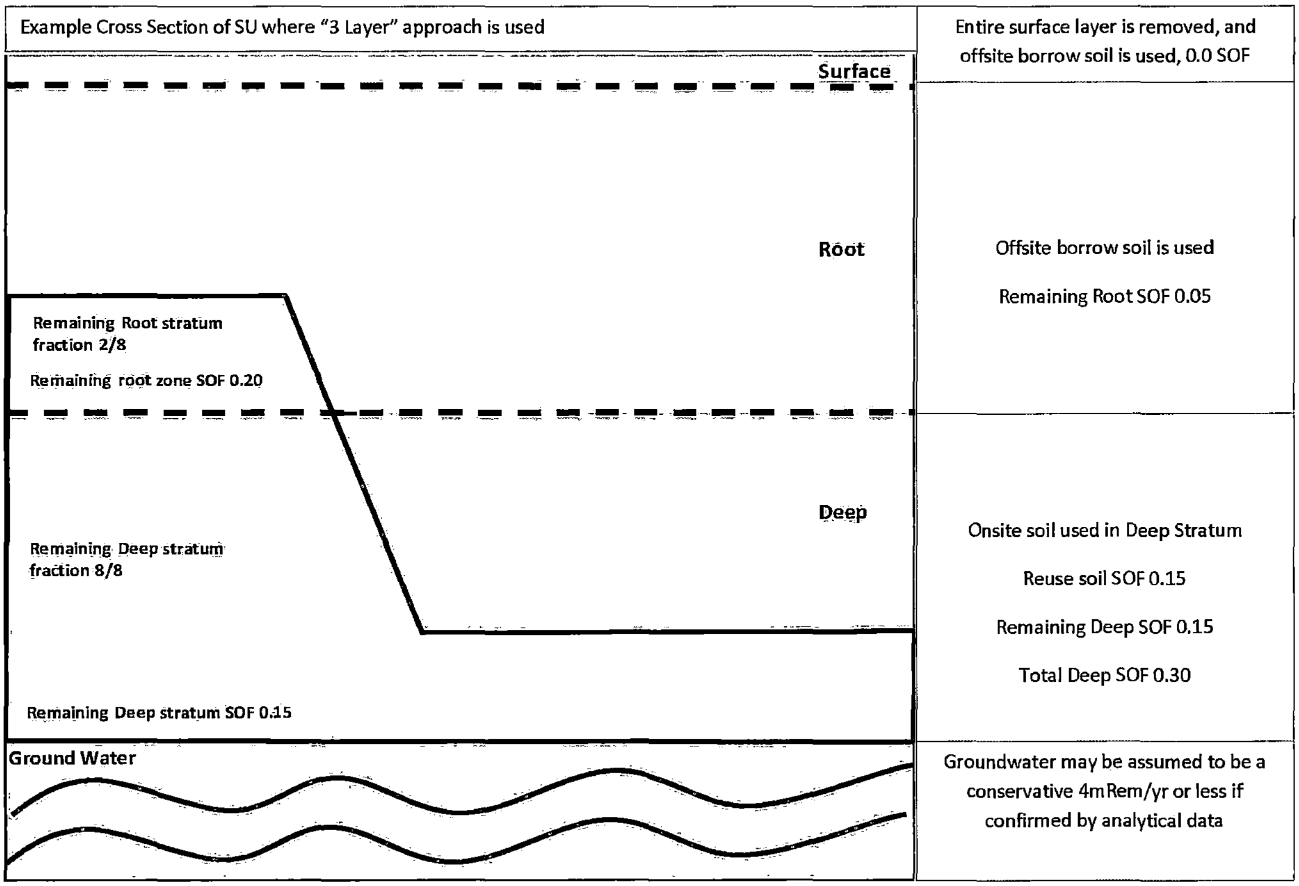
4. Groundwater is assumed to be a conservative 4 mrem per year (0.16 SOF) unless post remediation analytical sampling is available and the results have proven that the groundwater contribution can be accurately assigned a lower value.

5. Total SU dose is determined by SUM of all SOF's

SU Total SOF = Total Surface SOF + Total Root SOF + Total Deep SOF + Ground Water + EMC Dose +Piping

And in this hypothetical case:

SU Total SOF = 0.00 (total surface) + 0.05 (total root) + 0.30 (total deep) + 0.16 (groundwater)
+ 0.0 (no EMC dose) + 0.0 (no piping dose) = 0.51 SOF or 12.75 mrem/year



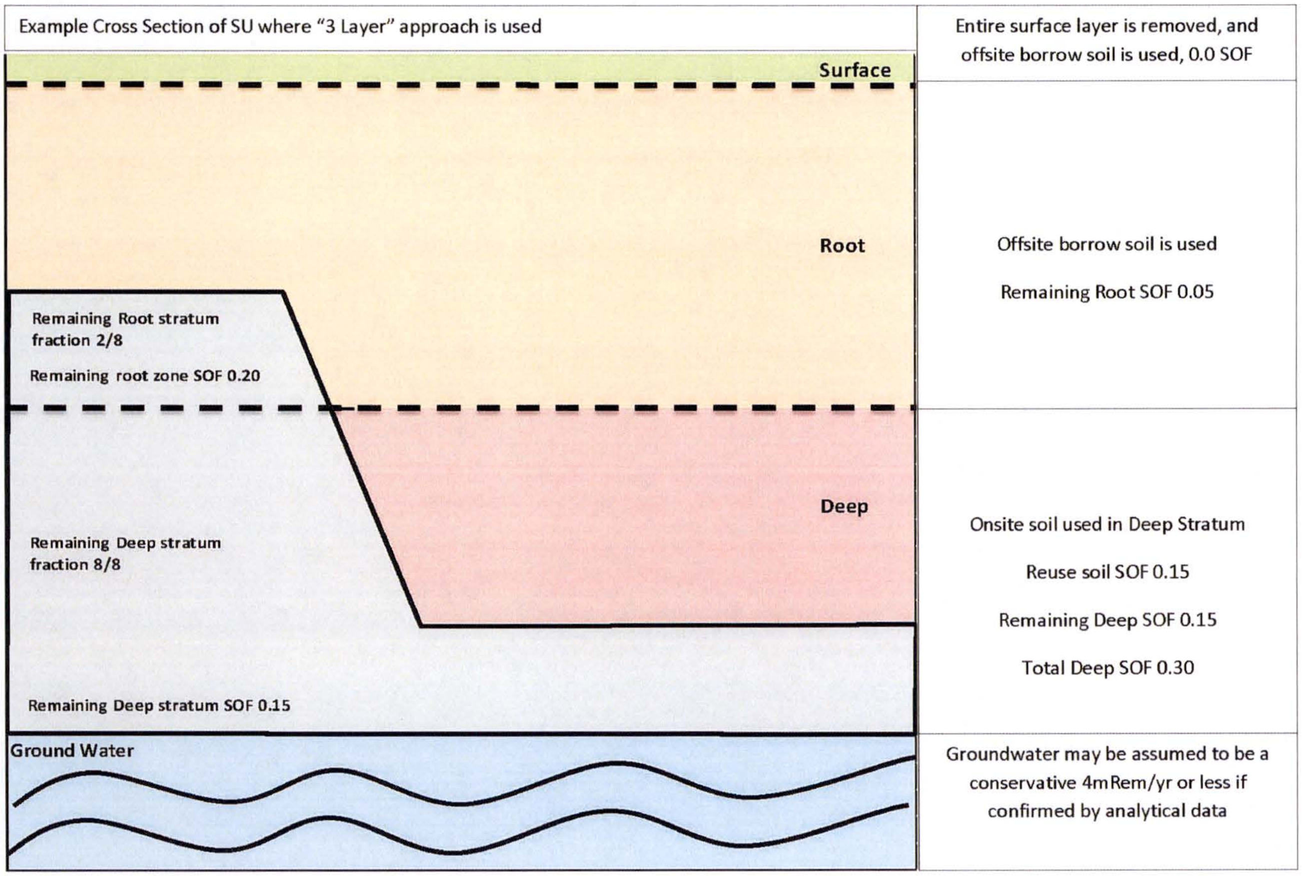
3.1.3 Elevated Areas

To address small areas of elevated radioactivity in a survey unit a simple comparison to an investigation level is used to assess the impact of potentially elevated areas rather than using statistical methods. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an Area Factor (AF) to account for the small area of the elevated radioactivity. An area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. (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 soil were developed by using the CSMs and adjusting the size of the contaminated zone. The AFs for the Surface, Root, and Uniform Soil strata are provided in Table 3-2a. The AFs for the Excavation CSM (and corresponding Deep $DCGL_W$) are provided in Table 3-2b and 3-2c.

And in this hypothetical case:

SU Total SOF = 0.00 (total surface) + 0.05 (total root) + 0.30 (total deep) + 0.16 (groundwater) + 0.0 (no EMC dose) + 0.0 (no piping dose) = 0.51 SOF or 12.75 mrem/year



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The AFs for soil were developed by using the CSMs and adjusting the size of the contaminated zone. The AFs for the Surface, Root, and Uniform Soil strata are provided in Table 3-2a. The AFs for the Excavation CSM (and corresponding Deep $DCGL_W$) are provided in Table 3-2b and 3-2c.

The areas factors were developed for each CSM by adjusting the size of the Contaminated Zone and dividing the resulting DCGL for the smaller areas of activity ($DCGL_{EMC}$) by the applicable site DCGL. AFs for the Surface, Root, and Uniform CSMs were determined for contaminated zone sizes ranging from 1 to 10,000 m^2, although it is not anticipated that AFs for areas greater than 300 m^2 will be utilized.

AFs for the Excavation CSM were calculated for contaminated zone sizes ranging from 1 to 100 m^2 in accordance with the MARSSIM using the RESRAD parameters which were modified for use with the excavation scenario (as detailed in Section 5.3.6). However, unlike for the development of the $DCGL_W$ values presented in Table 3-1, the elevated activity was not assumed to be mixed with the clean cover. Instead, the elevated activity was assumed to be excavated intact and brought to the surface, with the only modification to "flatten" the material from an assumed depth of 1.5 m to the excavation scenario depth of 0.9 m. The excavation scenario area factors are subject to the following constraints:

1. The excavation scenario for a small area of elevated activity must account for the increase in area after being excavated to the surface. An adjustment factor of 1.67 (1.5/0.9) was applied during modeling for geometrical transformation between the assumed excavation geometry depth (1.5 m) and the geometry modeled in RESRAD (0.9 m). For example, an elevated area of 1 m^2 and a depth of 1.5 m was assumed to be excavated to the surface and cover an area of 1.67 m^2 (1 $m^2 \times 1.67$) and a depth of 0.9 m. The modeled area factors are presented in Table 3-2b with the listed areas representing the post-excavation condition. This constraint limits the concentration within an area of elevated activity which remains contiguous during excavation to less than the $DCGL_{EMC}$.

2. The excavation scenario for a small area of elevated activity must also account for the fact that the scenario's excavation footprint is 200 m^2 and residual activity cannot exceed the $DCGL_W$ for post-excavation configurations that exceed 200 m^2. Therefore the calculated area factor is limited to the quotient 200 m^2 divided by the extent of the elevated area in square meters. This constraint limits the weighted average concentration over each 200 m^2 area to the $DCGL_W$.

The resultant area factors based on the two constraints discussed above are shown in Table 3-2c. The area factor selected for each radionuclide and area combination is the smallest of the values calculated based on each of the two constraints. Table 3-2b is provided for informational purposes only as Table 3-2c will always be used to determine AFs for use with the Excavation $DCGL_W$.

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 $DCGL_{EMC}$. The Elevated Measurement Comparison (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 DCGLw 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. 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.

3.2 Demonstrating Compliance with the Dose Criteria

3.2.1 Average SU Soil Dose

The average radioactivity within the survey unit will be determined from the systematic sampling and measurement results, excluding all biased measurements and any measurements (systematic or biased) within an elevated area, this is to ensure the proper statistical testing of the survey data without skewing the results of the evaluation. Biased sample results are excluded as these were not randomly selected.

$$f_{Avg} = \sum_{j=1}^{x} \frac{\delta_j}{DCGL_{w_j}} \qquad \text{Equation 3-1}$$

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,
$DCGL_{wj}$	=	Derived Concentration Guideline Level of contaminant j.

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:

$$Average\ SOF_{Weighted} = f_{SZ}\sum_{i=1}^{n}\left(\frac{\overline{C}_{i,SZ}}{D_{i,SZ}}\right) + f_{RZ}\sum_{i=1}^{n}\left(\frac{\overline{C}_{i,RZ}}{D_{i,RZ}}\right) + f_{DZ}\sum_{i=1}^{n}\left(\frac{\overline{C}_{i,DZ}}{D_{i,DZ}}\right) \qquad \text{Equation 3-2}$$

where:

n	=	Number of measured ROCs;
f_{SZ}	=	Fraction of survey unit area at the Surface stratum depth;
$\overline{C}_{i,SZ}$	=	Average concentration of ith measured ROCs in the Surface stratum layer;

$D_{i,\,SZ}$	=	Surface stratum $DCGL_W$ for the *i*th measured ROCs;
f_{RZ}	=	Fraction of survey unit area at the Root stratum depth;
$\overline{C}_{i,\,RZ}$	=	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.

3.2.2 Elevated Area Dose

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 higher and lower AFs will be determined from Table 3-3 and Table 3-2 for building and structural surfaces and soil, respectively, and an exponential interpolation will be performed using the following equation:

$$e\left[\frac{(\ln(actual\ area) - \ln(lower\ area))(\ln(higher\ AF) - \ln(lower\ AF))}{\ln(higher\ AF) - \ln(lower\ AF)} + \ln(lower\ AF)\ \right]$$

Equation 3-3

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 (Reference 8.5).

The additional dose fraction or contribution from each elevated area will be determined by calculating the average radioactivity within the elevated area (from biased and any systematic samples within the elevated area), subtracting the average radioactivity of the survey unit (from systematic samples not within the elevated area), 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.

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

Equation 3-4

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.

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. However, it is not anticipated that this will be necessary.

3.2.3 Groundwater Dose

Groundwater monitoring was performed prior to and during soil remediation efforts, and provides relevant data in regards to the anticipated groundwater conditions post remediation. FSSFR Volume 6, Chapter 1, Groundwater, {ML16041A340} describes the post remediation groundwater monitoring that is being conducted at HDP.

In summary, to support the NRC review of LSA survey area release records prior to the completion of the post remediation groundwater monitoring, the SOF for groundwater will be set at the conservative value of 0.16 in the survey area release records for each LSA. This value is based upon groundwater not exceeding the EPA drinking water standard of 4 millirem/year. To date, sampling within the sand/gravel, Jefferson City Cotter, and Roubidoux HSUs, which has been ongoing quarterly since June 2007, has not shown any contamination exceeding the EPA drinking water standard thus providing that the use of 0.16 SOF for groundwater is appropriate.

3.2.4 Buried Piping/Structure Dose

Buried piping that will remain at the time of License termination is also subject to FSS for the purpose of determining the additional dose, if any, that is required to be added to a particular survey unit total dose determination.

FSSFR Volume 5, Chapter 1, Piping Survey Areas (PSA) Overview, {ML16076A312} provides the specific details in regards to the survey methodology and evaluation to demonstrate that the piping is acceptable for unrestricted release. The dose determined for a piping survey area will be determined prior to submitting a survey area release record for a LSA in which the piping is located. The dose determined for each piping survey area is applied to each survey unit SOF in which the piping is located.

3.2.5 Reuse Soil Dose

A key component of remedial actions within open land areas of the site was to identify and separate soil that could be used for backfill (reuse soil) from soil that exceeded site cleanup criteria and had to be disposed as waste. FSSFR Volume 2, Chapter 1, Reuse Soil and Off-site Borrow Material Overview provides the specific details in regards to the history, survey methodology and Reuse Soil Release Criteria for each Reuse Stockpile of soil that was generated during remediation. In summary, based upon the FSS data for each individual Reuse Stockpile, an associated dose is derived for each individual Reuse Stockpile. The dose determined for a Reuse Soil Stockpile will be applied to each survey unit SOF in which re-use soil is placed based upon the use of either the Uniform or Three Stratum DCGL and its placement within the stratum.

3.2.6 Off-site Borrow Dose

FSSFR Volume 2, Chapter 1, Reuse Soil and Off-site Borrow Material Overview provides the specific details in regards to the historical sample methodology and evaluation to demonstrate that the soil from the various off-site borrow locations was not radiologically contaminated. The SOF dose contribution for soil from the off-site borrow used as backfill will be the value of 0.00 as offsite borrow soil has been determined to be non-radiologically contaminated soil.

3.2.7 Total Dose

Once all the dose contributions are determined, as described in the above sections, the SOFs are applied as follows:

$$f_{Avg} + f_{EMC} + f_{GROUNDWATE\ R} + f_{SOIL-OFFSITE} + f_{SOIL-REUSE} + f_{PIPING\ /\ STRUCTURE} \leq 1 \qquad \text{Equation 3-5}$$

Provided the SOF is less than or equal to unity (1), the survey unit is acceptable for unrestricted release.

FSSFR Volume 7, Chapter 1, Summary Report, will contain the summary report of all FSS data used to demonstrate compliance with 10 CFR 20.1402. FSSFR Volume 7, Chapter 1 will be compiled upon the completion of the post remediation groundwater monitoring period sample and analysis activity as described in FSSFR Volume 6, Chapter 1, Groundwater. At that time the groundwater SOF, as determined by post remediation groundwater monitoring sample result evaluation and dose determination, will be incorporated into the Total Dose calculation by replacing the 0.16 value.

4.0 DATA QUALITY OBJECTIVES (DQO)

The DQO process is thoroughly integrated within the DP and Hematite FSS procedures. The steps of the DQO process are most explicitly detailed in Section 9 of FSS procedure HDP-PO-FSS-700 *Final Status Survey Program* and correspond to the DQO steps described in

Chapter 14, Section 4.2.1 of the DP. The HDP DQO process reflects the recommendations given in MARSSIM, Chapter 2, Figure 2-2.

The FSS process consists of the following principal elements to which the DQO are applied: Planning and Design, Implementation, and Data Assessment. DQO allow for systematic planning, address situations that require a decision to be made, and provide a framework for selecting actions that result in obtaining data of sufficient quantity and quality. The DQO process is iterative and allows for incorporation of newly gained knowledge to enhance the effectiveness of subsequent actions. The seven steps of the DQO process are as follows:

1. State the problem,
2. Identify the decision,
3. Identify inputs to the decision,
4. Define the study boundaries,
5. Develop a decision rule,
6. Specify limits on decision errors, and
7. Optimize the design for obtaining data.

The DQO process is described below as it applies to the HDP.

1. State the Problem

The problem is the presence of residual radioactive material associated with previous licensed activities at HDP. The primary radionuclides of concern (ROC) and the extent of contamination were assessed in the HAS and the HRCR. The primary ROC are uranium-234 (U-234), uranium-235 (U-235+D), uranium-238 (U-238+D), technetium-99 (Tc-99), and thorium-232 (Th-232+C). Additionally, trace amounts of americium-241 (Am-241), neptunium-237 (Np-237+D) and plutonium-239/240 (Pu-239/240) may be present; however the latter are insignificant contributors to potential dose. Although Radium-226 (Ra-226+C) was identified as an ROC site-wide, it was only identified as an ROC due to radium plates found in the Documented Burial Pit Area. NOTE: The nomenclature "+D" above indicates that the dose contribution of short-lived progeny is accounted for by the parent nuclide and "+C" indicates that the dose contribution of the entire decay chain (progeny) in secular equilibrium is accounted for by the parent nuclide.

2. Identify the Decision

For the FSS, the principal study question is "Is residual radioactive contamination in the survey unit present in quantities which exceed the established $DCGL_W$ values?" The FSS Program is used to demonstrate that the HDP site meets the criteria for unrestricted release specified in 10 CFR 20.1402 of 25 millirem/year total effective dose equivalent (TEDE). Compliance with the release criteria is satisfied using the guidance in MARSSIM. The DP Section 14.4.2.1.1 and HDP-PO-FSS-700 Section 9.1 provide a discussion on the DCGLs, Unity Rule, Area Factors, and Background Measurements needed in order to make the decision relevant to the question "Does the survey unit meet the criteria for release?"

3. Identify Inputs to the Decision

Guidance provided in MARSSIM is the basis for this FSS Program. Inputs include sources of historical information, results of field and laboratory measurements, limitations on detectability, and the acceptable risk of a decision error. These inputs will be provided in each survey area release record. Survey Area Release Records are generated in accordance with HDP-PR-FSS-722, *Final Status Survey Reporting.*

4. Define the Study Boundaries

For the HDP FSS, 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. The initial site designations based on the potential for residual contamination, the determination of survey units based on size and similar characteristics, and survey unit classification, all contribute to defining the study boundaries. The boundaries for each individual survey unit are identified and described in the FSS Plan for each survey unit and provided in the Survey Area Release Record.

5. Develop a Decision Rule

The decision rule is the determination of whether residual radioactivity exceeds the established $DCGL_W$ values. If the SOF is less than, or equal to any applicable action limit and unity (1), then no additional investigation is required and the survey unit will be recommended for unrestricted release. If the SOF is greater than unity (1), then the RSO is consulted to determine the appropriate action(s). Potential actions include re-classification, additional data collection, and additional remediation. To implement the decision rule, HDP-PR-FSS-721, *Final Status Survey Data Evaluation* provides the information necessary to calculate the sum-of-fractions. To ensure the DQO process has been properly implemented, a checklist is provided in HDP-PR-FSS-721, Appendix G-1, *Final Status Survey Data Quality Objectives Checklist.* In addition, HDP-PR-FSS-721 states the following: "The purpose of this procedure is to provide guidance to interpret survey results using the Data Quality Assessment (DQA) process during the assessment phase of Final Status Survey (FSS) activities in support of the Hematite Decommissioning Project."

6. Specify Limits on 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 (alternate hypothesis) as defined and shown below:

- Null Hypothesis (H_0) – the survey unit does not meet the release criterion; or,
- Alternate Hypothesis (H_a) – the survey unit does meet the release criterion.

A Type 1 decision error (α) 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 α value will always be set at 0.05 unless prior NRC approval is granted for using a less restrictive value.

A Type II decision error (β) 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 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.

7. Optimize the Design for Obtaining Data

The results of characterization and/or Remedial Action Support Surveys (RASS) are evaluated and used to optimize the FSS design and ensure the DQOs are met. The RASS data and/or characterization data may include gamma scans, surface scanning surveys (alpha + beta), soil sampling, and surface activity measurements in impacted soil areas and SSCs. This data is evaluated and used to refine the scope of field activities to optimize implementation of the FSS design and ensure the DQOs are met. HDP-PR-FSS-701, *Final Status Survey Plan Development* provides the instructions to evaluate the data and refine the scope of field activities. This is accomplished in concert with HDP-PR-HP-601, *Remedial Action Support Surveys*.

5.0 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, 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 DP Chapter 7. 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 was 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;
- NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM);
- NUREG-1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions; and,
- NUREG-1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys.

Guidance for conducting an FSS on piping internals is outside the scope of MARSSIM. These special situations will be evaluated by judgment sampling and measurements. 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. FSSFR Volume 5, Chapter 1, Piping Areas Surveys contains further discussion regarding FSS of piping.

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;
- Design;
- Implementation; and,
- Data Assessment

The DQO and 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). For additional discussion on the DQA process see section 6.3.

Survey planning includes review of the Historical Site Assessment, 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 10 CFR 20.1402 (Reference 8.26) 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.

Before the FSS process can proceed to the implementation phase, turnover and control measures will be implemented for an area or survey unit as appropriate. A formal turnover process will ensure that decommissioning activities have been completed and that the area or survey unit is in a suitable physical condition for FSS implementation. Isolation and control measures are primarily used to limit the potential for cross-contamination from other decommissioning activities and to maintain the final configuration of the area or survey unit.

Survey implementation is the process of carrying out the survey plan for a given survey unit. This consists of scan measurements, total surface contamination measurements, and collection and analysis of samples. Quality assurance and control measures are employed throughout the FSS process to ensure that subsequent decisions are made on the basis of data of acceptable quality. Quality assurance and control measures are applied to ensure:

- DQOs are properly defined and derived;
- the plan is correctly implemented as prescribed;
- data and samples are collected by individuals with the proper training using approved procedures;
- instruments are properly calibrated and source checked;
- collected data are validated, recorded, and stored in accordance with approved procedures;
- documents are properly maintained; and,
- corrective actions are prescribed, implemented and followed up, if necessary.

The DQA approach is applied to FSS results to ensure the population of the data are complete, the data are valid, and to determine whether the objectives of the FSS have been met. The data quality assessment includes:

- verify that the measurements were obtained using approved methods;
- verify that the quality requirements for the methods were met;
- verify that the appropriate corrections were made to the gross measurements and the data are expressed in proper reporting units;
- verify that the measurements required by the survey design, and any measurements required to support investigation have been included;

- verify that the classification and associated survey unit design remain appropriate based on a preliminary review of the data;
- subject the measurement results to the appropriate statistical tests;
- determine if the residual radioactivity levels in the unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

In some cases, data evaluation will show that all of the measurements made in a given survey unit were below the applicable $DCGL_W$. If so, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the $DCGL_W$ are observed. In these cases, statistical tests must be performed to determine whether the survey unit meets the release criterion. The statistical tests that may be required to make decisions regarding the residual radioactivity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

The statistical tests will include the Sign test, or the Wilcoxon Rank Sum (WRS) test for instances when the measurement results are corrected for the contribution from background radioactivity. Typically, the use of the WRS test will be limited to the evaluation of results obtained within open land surveys where Ra-226 and Th-232 are identified in soil. The balance of the measurements of soil within open land areas, and 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 be reclassified and a re-survey performed consistent with the change in classification.

5.1 Surrogate Evaluation Areas (SEAs)

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 was performed to determine scaling factors that could be used in the FSS planning process by inferring the concentration of one or more radionuclides by the measurement of a surrogate radionuclide. Using the surrogate relationship during the FSS planning process allows Westinghouse to estimate a reduced instrument scan MDC that accounts for the presence of hard to detect nuclides. This reduced instrument scan MDC is then used to calculate prospective soil sample population size to ensure that a sufficient number of soil samples are collected during the initial FSS if instrument scan sensitivities are not sufficient. This reduces the risk of collecting an insufficient number of samples that would require re-sampling of the area, and helps to ensure that all DQO's for the survey area are met.

The surrogate study 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. The table below provides the distribution ratios for the use of U-235 as a surrogate to infer the Tc-99 concentration in soil within three 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 5-1.

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 5-1 that was adjusted for the contributions from insignificant radionuclides was further modified. This calculation was performed using Equation 4-1 of MARSSIM, the result for the Surface Soil stratum in the Plant Soil SEA using the distribution ratio of 9.24 is illustrated below:

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 8.37

$$DCGL_{U-235,\mathrm{mod}} = \frac{1}{\left(\frac{1}{102.3}\right)+\left(\frac{9.24}{151.0}\right)} = 14.1 \text{ pCi/g} \qquad \text{Equation 5-1}$$

where: 102.3 = adjusted and modified soil $DCGL_W$ (pCi/g) for U-235 (from Table 3-1), and
151.0 = adjusted and modified soil $DCGL_W$ (pCi/g) for Tc-99 (from Table 3-1).

Surrogate Evaluation Areas, and the distribution ratios above are utilized in FSS planning only. Tc-99 is analyzed in all FSS samples to demonstrate compliance with the dose based unrestricted release criteria.

5.2 Tc-99 Side Wall Sampling

During the commencement of FSS activities in early 2015, during a NRC Region III inspection, the NRC Inspector questioned the site staff in regards to the FSS program requirements for excavation sidewall sampling. The NRC Inspector was specifically interested in sampling for Tc-99. The site staff reiterated the requirements as provided in the HDP DP Chapter 14.4.4.1.6.2, Subsurface Soil, and provided an explanation of how the requirements were implemented within the FSS program and procedures. This topic was conveyed to NRC Headquarters and HDP was subsequently provided with three options for addressing the issue.

Westinghouse opted to provide a sampling plan for excavation sidewalls. The following excerpt from the NRC – Westinghouse August 11 and 12, 2015, Publicly Noticed Teleconference Summary {ML15230A324}, summarizes resolution of the issue;

> "*Westinghouse described a plan that it developed for sampling the sidewalls. This plan was documented in an internal memo (HEM-15-MEMO-039). In this plan, the sidewalls were sampled if any sample in the survey unit had a concentration of greater than 2.5 pCi/g of Tc-99, which corresponds to 10% of the DCGL. Westinghouse said that this was based on insignificant radionuclides being defined in the DP as those which contribute less than 10% to the 25 mrem/yr dose criteria. The NRC staff questioned whether there would be a correlation between the Tc-99 concentration at the bottom of an excavation and the sidewall. Westinghouse responded that the concentration of Tc-99 in sidewall samples taken to date agree with the systematic samples and that the sidewalls were no more likely to contain elevated Tc-99.*
>
> *NRC staff concluded that the method described in HEM-15-MEMO-039 was acceptable, with the exception that the NRC staff had concerns with the first bullet (i.e., samples would only be taken if the systematic or biased samples from the survey unit exceeded 10% of the applicable DCGLw). Westinghouse committed to revising the memo to delete this bullet and to revise its procedure to include this information. NRC staff also noted that it would also be clearer if a definition of "vertical or near vertical" were included in these documents.*

Westinghouse stated that this process would be used in the future and had been used in the past. Westinghouse noted, however, that there are some survey units that have already been backfilled for which this process was not followed. These survey units had shallower excavations and had low Tc-99. Westinghouse will provide justification for the characterization of Tc-99 in the FSSRs for these survey units. NRC staff stated that it will evaluate the information available for those survey units when those FSS reports are submitted to determine if they have been characterized adequately."

As a result the FSS procedures were revised to include judgmental sidewall samples for Tc-99 using guidance below:

- If vertical or near vertical sidewalls exist within the survey unit (i.e., surfaces that are greater than a 45 degree angle), and
- If those sidewalls exceed 12 inches in height vertically, and
- If those sidewalls exceed (in aggregate) 5 % of the total survey unit surface area (e.g., greater than 100 m^2 of sidewall in a 2,000 m^2 survey unit), then discretionary sidewall sampling is necessary.

Additional discussion is provided in section 7.2.3.

6.0 FINAL STATUS SURVEY

Near the conclusion of remediation activities and prior to initiating a final status survey, isolation and control measures were implemented. The determination of readiness for controls and the preparation for final status survey were based on the results of characterization and/or a RASS that indicated residual radioactivity was unlikely to exceed the DCGLs. The control measures were implemented to ensure the final radiological conditions were not compromised by the potential for re-contamination as a result of access by personnel or equipment. These measures consisted of both physical and administrative controls. Examples of the physical controls included rope boundaries and postings indicating that access was restricted to only those persons authorized to enter by health physics. Administrative controls included approved procedures and personnel training on the limitations and requirements for access to areas under these controls.

6.1 Gamma Walk Over Surveys

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.

The intent of the 100% GWS of accessible surfaces performed in Class 1 areas was to provide a basis that the survey unit met the remedial goal. The GWS was performed and documented in a manner that met the DQOs of the FSS program.

6.1.1 Instrumentation

The data quality objectives process included the selection of instrumentation appropriate for the type of measurement to be performed (i.e., total surface contamination measurement, scan or both), that were 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 was made to identify the ROC at levels sufficiently below the DCGL. Detector selection was based upon detection sensitivity, operating characteristics, and expected performance in the field. The specific DQOs for instruments were 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.

Instruments and detectors were calibrated for the radiation types and energies of interest or to a conservative energy source. Calibration was performed on-site using HDP procedures or off-site by an approved vendor. Instrument calibrations were documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels were also attached to all portable survey instruments. Prior to using any survey instrument, the current calibration was verified and all operational checks were performed.

Radioactive sources used for calibration were traceable to the National Institute of Standards and Technology (NIST) and were obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector was used, suitable NIST-traceable sources were used for calibration, and the software set up appropriately for the desired geometry.

Prior to use on-site, all project instrument calibrations were verified and initial response data collected. These initial measurements were used to establish performance standards (response ranges) in which the instruments were tested against on a daily basis when in use. An acceptable response for field instrumentation was an instrument reading within ±20 percent of the established check source value. Laboratory instrumentation standards were within ±3-sigma as documented on a control chart.

The DQO process determined the frequency of response checks, typically before issue and after an instrument had been used (typically at the end of the work day but in some cases this was performed during an established break in activity, e.g., lunch). This additional check was to expedite the identification of a potential problem before continued use in the field. Instrumentation was response checked in accordance with HDP Site procedures. If the instrument response did not fall within the established range, the instrument was removed from use until the reason for the deviation could be resolved and acceptable response again demonstrated. If the instrument failed a post-survey source check, all data collected during that time period with the instrument was carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that FSS data were discarded, replacement data would be collected at the original locations. However, during the course of FSS surveys at the site, no instruments failed a post-survey source check, and therefore no FSS data was subject to potential failure.

6.1.2 Scan MDC

One of the most important elements of a scan survey is to 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.

If the survey instrument scan MDC is less than the applicable $DCGL_W$ for the stratum (elevation) in which the soil resides, then scanning is the primary method for guiding the remediation. The average net count rate corresponding to the $DCGL_W$ is 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.

$$MDCR_{surveyor} = \frac{1.38\sqrt{10{,}000 \times \frac{1}{60} \times \left(\frac{60}{1}\right)}}{\sqrt{0.5}} = 1{,}512\ cpm \qquad \text{Equation 6-1}$$

The table below 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 for various Uranium enrichments provided in the table below are sufficient to measure concentrations at the $DCGL_W$ for field instruments used for scanning.

Instrument/Detector Type	Typical Background	Typical MDC 95 Percent Confidence Level
Ludlum Model 2360/Ludlum 44-10 or equivalent 2 in by 2 in NaI scintillation detector	**10,000 cpm**	**84 pCi/g (3 percent enriched Uranium) 99 pCi/g (20 percent enriched Uranium) 122 pCi/g (50 percent enriched Uranium) 140 pCi/g (75 percent enriched Uranium)**

The scan MDC for open land areas may be reduced further by using field instrumentation coupled with a GPS unit there by enabling the scan data to be logged, downloaded, and mapped. By logging and mapping the data, the scan data can 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, specific areas may be more readily identified for further investigation, survey, and sampling as necessary. This technology reduces the impact of surveyor efficiency (e.g., surveyor efficiency of 0.5 without data logging, surveyor efficiency of 0.75 when data logging is used), thereby reducing the scan MDC by approximately 17 percent when GPS data logging technology is used.

6.1.3 Investigation Action Level (IAL)

Due to the nature of remediation operations it is expected that radioactivity above background levels may be detected during the performance of FSS after all necessary remediation has been completed. It is for this reason that a Scan IAL is provided as part of the FSS plan. The IAL provides the HP Technicians performing field work a count rate above general area background that may indicate soils concentrations approaching the applicable $DCGL_W$ that should be investigated further.

To determine what an appropriate Scan IAL is for Class 1 areas of the site, all Final Remedial Action Support Surveys (RASS) of "Area 1" (i.e., LSA's 10-01, 10-02, 10-03, 10-04, and 10-12) were compiled calculating the mean count rate and standard deviation of the data set. All 5 of these LSA's are Class 1 areas where remediation was necessary and had proceeded until completion indicated that the areas were ready for FSS. Additionally all 5 of these LSA's did at one time contain contamination from all 6 of the contaminants of concern (i.e., U-234, U-235, and U-238, Ra-226, Th-232, and Tc-99), and were representative of a typical Class 1 area on the Hematite Site. The mean count rate of the population was calculated to be 10,698 cpm, with a standard deviation of 1,616 cpm. This is consistent with the average general area background observed on the Hematite site of approximately 10,000 cpm as reported in the HRCR.

MARSSIM Chapter 5.5.2.6 *Determining Investigation Levels* describes the industry standard practice of flagging areas for investigation that exceed background levels by a value of 3 sigma. For "Area 1" this would represent a count rate of 4,848 cpm above general area background. For ease of use in the field, and to provide some inherent conservatism, this value will be rounded down to 4,000 net cpm. Appendix D provides a Westinghouse memorandum HEM-15-MEMO-021 which documents the establishment of the 4,000 ncpm IAL.

Additionally, all GWS data will be "post-processed" and reviewed by the FSS Supervisor to determine if any additional areas require investigation based on the recorded GWS data.

GWS Post Processing is performed on each FSS GWS survey. Every data point logged within the survey unit is used to determine a sample population mean and standard deviation. Next the threshold for standard deviations above the mean is set (e.g., 3 standard deviations above the mean). The data is then presented in the form of a "Z-score map" showing only those locations within the survey unit that exceed 3 standard deviations above the mean or greater. Biased sampling locations are chosen both based on this data and also using the professional judgment of the Health Physics Staff.

The Z-score process aids in the identification of small pockets of elevated radioactivity that could potentially remain within the survey unit by identifying how an individual measurement differs from the rest of the data population. Prior to performing the Z-score analysis, the data population is reviewed by the Health Physics Professional Staff to identify any potential anomalies; a standard soil GWS data set collected at HDP will show a mean of approximately 10,000 cpm with a standard deviation of 1,000 cpm. If the data set appears to be free of any anomalies that may potentially skew the results, then the entire data population is used for the Z-score analysis. However if data anomalies do appear (e.g., multiple data populations within a single data set) performing the Z-score analysis on the entire data population may not be conservative, so a "population of interest" is identified for the purposes of Z-score analysis.

Multiple data populations in a single dataset most frequently occur due to dissimilar background materials, geometries, or instrument response curves. In order to eliminate errors caused by differing instrument response curves the Z Scores are calculated for each instrument separately. To evaluate potential geometry issues, elevated areas are compared to the physical layout of the survey unit (e.g., holes, sidewalls, trenches), and biased samples are collected from the highest activity sample within the physical area. In the case of dissimilar materials (e.g., survey areas containing more than one material type such as soil and gravel), the lower activity materials (usually gravel) are analyzed separately from the main data population (usually soil) to ensure that the Z-score defined by standard deviations do not potentially increase the likelihood of a false negative (e.g., ensuring that the "data bins" are not too wide as to obscure analysis).

While the Z-score is a useful tool in identifying potential biased sampling locations, it is important to note that it is most effective when used in conjunction with the Investigation Action Level (IAL) and the professional judgment of the Health Physics Staff.

6.1.4 Exposed Surfaces versus Accessible Surfaces

In the Westinghouse response to RAI's it was clarified that 100% of exposed surfaces would be subject to scanning, and procedures were implemented to direct a 100% scan of all accessible surfaces. During NRC Region III inspections conducted in 2015, an issue arose regarding exactly what "100% scan" meant and the distinction between "exposed surfaces" versus "accessible surfaces" when conducting scan surveys. This issue initially involved the fact that it has become a common industry standard to conduct land surveys wearing a GPS unit to track where the scan is being performed. While the scan itself is being conducted following accepted industry standards, the GPS unit is not always indicative of where the detector is being placed by the surveyor. Once the survey is completed and the GPS data printed out, it may appear to show an area was "missed", which may or may not have occurred. This evolved to a discussion of exactly what constitutes "100% scan". For example, in small areas where the soil surface may be difficult or impossible to access due to such things as standing water or worker safety concerns, an actual 100% scan may in fact not be feasible. (For large areas that could be construed as initially inaccessible, other means are normally employed such as dewatering the area or using man lifts to aid in getting the surveyors into areas safely.)

During an October 8, 2015, publicly noticed teleconference with the NRC, it was agreed by all parties that 100% GPS coverage was not a requirement or a standard, but, 100% scan of exposed surfaces is the expected goal, recognizing that the goal may be mitigated where accessibility becomes an issue. Westinghouse agreed to modify the procedures to document when the circumstances occurred where an area could not be scanned in its entirety. Procedure HDP-PR-FSS-711; Step 8.4.3, was modified to read:

> "For Class 1 areas, a 100% GWS of the exposed surface is required. For Class 2 and Class 3 areas, scan each survey area as specified in the survey instructions and any additional areas based on professional judgment using HDP-PR-HP-416, *Operation of the Ludlum 2221 for Final Status Survey* (Reference 5.8) (or equivalent instrument authorized by the RSO). If a prescribed survey location or area cannot be scanned in its entirety, indicate this and any other deviation in the Field Log of the FSS Plan and Sampling Instructions."

Subsequently, during an October 29, 2015, publicly noticed teleconference, it was agreed that the FSSFR should reflect that 100% GWS is the expected objective and that Westinghouse should provide a justification when 100% GWS coverage was not achieved, or did not appear to be achieved based on the output of the GPS software. The extent of the justification required may be greater for situations where 100% survey was not performed due to inaccessibility than is required when gaps appear due to artifacts of the GPS system. Where these situations occurred they were discussed within the FSSFR for that specific survey unit.

6.1.4.1 <u>GPS vs. GWS Coverage</u>

As discussed above the intent of the statement "100% GWS coverage" is understood to mean that the field of view of the gamma sensitive probe (e.g. 2x2 NaI probe) had passed over the entire exposed surface of the Class 1 soil survey unit, thereby identifying any potentially elevated areas or anomalies that may require further investigation. The absence of such elevated areas or anomalies indicates that the systematic soils samples will be representative of a homogenous soil survey unit, while the presence of elevated areas or anomalies indicates possible heterogeneous areas that require separate evaluation. Also as described above, it is recognized that the field of view of the survey probe is not always adequately represented by the position of the GPS antenna, or the size of the GPS "dots" that are displayed on the GWS figure. In some cases the field of view of the probe is much larger than the nominal 3ft diameter of the GPS "dots", and in some cases it could be smaller (e.g. due to obstructions such as sidewalls, or structures).

There are two predominant reasons for gaps in GPS and/or GWS coverage. The first reason is that the handheld GPS technology used to perform GWS may experience occasional "drift" or momentarily loose signal. To address this issue, training is provided to all Health Physics Technicians performing FSS to recognize these situations, and to attempt to correct these situations in the field by pausing to regain signal, or resurveying areas where the GPS signal may have drifted. Although pausing to regain the signal or resurveying an area has proven effective to address the issues common with using GPS while performing GWS, it is still extremely difficult to achieve 100% GPS coverage within a survey unit by even the most experienced Technicians.

The second reason for gaps in GWS as indicated by GPS is that there may be areas of a survey unit that are not accessible by traditional GWS means due to issues such as the surface of the survey unit containing steep slopes, or there may be excavation safety concerns for the Technicians performing the survey or other similar situations. In these situations the survey unit is surveyed to the maximum extent that is safe and/or practical. If it is not possible to achieve 100% GWS coverage then an evaluation of the survey unit GWS will be performed to determine if the results of the survey are sufficient in regards to GWS completion. The technical aspects of an assessment will include an analysis of the GWS count rates surrounding the inaccessible area, the size of the inaccessible area, or the amount of GPS coverage within the survey unit in order to determine the potential for the inaccessible area to contain an area of elevated activity that could potentially exceed the DCGL.

6.1.4.2 Post Survey Processing of GPS Data, and Determining GWS Coverage

While performing the GWS, if in the professional opinion of the Health Physics Technician, 100% GWS coverage has not yet been achieved, then the survey is repeated or supplemented with additional scans until the Technician believes that 100% GWS coverage has been achieved. Additionally the Technician listens and observes the 2x2 NaI instruments audio and visual scale response while performing the GWS and areas of elevated activity are first marked with paint or flags by the Technician in the field. Upon completion of the GWS the data from the GPS and GWS undergo post survey processing and the 2x2 NaI instrument log data is reviewed to determine if there are any additional areas of elevated activity within the survey unit that require follow up investigation (there by reducing surveyor error and increasing surveyor efficiency). The purpose of the GWS is to identify potential "hot spots" within the survey unit that exceed the IAL and thereby are believed to have the potential to exceed the DCGL. The justification for the IAL of 4,000 net cpm was documented in HDP-TBD-FSS-003 [Reference 8.38], and as documented in Westinghouse letter HEM-15-85, *Regional Response to U.S. Nuclear Regulatory Commission Review of Westinghouse Hematite Final Status Survey Issues and Associated Technical and Regulatory Bases and Paths Moving Forward*, Westinghouse previously provided to NRC Region III the technical basis document HDP-TBD-FSS-003 for review during NRC inspection activities.

As part of HDP-TBD-FSS-003, "Modeling and Calculation of Investigative Action Levels for Final Status Soil Survey Units" [Reference 8.38], Westinghouse developed the basis to determine when an elevated area or anomaly had the potential to exceed a DCGL. In this TBD Microshield© modeling was performed to support the calculations to determine the count rate that when collected with a 2x2 NaI probe would potentially exceed a DCGL. Area Factors were also applied to determine the maximum size of the potentially elevated area, and the calculated result determined that the IAL_{EMC} for a 300 m^2 area was 7,073 net cpm (Uniform DCGL) with a 2x2 NaI probe. The IAL_{EMC} for a 1 m^2 area was calculated to be 107,225 net cpm (Uniform DCGL) with a 2x2 NaI probe.

The result of the modeling demonstrates that an area of elevated activity would have to be physically very large (300 m^2 or greater) in order to potentially exceed the DCGL, as 300 m^2 is 15% of a typical 2,000 m^2 MARSSIM Class 1 survey unit, or very elevated (far exceeding the scan IAL of 4,000 net cpm) as readings in excess of 100,000 net cpm would skew high results in all surrounding areas much larger than 1 m^2.

When it is considered that the scan minimum detectable concentration (MDC) calculation, as well as the overall MARSSIM survey design, includes a 5% alpha error rate and a 10% beta error rate, (i.e. the expectation is not 100% accuracy, but rather a level of effort achieving the 95% confidence interval, at a minimum), it supports the conclusion that it is not necessary to achieve 100% GPS coverage in order to assess the potential for elevated areas or anomalies over 100% of the surface of the survey unit at the 95% confidence interval.

Next, in order to determine how much of the survey unit was covered by the GPS "dots" an assessment is performed by comparing the number of map pixels covered by GPS readings to the number of map pixels not covered by GPS readings to assign a percent value of GPS coverage.

HDP-TBD-FSS-003, "Modeling and Calculation of Investigative Action Levels for Final Status Soil Survey Units" [Reference 8.38] established that in order to potentially exceed a DCGL the area for a "hot spot" of 7,073 net cpm would cover an area of more than 300 m2 of a survey unit (approximately 15% of the surface area of the survey unit that is 2000 m2). As a conservative measure, HDP-TBD-FSS-003 established an Investigative Action Level of 4,000 net cpm.

Therefore, if 85 % or more of the survey unit undergoes a GWS, and no single un-surveyed area exceeds 300 m2, and there are no readings in the vicinity of any apparent GPS coverage gap approaching or exceeding the IAL of 4,000 net cpm there is assurance that there is not an area that would potentially exceed a DCGL that would go undetected.

If the GPS coverage is determined to exceed 95% with no readings approaching or exceeding the IAL of 4,000 net cpm in the vicinity of any apparent GPS coverage gaps, then the survey unit will be determined to meet the intent of the "100% GWS coverage" requirement. If the GPS coverage is less than 95%, or if elevated readings approaching or exceeding the IAL of 4,000 net are present in the vicinity of any apparent GPS coverage gaps then additional investigation will be performed and documented within the specific survey unit release record. These additional investigations will include an assessment of any supplemental manual GWS, investigative soil sampling, or other evaluations performed by the FSS group as necessary.

6.1.4.3 Determining FSS GWS Acceptability

In summary the modeling demonstrates that there is very little concern that small areas accounting for less than 5% of the survey unit would have the potential to contain elevated areas or anomalies that could potentially exceed the DCGL given that the scan Investigation Action Level for FSS surveys has been set at 4,000 net cpm. Therefore, if for all FSS GWS that were completed by a trained and qualified Health Physics Technician, verified by post survey processing to meet the 95% GPS coverage threshold, and not containing any readings approaching or exceeding the IAL of 4,000 net cpm in the vicinity of any apparent GPS coverage gap, then the GWS is adequate to demonstrate acceptability of the survey unit. For survey units in which the 95% GPS coverage threshold is not met, or elevated readings are identified in the vicinity of apparent GPS coverage gaps, then further assessment of GWS coverage will be performed.

6.2 Soil Sampling

6.2.1 Systematic Soil Sampling

All FSS soil sampling performed at HDP is analyzed for Uranium isotopes U-235, U-238, and U-234 (inferred), Technetium isotope Tc-99, Thorium isotope Th-232 (plus decay chain), and Radium isotope Ra-226 (plus decay chain). A Thorium background concentration of 1.0 pCi/g is subtracted from all soil sample results; and a Radium background concentration of 0.9 pCi/g when ingrowth is not used, and 1.07 pCi/g when ingrowth is used is subtracted from all soil sample results as well. No background values are used for Uranium or Technetium.

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 exposed areas 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 is adjusted to ensure that small areas of elevated radioactivity are detected. Special situations are evaluated by judgment sampling and measurements.

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. Systematic sampling and measurement locations for Class 1 and Class 2 survey units were located in a systematic pattern or grid. The grid spacing, *L*, was determined based on the SU size and the minimum number of sampling or measurement locations determined. Once the grid spacing was established, a random starting point was established for the survey pattern using a random number generator.

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 (excavation surface is within the Surface Stratum):
  - A surface sample from the ground surface to 15 cm bgs (Surface Stratum);
  - A composite sample from 15 cm bgs to 1.5 m bgs (Root stratum); and,
  - If the SOF in the sample obtained from the Root stratum exceeds 0.5, a sample from 1.5 m bgs to 1.65 m bgs (Deep stratum).

- Root Stratum Depth (excavation surface is within the Root Stratum):

  - A composite sample from the excavation surface (e.g., below 15 cm bgs) to 1.5 m bgs (Root stratum); and,
  - A sample from 1.5 m bgs to 1.65 m bgs (Deep stratum).

- Deep Stratum Depth (excavation surface is within the Deep Stratum):
  - Samples will be taken from the top 15 cm of the exposed surface (e.g., below 1.5 m bgs) and analyzed.

6.2.1.1 <u>Performance of the WRS Test</u>

The number of systematic sample locations collected during FSS sampling in a SU is dependent on a prospective statistical test (Wilcoxon Rank Sum Test) performed during the FSS planning phase which typically results in a minimum of eight sample locations. Following completion of the systematic sampling and analysis, a retrospective evaluation is performed to verify the sample size was appropriate based on the FSS data.

While the prospective WRS Test determines the minimum number of systematic sampling locations for the SU, multiple samples may be collected at each location within the SU if more than one stratum of soil remains at that location, and furthermore multiple $DCGL_W$'s may be used within the SU when the "3 Layer" approach is used. The use of multiple $DCGL_W$'s within the SU when the "3 Layer" approach is used prompted discussion between Westinghouse and the NRC regarding how the WRS Test should be applied. This discussion eventually expanded to include the application of the WRS Test when in the Uniform approach is used in a SU as well. During the publicly noticed teleconference help on January 26, 2017, at the request of NRC Headquarters, Westinghouse agreed to implement the WRS Test in the following fashion:

- Regardless of the $DCGL_W$ used, or the depth at which the systematic samples are collected, all systematically collected soil samples within a SU will be used for the performance of the WRS Test (this total number of samples will be equal to or greater than the number of sample locations that was determined by the prospective WRS Test evaluation). While including all systematically collected samples may increase the likelihood that the WRS Test will be successful, it will also increase the likelihood that the WRS Test will be determined to be required, and the NRC believes that this is also the most appropriate way to represent the data within the SU.

- When the "3 Layer" approach is used the most restrictive $DCGL_W$ within the LSA will be used to perform the WRS Test for both the activity concentrations of the background reference area samples, and the systematically collected samples regardless of the depth at which the sample was collected. This will typically mean the Root Stratum DCGLs will be used if there is soil remaining in the Root Stratum within the LSA. However, if the Surface and Root Stratums have been completely removed, then the Excavation Stratum DCGLs will be used.

In accordance with Step 7.8.3 of HDP-PR-FSS-721 *Final Status Survey Data Evaluation*, the WRS statistical test is required for a SU when the difference between the maximum SU data set gross SOF and the minimum background area SOF was greater than one (1.0) using the (most restrictive) $DGGL_W$ for the SU. However the WRS Test is typically performed and provided for illustrative purposes with each SU release record regardless if it is determined to be required or not. The 32 background area soil sample results that are discussed in Volume 1, Chapter 1, Section 5.1.3 are also used for the purpose of the WRS Test. These background soil samples were collected from two separate areas that are considered similar to the site, and therefore representative of background soil activity at the Hematite Site.

Gross SOF values were determined for the 32 background samples, and the systematically collected soil samples from the SU using the appropriate (and most restrictive) $DCGL_W$ for the SU. Next the $DCGL_W$ (i.e. 1.0) is added to the weighted sum of each background sample to obtain adjusted activity concentrations of the 32 samples collected within the Background Reference Area (e.g. adding unity to the gross SOF determination of the background reference area samples). The systematically collected samples in the SU are ranked against the adjusted activity concentrations of the 32 samples collected within the Background Reference Area. The SU passes the WRS Test if the ranked sum of the reference area ranks, or test statistic W_R, exceeds the critical value for the test. As such, when the WRS Test is successful the null hypothesis that the SU average concentration is greater than the $DCGL_W$ is rejected.

6.2.2 <u>Biased Soil Sampling</u>

As discussed above in Section 6.1.3, there are three key methods for identifying areas for biased soil sampling, the IAL, the Z-score of the FSS GWS, and the professional judgment of the HP Staff. These three methods are best when used in conjunction with each other to determine when an area of a survey unit may be non-homogenous, and require separate evaluation (i.e., biased sampling).

When biased soil sample locations are selected, a sample will be taken from the top 15 cm of the exposed surface and analyzed at the offsite laboratory for the same parameters as the systematic FSS samples (e.g., Gamma Spec and Tc-99 by ICP-MS). Biased samples that do not exceed a SOF of 1.0 are not used to calculate final survey unit dose and are not included in the systematic mean. If a biased sample does exceed a SOF of 1.0, then an EMC will be performed to determine if the survey area is still suitable for release as discussed further in Sections 3.1.3 and 3.2.2.

6.2.3 <u>Judgmental/Tc-99 Side Wall Soil Sampling</u>

Although not addressed in the RAI's for Revision 0 of the DP, the NRC raised the concern for the potential lateral movement of Tc-99 (a hard to detect nuclide), and requested that sidewalls within excavated survey areas be evaluated above the requirements for systematic and biased sampling as stated in the DP.

The NRC presented the following as one potential option for addressing these sidewall samples:

"The area of the excavation floor would be compared to the area of the sidewalls of the excavation, and the total number of samples would be increased proportionally. An appropriate number of samples would be determined for the excavation floor per MARSSIM, and a random start sampling grid would be developed for the floor. A random start sampling grid would also be developed for the sidewalls, and the number of samples would be determined as a proportion of the number of samples for the excavation floor. For example, if the floor encompasses 2000 m^2 and the sidewalls represent a 10% areal increase to 2200 m^2, an additional 10% of samples should be taken from the sidewalls. It may also be appropriate for the licensee to set a discretionary number of samples for the sidewalls to ensure that the measurements are representative (i.e., there will always be a minimum of x number of samples from the sidewalls)."

In response to this request from NRC Headquarters, the HDP FSS procedures were revised to include judgmental sidewall samples for Tc-99 using the guidance below:

- If vertical or near vertical sidewalls exist within the survey unit (i.e., surfaces that are greater than a 45 degree angle), and
- If those sidewalls exceed 12 inches in height vertically, and
- If those sidewalls exceed (in aggregate) 5 % of the total survey unit surface area (e.g., greater than 100 m^2 of sidewall in a 2,000 m^2 survey unit), then discretionary (aka. judgmental) sidewall sampling is necessary.

If judgmental sidewall samples are necessary:

- Determine the number of samples to be collected based on the sidewall surface area compared to the two dimensional systematic surface area (e.g., 8 systematic samples were collected over 2,000 m^2, then collect 1 sample per 250 m^2 of sidewall).
- Collect a judgmental sample(s) at sidewall location(s) not based on radiological scans, but selected at the discretion of the Health Physics Technician performing soil sampling.

6.2.4 Quality Control Soil Sampling

During the FSS within an open land survey unit, the laboratory was assessed through the analysis of field and laboratory duplicate samples. Field duplicate samples consisted of splitting a homogenized sample into two or more separate samples for analysis. Field duplicates were obtained from one location, homogenized, divided into separate containers, and treated as separate samples. Laboratory duplicate samples consisted of the re-analysis of the same sample at the laboratory. Both types of quality assurance samples were analyzed at a frequency of one sample per 20 final status survey samples collected (5%). Field duplicate samples were evaluated per guidance in MARLAP. Laboratory duplicates were evaluated internally by the laboratory and were part of the laboratory's QA program.

6.2.5 Off-site Laboratory

Test America in St. Louis, Missouri is the laboratory selected by Westinghouse to conduct the sample analysis for the Hematite decommissioning project. Test America was 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, such as NELAP, NVLAP, and ISO 9001:2000 were also considered during their evaluation. Maintenance of applicable accreditations is imposed as a quality requirement on the purchase order. Methods used were standard industry methods from the U.S. Environmental Protection Agency (EPA) and the Environmental Measurements Laboratory (EML).

Following receipt of the results of laboratory analyses, HDP staff performed a data review to assess the validity of the data for use in the final status survey. This review included an evaluation of the data to ensure that all of the DQOs were met.

As previously discussed with the NRC (and provided to the NRC in Westinghouse letter HEM-11-96 dated July 5, 2011, (Reference 8.20)), the contract laboratory performed data review, verification, and reporting in accordance with approved standard operating procedures (SOPs). In accordance with these SOPs, analytical data was reviewed by the analyst performing the task, followed by a secondary review by a department supervisor/lead analyst or their designee, and then review by the associated project manager. The vendor QA department performed an independent random review as oversight of the process. This review was documented on a data review checklist specific to each analytical method. Following receipt of laboratory data for use in the final status surveys, HDP staff performed an additional data review to assess the validity of the data. This review included an evaluation of the data to ensure that all of the DQOs were met. Essentially, this meant that the program had been structured to place a high degree of responsibility on Test America to verify the validity of the data prior to it being provided to Westinghouse. Therefore, there was no specific basis for HDP personnel to re-validate the analytical results from the backfill soil and off-site borrow locations.

In Westinghouse letter HEM-14-31, dated March 13, 2014, (Reference 8.36), Westinghouse sent the NRC the results of radiological testing of backfill soil from an off-site borrow location. That submittal contained radiological data for Tc-99 in which 2 of the 16 samples showed concentrations that were slightly above their minimum detectable concentrations (MDCs). The data, provided by Test America, was identified as questionable by the NRC, indicating the soil samples shouldn't identify any Tc-99 in excess of the MDC. After conducting an initial review of the data, Westinghouse confirmed that the Tc-99 data was anomalous. As a result of that finding, Westinghouse entered the issue into the Westinghouse Corrective Action Prevention and Learning (CAPAL) Program for follow up.

The CAPALs had two primary areas of focus. The first area focused on assessing the Test America program and its effectiveness in meeting expectations and commitments for data acceptance, quality control, and verification. An investigation of the Test America program included an independent audit by outside experts. The audit identified numerous deficiencies and issues, including: use of a backup counting instrument that laboratory personnel failed to identify as providing biased high results for Tc-99 (for a total of 90 out of 190 samples); a laboratory information management system that contained a Reporting Limit in mg/kg metals concentration that did not meet the HDP Requested Limit of 1 pCi/g Tc-99 activity, and; a failure by laboratory personnel to adequately validate the final sample results that were included in the final report to Westinghouse. All these issues were addressed by Test America. Corrective actions included: performing an Extent of Condition that validated the integrity of historical data produced on the backup instrument and confirmed no impact to off-site borrow soils used for backfill at HDP; the primary instrument which was returned to service after repair and used for Tc-99 analysis results was verified to be providing correct results; conducting intra laboratory comparisons to validate data results: and, conducting numerous meetings with Test America to ensure expectations and contract obligations would be met going forward.

The second area of focus was to understand why HDP personnel did not identify the anomalous sample results before they were transmitted to the NRC. While HDP did have an opportunity to identify the errant data, a number of factors contributed to the failure to do so. In accordance with the contract laboratory SOPs, analytical data is reviewed by the analyst performing the task, followed by a secondary review by a department supervisor/lead analyst or their designee, and then review by the associated project manager. The vendor QA department performs an independent random review as oversight of the process. This review is documented on a data review checklist specific to each analytical method. Following receipt of laboratory data for use in the final status surveys, HDP staff performs an additional data review to assess the validity of the data. This review includes an evaluation of the data to ensure that all of the DQOs have been met. This means that the program was structured to place a high degree of responsibility on Test America to verify the validity of the data prior to it being provided to Westinghouse. Therefore, there was no specific basis for HDP personnel to re-validate the analytical results from the backfill soil and off-site borrow locations.

Regardless, in manipulating the data to perform the statistical analyses, there was an opportunity to identify that the data was questionable. However, that would depend in large part on the experience and knowledge of the individuals conducting the work. Since Tc-99 is normally not found in native soils, and most of that which is present, if detectable, is from atomic bomb testing in the 1950's and 1960's, not everyone who saw Tc-99 in testing of native soil would recognize that it may not be valid. In this case, no one at HDP identified that having any Tc-99 in the soil sample results was suspect. Lastly, the HEM-14-31 letter (Reference 8.36) which contained the questionable data was issued during a turnover of decommissioning contractors at HDP. During that time frame, there was an effort to complete specific tasks before a change in personnel occurred and this may have hampered giving this issue the attention it needed.

Since the beginning of remediation activities in March, 2012, and through 2015, Test America analyzed over 20,000 radiological soil samples for the Westinghouse HDP. The sample results identified as biased high in regard to the use of the backup counting instrument were the only results that were identified as questionable. Since that data represented an extremely small error rate overall by Test America, Westinghouse believes the program as structured was acceptable.

6.3 Data Quality Assessment

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). 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 DCGLw); 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 DCGLw 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.

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 MARSSIM (NUREG-1575), 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.

A detailed statistical review of analytical data, as presented in Chapter 14 of the Hematite Decommissioning Plan, will be provided in each subsequent chapter containing LSA Survey Area Release Records.

7.0 SURVEY AREA RELEASE RECORD ORGANIZATION

In accordance with HDP-PO-FSS-700, *Final Status Survey Program*, documentation of the FSS will transpire in two types of reports, FSS Survey Area Release Records and an FSS Final Report, and will be consistent with Section 8.6 of NUREG-1757, Volume 2, Consolidated Decommissioning Guidance – Characterization, Survey, and Determination of Radiological Criteria.

The FSS Final Report will incorporate multiple Volumes. The first Chapter of Volumes 2 through 5 will include general Site information and an overview of the FSS Program for that subject area. (Volume 2, *Reuse Soil*; Volume 3, *Land Survey Areas*; Volume 4, *Building Survey Areas*; Volume 5, *Piping Survey Areas;* Volume 6, *Groundwater*) Subsequent Chapters within these Volumes will contain FSS Survey Area Release Records.

Survey Area Release Records are prepared to provide a record of the composition and location of the survey area; the measurements obtained during the FSS; the number and location of any small areas of elevated concentration; and a summary of the data that represents the final radiological condition, including a determination that an individual survey area/unit meets the release criteria.

The Survey Area Release Records will be formatted to contain the following information:

a. An Introduction section which will include Survey Unit specific information (e.g., geographical description, summary of historical radiological data).

b. A description of the specifics of FSS Protocol and DQOs, including but not limited to:
   - Survey Unit designation and classification.
   - Background determination.
   - Instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration).
   - Survey methodology and protocols.
   - QC surveys.
   - A discussion of any changes that were made in the FSS from what was proposed in the DP, including its classification.

c. Conclusion, including dose estimate from all pathways and estimated dose contribution from groundwater.

d. Supporting documents (e.g., spreadsheets, statistical analyses, figures, tables).

Appendix C, HDP FSSFR LSA Document Matrix, provides a status of the FSS documents that will be submitted as supporting information to FSSFR Volume 3. This table will be updated as FSS Release Reports are generated.

8.0 REFERENCES

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8.2 NRC letter 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)(4), and Issuance of Hematite License Amendment 57* {ML112101699}

8.3 License SNM-33 Amendment 57 {ML112101640}

8.4 Hematite Decommissioning Plan Safety Evaluation Report {ML112101630}

8.5 NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual

8.6 NUREG-1757, Volume 2, Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria

8.7 NUREG-1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions

8.8 NUREG-1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys

8.9 Code of Federal Regulations, Title 10, Part 20.304, "Disposal by Burial in Soil," 1964

8.10 Gulf United Nuclear Fuels Corporation, Nuclear And Industrial Safety, Commercial Products Division, Memorandum NIS:DGD-70-332, Peter Loysen, "AEC Inspection Wrap-Up Meeting, November 5, 1970"

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8.14 U.S. Nuclear Regulatory Commission, Letter to ASEA Brown Boveri/Combustion Engineering, C. J. Haughney to J. A. Rode, "Authorizes Backfill of Area of Bldg. 253 Construction Site," July 2, 1990 {ML052550383}

8.15 Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, "Spent Limestone Results," April 7, 1989 {ML053640233}

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8.17 Gateway Environmental Associates, Letter to R.W. Sharkey, "Exploratory Probehole Investigation for the Evaporation Ponds at the ABB Combustion Engineering Hematite Facility," April, 17, 1997

8.18 Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, "Evaporation Pond Characterization," October 22, 1997

8.19 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, "Hematite License Amendment Application Dated May 30, 1997," January 26, 1998.

8.20 Westinghouse letter HEM-11-96, Final Supplemental Response to NRC Request for Additional Information on the Hematite Decommissioning Plan and Related Revision to a Pending License Amendment Request, July 5, 2011{ML111880290}

8.21 NRC Inspection Report 07000036/2014005(DNMS) {ML15054A418}

8.22 Westinghouse Electric Company Document No. HDP-PO-WM-900, "Waste Management and Transportation Plan"

8.23 ASEA Brown Boveri/Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, J.A. Rode to D.J. Sreniawski, "Building 253 Construction Site Soil," May 14, 1990 {ML052550381}

8.24 U.S. Nuclear Regulatory Commission, Letter to ASEA Brown Boveri/Combustion Engineering, C. J. Haughney to J. A. Rode, "Authorizes Backfill of Area of Bldg. 253 Construction Site," July 2, 1990 {ML052550383}

8.25 Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, "Spent Limestone Results," April 7, 1989 {ML053640233}

8.26 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use"

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8.33 NRC letter dated January Westinghouse Electric Company (Hematite) Acknowledgement of Disputed Violation of NRC Inspection Report 0700036/2015003(DNMS) {ML16020A093}

8.34 NRC letter dated October 8, 2015, U.S. Nuclear Regulatory Commission Conclusions Associated with the Utilization of Off-site Borrow Material at the Westinghouse Hematite Site {ML15275A428}

8.35 Westinghouse letter HEM-15-39, Additional Statistical Analysis for Backfill Soil from an Off-site Borrow Location {ML15117A151}

8.36 Westinghouse letter HEM-14-31, Radiological Testing of Backfill Soil from an Off-site Borrow Location {ML14072A485}

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Table 3-1

Adjusted and Modified Soil DCGLw Values for Demonstrating Compliance, TC-99 Surrogate Evaluation Area (SEA)

Radionuclide	DCGL$_W$ (pCi/g) By Conceptual Site Model									
	Surface Soil		Root Stratum		Deep Volumetric [a]		Uniform [b]		Excavation [a]	
	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99
Plant Soil SEA										
Total Uranium [c]	394.3	191.7	202.4	52.8	2917	2895	170.2	44.1	706.3	202.8
U-234	508.5	508.5	235.6	235.6	2890	2890	195.4	195.4	872.4	872.4
U-235	102.3	14.1	64.1	3.0	3034	2565	51.6	2.5	208.1	11.8
U-238	297.6	297.6	183.3	183.3	3028	3028	168.8	168.8	551.1	551.1
Tc-99	151.0	N/A	30.1	N/A	98649	N/A	25.1	N/A	74.0	N/A
Th-232 + C	4.7	4.7	2.0	2.0	9279	9279	2.0	2.0	5.2	5.2
Ra-226 + C	5.0	5.0	2.1	2.1	13029	13029	1.9	1.9	5.4	5.4

a The distribution ratio for Deep Stratum soil was used to calculate the DCGL$_W$ for Total Uranium and U-235 when inferring Tc-99

b The distribution ratio for Root Stratum soil was used to calculate the DCGL$_W$ for Total Uranium and U-235 when inferring Tc-99

c Total Uranium DCGL$_W$ values were calculated using Equation 4-4 of MARSSIM, adjusted DCGL$_W$ values from Table 14-4 of attachment 4 to HEM-11-96, modified U-235 DCGL$_W$ values from Table 14-9 of attachment 4 to HEM-11-96, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4% in soil.

Table 3-1 (continued)

Adjusted and Modified Soil DCGLw Values for Demonstrating Compliance, TC-99 Surrogate Evaluation Area (SEA)

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 of attachment 4 to HEM-11-96, modified U-235 DCGL$_W$ values from Table 14-9 of attachment 4 to HEM-11-96, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4%.

Table 3-1 (continued)

Adjusted and Modified Soil DCGLw Values for Demonstrating Compliance, TC-99 Surrogate Evaluation Area (SEA)

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 of attachment 4 to HEM-11-96, modified U-235 DCGL$_W$ values from Table 14-9 of attachment 4 to HEM-11-96, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4%.

Table 3-2a

Area Factors for Soil Contamination

Radionuclide	Elevated Measurement Area (m2)									
	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 3-2b

Calculated Area Factors Based On Excavation Scenario Constraints 1 And 2

Radionuclide	**Contiguous Elevated Area after Excavation (size of elevated area shown in m^2)***					
	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 3-2c

Effective Area Factor For Use With Excavation DCGLs

Radionuclide	**Size of elevated area shown in m^2**					
	148	**100**	**30**	**10**	**3**	**1**
U-234	1.0	2.0	6.7	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	2.0	6.7	20	67	200
Th-232 + C	1.0	1.9	3	4	7	14
Ra-226 + C	1.0	2.0	4	5	10	20
Np-237 + D	1.0	2.0	6.7	17	37	79
Pu-239/240	1.0	2.0	6.7	20	67	117
Am-241	1.0	2.0	6.7	17	32	58

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

Table 3-3

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 3-4

Groundwater DSRs

Radionuclide	Well Water Concentration (pCi/L)	TEDE (mrem/yr) [a] For Water-Dependent Pathways	DSR_{GW} (mrem/yr per pCi/L)
U-234	5.707 E+00	8.744 E-01	0.1532
U-235 + D	5.707 E+00	8.261 E-01	0.1448
U-238 + D	5.707 E+00	8.302 E-01	0.1455
Tc-99	9.415 E+00	8.826 E-03	9.374 E-04
Th-232 + C	3.030 E-01	1.007E+00	3.323
Ra-226 + C	6.346 E-03	4.786E-01	75.42
Np-237 + D	3.966 E+01	1.118 E+02	2.819
Pu-239/240	8.332 E-01	1.744 E+00	2.093
Am-241	1.190 E-01	3.098 E-01	2.603

[a] At t = 0 years

Table 5-1

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 accounting for the dose contribution from insignificant radionuclides (see Equation 14-1 in Section 14.1.3.2 of the Hematite DP).
[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.

Figure 2-1
Area of Documented Burial Pits Based on Historical Information

Figure 2-2
Burial Pits Easily Identified by Change in Soil Color

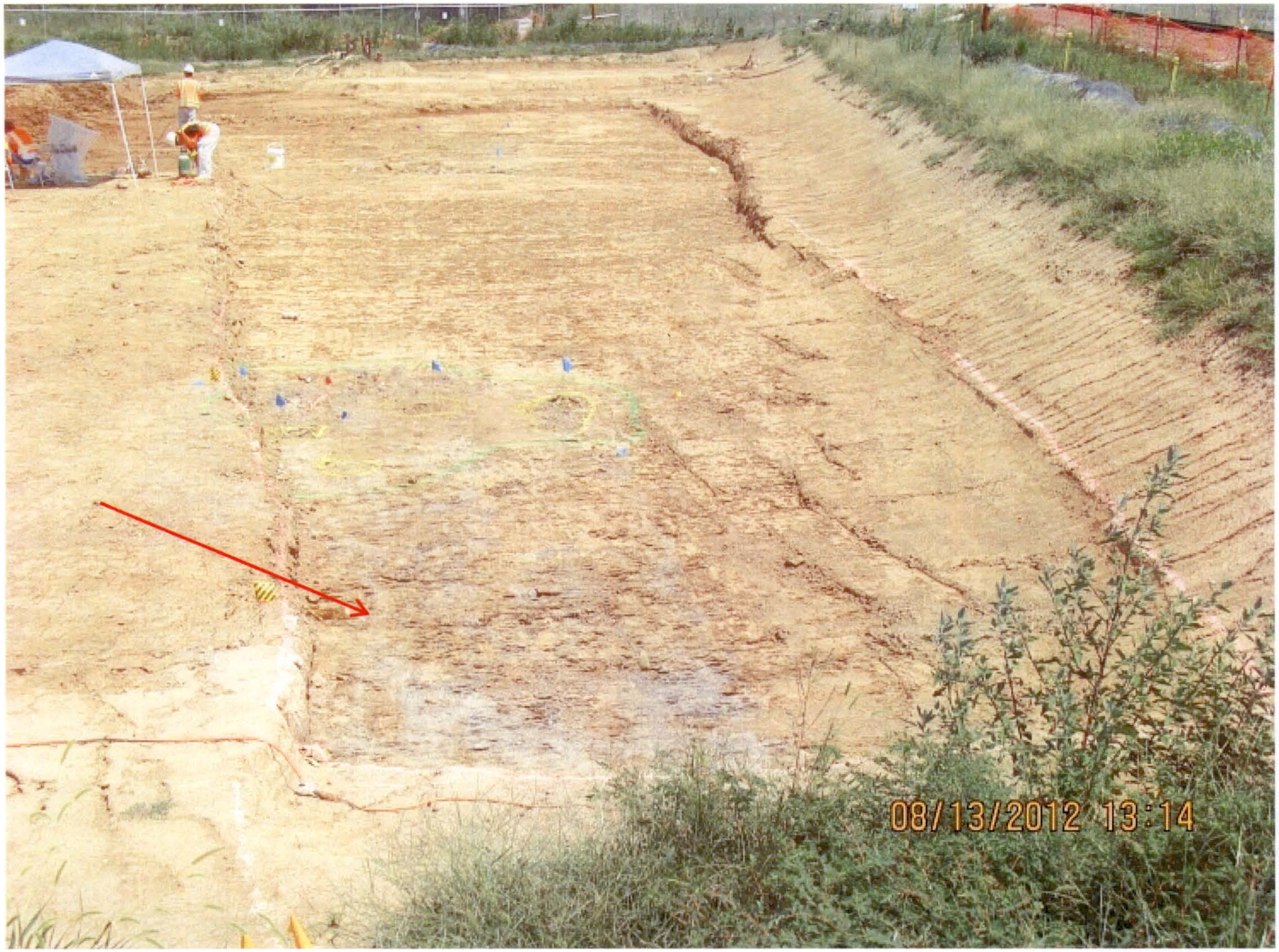


Figure 2-3
Burial Pit Easily Identified by Change in Soil Color

Figures 2-4
Example of Worker Conducting a Gamma Walkover Survey (GWS) in the Burial Pit Area with a NaI 2 x 2 Detector

Figure 2-5
Examples of Radium Contaminated Filter Plates Being Unearthed While Excavating in the Burial Pit Area

Figures 2-6
Examples of Radium Contaminated Filter Plates Being Unearthed While Excavating in the Burial Pit Area

Figure 2-7
Examples of Debris that was Exhumed During Remediation Excavation of the Burial Pit Area

Figure 2-8
Examples of Debris that was Exhumed During Remediation Excavation of the Burial Pit Area

Figure 2-9

Burial Pit Area March 2012 (Looking NW)
The Removal of Overburden Soil Has Commenced

Figure 2-10

Burial Pit Area September 2012 (Looking SE)
The Top Layer of Overburden Which Contained the Sod Has Been Removed and Remediation Is Well Under Way

Figure 2-11

Burial Pit Area March 2013 (Looking SE)
Remediation in Some Areas Is to Significant Depths Much of the Burial Pit Area Elevation Is Below That of the Northeast Site Creek

Figure 2-12

Burial Pit Area September 2013 (Looking SE)
A Significant Portion of Remediation of the Burial Pit Area Proper Has Been Completed: Remediation of Non-Burial Pit Areas Has Been Expanded To the Northeast Site Creek to the Left and Towards the Former Process Building Area to the Right

Figure 2-13

Burial Pit Area November 2013 (Looking SE)
The Geometric Shape of the Excavations Indicate the Ability of the Excavation Process to Identify Burial Pits and Remediate to a Depth that Ensures the Burial Pit Has Been Removed

Figure 2-14

Burial Pit Area September 2014 (Looking North)
Remediation of the Burial Pit Area Proper Is Nearly Complete

Figure 2-15

Burial Pit Area October 2014 (Looking South)
Shortly After This Photograph Was Taken All Visual Inspections and Radiological Surveys Associated With the Burial Pit Area Were Completed and the Data Indicated That the Burial Pit Area Remediation Was Complete

Figure 2-16

Burial Pit Area May 2015 (Looking North)
Photograph of the Burial Pit Area Prepared for Final Status Survey
The Silt Fence Is In Place for Isolation and Control of the Area

Figure 2-17

Burial Pit Area September 2015 (Looking SE)
Backfill Operations in the Burial Pit Area

Figure 2-18

Burial Pit Area September 2015 (Looking South)
NRC Region III Inspector Inspecting the Backfill of the Burial Pit Area

Figure 2-18a
Example of Auger Types Used to Perform NCS Borings

Figure 2-18b

Change in Color of Soil in Bottom of Burial Pit

Figure 2-19

Core Bore Locations Performed for Nuclear Criticality Safety Control Purposes

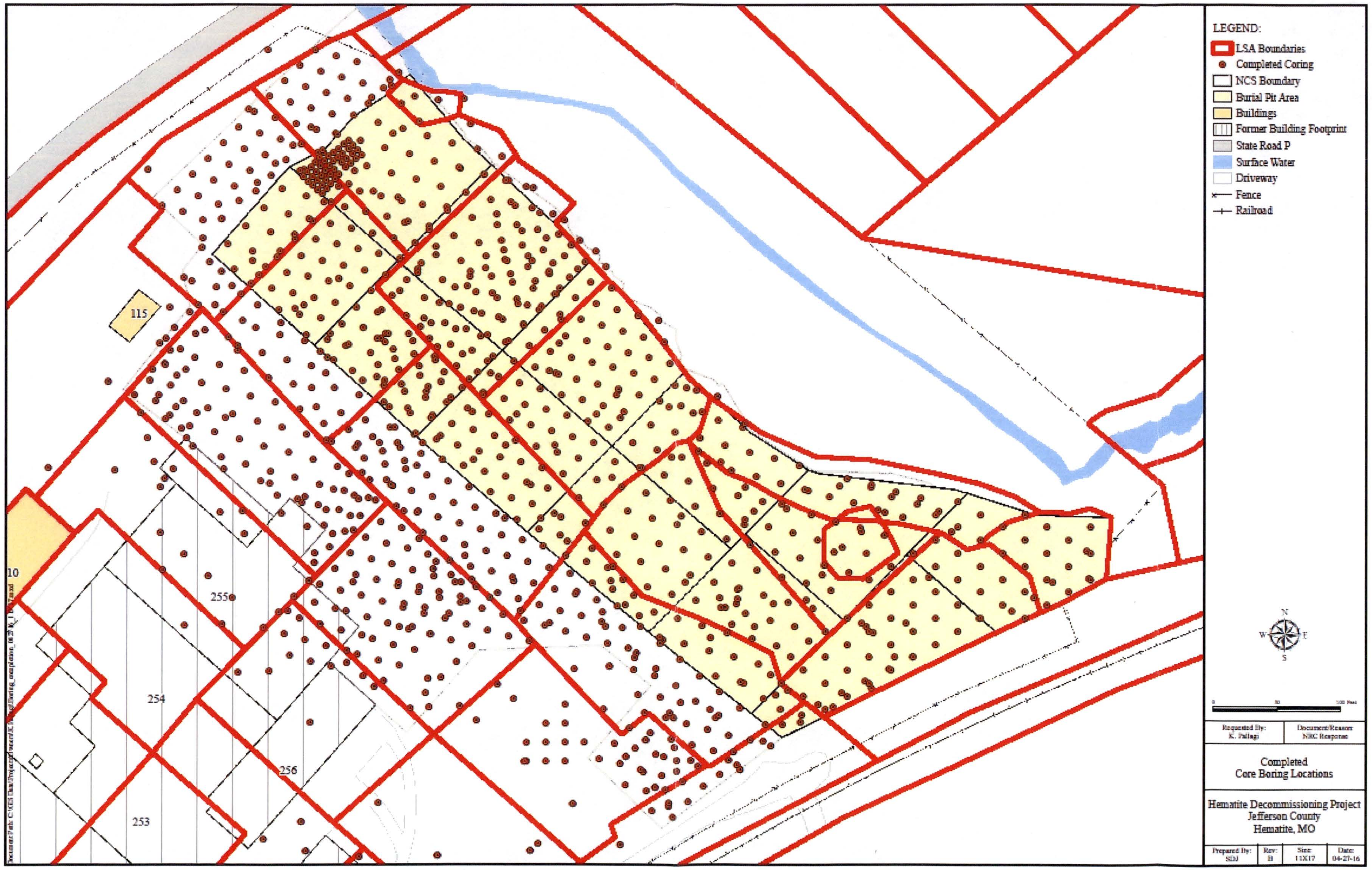


Figure 2-20

Surface and Sub-surface Soil Sampling Locations in the Burial Pit Area

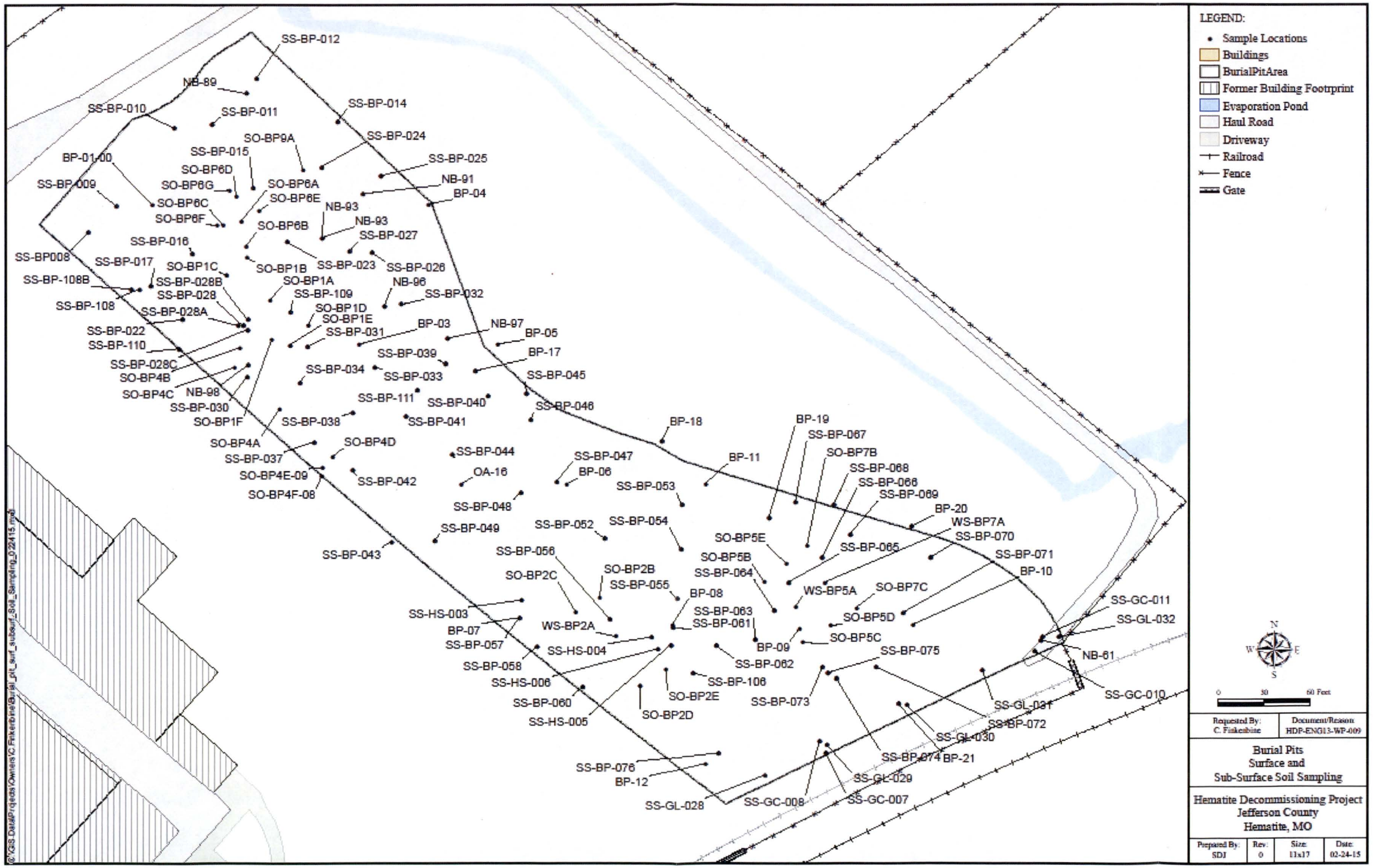


Figure 2-21
Excavated Elevations of the Burial Pit Area Post Remediation

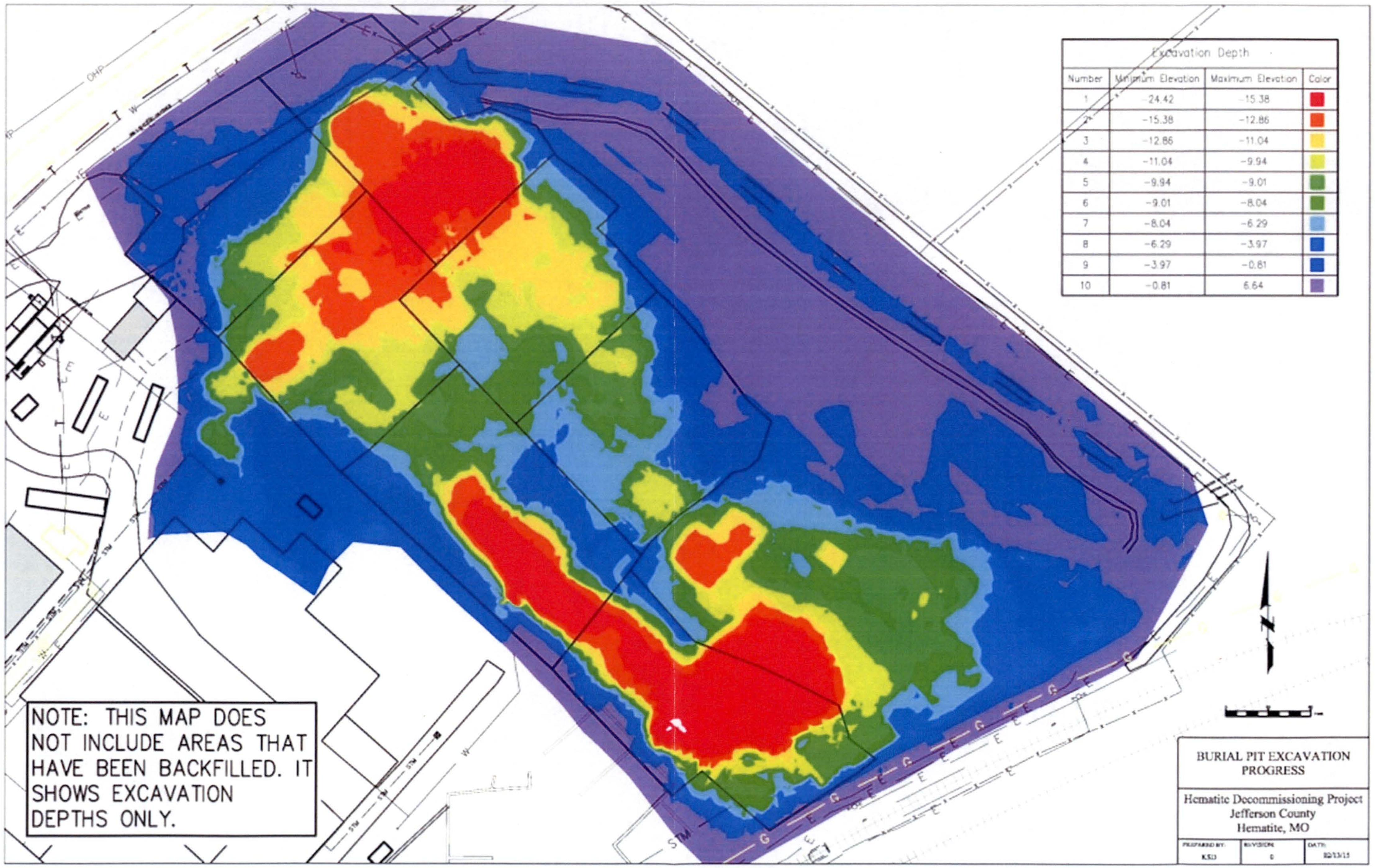


Figure 2-22

Documented Burial Pit Area Boundary and the Outer Boundary of the Area Defined By The Extent of the Undocumented Burials

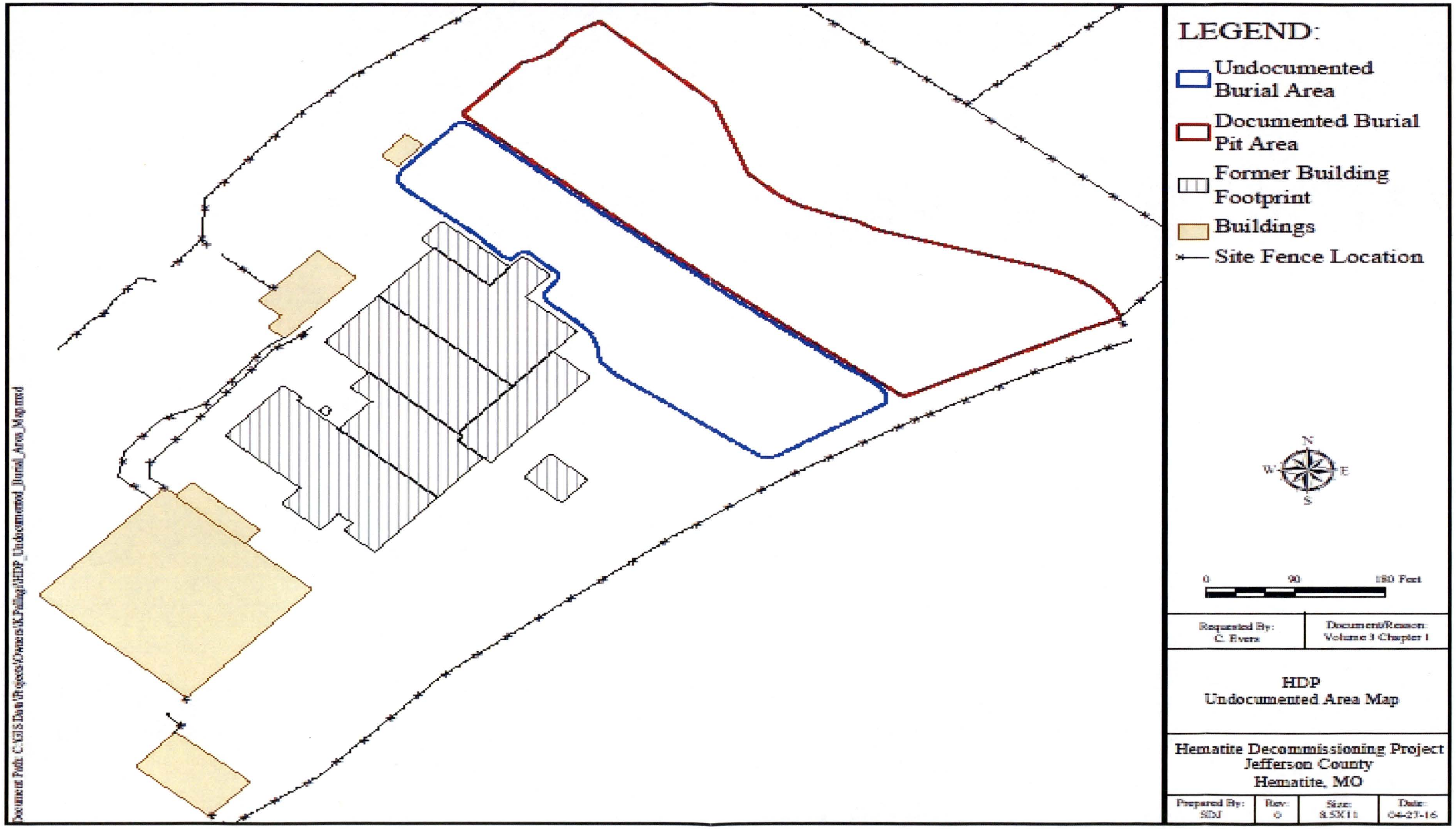


Figure 2-23
Location of Hematite Buildings Prior to Demolition Activities

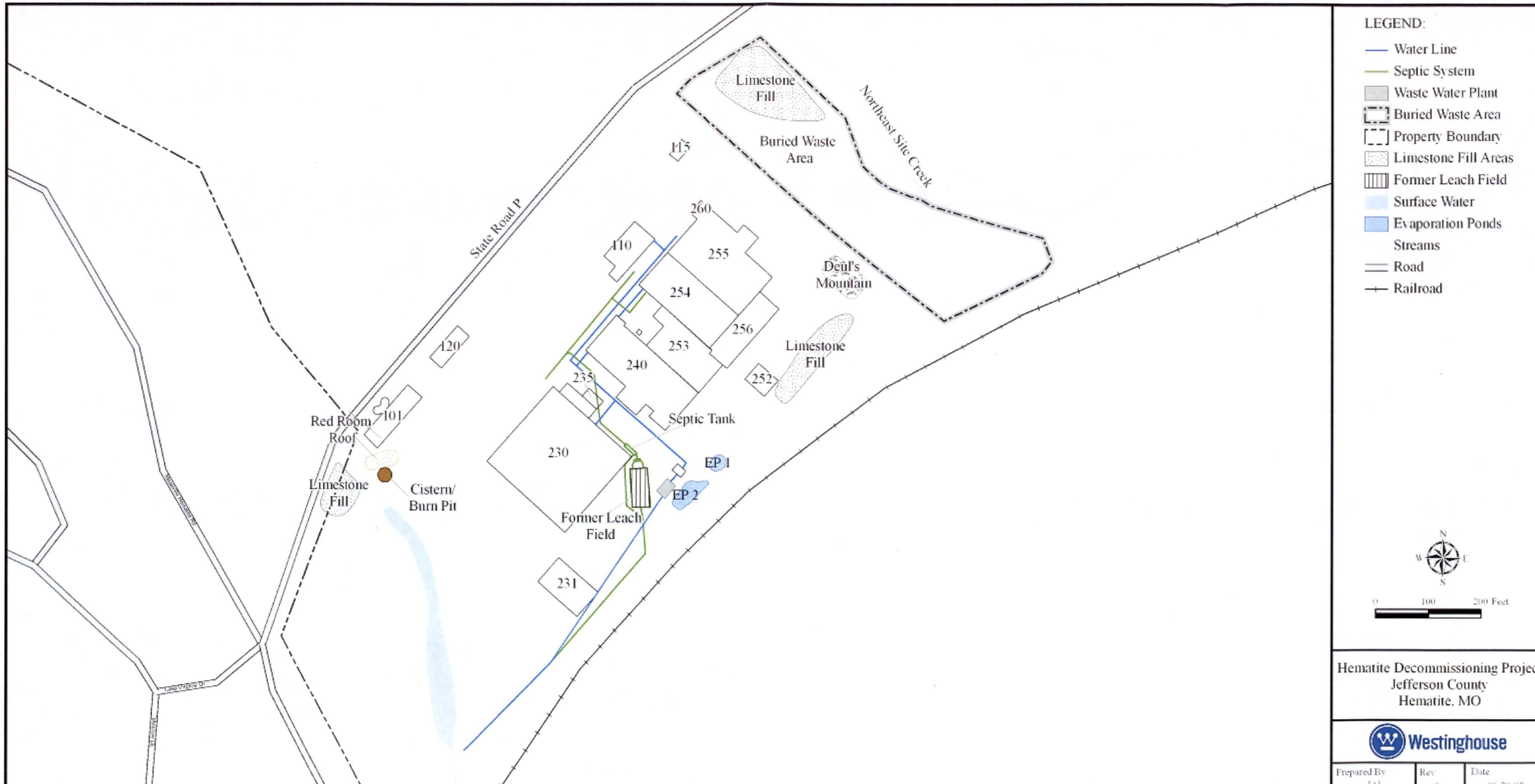


Figure 2-24
Hematite Circa 2006

Figures 2-25 through 2-36
Process Building Demolition Photographs

DELETED

(see FSSFR Volume 4, Chapter 1, Building Survey Areas Overview)

Figure 2-37
Process Building Slab with Process Building Removed - August 2011 (Looking East)

Figure 2-38

Process Building Area Showing the Remediation Removed All of the Subterranean Process Piping Except Those Portions of the Piping That Resided Under the Haul Road (Former Building Slab) (Looking NE)

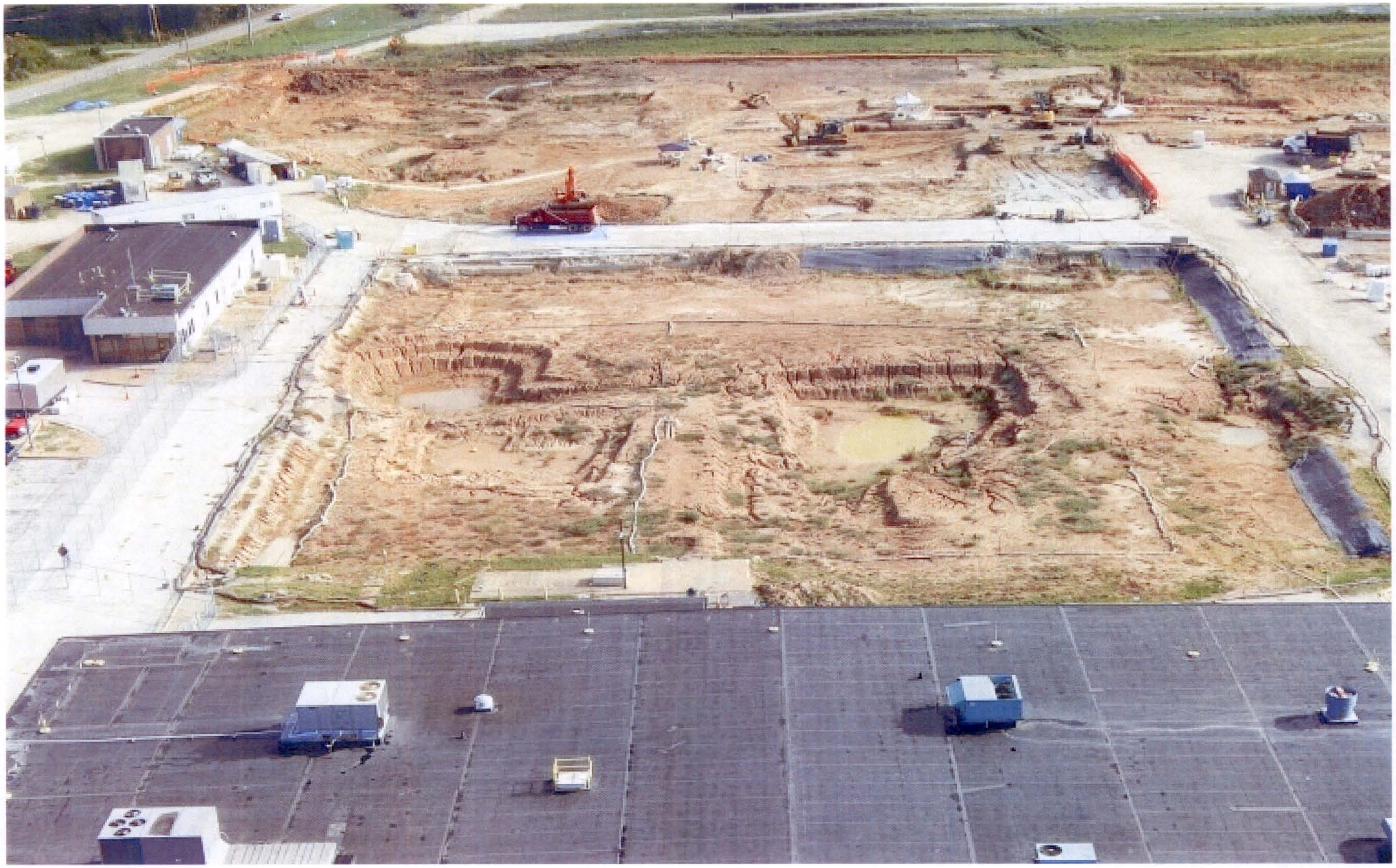


Figure 2-39

Photograph Taken On November 7, 2015, Showing the Removal and Remediation of the Haul Road (Former Building Slab) Along With Remaining Subterranean Process Piping (Looking NE)

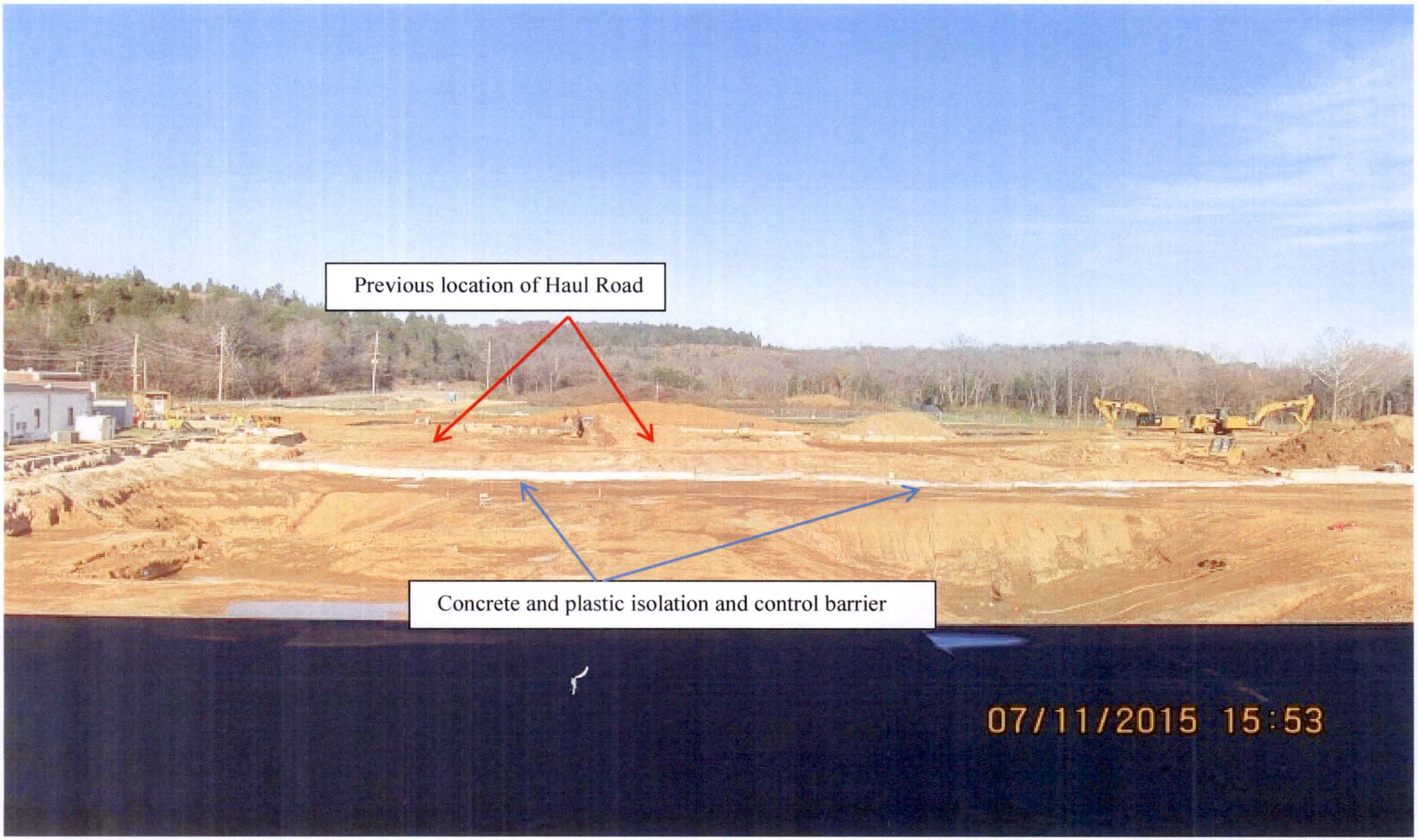


Figure 2-40

Evaporation Ponds (The Second Pond is Below the Pond Shown and Covered with Soil)
The Red Line Shows Where an Eight Inch Natural Gas Pipeline is Located

Figure 2-41

Evaporation Ponds (The Second Pond is Below the Pond Shown and Covered with Soil)
(Looking NE)

Figure 2-41a
Sampling Initiative for the Evaporation Ponds

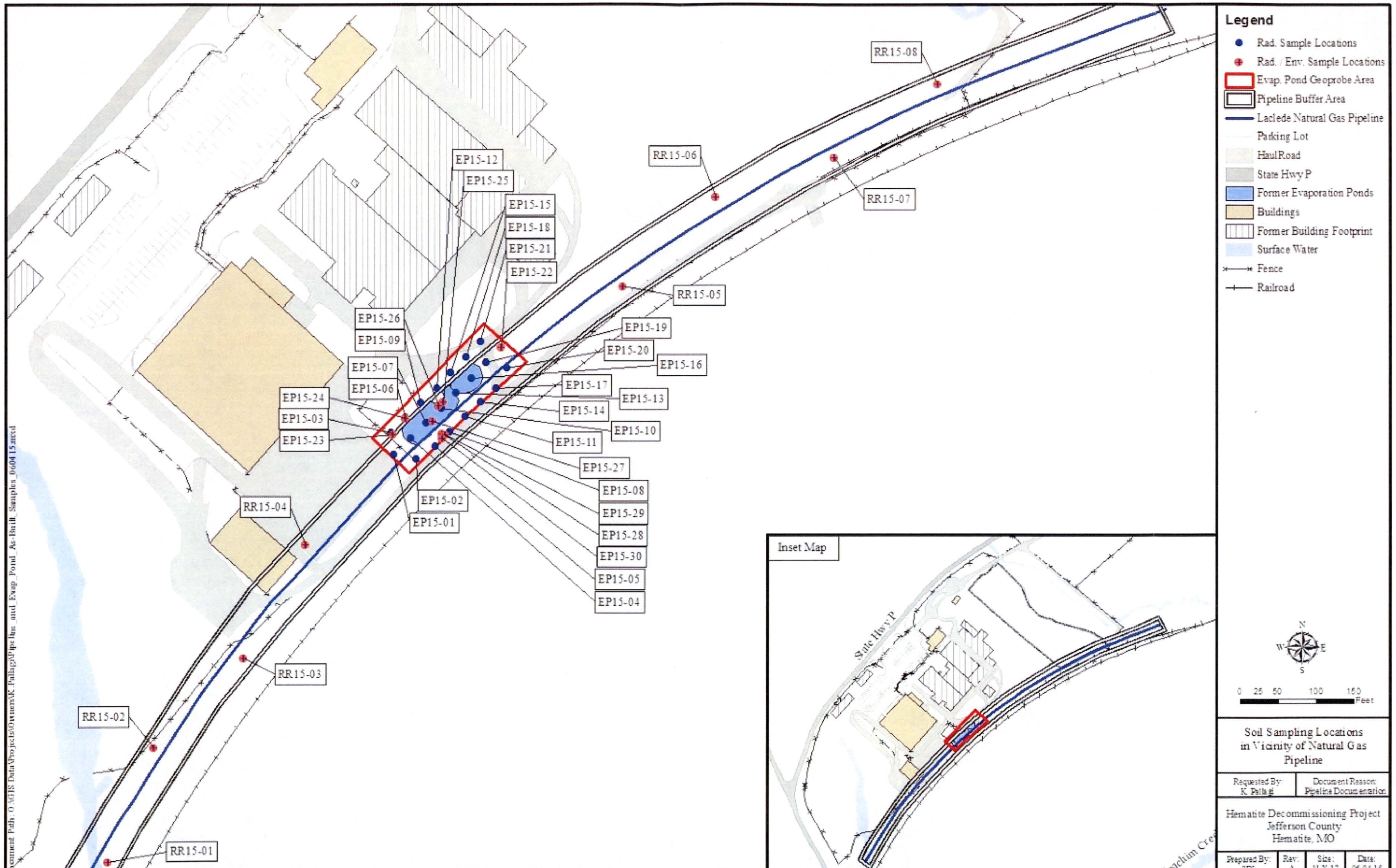


Figure 2-42
Excavation of the Evaporation Ponds (Looking SW)

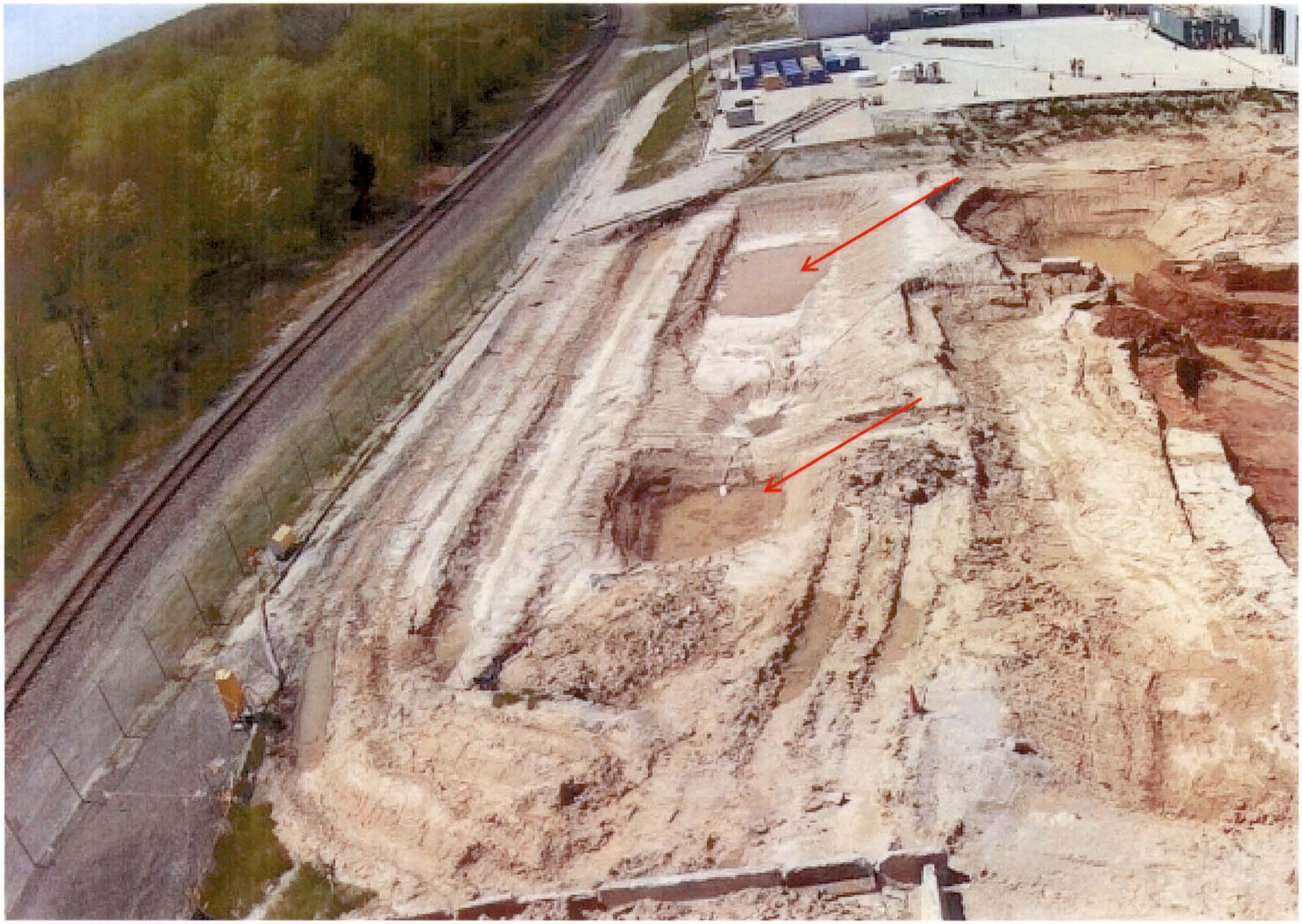


Figure 2-43
Natural Gas Pipline with Vertical Measurements Used to Determine It's Depth

Figure 2. Vertical Measurements Used to Determine the Depth of the NGP

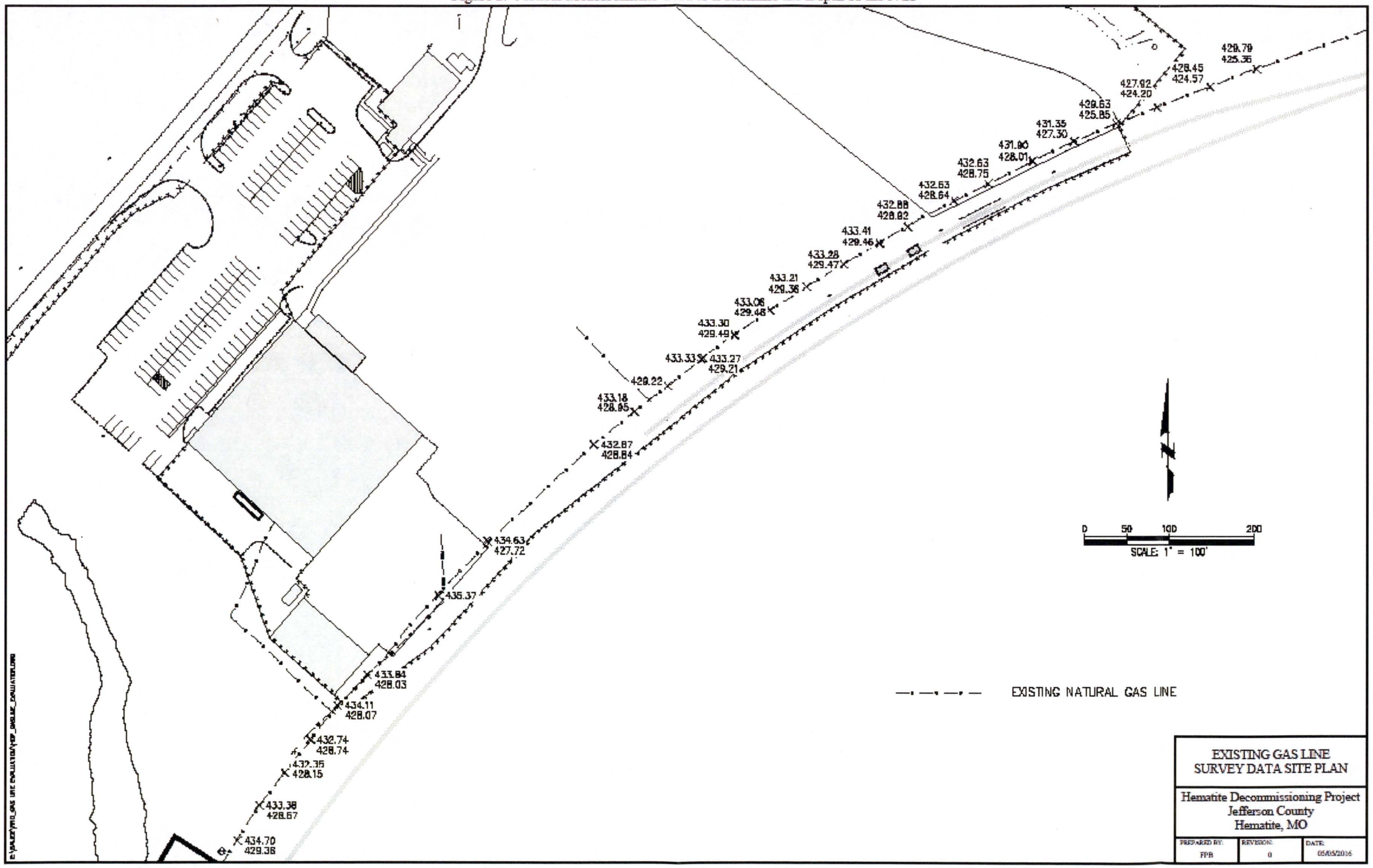


Figure 2-44

Remediation Activities at the Natural Gas Pipeline (Looking NE)

Figure 2-45

The Wood and Tile Barns Prior To Demolition (Looking W)

Figures 2-46 through 2-51
Barns Demolition Photographs

DELETED

(see FSSFR Volume 4, Chapter 1, Building Survey Areas Overview)

Figure 2-52
Barns Area Being Backfilled (Looking N)

Figure 2-53
February 2013 - Remediation of the Red Room Roof Burial/Cistern Burn Pit Area (Looking NE)

Figure 2-54

Concrete Slabs Covering the Location of the Septic Tank At the Southeast Corner of Building 230

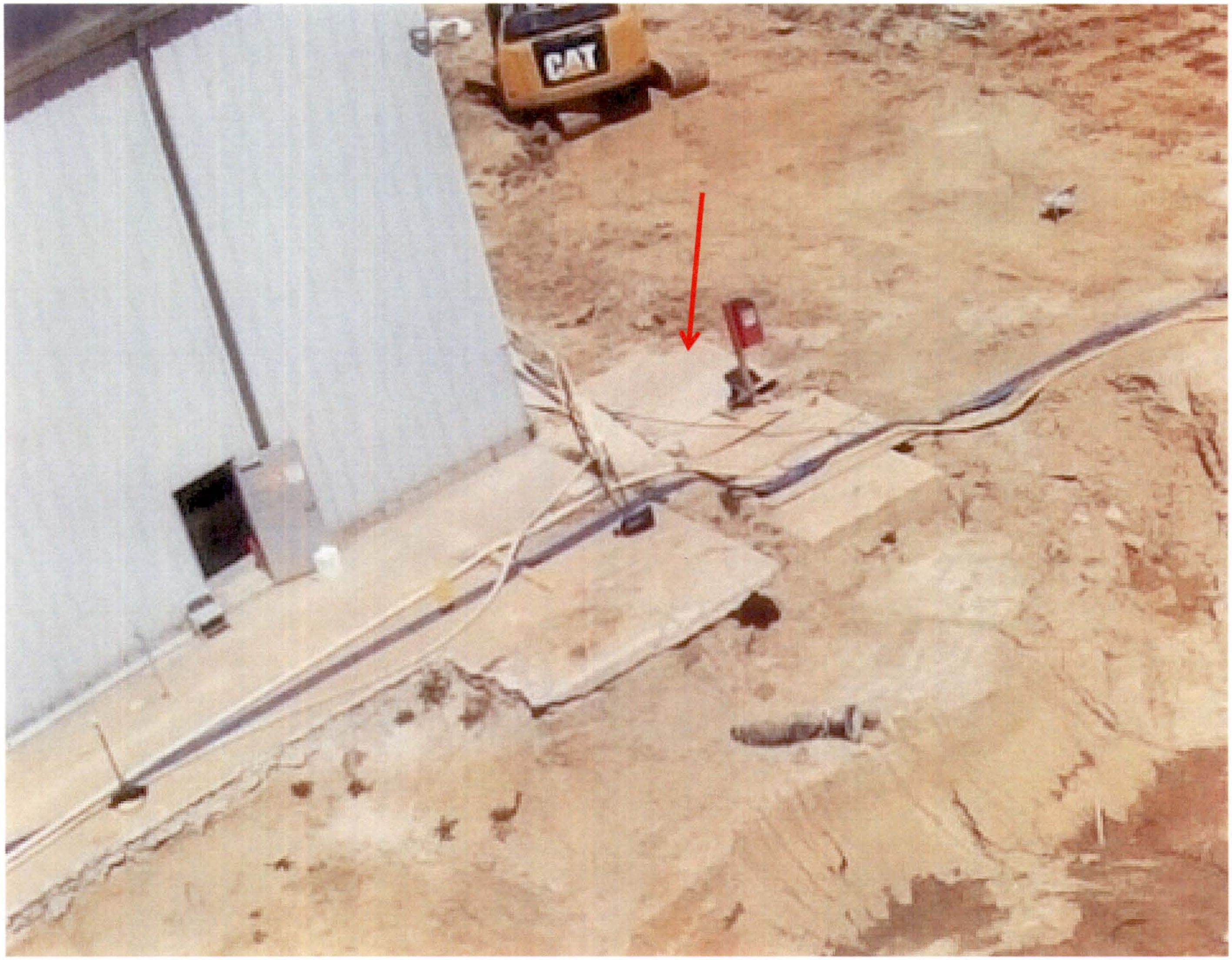


Figure 2-55

Interior of the Septic Tank

Figure 2-56
Excavation of Leach Field (Lower Right) (Looking W)

Figure 2-57
The SWTP Tanks Location Prior To Demolition and Removal (Looking NE)
(Between Building 230 to the Left and the Evaporation Ponds on the Right)

Figure 2-58

Sanitary Wastewater Tanks (Looking SW)

Figure 2-59

Excavation Remaining After Demolition and Removal of the SWTP Tanks

Figure 2-60

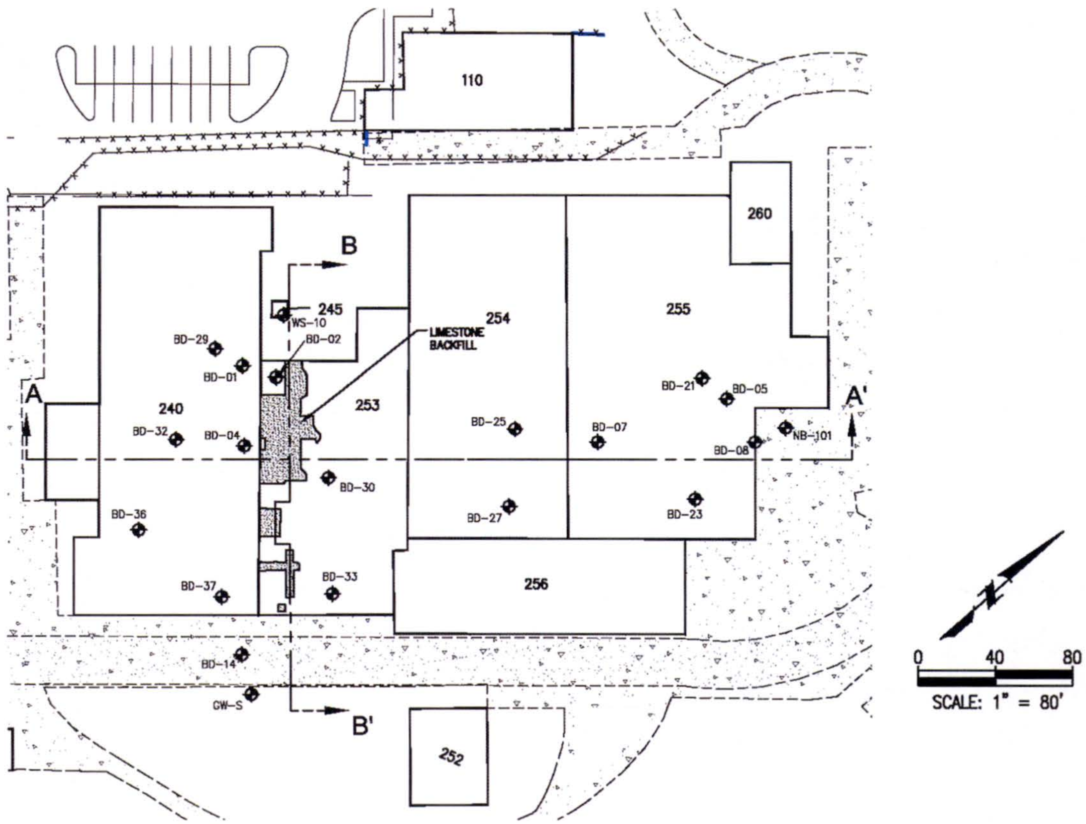
Site Pond Prior to Excavation and Remediation (Looking NE)

Figure 2-61
Installation of the Site Pond Diversion Ditch Which Was Constructed To Divert Water From The Site Spring Around And Downstream Of The Site Pond (Looking N)

Figure 2-62

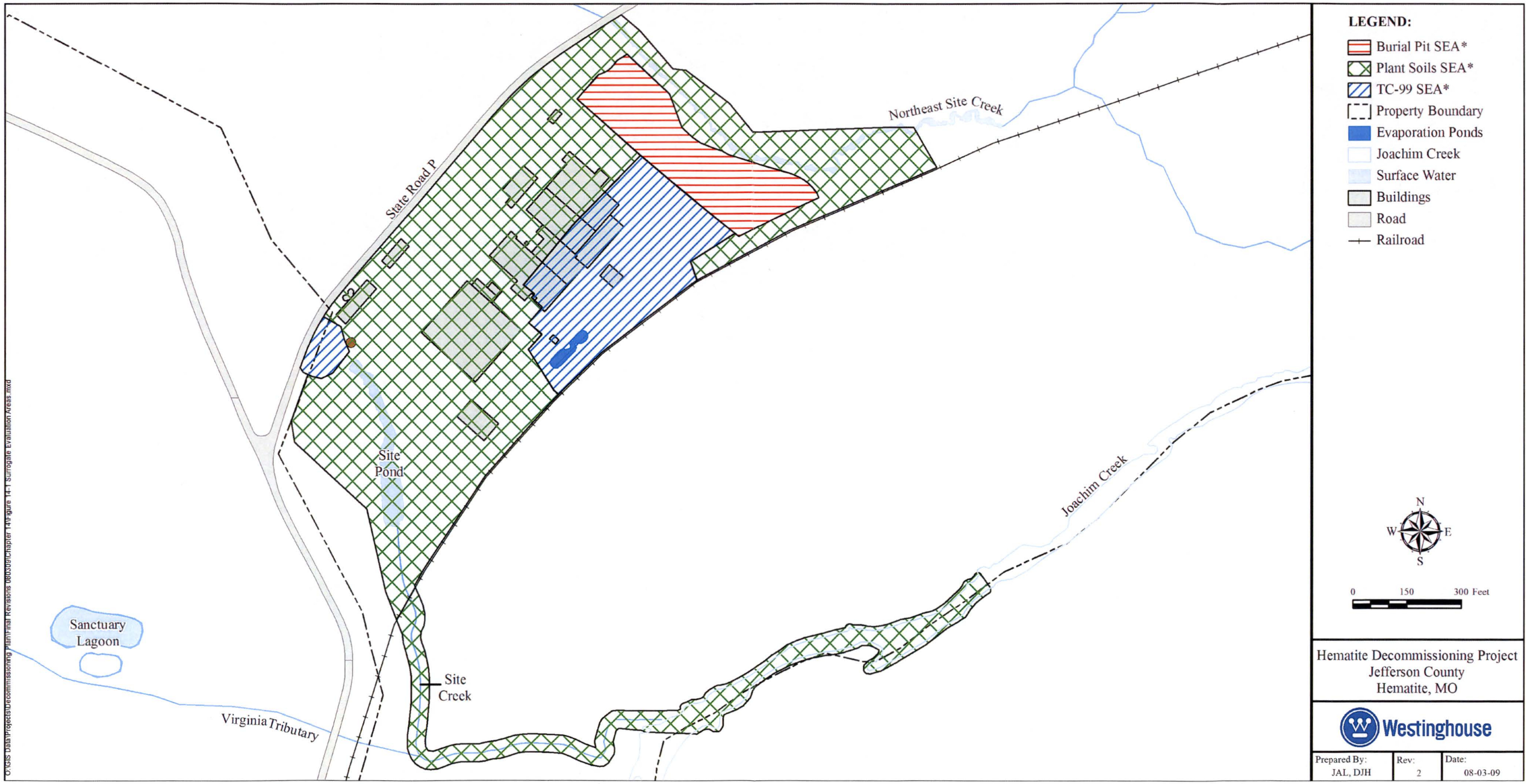
Site Pond Excavation and Remediation Nearing Completion (Looking S)

Figure 2-63
Location of Limestone Fill Under Building 253



SITE PLAN

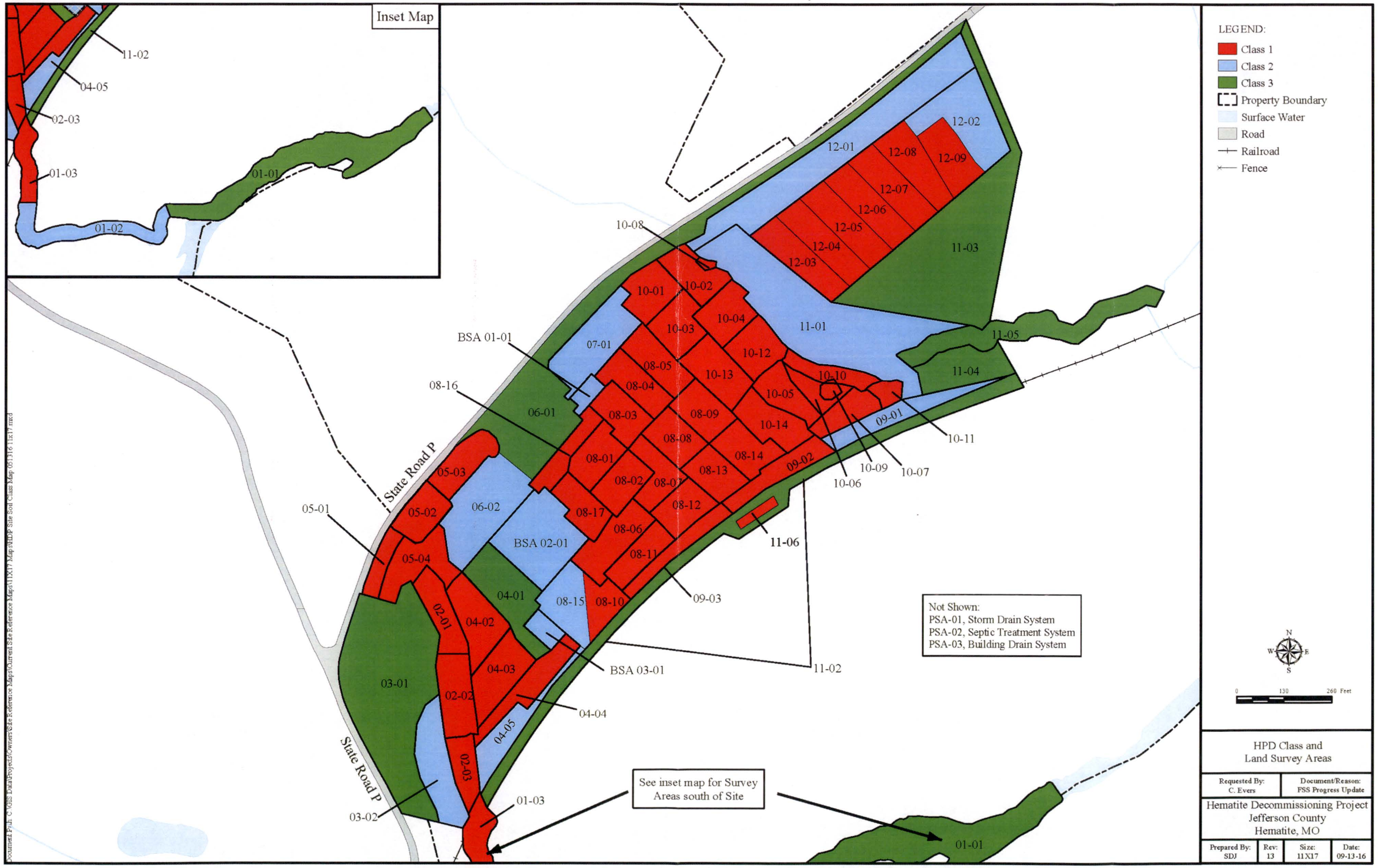
Figure 5-1
HDP Site Surrogate Evaluation Area Map



*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.

Appendix A
Final Configuration of HDP Land Survey Areas



Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PO-FSS-700
Final Status Survey Program

Revision Number	Effective Date	Summary of the Revision
0	01/16/2012	Initial issuance of the policy. Implements the requirements of the Final Status Survey plan contained within the DP.
1	12/04/2012	Updated policy to include soil sampling requirements during the abandonment of hybrid wells, per Section 14.4.3.4.2 of the DP.
2	01/31/2013	Updated Section 13, Final Status Survey Reporting to reflect the changes of Section 14.6 in the DP.
3	11/26/2014	Revision included clarifications and enhancements.
4	02/19/2015	Added information regarding scan MDCs and revised wording to be consistent with Westinghouse letter HEM-11-56.
5	10/28/2015	Changed policy to Westinghouse Proprietary Class 2. No technical changes.
6	04/07/2016	The revision incorporates a new section 15, "Surveillance Following FSS" as a component of a corrective action to a Notice of Violation. Clarifications have been made to section 9.7 regarding isolation and control.

HDP-PR-FSS-701
Final Status Survey Plan Development

Revision Number	Effective Date	Summary of the Revision
0	01/16/2012	Initial issuance of the procedure.
1	02/04/2013	Provided clarification on soil sampling by stratum as indicated in Decommissioning Plan Table 14-24.
2	02/12/2013	Provided additional instructions for creating FSS Plans for Reuse Soil.
3	11/26/2014	Significant revision for clarification and minor corrections.

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Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-701 Final Status Survey Plan Development		
Revision Number	**Effective Date**	**Summary of the Revision**
4	01/07/2015	Subsequent to comments received by NRC Region III regarding content of this procedure a technical readiness review was performed and the procedure revised accordingly.
5	02/11/2015	Updated the scan MDCs for U, Th-232 and Ra-226.
6	03/25/2015	Clarification of guidance for background ranges including acceptable ranges for use of a 10,000 cpm background for calculations and direction when background values are outside that range or survey parameters differ from those in HDP-TBD-FSS-002.
7	06/15/2015	Added information regarding piping survey plans, updated Ra-226 in-growth background value, clarified mean of SO equation, added direction on adjusting grid spacing to account for potential Tc-99 hotspots.
8	08/21/2015	The revision is initiated upon an agreement between the NRC and Westinghouse HDP in regards to Tc-99 sidewall sampling. The agreement was reached during a NRC Public Teleconference Meetings held on August 12, 2015, and August 19, 2015
9	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.
10	11/19/2015	Resolution and clarification of 100% GWS based on discussions with NRC Headquarters.

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HDP-PR-FSS-703
Final Status Survey Quality Control

Revision Number	Effective Date	Summary of the Revision
0	01/16/2012	Initial issuance of the procedure.
1	11/26/2014	Clarification for SSC Survey units.
2	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

HDP-PR-FSS-711
Final Status Surveys and Sampling of Soil and Sediment

Revision Number	Effective Date	Summary of the Revision
0	01/16/2012	Initial issuance of the procedure.
1	02/04/2013	Updated reference to HDP-PR-FSS-701.
2	02/13/2013	Typographical correction. The procedure still had "Draft A, Proposed 1" listed under the revision heading on the title page when it was placed onto SharePoint.
3	04/25/2013	Provided clarification when survey instructions cannot be followed as written.
4	11/26/2014	Added instructions for notifications when working in or near physical security structures and components.
5	02/09/2015	Added steps to provide further details on where to document background readings and how to apply the backgrounds during the survey.
6	04/15/2015	Added detail for the notification and direction in the event that isolation controls are breached in a survey unit.
7	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.
8	11/19/2015	Added clarification to Step 8.4.3 that a 100% GWS is required in Class 1 areas, and that professional judgment may be used to scan additional areas than the minimum requirements for Class 2 and Class 3 areas.

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HDP-PR-FSS-720 **Final Status Survey Data Integrity and Database Management**		
Revision Number	**Effective Date**	**Summary of the Revision**
0	09/13/2011	Initial issuance of the procedure.
1	01/13/2015	Revised section for Data Download to update the process for downloading data to a FSS computer that is password protected. Inclusion of a section to review and store GWS data. Addition of new Data Download Report Form. Changed back up of FSS data from being backed up as part of the routine site media backup process to being backed up on a password protected external hard drive and included a minimum frequency of weekly for hard drive backup.
2	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

HDP-PR-FSS-721 **Final Status Survey Data Evaluation**		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the procedure.
1	02/06/2013	Administrative revision only, forms changed to appendices.
2	02/20/2013	Administrative revision only, references updated.
3	11/26/2014	Provided clarification SOF, Corrected typographical error in SOF equation and other steps.
4	01/06/2015	Minor revision to insert procedure number in front of procedure title in Section 5.0 References.
5	02/09/2015	Clarifications on the calculation of statistics in Section 8.3 and SOF in Section 8.4, Additional guidance on completion of the WRS test and correct terminology used for the Test Statistic, Clarifications between the requirements for soil and structural survey units, added notes to clarify using the DCGLs used to develop the FSS plan unless instructed otherwise by the RSO, added step to Appendix G-1 to verify Laboratory quality control parameters are within acceptable

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-721 Final Status Survey Data Evaluation		
Revision Number	**Effective Date**	**Summary of the Revision**
		limits.
6	03/25/2015	Clarifications on the calculation of statistics to correct negative values to zero when listing basic statistical data and listing the background values. Replaced Appendix C with table 14-4 from Attachment 4 to HEM-11-96 as this table is more appropriate since inferred Tc-99 is prohibited for use to demonstrate compliance. Added a note to clarify that it is prohibited to use U-235 to infer Tc-99 to demonstrate compliance. Removed references to SEA in the procedure and in the acronym list. Various terminology corrections. Added the calculation to use for obtaining a U-234 value when not measured for the calculation. Changed scan action level for Class 1 to be consistent with change to HDP-PR-FSS-701. Added underground piping.
7	04/15/2015	Added detail in Step 7.5.4 as to notification and direction in the event that isolation controls are breached in a survey unit.
8	06/15/2015	Updated the Ra-226 soil background value. Added instruction for use of other DCGL conceptual site models rather than only the uniform DCGL.
9	08/13/2015	This revision implements clarification on the determination of dose for survey units. The changes are based upon an agreement between the U.S. NRC and Westinghouse in regards to calculating dose. The technical changes were agreed upon during the NRC-Westinghouse teleconference held on August 12, 2015.
10	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

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HDP-PR-FSS-722
Final Status Survey Reporting

Revision Number	Effective Date	Summary of the Revision
0	09/13/2011	Initial issuance of the procedure.
1	02/05/2015	Updated to reflect changes to HDP-PO-FFS -700 regarding FSS reporting.
2	01/31/2014	Removed the formatting guidelines which were contained in Appendix A and B.
3	01/13/2015	Editorial corrections.
4	11/06/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

HDP-PR-HP-416
Operation of the Ludlum 2221 for Final Status Survey

Revision Number	Effective Date	Summary of the Revision
0	01/16/2012	Initial issuance of the procedure.
1	02/09/2015	Clarified instruction that the probe be kept as close to the ground as possible, nominally 1 inch from the surface and not to exceed 3 inches. Clarified instruction to scan at 1 foot per second and that head phones should be used when it is noisy and interferes with hearing audible count rate. Revised wording consistent with current wording in FSS instructions. Added a step to stop and pause when elevated count rates are identified to determine if further investigation is needed. Added clarification that documentation using HDP-PR-FSS-701 is following directions as specified in the FSS instructions.
2	11/19/2015	Administrative changes. Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
0	03/14/2012	Initial issuance of the procedure.
1	04/09/2012	Updated procedure to accommodate performing 6 inch excavation lifts instead of 12 inch excavation lifts, updated Table 1 Column B screening count rate to 46k cpm for saturated soil, which is more conservative than the 58k cpm and is associated with a lump of material, Updated flow chart in Appendix B to move evaluation of items after the initial survey, and to include pipe surveys, Section 8.1.6.2 updated to identify items for further evaluation and evaluate them after the initial survey to determine if they are NCS Exempt. Clarifications for the count rates are less than or equal to (≤) instead of "less than", consistent with wording in Section 8.7 of this procedure and the applicable NCSA, removed "general area" to clarify that the background count rate used will be obtained from the daily background check. Added reference to NSA-TR-HDP-11-06 for surveys of pipe and added section for surveying piping to determine if it is NCS Exempt. Specified that the screening level for reuse is on Form HDP-PR-HP-100-2.
2	06/11/2012	Administrative changes and clarifications.
3	06/26/2012	Updated procedure to use the general area background count rate when performing surveys or items and containers and Sections 8.4.3 and 8.7 to specify the external surfaces of containers need to be surveyed, not only the bottom of the containers.
4	07/05/2012	Updated Section 8.2 to clarify that the radiological surveys do not need to be formally documented prior to determining if an area meets NCS Exempt criteria and can be excavated. Additionally, 2 independent HP Technicians can verify all hotspots/items have been evaluated and handled required to ensure the surveyed area meets NCS Exempt Material criteria. Updated Section 8.4.3 to remove the requirement to have intact containers in a vertical position when surveying them because this is not required as all external surfaces of

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Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		the container are surveyed. Updated Section 8.7 to clarify the criteria for Field Containers with net count rates between 300,000 and 1,400,000 cpm.
5	08/14/2012	The requirement to identify areas that are NCS Exempt for a 12 inch cut depth (i.e., areas that are 19k ≤ ncpm < 66k dry soil, marked as blue) was removed because the maximum permitted cut depth will be 6 inches. Blue will no longer be used to mark areas as it is no longer applicable since it is associated with the NCS Exempt criteria for a 12 inch cut depth. This change includes the removal of Column A in Table 1. Updated the count rates in Table 1 for Columns C & D, and in associated locations throughout the procedure. These count rates were previously based on a cut depth of 7 inches (an extra 1 inch greater than the cut depth for conservatism). The count rates were adjusted in this revision to the count rates associated with a cut depth of 6 inches. Based on operational experience and the current maximum permitted cut depth (i.e., 6 inches), an extra 1 inch of conservatism is not necessary. Updated the NCS Field Container Loading Limit from 1,400k ncpm to 1,500k ncpm. 1,400 ncpm was based on 102 g235U, assuming up to 3 field containers in a collared drum (total mass in a collared drum <350 g235U). Since only 2 field containers can fit into a collared drum, the count rate was increased to 1,500k based on 150 g 235U.
6	08/23/2012	Corrected misnumbered step references.
7	09/06/2012	Added note to include the segregation of material with substantially differing characteristics (i.e., radiological, media type) when loading a field container.
8	12/03/2012	NSA-TR-09-15 Revision 4 was issued. Changes to the CSCs in Revision 4 were incorporated throughout the procedure. Sections 7.12 and 8.9 were revised to include instructions on assigning a 235U gram quantity based on close proximity radiological survey results.
9	03/07/2013	This was a significant revision which incorporated nuclear criticality information and requirements based upon NSA-

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		TR-09-08, NCS of Subsurface Structure Decommissioning at Hematite Site and NSA-TR-11-11, NCS Assessment of the USEI Site for Landfill disposal of Additional Decommissioning Waste from Hematite Site.
10	04/08/2013	Added to Step 8.1.7.2 "NOTE: Sections of subterranean pipe less than the NCS Exempt Piping Limit may be crushed and treated as soils, in accordance with HDP-PR-HP-607 (Reference 5.18); however, depending on the size, thickness and material type of the subterranean piping pieces (i.e., Bulky pieces with linear dimensions exceeding the permitted cut depth, and thick metallic items) should be treated as Non-Conforming Items. (NSA-TR-09-08 CSC 11)".
11	05/01/2013	Instructions were added for evaluating hot spots using the LaBr3 probe.
12	05/22/2013	A clarification was made in regards to evaluating intact containers.
13	06/21/2013	Added instruction to ensure Pre and Post Assay QC checks are performed in accordance with HDP-PR-NC-008, Probe, when the LaBr3 probe is used to determine if an area meets NCS Exempt requirements.
14	07/22/2013	Added instruction to allow for Ex-situ scanning of soil in less than 6 inch layers. Added instruction for surveying material underneath the Former Building 240 Slab to allow excavation of material up to 0.8 g235U/L as NCS Exempt.
15	07/25/2013	Instructions added for performing measurements using a NaI 2x2 detector with a "Closed Window" setting of 75 to 250 keV and added option to survey a greater than 5L container to less than or equal to 34k to determine if it meets NCS exempt criteria.
16	08/07/2013	Revised to incorporate the requirements of HEM-13-MEMO-067, Remediation of Soil Under the Concrete Slabs.
17	08/15/2013	Revised to allow the survey of piping from discovered Burial

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		Pits to be performed in one foot increments.
18	09/11/2013	Redundant FSS design requirements were removed from as they are captured in FFS procedure. Introduced the terms "Large Volume Container" and "Small Volume Container" throughout procedure. Removed all instructions for Inspector 1000 with LaBr3 probe as it will no longer be used to evaluate "hot spots" as part of the initial radiological survey. Extensive changes were made to this procedure to remove directions on performing measurements using a NaI 2x2 detector with a "Closed Window" setting of 75 to 250 keV. Steps were added to Section 8.5 to satisfy expanded NCS Controls to perform a visual inspection of Ex-situ excavated material in layers prior to consolidation at the WCA.
19	09/12/2013	Step 8.5.2 was revised to clarify the requirement of NSA-TR-09-15 CSC 25, to "Identify and demarcate an area, using paint, flags, or other appropriate means, adjacent to the excavation area that does not contain any Non-NCS Exempt Material (identified in accordance with Section 8.1)".
20	09/30/2013	The term Inspection Items was introduced. References to the use of GPS in section 8.1 were removed since GPS use is considered optional. References to DCGL were replaced with Reuse Material Screening Level (RML) and a count rate of 12k ncpm is now specified. Section 8.4 was revised to include detailed direction on how intact containers are to be emptied in the field. Section 8.9 title was changed to "Surveys of Non-Conforming Items or Items Potentially Containing Fissile Material"
21	10/01/2013	Minor typographical errors were corrected from the previous revision.
22	10/23/2013	Step added to "Identify the area to be excavated" prior to excavation of an area. Added that the adjacent working surface should be demarcated with pink paint or flags. Step was revised to specify that a GWS with Visual Inspection

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		will be performed of the ex-situ soil layer regardless if visible Non- Conforming Items were identified.
23	11/22/2013	Added instruction for remediation of soils surrounding radium filter press plates in the Burial Pits.
24	01/10/2014	Revised the scope of Section 8.10 to apply to all pavement and building slabs covering soil under NCS controls. Added saturated soil numbers to Sections 8.11 and 8.12. Added step to address the use of ISOCS measurements performed in the field as part of evaluation of radium contaminated soil areas.
25	05/19/2014	Incorporated the requirements of Memo HEM-13-MEMO-099, Radiological Requirements for the Handling of Re-Use Soils During Development of Stockpile 8. Added the requirements of NSA-TR-09-08 CSC 17, and CSC 20 and NSATR- HDP-11-11 CSC 15 related to septic and sewage treatment tanks. Removed references to remediation contractor.
26	06/02/2014	Updated work package references throughout procedure, as work package numbers have changed.
27	07/10/2014	Added potential NCS Evaluation for Non-Conforming Items which are > 0.1 g U-235/L to Appendix C.
28	08/11/2014	Clarified the requirements for excavation cut depth outside of NCS controlled areas. Clarified how "areas where buried debris has been identified" is defined by NCS.
29	09/18/2014	Added instruction to allow the third visual inspection and processing of potentially NCS exempt material to take place on a working surface that will no longer be required to be located adjacent to the excavation area.
30	11/05/2014	A step was added to provide guidance when performing surveys and visual inspections under artificial lighting.
31	11/06/2014	Administrative changes.

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Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
32	12/05/2014	Added instruction to implement the enhanced work controls developed to ensure compliance with the USEI WAC when excavation in the Site Pond. Added Reference to HEM-14-MEMO-112, NCS Analysis of the HDP Site Pond Remediation Activities.
33	02/04/2015	Added instruction on the direct loading of waste for disposal at a NRC licensed burial facility up to a concentration of 0.8 g235U/L.
34	11/19/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

Appendix C
HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-01 South Site Waterways					
LSA-01-01	Site Creek/Joachim Creek	3	12/16/15	20	
LSA-01-02	South Section of Site Creek	2	1/14/16	20	
LSA-01-03	North Section of Site Creek	1	1/14/16	20	
LSA-02 Site Pond					
LSA-02-01	North Section of Site Pond	1	9/29/15	20	
LSA-02-02	Central Section of Site Pond	1	9/29/15	20	
LSA-02-03	South Section of Site Pond	1	9/29/15	20	
LSA-03 West Open Land Area					
LSA-03-01	Area West of Site Pond	3	11/6/15	20	
LSA-03-02	Area Southwest of Site Pond	2	11/12/15	20	
LSA-04 Southwest Open Land Area					
LSA-04-01	Area between Buildings 230/231 and Site Pond	3	4/5/16	15	
LSA-04-02	Area East of North Section of Site Pond (west soil laydown area)	1	4/22/16	15	
LSA-04-03	Area East of Central Section of Site Pond (west soil laydown area)	1	4/21/16	15	
LSA-04-04	Area South of Building 231	1	4/12/16	15	
LSA-04-05	Wooded Area South of Building 231	3	6/21/16	15	
LSA-05 Barns and Cistern Open Land Area					
LSA-05-01	Site Spring Area adjacent to State Road P	1	3/9/14	16	
LSA-05-02	Tile Barn and Red Room Roof	1	9/13/13	16	

Appendix C
HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-05-03	Wood Barn	1	11/7/13	16	
LSA-05-04	Site Spring and Cistern	1	4/27/16	16	
LSA-06 North Open Land Area					
LSA-06-01	Main Parking Lot	3	6/24/16	17	
LSA-06-02	West Parking Lot	2	6/17/16	17	
LSA-07 North Central Open Land Area					
LSA-07-01	Truck Scale Area	2	5/3/16	17	
LSA-08 Central Open Land Area					
LSA-08-01	Process Building Area Section 1	1	3/15/16	12	
LSA-08-02	Process Building Area Section 2	1	2/4/16	12	
LSA-08-03	Process Building Area Section 3	1	3/18/16	11	
LSA-08-04	Process Building Area Section 4	1	4/7/16	10	
LSA-08-05	Process Building Area Section 5	1	4/12/16	10	
LSA-08-06	Process Building Area Section 6	1	1/6/16	11	
LSA-08-07	Process Building Area Section 7	1	1/7/16	11	
LSA-08-08	Process Building Area Section 8	1	4/7/16	10	
LSA-08-09	Process Building Area Section 9	1	4/28/16	12	
LSA-08-10	Process Building Area Section 10	1	7/13/16	14	
LSA-08-11	Process Building Area Section 11	1	12/16/15	13	
LSA-08-12	Process Building Area Section 12	1	4/22/16	12	
LSA-08-13	Process Building Area Section 13	1	4/21/16	12	
LSA-08-14	Process Building Area Section 14	1	5/24/16	10	
LSA-08-15	Process Building Area Section 15	1	7/18/16	14	
LSA-08-16	Process Building Area Section 16	1	3/17/16	11	
LSA-08-17	Process Building Area Section 17	1	1/19/16	11	

Appendix C
HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-09 Rail Spur open Land Area					
LSA-09-01	East Rail Spur Area	2	5/24/16	18	
LSA-09-02	Central Rail Spur Area	1	7/5/16	18	
LSA-09-03	West Rail Spur Area	1	5/24/16	18	
LSA-10 Burial Pits Open Land Area					
LSA-10-01	Burial Pit Area Section 1	1	6/17/15	2	10/27/2016
LSA-10-02	Burial Pit Area Section 2	1	6/17/15	2	10/27/2016
LSA-10-03	Burial Pit Area Section 3	1	6/17/15	3	11/07/2016
LSA-10-04	Burial Pit Area Section 4	1	6/17/15	3	11/07/2016
LSA-10-05	Burial Pit Area Section 5	1	2/13/14	6	
LSA-10-06	Burial Pit Area Section 6	1	1/10/14	6	
LSA-10-07	Burial Pit Area Section 7	1	1/10/14	6	
LSA-10-08	Burial Pit Area Section 8	1	9/2/15	6	
LSA-10-09	Burial Pit Area Section 9	1	10/21/13	6	
LSA-10-10	Burial Pit Area Section 10	1	2/20/14	6	
LSA-10-11	Burial Pit Area Section 11	1	5/21/15	7	12/12/2016
LSA-10-12	Burial Pit Area Section 12	1	6/17/15	4	11/14/2016
LSA-10-13	Burial Pit Area Section 13	1	6/10/15	5	11/16/2016
LSA-10-14	Burial Pit Area Section 14	1	6/10/15	5	11/16/2016
LSA-11 East Open Land Area					
LSA-11-01	Northeast Site Creek	2	10/29/15	7	12/12/2016
LSA-11-02	Rail Road Line	3	7/5/16	19	
LSA-11-03	East Site Wooded Area	3	6/24/15	19	
LSA-11-04	Small East Site Wooded area	3	4/24/15	19	
LSA-11-05	Northeast Site Creek East Section	3	6/3/15	19	
LSA-11-06	Rail Road Line Elevated Area	1	7/5/16	19	

Appendix C
HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-12 Lay Down Area					
LSA-12-01	Reuse Soil Laydown Area Section 1	2	7/13/16	8	01/05/2016
LSA-12-02	Reuse Soil Laydown Area Section 2	2	7/13/16	8	01/05/2016
LSA-12-03	Reuse Soil Laydown Area Section 3	1	7/12/16	9	01/05/2016
LSA-12-04	Reuse Soil Laydown Area Section 4	1	7/12/16	9	01/05/2016
LSA-12-05	Reuse Soil Laydown Area Section 5	1	7/12/16	9	01/05/2016
LSA-12-06	Reuse Soil Laydown Area Section 6	1	7/12/16	9	01/05/2016
LSA-12-07	Reuse Soil Laydown Area Section 7	1	7/12/16	9	01/05/2016
LSA-12-08	Reuse Soil Laydown Area Section 8	1	7/12/16	9	01/05/2016
LSA-12-09	Reuse Soil Laydown Area Section 9	1	7/12/16	9	01/05/2016

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INTER-OFFICE MEMORANDUM

Date: March 2, 2015 HEM-15-MEMO-021

To: Brian Miller, FSS Manager
Ellen Jakub, FSS Data Manager

From: W. Clark Evers, Radiation Safety Officer

Cc: Steven Grice, FSS Task Manager

Subject: Evaluation of the Scan Investigation Action Level (IAL) for Class 1 areas at the Westinghouse Hematite Site

Class 1 areas at the Hematite site are impacted areas requiring remediation where historic sampling, characterization, or remediation surveys have detected contamination above the applicable DCGLw. Therefore radioactivity above background levels may be detected during the performance of Final Status Surveys (FSS) after all necessary remediation has been completed. It is for this reason that a Scan IAL is provided as part of the FSS plan. The IAL provides the HP Technicians performing field work a count rate above general area background that may indicate soils concentrations approaching the applicable DCGLw that should be investigated further.

Attempting to calculate the Scan IAL using theoretical values poses significant challenges since the Sum of Fractions (SOF) must be calculated based on several different isotopes, all with different physical characteristics, specific activities, and DCGLw values. For this reason a review of empirical data was performed.

To determine what an appropriate Scan IAL is for Class 1 areas of the site, all Final Remedial Action Support Surveys (RASS) of "Area 1" (i.e. LSA's 10-01, 10-02, 10-03, 10-04, and 10-12) were compiled calculating the mean count rate and standard deviation of the data set. All 5 of these LSA's are Class 1 areas where remediation was necessary and had proceeded until completion indicated that the areas were ready for FSS. Additionally all 5 of these areas did at one time contain contamination from all 6 of the contaminants of concern (i.e. U-234, U-235, and U-238, Ra-226, Th-232, and Tc-99), and were representative of a typical Class 1 area on the

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Hematite Site. The mean count rate of the population was calculated to be 10,698 cpm, with a standard deviation of 1616 cpm. This is consistent with the average general area background observed on the Hematite site of approximately 10,000 cpm as reported in the Hematite Radiological Characterization Report (HRCR). The Gamma Walkover Survey (GWS) data for Area 1 is attached in Figure 1.

MARSSIM Chapter 5.5.2.6 Determining Investigation Levels describes the industry standard practice of flagging areas for investigation that exceed background levels by a value of 3 sigma. For "Area 1" this would represent a count rate of 4848 cpm above general area background. For ease of use in the field, and to provide some inherent conservatism, this value will be rounded down to 4,000 net cpm.

While count rates observed in the field above the Scan IAL of 4,000 ncpm could be an indication of soil concentrations approaching the DCGL, it could also be an indication of a change in soil surface features (e.g. sidewall or trench) or fluctuating background. It is for this reason that when the HP Technician identifies areas above or approaching the Scan IAL in the performance of FSS that site procedures instruct the HP Technician to pause and investigate the area further. If the HP Technician determines based on professional judgment that the area does in fact exceed 4,000 net cpm above the general area background then the area should be "flagged" for further investigation. Chapter 14.3.2 of the Hematite Decommissioning Plan states "The average net count rate corresponding to the DCGLw 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". Therefore the Scan IAL of 4,000 ncpm was compared to the biased sample results that were collected during Final RASS of "Area 1".

LSA 10-02 provides a good example of a Land Survey Area that presents a favorable survey geometry (e.g. area is relatively flat, lacking sharp side walls), so soil areas with elevated count rates could be sampled for comparison, and have no concern for a "low bias" (e.g. sampling clean soil area due to unfavorable counting geometry). As can be seen in Figure 1 there are three distinct soil areas that are elevated above 3 sigma from the mean (circled in Figure 1), and one biased soil sample was collected at each of these locations. The results were compared to the Uniform DCGLw using the Tc-99 surrogate value for U-235 since Tc-99 has no gamma emitter and cannot be readily identified using field scanning techniques. To be consistent with FSS data analysis, values less than the minimum detectable activity (MDA) were set to zero, U-234 was calculated based on enrichment, and the 0.9 pCi/g Ra-226 and 1.0 pCi/g Th-232 background values were subtracted. The biased sample results are presented in Table 1, the maximum sample SOF was 0.44 of the Uniform DCGLw using the conservative Tc-99 surrogate value.

Lastly, consideration should be given to the potential presence of discrete material. While the Uniform DCGLw for soil is based on diffuse contamination, discrete contamination (e.g. radioactive debris or pellet fragments) must also be identified and removed to the maximum extent practical. A previous FSS survey was performed in an area where a fragment of a

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Uranium fuel pellet was identified. This event was documented in CAPS Item # 13-155-W008.01, and it was identified in the findings of the CAPS investigation that small discrete items are difficult to detect when buried beyond 3 inches beneath the soils surface. However since at the time FSS is to begin, all remediation activities in the area have been completed it is reasonable to assume that any remaining discrete items will be within the top 3 inched of soil. The small pellet fragment that was identified during FSS previously was evaluated to theoretically result in an additional dose to the Survey Unit (SU) of 0.08 milliRem per year, and exhibited a count rate of 5,445, ncpm with the probe held 1 inch above the soils surface, and the pellet fragment buried at approximately 3 inches. This count rate is significantly greater than the proposed Scan IAL of 4,000 ncpm and therefore would provide a high level of confidence that any potential remaining discrete items would be identified and flagged for further investigation and subsequent removal.

It is the conclusion then that the proposed Scan IAL of 4,000 ncpm is appropriate for use in Class 1 survey units as it represents a value that is conservatively less than the DCGLw and thus will also be significantly less than the DGCLEMC. Additionally, all GWS data will be "post-processed" and reviewed by the FSS Supervisor to determine if any additional areas require investigation based on the recorded GWS data.

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Figure 1 – Area 1 GWS

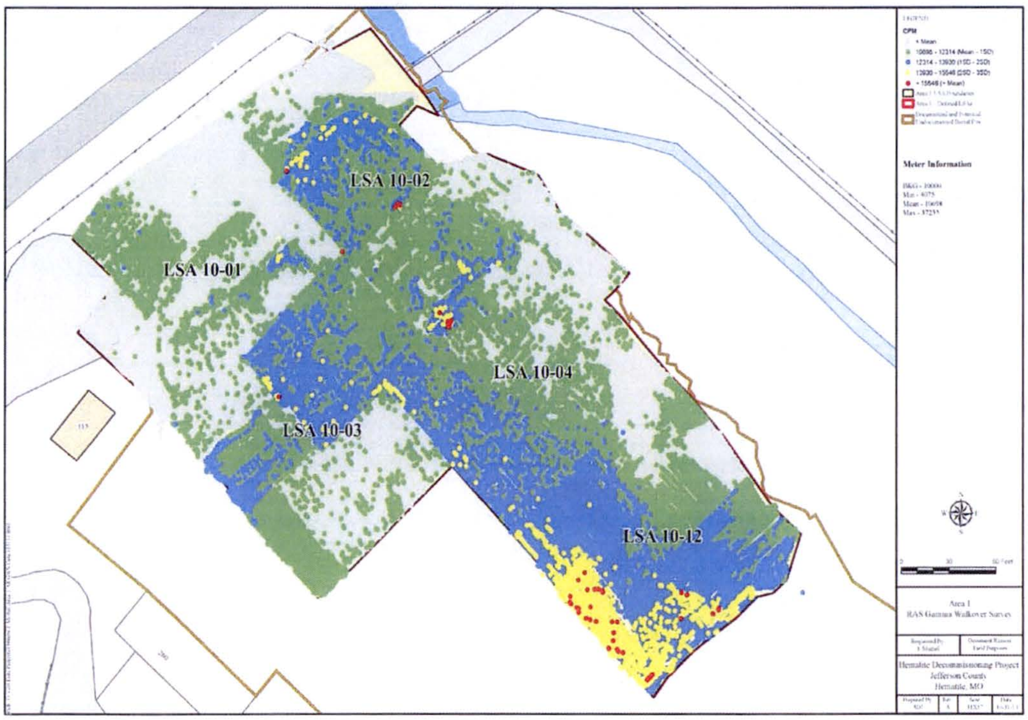

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Table 1 – LSA 10-02 Biased Results

SOF Calculation

	RA226	Tc99	Th232	U234	U235	U238
Avg. Conc	0.27	0.00	0.05	7.70	0.38	1.29
DCGL	1.9	25.1	2	195.4	5.8	168.8
SOF	0.14	0.00	0.02	0.04	0.06	0.01

SOF=	**0.28**	**MAX_Sample (SOF)**	**0.44**

RaBKGD	0.90
ThBKGD	1.00

	Ra-226	Th-232	U-235	U-238	U-238 / U-235	% En.	U-234 Ratio (enrichment) U234 /U-235	U-234	Sample SOF
0183-SS-141114-05-01	1.27	1.16	0.00	0.00	0.00	HEU	32.50	0.00	0.28
0183-SS-141114-05-02	1.02	1.08	0.18	0.00	0.00	HEU	32.50	5.95	0.17
0183-SS-141114-05-03	1.21	0.90	0.95	3.86	4.08	3.7	18.12	17.15	0.44

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Owner/Author:

W. Clark Evers — 3/2/15

W. Clark Evers
RSO

Date

Reviewed & Approved By:

Steven A. Grice — 3/2/15

Steven A. Grice
FSS Task Manager

Date

Attachment 2

Final Status Survey Final Report Volume 3, Chapter 1

Land Survey Areas (LSA) Overview, Revision 3

Track Change Version

Westinghouse Electric Company LLC, Hematite Decommissioning Project

Docket No. 070-00036

Westinghouse

Final Status Survey Report

Hematite Decommissioning Project

Final Status Survey Final Report Volume 3, Chapter 1

TITLE: **Land Survey Areas (LSA) Overview**

REVISION: **3**

EFFECTIVE DATE: FEB 13 2017

TRACK CHANGE VERSION

Approvals:

Author: Kenneth E. Pallagi — Date: 02-13-2017

Owner/Manager: W. Clark Evers — Date: 2/13/17

REVISION LOG	
Revision No. Effect. Date	**Revision**
0 01/27/2016	Revision 0 is the initial issuance of the Land Survey Areas Overview.
1 05/17/2016	The NRC provided via email a Pre-Audit Submittal Table for Final Status Survey Final Report Volume 3, Chapter 1 on March 8, 2016, which contained comments generated by the review. During subsequent recurring weekly publicly noticed teleconferences, Westinghouse and the NRC discussed the path forward and resolution of the NRC comments for Final Status Survey Final Report Volume 3, Chapter 1. This revision implements the resolution of the comments.
2 10/27/2016	The NRC provided comments on the review of Final Status Survey Final Report Volume 3, Chapter 1, Revision 1, during recurring weekly publicly noticed teleconferences. Westinghouse and the NRC discussed the path forward and resolution of the NRC comments for Final Status Survey Final Report Volume 3, Chapter 1. This revision implements the resolution of the comments. The revision includes discussion of 1) GPS vs GWS Coverage, 2) clarification of section 3.1.2, Three Stratum DCGLs, and 3) addition of Appendix D.
3 See Cover Page	During the NRC review of Survey Area Release Record the NRC, during recurring weekly publicly noticed teleconferences, provided feedback in regards to the application of the WRS Test when applied to the Three Stratum approach. Westinghouse and the NRC discussed the path forward and resolution of the NRC comments. This revision implements the resolution of the comments.

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LIST OF ACRONYMS AND SYMBOLS

AEC	Atomic Energy Commission
AF	Area Factor
ALARA	As Low As Reasonably Achievable
AMSL	above mean sea level
bgs	below ground surface
CFR	Code of Federal Regulations
cm	centimeter(s)
cpm	count(s) per minute
CSM	Conceptual Site Model
DCGL	Derived Concentration Guideline Level
$DCGL_{EMC}$	DCGL for small area of elevated activity
$DCGL_{w}$	DCGL for average concentrations over a survey unit, used with statistical tests. ("W" suffix denotes "Wilcoxon")
DGPS	Differential Global Positioning System
DP	Hematite Decommissioning Plan
EMC	Elevated Measurement Comparison
EPA	U.S. Environmental Protection Agency
ft	foot (feet)
FSS	Final Status Survey
FSSP	Final Status Survey Plan
FSSFR	Final Status Survey Final Report
gcpm	gross count(s) per minute
GPS	Global Positioning System
GWS	Gamma Walkover Survey
HDP	Hematite Decommissioning Project
HP	Health Physics
HRCR	Hematite Radiological Characterization Report
HRGS	High Resolution Gamma Spectroscopy
HSA	Historical Site Assessment
I & C	Isolation and Control
IAL	Investigation Action Level
LSA	Land Survey Area
m	meter(s)
m^2	square meter(s)
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCL	Maximum Concentration Limit
MDC	Minimum Detectable Concentration
mrem	Milliroentgen Equivalent Man
NAD	North American Datum
NaI	Sodium Iodide
ncpm	net count(s) per minute
NCS	Nuclear Criticality Safety

NRC	U.S. Nuclear Regulatory Commission
pCi/g	picocurie(s) per gram
QC	Quality Control
Ra	Radium
RASS	Remedial Action Support Survey
RG	Remediation Goal (specific to chemical contaminants)
RML	Reuse Material Screening Action Level
ROC	Radionuclides of Concern
RSO	Radiation Safety Officer
SEA	Surrogate Evaluation Area
SNM	Special Nuclear Material
SOF	Sum of Fractions
SSC	Structures, Systems and Components
SU	Survey Unit
Tc	Technetium
TEDE	Total Effective Dose Equivalent
Th	Thorium
U	Uranium
UF_6	Uranium Hexafluoride
WRS	Wilcoxon Rank Sum
yr	year

1.0 INTRODUCTION

The objective of the Final Status Survey (FSS) is to demonstrate that the dose from residual radioactivity at the Hematite Decommissioning Project (HDP) Site does not exceed the annual dose criterion for license termination for unrestricted use as specified in U.S. Nuclear Regulatory Commission (NRC) Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination," and that the levels of residual radioactivity are As Low As Reasonably Achievable (ALARA). To demonstrate that this objective is achieved, an FSS will be performed on all impacted open land areas that are to remain at the time of license termination. The principal requirement is that the dose to future site occupants will be shown to be less than 25 millirem/year.

The conduct of remedial activities in land survey areas (LSAs) is described in the HDP Decommissioning Plan (DP) DO-08-004, *Hematite Decommissioning Plan* [Westinghouse 2009] (Reference 8.1) and associated documents as approved on October 13, 2011, by NRC letter (Reference 8.2) with the issuance of License SNM-33 Amendment 57 (Reference 8.3) and the DP Safety Evaluation Report (Reference 8.4). The goal of the decommissioning project is to release the facility for unrestricted use in compliance with the requirements of 10 CFR Part 20.1402. The FSS process described in the DP and associated documents, and discussed in this section, adheres to the guidance provided in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Reference 8.5) for planning, conducting, and evaluating surface soil final status radiological surveys. In addition to MARSSIM, 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 8.6);
- NUREG 1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions (Reference 8.7); and,
- NUREG 1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys (Reference 8.8).

The conduct of FSS activities for LSAs was carried out through the implementation of the following HDP procedures and their current Revision number:

- HDP-PO-FSS-700, Final Status Survey Program
- HDP-PR-FSS-701, Final Status Survey Plan Development
- HDP-PR-FSS-703, Final Status Survey Quality Control
- HDP-PR-FSS-711, Final Status Surveys and Sampling of Soil and Sediment
- HDP-PR-FSS-720, Final Status Survey Data Integrity and Database Management
- HDP-PR-FSS-721, Final Status Survey Data Evaluation
- HDP-PR-FSS-722, Final Status Survey Reporting

The current site LSA figure is provided for reference in Appendix A. A procedure revision history for the above procedures and other procedures discussed in this overview document is provided in Appendix B.

1.1 FSSFR Volume 3, Chapter 1, Revision 1

This revision (Revision 1) to FSSFR Volume 3, Chapter 1 has been issued to address and respond to pre-audit submittal comments provided by the NRC in regards to FSSFR Volume 3, Chapter 1, Revision 0. The NRC comments are provided in the document titled Pre-Audit Submittal Table for Final Status Survey Report Volume 3, Chapter 1 – Land Survey Areas (LSA) Overview {ML16068A239}. Westinghouse's response to the comments and the corresponding revisions made within this document accompanies the submittal of FSSFR Volume 3, Chapter 1, Revision 1, as submitted to the NRC in Westinghouse letter HEM-16-50.

2.0 REMEDIATION ACTIVITIES

Prior to the implementation of an FSS for any LSA, it was necessary to remove all structures, concrete slabs, buried waste and piping, vegetation, and/or any other obstructions, to the extent practicable, to allow the remaining soil to be assessed for radiological contamination and disposed offsite as appropriate. To achieve this objective, the scope of decommissioning activities conducted included the following:

- Installation of additional site infrastructure, including: temporary utilities, security equipment, rail spur and loading pad, soil treatment facility, water treatment system, equipment and soil staging areas, and temporary haul roads;
- Decontamination of structures, systems, and equipment intended to remain at the time of license termination;
- Performing radiological surveys of buried and embedded piping to assess nuclear criticality safety requirements, designing remediation "cut-plans", and developing waste disposition strategies;
- Demolishing and packaging for off-site disposal, site buildings and infrastructure not designated for unrestricted release;
- Excavation of soil, buried waste, and concrete foundations within impacted areas while segregating soil that was acceptable for re-use as backfill; and,
- Packaging and coordinating transportation for disposal of radioactive, hazardous and mixed waste.

These activities took over ten years to complete and included the removal and offsite disposal of the former Process Buildings and their contents, documented onsite burials, undocumented onsite burials, subterranean piping, a sanitary wastewater treatment plant, paved areas, concrete slabs, and other miscellaneous site structures. In some cases remediation, excavation, and surveys were all accomplished concurrently. In other cases, such as with the removal of the Process Buildings and the underlying concrete slabs, the radiological assessment of the underlying soil could not begin until the buildings and slabs were removed. Other areas onsite, such as the Site Pond, created its own set of challenges as incoming water had to be diverted and the pond dried before it could be fully evaluated and remediated.

Due to the significant diversity of tasks and activities that were required to be addressed before FSS of LSAs could be conducted, an overview is provided discussing each of these areas. This includes the Documented Burials, Undocumented Burials, Process Buildings, Vaults, Evaporation Ponds, Natural Gas Pipe Line, Red Room Roof Burial and Barns Areas, Sanitary Wastewater Treatment Plant, Site Pond/Site Creek Area, Tc-99 Area, Class 2 and 3 Survey Areas, Waste Disposal, and Backfill Operations.

2.1 Documented Burials

On-site burial was used as a disposal method for contaminated materials and wastes at Hematite from 1965, until 1970, in accordance with regulatory requirements and specific license authorizations. Detailed logbooks of these waste burials documented that there were 40 unlined pits located east of the site buildings as shown on Figure 2-1. These documented Burial Pits (collectively referred to as the Burial Pit Area) were used to dispose of waste materials generated by the fuel fabrication processes. These on-site burials were created under the governance of AEC regulations contained in 10 CFR 20.304 (1964, Reference 8.9). These regulations described the spacing of the pits, the thickness of the cover, and the quantity of radioactive material that could be buried in each pit. The nominal dimensions of each Burial Pit were 20 feet (ft) wide by 40 ft long by 12 ft deep and the regulations provided that these were supposed to include an approximate cover depth of 4 ft.

The site owner at the time, United Nuclear Corporation (and later Gulf United Nuclear Corporation) maintained detailed logs of waste burials occurring between July, 1965, and November, 1970. Each entry contains a date, a description of the waste buried, the weight of the Uranium measured or estimated for that waste, and a cumulative total of the Uranium buried in that particular pit. The weight of the contaminated item measured or estimated was determined to the nominal value of 1 gram which likely resulted in an over-estimate of the actual amount. Some entries also list the percent enrichment for the Uranium. The Burial Pit logs show a wide variety of wastes being buried in the pits; the majority of the listed waste is non-SNM waste, such as contaminated trash, drums, pails, bottles, rags, etc. Additional waste materials listed include Uranium process metals of various enrichments, metal wastes, liquid and solid chemical wastes, and HEPA filters.

The on-site burial of radioactive waste materials was terminated in November, 1970, as a result of an AEC violation issued to the Hematite facility for failure to adhere to revised AEC regulations concerning the quantity of material which could be buried onsite. An AEC Inspection Wrap-up Meeting memo (Reference 8.10), stated that a revision of 10 CFR 20 was enacted in June, 1970, that reduced burial limits for enriched Uranium. The licensee at the time had continued burials based upon the limits prior to June, 1970, resulting in the above AEC violation. It is noted that the Burial Pit logbook records, employee interviews, and the operational Uranium recovery process used during this time period consistently show efforts to maximize recovery and utilization of Uranium material whenever possible. Based on these records, Westinghouse believed that there was little likelihood the Documented Burial Pits contained significant quantities of recoverable SNM.

HDP developed consistent generic screening and handling approaches in preparation for the excavation and removal of buried wastes, contaminated soils, and sub-surface structures (e.g., concrete slabs, buried piping) in areas where it was determined fissile materials had a reasonable possibility to exist based on characterization data and historical knowledge. These approaches were analyzed from a Nuclear Criticality Safety (NCS) perspective in NCS Assessments (NCSAs) specific to buried waste exhumation, contaminated soil remediation, and sub-surface structure decommissioning. Although not an element of the FSS, screening for fissile materials and the required NCS controls were important due to the inherent safety significance of its performance during the remediation process. Screening typically involved duplicate performance of radiological surveys using sodium iodide scintillation detectors, and defined appropriate volumes of material to ensure that NCS limits were not exceeded. The objective of the *in-situ* radiological surveys was to identify any item or region of soil/waste with a fissile concentration exceeding 1 gram U-235 in any contiguous 10 liter volume. This provided a high degree of assurance that any items with elevated (i.e., nontrivial) levels of U-235 contamination would be identified. The in-situ radiological surveys were to be complemented by visual inspection of the survey area with the aim of identifying:

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

2) Items that resembled intact containers;

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

4) Metallic items.

While carrying out the fissile material screening and handling process the secondary remedial action objectives to identify hazardous materials (e.g., Volatile Organic Compounds or VOCs) and verify radioactivity concentrations of soil for potential use as backfill were also completed. In areas where fissile materials were suspected to exist based on historical knowledge, excavation continued until both visible and radiological evidence indicated that suspect materials had been removed. Once fissile material screening determined that fissile materials were not present in the remediation area in excess of the NCS Exempt Material Limit, NCS controls were curtailed. By making this the initial goal, remaining remediation, Health Physics, and Final Status Survey activities could proceed unencumbered by NCS controls. Identified items that exceeded NCS limits were segregated into designated Field Containers, which were placed in transport containers such as Collared Drums (CDs). Individual designated containers were then handled and stored in accordance with NCSA requirements.

In addition to the documented AEC authorized onsite burial pits, undocumented on-site burials were also conducted for disposal of general trash and items that may have been slightly contaminated. Prior to commencing excavation it was estimated that 20-25 of these non-AEC burials could exist for which there were no records. No written information was available that indicated the specific nature of the waste material buried in the undocumented burials. To provide clarity, the term "documented burial pit" refers to the AEC authorized burials that were identified in the site log books. The term "undocumented burials" refers to all other burials. The distinction is the "undocumented burials" were *not pits*, but shallow burials used to dispose site trash and debris. Both the documented burial pits and the undocumented burials were located within the larger area that is formally designated as the Burial Pit Area.

2.1.1 Remediation and Excavation

Excavation and removal of the Burial Pit Area soil was initially planned to begin at the northwest corner and continue towards the east and south. The soil excavation was to be performed in multiple burial pits concurrently ensuring sufficient space for heavy equipment to operate safely, while maximizing the handling of material available for re-use as backfill and minimizing cross-contamination. However, early in the excavation of the Burial Pit Area, Westinghouse recognized and made the decision that it was more efficient, provided a more comprehensive remediation process, and was more cost effective to excavate the entire area as opposed to trying to excavate each individual burial pit. This also had the added benefit of eliminating any need to correlate the size and content of the individual burial pits since not only were the pits being excavated, but so was any soil in between the pits.

The majority of materials buried in the Burial Pit Area were anticipated to be contaminated soil and trash, some laden with VOCs, floor tiles, glass wool, and laboratory glassware. Minor components of the buried waste volume were anticipated to include: acid-insoluble residue; filters; metallic debris; and, metallic oxides. However, because of the potential that fissile quantities of material could be found in the documented burial pits, excavations were performed in accordance with the limitations of the nuclear criticality safety assessment(s). In general, the order of techniques to be employed in removing soil and debris from the Burial Pit Areas was:

- Evaluate soil using in-situ gamma walkover surveys (GWS), VOC monitoring (with a Photo-Ionization detector), and visual inspection of the exposed surface, repeated for each newly exposed surface following the removal of each lift;
- Excavate and remove soil in nominal 6-inch lifts when under NCS controls, otherwise in 1-foot lifts;
- Excavate and segregate surface and subsurface soil based on: visual inspection; radiological and chemical survey/screening; supplemental sampling and analysis; the appropriate DCGLs; chemical Remediation Goals (RGs); and, the NCS Exempt Material Limit for potentially fissile material;
- Stockpile excavated soil at a safe distance adjacent to the excavation, or load into a haul truck for transfer to a Waste Consolidation Area (WCA) for further visual inspection;
- Using heavy equipment, excavate objects encountered in the soil; however, if deemed appropriate based upon GWS or NCS evaluation more precise methods and equipment could be used to excavate an object (e.g., hand-shoveling, small bucket excavator); and,
- Employ sloping and benching during the excavation process, as required, and continue until visible wastes are removed and in-process surveys and soil sampling meet specified acceptance criteria.

In March, 2012, excavation and remediation of the Burial Pit Area soil began in both the northwest and southeast corners of the Burial Pit Area simultaneously. Excavation and removal of soil was performed concurrently, which optimized the efficiency in handling of equipment and movement of material. The burial pit waste excavation activities concluded in December, 2014.

Initially the removal of the overburden began in 1 foot lifts as specified in the HDP DP. The intent of the 1 foot lifts was to ensure potentially fissile quantities of material could be identified for appropriate handling, and to enable waste to be identified and removed so a maximum volume of soil could be saved for reuse. This process was achieved through a GWS and visual inspection; any lumps of material that potentially exceeded 40g U-235 total were manually separated out and underwent a more thorough evaluation utilizing close proximity radiological surveys to determine if further handling requirements were necessary to maintain NCS compliance. At this point any material that was identified ex-situ to be greater than 15g U-235 required segregation from the waste stream, and special handling for NCS Control purposes. The NCS evaluation performed at the time identified that material in-situ less than 40g U-235 could go unidentified with a 12 inch lift, but any of the material identified ex-situ greater than 15g U-235 still required segregation. It had been determined that these were adequate ex-situ survey controls to provide a wide margin of Criticality Safety, and also demonstrate compliance with offsite waste disposal acceptance criteria. These activities were conducted under Procedure HDP-PR-HP-601, Remedial Action Support Surveys.

Within the first few months after excavation activities began, the NRC expressed concerns as to whether a NaI 2x2 detector could identify lumps of U-235 via in-situ scanning through a foot of soil, therefore potentially allowing material between 15g and 40g U-235 to be excavated prior to segregation. After further review and discussion with the NRC, HDP made the decision to perform subsequent NCS controlled excavations in 6 inch lifts. Soil/debris in areas not subject to NCS controls was still excavated in 12 inch lifts. This change was initiated in August, 2012, approximately four months after site excavation activities began and resulted in a revision to Procedure HDP-PR-HP-601.

The general locations of the former burial pits were known based on reviews of historical records, such as site aerial photographs taken during the time period the burials occurred, and the field observations of depressions that were visually discernible in the Burial Pit Area ground surface. The localization of the pits was further helped based on visual clues, and physical work activities performed during the excavation activities. Specifically, as the overburden soil was removed it was easy to visually identify the location of a burial pit based on a change in soil color. Even the undocumented burials could be easily identified by a change in soil color in spite of the fact that their size and shape was not as well defined as the documented pits. See Figures 2-2 and 2-3. Additionally, the equipment operators conducting the excavation could distinguish when they were digging in a burial pit based on the difference in the hardness of the soil. Workers could even detect the difference in the soil hardness when walking over a burial pit, which tended to be soft and spongy. Adding to the visual and soil hardness cues, the burial pit was also radiologically identifiable based on a GWS once reaching the contaminated layer. Figure 2-4 shows a radiation technician conducting surveys with a NaI 2x2 detector during burial pit remediation activities.

Overall the process of locating, assessing, and removing debris from the burial pits was very labor intensive. For example, on several occasions hundreds of small plastic vials were found. In each case, every vial had to be separately surveyed by two independent technicians using different instrumentation. While the majority of materials located within the burial pits were as anticipated, 215 radium contaminated filter plates made of steel, cast iron, and plastic, about 3 feet by 3 feet in size, were unexpectedly found. See Figures 2-5 and 2-6. It was determined that these were brought to the Hematite site from an offsite entity and did not originate from any onsite process or operation. These plates required a significant amount of time and effort to be decontaminated prior to disposal which resulted in the single largest contributor to worker dose (603 mrem max individual annual exposure) since the DP was approved in 2011. Figures 2-7 and 2-8 also show several examples of some of the waste removed from the burial pits.

As excavation and remediation of the Burial Pit Area progressed, it became apparent that most of the buried debris was located in the north and south ends and typically in closely aligned pits, while the middle area had minimal debris and contamination. As sloping and benching practices were employed, and due to the close nature of the pits, large areas ended up being remediated as opposed to individual standalone pits. This had the advantage of providing additional assurance that any radiological contaminate that could have migrated laterally from a pit was also likely remediated. Also as expected, the burial pits were generally of similar depth. This was expected because of the equipment that would have in all likelihood been used to dig the pits. A normal sized backhoe would have been the expected heavy equipment employed, which has a typical maximum reach of about 10 feet (plus or minus) when digging a trench. In addition, the depth of the pit was constrained by the regulatory requirement of 12 feet deep with 4 feet of cover. This knowledge helped when excavating a burial pit as it provided an informal bound of 16 feet for the depth of the pit, as well as providing a solid basis for not expecting two burials to be placed one on top of another.

Figures 2-9 through 2-18 provide a chronological aerial view of the Burial Pit Area excavation progress from March, 2012, when the removal of overburden soil had commenced through September, 2015, during backfilling of the area.

2.1.2 Ceasing Excavation

As excavation progressed in the Burial Pit Area, five activities came into play that determined the extent of remediation in a given survey unit (SU). These were: 1) ongoing remedial action support surveys (RASS), 2) conducting core bores to support release from criticality controls, 3) performing a final RASS, 4) sampling for VOC concentration in soils to determine if the chemical cleanup Remediation Goal (RG) had been met, and 5) conducting final status surveys (FSS). The RASS was conducted to: guide remediation activities, determine when an area or survey unit had been adequately prepared for FSS, and provide updated estimates of the parameters to be used for planning the FSS. During soil excavation, the RASS would serve to assess the potential concentration and amount of U-235 for comparison to the NCS Exempt Material Limit. In areas subject to NCS controls, a GWS with visual inspection was required to be performed independently by two different HP Technicians using two separate instruments prior to excavation of each layer of soil. In conjunction with the GWS, remediation areas were visually inspected prior to exhumation of any material. Once excavation of a SU reached a point where there was no longer any indication of burial pit waste, and the RASS indicated the soil met the NCS Exempt Limits, core borings were conducted. The core bores were required to be dug 7 feet deep below the original pre-excavation surface, with the additional constraint that it must also extend at least 3 feet below the excavated surface. This meant that any excavation greater than 4 feet deep would require a core bore of 3 feet below the excavated depth. While the purpose of the core bores was only to verify that NCS controls could be reduced, it also provided an additional opportunity to support the remediation objectives by allowing for a visual inspection to ensure that there was no longer an indication of a burial pit below the excavated elevation (surface). Core bores were performed using an auger of anywhere from 3 to 8 inches in diameter either manually or as an attachment to a skid steer. See Figure 2-18a.

These core bores were systematically placed based on a maximum 20 foot grid, eventually covering the entire Burial Pit Area. Core cuttings and the core bore hole were surveyed to provide radiological data to determine whether NCS controls could be suspended for that SU. The data was evaluated against pre-determined NCS Limits. The core cuttings were evaluated against a value of 47k net counts per minute (ncpm), and the core bore holes were evaluated against a value of 63k ncpm. These values were derived for a lump of uranium under two and six inches of soil respectively. The values differ because the cuttings were spread out in a layer while the bore hole core measurement was taken from the wall of the bore hole. The methodologies and results are detailed in NSA-TR-10-12, *Calibration Analysis for ^{235}U Response from Burial Pit Waste Materials at the Hematite Facility*. If the core bore data was less than the above criteria and no waste material indicative of a burial pit was found, it was used as a decision point that assumed the bottom of the burial pit had been reached and criticality controls were no longer necessary. Even though additional radiological remediation may still have been required for the soil surrounding a burial pit, by eliminating the NCS controls, work could progress at a faster pace and with fewer resources. (As a note, core bore data was not used in the development of the FSS Plan. It was only used in the assessment of the need for NCS controls.)

Following removal of NCS controls, remediation would often continue to remove any remaining radiological contamination in the soil surrounding the burial pit debris field. This work would continue in conjunction with RASS until it was determined the DCGLs for that SU were be met. At this time, radiological remediation was complete unless additional remediation was required based on subsequent FSS activities. However, excavation in many cases continued after radiological concerns were addressed due to the need to remove VOCs in the soil. In a number of cases, VOC remediation resulted in the removal of soil down to the phreatic surface. Once all radiological and VOC remediation activities were completed, a final RASS survey was performed to ensure the area met required DCGLs and to obtain data for FSS design. The FSS was then performed to demonstrate the SU met the NRC unrestricted release requirements based on a GWS and soil sampling results.

2.1.3 Retrospective Review

As the remediation of the Burial Pits neared completion, two issues were identified for examination as a retrospective review to provide assurance that all contaminates related to the burial pits were adequately removed. The first issue involved the potential for a radioactive contaminate to migrate through the soil and collect in an area below where a burial pit or radioactivity was known to have existed. In this case, because an area of elevated contamination could pose a long term risk factor, it was deemed sensible to examine the potential that an area like this could still exist in spite of the extensive remediation activities that had been performed. The second issue was whether another burial "pit" could potentially exist below where the current excavation activities ceased.

To address the first issue, Westinghouse evaluated the potential for the existing radionuclides identified to migrate through the soil. As discussed in Volume 1, Chapter 1 of the FSSFR, the primary nuclides of concern at Hematite were U-234, U-235, U-238, Ra-226, Th-232, and Tc-99. Of these radionuclides, only the Tc-99 had the potential to move downward through the soil and collect within a timeframe consistent with existence of the Hematite facility. The other nuclides are generally non-soluble and more readily bind to soil particles which retards their movement. Because Tc-99 is less likely to bind to soil and is more mobile it will travel more readily with water. A detailed discussion of Tc-99 and limestone as its source is provided in Section 2.10.

A review was conducted of the Burial Pit Area surface and sub-surface soil sampling data collected between 2004, and 2008, and summarized in the Hematite Radiological Characterization Report. The review indicated that 94 surface soil samples (at 0.0-0.5 feet) were analyzed for Tc-99. Fifty-five of the 94 samples had Tc-99 activity above the Minimum Detectable Concentration (MDC) with a maximum result of 68.3 picocurie per gram (pCi/g) and an average of 2.6 pCi/g. All of these areas were remediated when the overburden was excavated.

There were 89 soil samples within the root zone (0.5-5.0 feet) that were analyzed for Tc-99. One sample had a value of 33.8 pCi/g, and this area was remediated. Twenty-one of the 89 samples had Tc-99 activity above the MDC with a maximum result of 14.8 pCi/g and an average of 1.0 pCi/g. None of the results were in excess of the Tc-99 Uniform DCGL of 25.1 pCi/g or the Root Stratum DCGL value of 30.1 pCi/g. Although below the DCGL, many of these areas were still remediated in conjunction with excavation activities.

There were 144 soil samples within the deep zone (> 5.0 feet) that were analyzed for Tc-99. Twenty-seven of the 144 samples had Tc-99 activity above the MDC with a maximum result of 38 pCi/g and an average result of 0.75 pCi/g. None of the results were in excess of the Tc-99 Excavation DCGL value of 74.0 pCi/g. Regardless, many of the areas where these samples were collected were also remediated. Also reviewed were 106 soil sample results of Tc-99 in the deep zone from 16 feet down to a maximum of 35 feet. The highest value recorded was 9.34 pCi/g, which indicated the limestone had contributed very little Tc-99 at depth in the Burial Pit Area. While Tc-99 can migrate readily in soil the data shows that it was not an issue in the Burial Pit Area based on characterization results obtained between 2004, and 2008.

There was also a concern that previous hybrid groundwater monitoring wells could have provided a conduit for Tc-99 to migrate downward. A hybrid well was a well that had been installed with a screen that extended from the overburden clay to the sand-gravel aquifer, which could allow transport between layers. Seven hybrid wells (BP-17, BP-20A, BP-21, NB-61, WS-25, WS-27, WS-29) were installed in the Burial Pit Area in 2004. From each of these seven wells, Tc-99 water sample results were evaluated to determine if any well analytical results for Tc-99 exceeded a threshold value of the MDC plus any measurement error. Only well BP-17 exceeded this threshold value, with a value of 43.2 pCi/L Tc-99 on 9/15/2008. (The EPA drinking water standard for Tc-99 is about 900 pCi/L.) This well was located in LSA 10-12, in the area where the limestone was buried so the higher level of Tc-99 in this well was not unexpected.

In 2011, Westinghouse committed to investigate the potential for a preferential pathway of Tc-99 and uranium along a monitoring hybrid well, and to determine whether contaminated soil existed in proximity to a hybrid monitoring well (HEM-11-56, May 5, 2011, Attachment 1). When hybrid wells were abandoned they would 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 would continue until reaching refusal (typically bedrock) or a depth of 35 feet. The soil cuttings that were removed during the boring process would be surveyed for indications of elevated radioactivity as a qualitative measure and sampled by laboratory analysis.

In 2006, hybrid well NB-61, which was located in LSA 10-11 and was close to the natural gas pipeline was abandoned. Since this pre-dated HEM-11-56, cuttings from this well were not collected for analysis. In 2012, the remaining six hybrid wells (BP-17, BP-20A, BP-21, WS-25, WS-27, WS-29) plus three groundwater monitoring wells in the Burial Pit Area were abandoned. When the wells were abandoned they were over drilled as required. Soil cuttings that were removed during the boring process were surveyed for indications of elevated radioactivity as a qualitative measure and sampled for laboratory analysis. For the six hybrid wells and three groundwater monitoring wells abandoned in 2012, radiological samples were collected in the cuttings of each 5 foot interval to the bottom of the boring which was in the range of 28-35 feet below ground surface. A total of 62 samples were collected. The maximum Tc-99 concentration identified was 21.1 pCi/g from the 5-10 foot interval of the BP-17 well cuttings. This area was subsequently excavated. In 2013, to better characterize the area around BP-17 and assess the potential extent of Tc-99 contamination four core borings were conducted surrounding BP-17. The highest reading was in the boring east of BP-17, which showed a maximum Tc-99 value of 59.6 pCi/g at 28-30 feet.

Subsequent to the initial four core borings, another six investigation borings were conducted in 2013, in the area where BP-17 was located. A total of 86 soil samples were collected from the ground surface to bedrock from the ten borings conducted. None of the sample results from the four initial cores exceeded the applicable Tc-99 Excavation DCGL of 74 pCi/g. The maximum Tc-99 value identified was 59.6 pCi/g at 28-30 ft on the boring collected just east of BP-17. The remaining samples collected in the other three borings contained a maximum value of 4.25 pCi/g. The maximum values of Tc-99 identified from the additional six investigation borings was 33.1 pCi/g at 16-20 feet.

Surficial deposits of spent limestone were located in the central portion of the Burial Pit Area along the eastern slope. The spent limestone was remediated and the sent to US Ecology for disposal. The overall depth of the limestone was approximately 2 – 4 feet below original grade.

In summary, based on sample results, it has been determined that there is very little probability that Tc-99 could exist at levels that exceed the Excavation DCGL at a depth below where excavation activities ceased because there was very little Tc-99 that existed prior to excavation and remediation activities in the majority of the burial pit area. The Tc-99 that was identified was predominately in the areas where limestone had been placed, which was expected since the limestone was the path by which the Tc-99 was introduced into the waste stream. In addition, the areas where the Tc-99 was identified were ultimately excavated.

To address the second issue of whether another burial pit could potentially exist below where the excavation activities ceased, a number of factors were considered. In some locations it is clearly demonstrated that another burial pit could not exist because the excavation went to the phreatic surface or sample results from core borings that went to bedrock showed there was no additional radiological contamination, while in other cases it is demonstrated through a combination of a number of factors.

The first things considered were the nature of the burial pits themselves. The documented licensed burial pits were fairly easy to identify by location based on historical records, examination of site aerial photographs taken during the time period the burials took place, and depressions visually discernable in the ground surface. In none of the historical records, photographs, or anecdotal evidence, including input from former Hematite workers, was it indicated, observed, or implied that burial pits were stacked one on top of another.

In regards to the contents of the burial pits, the characterization studies performed prior to the development of the DP had identified numerous radionuclides of concern (ROC) that were used to define the site DCGLs, which included radium as a ROC. While there was no documented historical process or other licensed activity onsite that would explain the presence of radium in the burial pits, it had none-the-less been identified. It was therefore not unexpected and it was anticipated that radium would be encountered during the remediation of the Burial Pit Area.

As could be expected, based upon the documented history of site operations and the information contained in the logbooks, the number and size of the radium contaminated filter plates found during the Burial Pit Area remediation was not anticipated. It is understandable then to assume that previous owners of the facility may have allowed the radium contaminated filter plates generated at another location to be buried in the Burial Pit Area. No other unanticipated waste was identified during remediation. The decision to remediate the Burial Pit Area in its entirety rather than locating individual burial pits for subsequent remediation proved to be a prudent and invaluable decision. The process provided an extremely high degree of assurance that all radioactive wastes were identified and removed from the Burial Pit Area, regardless if the specific type of waste or radionuclide was expected or not.

Ultimately, the excavation remediation activities bore out what was expected in regards to burial pit configuration, size, location, and the radionuclides of concern based on all the above sources of information. No burials (documented or undocumented) of any kind were found in any other location on the site, or one burial pit on top of another burial pit anywhere within the excavated

Burial Pit Area during excavation remediation work.

The second factor to consider is the visual nature of the burial pits. At the start of the Burial Pit Area excavation the overburden layer of soil was removed. Once this overburden layer was gone, the location of a burial pit was visually observable based on the discernable change in soil color. They could also be identified by the equipment operators conducting the excavation by the difference in the hardness of the soil, and even the workers could detect the difference in the soil hardness when walking over the burial pit. In essence, the burial pits were very easy to visually identify. Also of significance is the burial pit could be radiologically identified, even before it was fully uncovered. In some instances the actual buried debris was encountered a few feet below the observable change in soil color/texture, but was still identifiable using a 2 x 2 NaI detector. Once the bottom of the burial pit was reached it was evident by the change in color of the soil, the hardness difference between the softer burial pit material and the hard native soil, a disappearance of debris, and, a sharp decline in radiological readings. Figure 2-18b is an example of the evident change in color of the soil as excavation is nearing the bottom of a burial pit. It is important to be aware that had another burial existed below an already excavated burial pit, the soil between the "lower" and "upper" pit would have to have been disturbed, which would have been visually evident, and this was never identified to occur. Overall, the visually observable change in the soil conditions provided a very good confirmation that there was no buried waste located further down.

As excavation/remediation was conducted, the depth at which the remediation was necessary to remove radiological debris varied throughout the burial area. Radiological surveys were conducted continually as the burial pits were remediated. Prior to NCS controls being suspended, core bores were taken to verify the burial pit had ended as evidenced by visual inspection, the radiation readings from the bore holes, and the bore hole cuttings. These core bores were systematically placed based on a 20 foot grid over the entire burial pit area. A technical basis document (HDP-TBD-NC-205, *Assessment of the Adequacy of Lateral Subsurface Soil Sampling in the Burial Pit Area*) was developed to statistically verify the 20 foot spacing was acceptable to ensure that an undiscovered burial pit should be identified with a high level of confidence. Figure 2-19 provides a map showing the locations of all the core bores that were conducted for criticality control purposes. While this process was not conducted for the purpose of demonstrating a SU was ready for FSS, the fact that over 600 bore holes were drilled did provide additional assurance that no other burial pit existed below the depth of an already remediated pit.

Separate from the radiological remediation activities, excavation was also necessary to comply with State of Missouri regulatory requirements for VOCs. To meet the RGs for VOCs, a large portion of the Burial Pit Area was remediated beyond what was required to be remediated for radiological contamination. In some cases, this resulted in the entire core boring previously conducted for criticality concerns to be dug out as the excavation activities continued. In several locations, soil was excavated down to the phreatic zone. After all these activities, not one instance was identified where a second burial pit was located below an existing burial pit, or where radiological remediation activities had ceased.

A review was conducted to examine sample results from soil borings carried out from 2003, to 2014, for site characterization purposes. Soil samples from these borings were analyzed using gamma spectroscopy for any radionuclides and separately assessed for the presence of Tc-99. These borings provided evidence of both buried waste and the lack of buried waste. Due to criticality concerns, many initial borings were intentionally placed to avoid the burial pits. In subsequent characterization efforts, borings were conducted in the burial pits as well. Overall this provided a good mix of data that was representative of the entire Burial Pit Area. Figure 2-20 provides a map showing the locations of characterization core borings in the Burial Pit Area. Based on the review, none of the sample results showed there was contamination at a depth that would be indicative of a second burial pit below an existing burial pit. The core boring data was also examined against final excavation elevations. While these results are summarized within the FSSFR Report for each individual SU, in summary, all areas of contamination identified through characterization surveys were remediated entirely or at a minimum, to levels below the applicable DCGL for that stratum and SU.

Lastly, following the completion of all remediation activities, and in accordance with the DP, final status surveys were conducted that involved a 100% walkover scan performed with a 2 x 2 NaI detector, and a prescribed number of systematic soil samples (and additional biased soil samples as necessary) collected for analysis. This was completed for each SU within the Burial Pit Area.

Figure 2-21 provides a map of the post remediation excavation depths in the Burial Pit Area. The depths shown are the feet of material excavated from the original surface elevation at that location. The maximum excavated depth was approximately 24 ½ feet.

Based on the above retrospective review, all the above factors in combination provide an extremely high level of assurance that there are no undiscovered burial pits or localized elevated areas of contamination that were not remediated. This is based on, 1) the ability to visibly observe the locations of the burial pits during remediation activities, 2) criticality related core borings that were performed, 3) additional excavation activities to remove VOCs, 4) not one instance was documented in the log books, stated by former plant workers, or identified during excavations throughout the burial area that a second burial pit was located below an existing burial pit, 5) characterization data based on borings in the Burial Pit Area demonstrates all known areas of radiological contamination were already below or were remediated below the applicable DCGLs, and 6) the acceptable analytical results of FSS soil sampling and 100% gamma walkover surveys.

2.2 Undocumented Burials

2.2.1 History

Interviews with former employees indicated that undocumented on-site burials may have occurred as early as 1958, or 1959. Available employee interview records indicate that three or four burials may have been performed each year, prior to 1965, for disposal of general trash and items that may have been slightly contaminated. Accordingly, it was estimated that 20-25 of these non-AEC 10 CFR 20.304 burials could exist for which there were no records. Burials prior to 1965 were not documented (logged), as they were not considered to contain significant quantities of SNM, and were not known to contain radioactive wastes. No information was available that indicated the specific nature of the waste material buried in the undocumented pits. Additionally, no evidence was found that indicated that burial of known uranium-bearing materials occurred prior to 1965.

These undocumented burials were also believed to have been in the same general area as the Documented Burial Pits, and/or were in the proximity of site buildings in the eastern portion of the Central Tract (see Figure 2-12). Also, no specific information was located that indicated the specific nature of the waste material buried in these undocumented burial pits. Additionally, no evidence was found to indicate that burial of known uranium bearing materials (i.e., levels greater than free-release criteria) occurred during this time period.

2.2.2 <u>Remediation and Excavation</u>

Remediation and excavation activities for the Undocumented Burials were a continuation of and conducted in the same manner as the Documented Burial Pits. Initially, the entire area where documented Burial Pits were expected to be found was excavated to a depth of at least 4 ft to identify any burials. This depth was based on the AEC requirement that at least 4 ft of cover be placed over any regulated burial pits. As this process proceeded, undocumented burials, mostly to the west of the documented burial pit area, were also being uncovered. This was consistent with interviews with former licensee employees. These undocumented burials were found to have a minimal amount of cover, often just one or two feet, and there was no consistency to the shape and depth of these burials.

Although there was no evidence that any of these undocumented burials would contain fissile quantities of material, excavation practices including NCS controls remained in use until such time as it could be demonstrated the controls could be lifted. Figure 2-22 depicts the original Documented Burial Pit Area and the outer boundary of the area defined by the extent of the locations of the undocumented burials that were eventually uncovered. No documented burials were located outside of the area where the burial pit logs indicated they would be, but some undocumented burials were located within the documented Burial Pit Area.

Debris remediated from the undocumented burials was generally less contaminated than debris from the Documented Burial Pit Area and consisted of mostly trash and construction debris. Radiation surveys of this material identified the same nuclides identified in the Documented Burial Pit Area but typically at lower concentrations. These materials were dispositioned as required following the same practices employed for the documented burials. From an operational standpoint the excavation and remediation of the undocumented burials was conducted in the same manner as the documented burials.

2.3 Process Building

2.3.1 History

The primary structure removed during site remediation/demolition activities was the Process Building, which collectively included Building 240, Building 253, Building 254, Building 255, Building 256, and Building 260. The Process Building housed equipment associated with the chemical conversion of Uranium into compounds, solutions, and metals, and for the fabrication of Uranium compounds into physical shapes. The former location of these buildings is shown in Figure 2-23. An aerial view of the Process Building prior to demolition can be seen in Figure 2-24.

In June, 2001, Hematite ceased principal activities and shut the facility down. Although License Amendment No. 52 was issued in June, 2006 (Reference 8.12), authorizing the dismantlement and demolition of buildings, Westinghouse utilized the period between ceasing licensed operations in 2001, and initiating building demolition in 2011, by conducting decontamination of the various Hematite buildings to facilitate demolition. This included the removal and disposal of all Process Building process equipment and components, and remaining product.

2.3.2 Demolition

Demolition sequence of the Process Building included the removal of the processing equipment, then building demolition, and then the removal of the foundations, floor slabs, and associated drains, with all debris being shipped off-site for disposal.

Demolition of the Process Buildings took place during May, and June, 2011. To minimize the potential spread of contamination during demolition, the entire interior surface of the Process Building was sprayed with a chemical control fixative. An additional measure included spraying water on demolition debris to reduce dust and other airborne particulates.

Between May, 2013, and November, 2015, the Process Building slab was removed. The mid-easterly portion of the slab was left in place to be used as a haul road for the movement of vehicles and heavy equipment around the site in support of decommissioning activities. The haul road was removed in early November, 2015. Removal of the Process Building slab (except Haul Road) then allowed for access to the underlying soil for remediation. Excavation and remediation activities on the soil under the slab commenced in May, 2013, and all remediation of the slabs including under the Haul Road was completed in January, 2016.

2.3.3 Remediation and Excavation

Based on historical records and interviews it was identified that non-native material was introduced under the existing Process Buildings near the existing northwestern side of Building 253. As part of preparing for the construction of Building 253, native soil was excavated due to the presence of gross alpha radioactivity exceeding 30 pCi/g and spent limestone was used as backfill. Because of concerns about undermining Building 240, the excavation was stopped before all soil exceeding this limit was removed. The average alpha concentration in soil on the surface at the conclusion of this excavation was 17 pCi/g. The maximum alpha concentration was 82 pCi/g. Per "Building 253 Construction Site Soil" (Reference 8.13), this historical excavation area reached a maximum depth of approximately six feet below the planned floor level. Based on this description, the maximum depth of the excavation was likely to have been approximately eight to ten feet. However, based on interviews with personnel that were present when Building 253 was constructed, the excavation depth may have extended to a depth of 10-15 feet.

Prior to placing backfill, the then owner, Combustion Engineering, requested that NRC allow spent limestone that was stored on-site to be used as backfill material within the excavation area under Building 253. Per NRC letter to ASEA Brown Boveri/Combustion Engineering, "Authorizes Backfill of Area of Building 253 Construction Site" (Reference 8.14), dated July 2, 1990, the NRC allowed spent limestone from two piles meeting a 30 pCi/g limit (alpha) to be used as fill below Building 253 with the understanding that the fill would have to be removed upon facility decommissioning. Per a Combustion Engineering letter to NRC, "Spent Limestone Results" (Reference 8.15), dated April 7, 1989, the two limestone piles had average gross alpha concentrations of 7 pCi/g and 8 pCi/g.

Due to the presence of residual radioactivity in soil that could not previously be removed, and due to the presence of spent limestone as backfill, these areas were required to be excavated to the extent that all limestone was removed. This issue was addressed in a letter to the NRC, Westinghouse (E. K. Hackmann) letter to NRC (Document Control Desk), HEM-11-56, dated May 5, 2011, "Evaluation of Technetium-99 Under the Process Buildings" (Reference 8.16).

Remediation of the soil under the slab was conducted with the same procedures and practices employed for all other open land areas. All limestone used as backfill to support the construction of Building 253 was removed and shipped as waste, as were any other soils/materials identified as waste requiring offsite disposal.

The Process Building area also contained subterranean process piping. All of this piping was removed and disposed as waste. While additional areas of contamination were identified in the soil during the removal of the Process Building slab and subterranean piping, these areas were due to spills originating from the former Process Building operations and not due to burials. Figure 2-38 is a photograph of the Process Building Area showing the excavation and remediation that occurred under the Process Building slab. Figure 2-39 is a photograph taken on November 7, 2015, showing the area after removal and remediation of the haul road along with the remaining subterranean piping.

FSSFR Volume 4, Chapter 1, Building Survey Areas Overview, provides further photographs and discussion regarding the history, demolition and disposal of the Process Building.

2.4 Storage Areas/Vaults

2.4.1 History

The two storage areas/vaults constructed at the Hematite facility were the South Storage Area and the West Storage Area.

The South Storage Area was a standalone building (Building 252, see Figure 2-23). Constructed in 1960, in response to an increased need for storage, the building was used for storing process materials as well as final product storage. In 1974, all equipment was removed from the South Storage Vault and this area was decontaminated. Subsequently, during the commercial nuclear era, this building was used for radioactive waste storage.

The West Storage Area was located in Building 235, which was also a standalone building. The original building was constructed in 1956, housed the West Storage Area, and was utilized as the outgoing storage building where final Uranium products were stored. During the commercial nuclear era this building stored source material. In 1992, when Building 230 was constructed, Building 230 was placed such that the two buildings shared a common wall.

2.4.2 Demolition

The West Storage Area was not demolished until March, 2015. The demolition of this vault was delayed in case sufficient fissile material was identified during excavation of the Burial Pit Area that required the vault to meet storage requirements. No quantities of fissile material were ever found that met this requirement. Once the Burial Pit Area excavation and remediation was completed, the West Storage Area was demolished.

Building 252 was demolished in May, 2001, during the demolition of the Process Building.

Prior to demolition both vaults were sprayed with a chemical control fixative to limit airborne dust, as well as being sprayed with water during demolition activities. Neither facility contained a drain or underground piping, nor was significant radiological contamination identified in the soil beneath the buildings following demolition and removal.

FSSFR Volume 4, Chapter 1, Building Survey Areas Overview, provides photographs and further discussion regarding the history, demolition, and disposal of the Storage Buildings.

2.4.3 Remediation

The soil underlying Building 252 was located in LSA 08-12 and was remediated during the remediation of that survey unit. The soil underlying Building 235 was located in LSA 08-17 and was remediated during the remediation of that survey unit.

2.5 Evaporation Pond Area

2.5.1 History

The Hematite facility had two Evaporation Ponds that were placed into operation between 1962, and 1964, that were used for on-site disposal of process filtrates, low-level contaminants, and high-enriched and low-enriched Uranium materials. Use of the evaporation ponds was discontinued in 1978. In 1979, 1985, and again in the 1990's, these ponds were partially remediated. The Ponds had been known to overflow during periods of high precipitation, thereby impacting the soils around the Evaporation Ponds. Employee interviews and recent experience confirmed overflows have occurred, which made it an expectation that remediation of this area would be required.

Figure 2-40 is from an aerial photograph taken in September, 2012, showing one of the Evaporation Ponds (with a tarp covering to minimize water intrusion) and the location of an adjacent 8 inch natural gas pipe line (red line). The second pond has been covered with soil, but is located adjacent to and below (towards the bottom of the photo) the visible pond. Figure 2-41 provides a second view that shows the extent of the second pond (covered over) below the first pond. A historical site review determined that the ponds were placed into operation after the Natural Gas Pipeline (NGP) was constructed. There was a concern that contamination from the Evaporation Ponds could have migrated under the NGP. Over most of its onsite traverse the depth of the NGP is approximately 4½-5 ft, but in the vicinity of the evaporation ponds the NGP is 4-7 ft in depth. The close proximity of a Union Pacific rail line is also seen in Figures 2-40 and 2-41.

The Evaporation Ponds consisted of a primary Pond and a larger, secondary/overflow Pond with a 1.5 foot berm around each Pond. The Ponds were originally lined with approximately 10 in. of rock (nominal diameter of 0.5 to 3 inches). The size of the primary Pond was approximately 30 ft by 40 ft, and the secondary Pond was 30 ft by 85 ft. While the Evaporation Ponds were designed and built to receive filtrates from the low enrichment processes, they were also used for the retention of both high and low enrichment recovery waste liquids. Historical documentation also indicates retention of other liquid waste solutions in the Evaporation Ponds. Examples of these waste liquids include acidic cleanup solutions, organic solvent solutions (perchloroethylene and trichloroethylene), oils, building sump contents, and mop water. The precipitates and solids were allowed to settle and the water evaporated naturally. As additional liquids were added to the primary Pond, the overflow flowed through a pipe into the secondary Pond.

After CE purchased the Hematite facility in 1974, use of the Evaporation Ponds was curtailed to allow only the retention of spent potassium hydroxide scrubber solution from the Uranium dry recycle process and liquids from startup testing of the wet recovery process. Use of the Ponds was discontinued altogether in September, 1978. In 1979, 700 ft^3 of sludge was pumped from the primary Pond. The sludge was dried and shipped to a licensed burial facility between 1982, and 1984. Additional decommissioning efforts for the Evaporation Ponds were undertaken by CE in 1984, in response to NRC directives. As a result, in 1985, CE removed approximately 2,800 ft^3 of sludge, rock, and soil from the primary Evaporation Pond. Detailed sampling following the remediation effort determined the average total Uranium contamination of soil in the Evaporation Pond was below the 250 pCi/g total Uranium decontamination limit set by the NRC, however, spot contamination levels in excess of the limit remained. Approximately 1,200 ft^3 of soil and rock were also removed from the secondary Evaporation Pond in 1987. Subsequent soil/sediment samples collected from the Evaporation Ponds following these remediation efforts revealed an average concentration of Uranium in the Evaporation Ponds below the 250 pCi/g limit; however, individual sample results showed that soil/sediment contamination levels in excess of the limit remained.

On May 4, 1995, a decommissioning plan for the Evaporation Ponds was incorporated by amendment into the site license. Following additional characterization of the Evaporation Ponds, this decommissioning plan was revised based on more extensive characterization results (References 8.17 and 8.18). The Evaporation Ponds' decommissioning plan was implemented over the next four years and resulted in the removal of approximately 6,000 ft^3 of additional soil/sediment for disposal. Surveys and sampling of the Evaporation Pond area conducted in 1999 indicated an average concentration of 170 pCi/g U-235, with several samples yielding higher results, up to 745 pCi/g U-235. In addition, Uranium concentrations of approximately 100 pCi/g were detected at depths of 10 ft below ground surface (Reference 8.19). Remediation efforts associated with the Evaporation Ponds were suspended in 1999, to evaluate additional remediation techniques and options.

Because of the known levels of contamination in the area of the Evaporation Ponds, additional characterization was performed between 2004, and 2014. The objective of these characterizations was to fully understand the extent of the contamination to ensure that excavation and remediation could be conducted safely and effectively. A total of 136 samples were collected from the surface to depths of over 30 feet. Because the NGP line runs through the Evaporation Ponds, remediation activities in this area created a potential risk of damage to the 58 year old line. While this risk was low, the resulting consequences if the natural gas were ignited would be unacceptably high when considering the safety of workers and members of the public. The primary concern with excavating near the NGP was the need to provide a required slope (1:1) to any excavation, or use some type of side-wall shoring. The deeper the excavation the further out a slope would need to extend to ensure required worker safety. To achieve the same level of remediation of the Evaporation Ponds as conducted elsewhere on-site, the depth of the excavation would require sloping that could potentially undermine the NGP.

Based on the results of samples collected from 2004, to 2014, it was determined that in most cases, identified contamination was between the surface and a depth of 10 feet and could be remediated without significant concern for the NGP. However, several sample locations at depths of 20 to 30 feet showed Tc-99 in excess of the DCGL. The maximum value identified was 221 pCi/g, at a depth of 26 to 28 feet, in the southwest portion of the Evaporation Ponds (Sample location EP-2014-6). (Tc-99 was the only nuclide of concern that was identified in excess of the DCGLs at depth in the vicinity of the Evaporation Ponds.) Due to these sample locations' close proximity to the NGP, normal excavation practices were not possible to reach that depth while maintaining a 1:1 sidewall slope to ensure worker safety. For this reason an additional systematic sampling initiative was undertaken for the Evaporation Pond area in April, and May, 2015, which included collecting samples where the previously high Tc-99 samples had been identified at depth.

The initiative was comprised of 21 core borings in a systematic grid and nine additional core borings located where previous sample results had shown elevated Tc-99 contamination. (Figure 2-41a shows the locations of the core borings that were conducted during the initiative.) The borings were conducted to collect soil samples in four foot increments to a depth of 35 feet or until refusal (e.g., bedrock), which typically occurred at a depth close to 32 feet. The grid itself was laid over LSA 08-11, which had been previously adjusted to cover just the area of the

Evaporation Ponds. Sample EP-15-27 was collected in the same location as EP-2014-6, where the sample containing 221 pCi/g of Tc-99 was taken. At a depth of 16-20 feet, EP-15-27 showed 82.0 pCi/g of Tc-99. In the 20-24 and 24-28 foot sections the results showed 4.34 and 4.19 pCi/g respectively. The sections of soil above 16 feet were all between 2 and 3 pCi/g. The value of 82.0 pCi/g was the highest sample collected from the total 212 samples collected during the initiative. Core bore EP-15-07, which was drilled very close to EP-15-27 showed 26.0 pCi/g at 16-20 feet. The second highest reading collected was 50.50 pCi/g from EP-15-04, also at 16-20 feet. EP-15-04 was southwest of EP-15-07 and EP-15-27, and at the southwest end of the Evaporation Pond. Sample locations EP-15-01, -02, -05, -08, -11, -28, -29 and -30, all to the southwest, south and southeast of EP-15-04, -07 and -27 (the prevailing direction of groundwater flow) were less than or equal to 14.0 pCi/g

While this initiative was not the final FSS for LSA 08-11, it was performed with the intent that the data would be used in support of demonstrating that the dose from LSA 08-11 would be less than 25 mrem. Additional detail supporting the release of LSA 08-11 is provided in the FSSFR for LSA 08-11.

2.5.2 Remediation

With the information provided by the April, and May, 2015, core bore sample results the extent of excavation for the Evaporation Ponds was further defined as well as providing the necessary information to ensure safe remediation near the natural gas pipe line. Remediation was completed in the same manner as the other LSAs at HDP. Figure 2-42 shows the excavated Evaporation Ponds.

2.6 Natural Gas Pipeline Area

2.6.1 History

An 8 inch diameter high pressure natural gas transmission pipe line runs the length of the southeast (SE) border of the Hematite Site. The NGP was installed in 1956, and is a sole source of natural gas for local communities. In 2008, Westinghouse used air-knife equipment to unearth and positively identify the precise location and depth of the NGP at approximately 40 ft. intervals. Markers were installed at these locations to provide surface level visual indications of the buried NGP. Civil Survey techniques were used to record coordinates and the surface elevation and depth of the NGP at each interval. The location of the NGP is identified in Figure 2-43 by the depths documented in the diagram. The top value is ground surface and the bottom value is top of the NGP.

In Westinghouse letter HEM-11-96 (Reference 8.19), Westinghouse made the commitment to contact the Laclede Gas Company, owners of the NGP, in advance of excavation and/or sampling within five feet of the NGP to discuss any necessary precautions or controls. On November 6, 2014, HDP staff met with Laclede Gas Company representatives to get their input on safely working around the NGP. The Laclede Gas Company representatives considered that excavating down to the top of the NGP was acceptable. The first choice of the representatives for protecting the structural integrity of the NGP was to not excavate if at all possible closer than two feet of the sides and not under the NGP. If excavation was to be conducted within two feet of the sides of the NGP, then an appropriate 1:1 slope should be employed. If excavation could not be avoided within that structural support area for the NGP, then shoring of the NGP would be required at a minimum of every 20 ft.

Soil impacted by former Site operations was located near the NGP. Specifically, the Hematite facility had two former evaporation ponds that were used for the retention of process filtrates, low-level liquid wastes, and high- and low-enriched Uranium-containing materials. Figures 2-40 and 2-41 show the proximity of the Evaporation Ponds to the NGP. In this area radiological contamination was identified close to the NGP on the northwest side of the NGP as expected, due to the Evaporation Ponds, but also on the SE side between the NGP and the rail line. Contamination requiring remediation SE of the NGP was close enough to the surface to enable removal through excavation while maintaining a required 1:1 sloping ratio for sloping sidewalls.

Early in the project there was a concern in regards to how remediation could successfully be accomplished near the NGP, as characterization data identified Tc-99 contamination at depth where the Evaporation Ponds were located, adjacent to the NGP. To excavate to that depth and maintain a 1:1 sloping ratio would not have been feasible that close to the NGP, and might have required shoring or other means to enable remediation. However, subsequent core bore sampling determined that the extent of contamination in the Evaporation Pond area was much less than originally believed, which allowed excavation and remediation activities to proceed in a normal manner and without potential risk to the NGP.

2.6.2 Remediation

For removal of contamination close to the NGP, hand digging with shovels was conducted. The Laclede Gas Company representatives were onsite during portions of the excavation activities around the NGP. Only a few small areas of surface soil contamination were identified elsewhere along the NGP and were remediated. Figure 2-44 is a photograph of remediation activities at the Natural Gas Pipeline.

2.7 Barns Area and Red Room Roof Burial Area

2.7.1 History

The Barns Area consisted of two barns, one wood and one tile, and two concrete silos. Figure 2-26 shows the location of the barns in relation to the entire site (lower right corner). Figure 2-45 shows the barns prior to demolition. The Wood Barn (Building 120) existed on the property prior to purchase by Mallinckrodt. It had a dirt floor and was used to store both clean and contaminated equipment throughout the facility's operating period. The Tile Barn (Building 101) also existed on the property prior to purchase by Mallinckrodt and was a former dairy barn. It was designated as a temporary storage facility and was used to store both clean and radiologically contaminated equipment. The Tile Barn's main floor was concrete. The walls were constructed of hollow tile to an elevation about 12 feet off the floor. This building was also used to store emergency equipment during the commercial nuclear phase of operations. The silos were not used in conjunction with any licensed activities.

The Area 240-2, Red Room area of Building 240, was used for UF_6 conversion of highly-enriched Uranium. The Red Room roof was replaced in the late 1984-85 timeframe and the old roof was buried in an area south of the Tile Barn, otherwise known as the Red Room Roof Burial area. The Cistern Burn Pit area, southwest of Building 101 and adjacent to the Red Room Roof Burial area, was used to burn wood pallets that may have been contaminated. This general area was also known to have been used for temporary storage of scrap materials. In 1993, soil contamination was discovered in the Red Room Roof Burial/Cistern Burn Pit area during renovations to the Tile Barn.

All equipment, materials, and debris were removed from the Barns and disposed prior to demolition.

2.7.2 Demolition

Demolition of the Barns began in late March, 2011, and the Barns were the first buildings to be demolished as part of the building demolition project. All Barn building demolition debris was loaded onto trucks and shipped offsite for disposal. To minimize the potential for the spreading of contaminated dust, water was sprayed on the debris as demolition was conducted.

FSSFR Volume 4, Chapter 1, Building Survey Areas Overview, provides photographs and a further discussion regarding the history, demolition and disposal of the Barns.

2.7.3 Remediation

In December, 2012, the concrete slab comprising the floor of the Tile Barn was removed. Upon removal of the floor slab a previous foundation was discovered, indicating that the Tile Barn had replaced another barn/structure on the same location. In December, 2012, excavation and remediation of soils in the Barns Area was started. These areas were not subject to the same criticality controls as the burial pits because radiological characterization work performed had shown little to no fissile material present in these areas. Therefore, the survey and soil excavation process was different than that of the Burial Pit Area. Overall no widespread contamination of the soil in the Barns Area was identified. Localized hot spots and some fuel pellets were identified and remediated; otherwise the area was generally clean. Following soil remediation, a FSS was performed.

Subsequent to the FSS the NRC with support from ORAU conducted a confirmatory survey of the Barns Area. The NRC survey results were documented in NRC Inspection Reports 07000036/13001 {ML13154A125} and 07000036/2014001 {ML14084A566}. The reports stated, in part, that the plan and instructions for survey unit LSA-05-01 were in accordance with MARSSIM and the DP, and the results for walkover surveys, concrete surface scans and activity measurements, and radionuclide concentration in soil samples and smears were consistent with the license and below the DCGLs, with a few exceptions in LSA-05-01, in which three samples exceeded the DCGLs. The NRC Inspector who was on-site and also performing surveys during FSS of the area discussed these results with the HDP RSO and determined the HDP soil sample results were also elevated in those same areas. HDP had remediated to the extent possible in those specific areas per the direction of the Missouri Department of Transportation. LSA 05-01 contains a portion of the base of Missouri State Road P and any further excavation could have destabilized the roadway.

Based on the HDP sample results and knowledge that the NRC's samples results were consistent with the on-site survey results it was determined that an EMC evaluation would be performed for the survey unit that would demonstrate compliance with the release criteria. To facilitate remediation in the southeasterly section of LSA 05-01, while completing the work necessary to return stability back to the base of Missouri State Road P, the survey unit was separated into two individual Class 1 survey units. The section that contains the base of Missouri State Road P remained designated as LSA 05-01 and the section yet to be completed was designated LSA 05-04. To restore stability to the roadway LSA 05-01 was immediately backfilled after the completion of FSS. One layer of backfill soil came from Reuse Stockpile 2 (which will be discussed in FSSFR Volume 2, Chapter 2, regarding the stockpiled soils) and the rest of the backfill was brought in from offsite (and will be included in FSSFR Volume 2, Chapter 9, regarding the off-site borrow soils). Figure 2-52, which was taken during backfill of the Barns Areas shows the location of LSA 05-01, LSA 05-02, LSA 05-03 and LSA 05-04.

The Red Room Roof Burial/Cistern Burn Pit Area was remediated following the remediation of the Barns Area. Remediation proceeded as expected with no widespread contamination of the soil in the Red Room Roof Burial/Cistern Burn Pit Area identified. Figure 2-53 shows remediation of the Red Room Roof Burial Area.

2.8 Sanitary Wastewater Treatment Plant

2.8.1 History

The original Sewage Treatment System was designed such that drains inside buildings were directed to a buried holding (septic) tank connected to a leach field. Liquid wastes from personnel showers, mop water, and small spills in process areas were directed to various floor drains leading to the septic tank and leach field of the former system. Between 1977, and 1978, the septic tank and leach field were bypassed and abandoned in place. The modified Sewage Treatment System was connected to new wastewater treatment equipment located just northwest of the Evaporation Ponds and designated as the Sanitary Wastewater Treatment Plant (SWTP). This modified SWTP utilized the existing Sewage Treatment System discharge pipe and discharged to the Site Creek just below the dam. Degradation of the buried SWTP discharge pipe was identified in 2007, during remediation, when it was discovered that no flow existed at the discharge outfall effluent sampling point. Degradation of the effluent pipe had progressed to the point that the majority of the liquid effluent entering this line did not reach the discharge point at the Site Creek. Evidence of subsurface contamination indicated that liquids from the degraded pipe leaked through cracks or breaks in the pipe, resulting in effluent migrating into the surrounding soils. Since the effluent of the SWTP may have contained residual radioactivity (within approved regulatory release limits), remediation of this system and associated effluent piping and adjacent soil was addressed during decommissioning.

2.8.2 Remediation

Figure 2-54 shows the concrete slab covering the location of the abandoned septic tank. Below this slab was the septic tank with a separate prefabricated concrete cover that was removed and disposed. Due to the close proximity of the septic tank to the foundation of Building 230, it was only partially removed so the structural integrity of the building would not be compromised. All piping associated with the original septic tank and leach field was removed as part of remediation. Figure 2-55 shows the interior of the septic tank. The construction of the septic tank was such that should there have been any transport of a discrete quantity of SNM it would have settled to the bottom of the septic tank. The septic tank was remediated using conventional methods such as power washing and scabbling. Radiological surveys and sampling of the septic tank only identified low levels of fixed residual contamination with no removable contamination identified. No discrete or recoverable quantities of SNM were found during remediation of the septic tank. FSS surveys conducted on the portion of the septic tank that remains showed 17% of the DCGL for systematic samples and 50% to 90% of the DCGL for biased samples. All smears were less than the MDA. The radiological data for the septic tank (BSA 04-02) will be included in the survey area release record for LSA 08-17, the survey unit in which it resides.

Figure 2-56 shows the excavation of the leach field in the lower portion of the photograph. The discharge pipe from the leach field and the SWTP went southwest (towards the upper left in the photo) under the concrete slab, and to the south (left) of the smaller building (Building 231). The line continued to the Site Creek where it discharged just below the dam (Outfall 1). Between Building 231 and the Site Creek the discharge line passed under the NGP. Between the leach field/SWTP and the concrete slab, the discharge line was removed. Between the Site Creek and the NGP the discharge line was also removed. The remaining portion of the line was left intact. This section, which could be accessed via a manhole in the concrete slab as well as the ends, was power washed. The radiological data for the SWTP Discharge Line (SAN-1) will be included in the survey area release record for LSA 08-10, LSA 08-15 and LSA 04-04, the survey units in which it resides.

Figure 2-57 shows the SWTP Tanks prior to demolition and removal, located between Building 230 to the north (left) and the Evaporation Ponds to the south (right). Figure 2-58 shows the SWTP Tanks exposed after excavating around them prior to demolition. The original intent was to decontaminate the tanks and dispose of them generally intact at US Ecology Idaho. However, to adequately characterize and decontaminate the interior of the tanks would have required a difficult confined space entry that raised numerous safety concerns. Based on commitments to the NRC, the interior of the tanks required a systematic grid sampling of concrete samples of the walls to obtain sufficient radiological data to demonstrate the material would meet the waste acceptance criteria (WAC) at US Ecology. This would have been time consuming and the structural integrity of the walls was questionable. It was therefore determined it was cost effective to simply demolish the tanks in place and ship the rubble to EnergySolutions in Utah, which had waste acceptance criteria that would accommodate disposal at that location, although at a higher cost of disposal, but eliminating the need to enter the tanks. Figure 2-59 shows the excavation remaining after demolition and removal of the SWTP Tanks.

2.9 Site Pond/Site Creek Area

2.9.1 History

The Site Pond is fed by a natural spring located on the northwest portion of the site and generally contains water year around. The Site Pond receives storm water runoff from the plant area. The Site Creek is the effluent from below the dam of the Site Pond, and receives discharge from the Sanitary Wastewater Treatment Plant (SWTP) directly below the dam. It flows through a culvert beneath the railroad track, and joins the effluent from the Lake Virginia drainage basin; the combined stream discharges to Joachim Creek. The Site Creek normally flows year round. Figure 2-60 is a photograph of the Site Pond prior to remediation.

In 1995, it was identified that occasional upsets in the operation of the SWTP over a period of time had resulted in contamination collecting in the Site Creek sediments. The contamination sediment had settled between the dam and the point where the Site Creek passes beneath the railroad tracks. Prior to remediation, sediment samples showed total Uranium concentrations within the range of 40 pCi/g to 800 pCi/g.

A Water Treatment System (WTS) was maintained onsite during decommissioning to process potentially contaminated water that resulted from decommissioning operations, from precipitation that entered work areas, or from excavations that encountered ground water. Collected water was directed to the WTS for analysis, and treatment as required. A portion of the system was comprised of an ion exchange resin bed to remove metals, including uranium and Tc-99 that may have been in solution and not removed by the granulated activated carbon filters in the system. On August 16, 2014, the presence of resin beads was identified in Tank T-5 of the WTS, the final holding tank before water was discharged into the Site Pond (through Outfall #003a). In addition, a small amount of resin beads was identified in a water sample collected at Outfall #003a. Upon discovery, a Stop Work Order was issued and all water treatment and discharge activities were paused until the problem was located and the WTS component that had failed was taken out of service. Restoration of the system included resin recovery efforts up to the point of discharge into the Site Pond. HDP routinely updated the NRC on recovery of the system and resin recovery. An assessment of the WTS estimated that 1.627 cubic feet of resin was unrecoverable and determined to be in the Site Pond as it would not travel downstream into the Site Creek due to the presence of the Site Pond Dam. This volume of resin was evaluated to contain below the NCS limit of 0.1 g/L U-235. This volume of resin would be recovered during the remediation of the Site Pond and disposed, along with sediment.

NRC Inspection Report 07000036/2014005(DNMS) dated February 20, 2015, (Reference 8.21), stated in part that, "During the collection of water and sediment radiological sampling, neither the licensee nor the NRC identified resin beads downstream of the outfall separating the Site Pond from the Site Creek which feeds into Joachim Creek. The NRC Inspector completed walk-over surveys and collected soil/sediment and water samples of potentially impacted areas. Walk-over surveys/sample results did not identify abnormal radiation levels or radioactive concentrations. Consequently, the NRC determined that the failure of the resin-tank did not result in any significant release of resins, and associated licensed material that may have been collected on it, to be released from the site."

2.9.2 Remediation

Remediation of the Site Creek was accomplished by diverting the Creek and then removing the sediment with a backhoe to a depth of approximately 0.5 ft to 3 ft between the site dam and the railroad tracks. The removed material was dried and shipped to an offsite licensed disposal facility. Sediment was removed until the average remaining contamination was less than 30 pCi/g, with no single sample above 90 pCi/g. Remaining residual radioactivity after remediation of the Site Creek averaged 22 pCi/g, with a maximum concentration of 85 pCi/g. Samples taken at the confluence of the Site Creek and Joachim Creek indicated contamination had not extended to Joachim Creek.

Remediation of the Site Pond and surrounding area required diverting inflow to the Site Pond, followed by draining, excavating, and removing sediments and soil. A water-inflow bypass basin was constructed to divert the Site Spring and storm water discharge during remediation of the Site Pond. Figure 2-61 shows the construction of the bypass diversion. Once the inflow-bypass system was operating and diverting the inflow of the Site Spring and storm water discharge, the Site Pond was drained. Site Pond water was drawn down in accordance with site discharge permits, sampled, and processed as necessary for discharge under the Water Management Plan. Figure 2-62 shows the Site Pond near the completion of excavation and remediation.

Remediation of the concrete dam required limited decontamination (power washing and scabbling) to meet the appropriate DCGLs and RGs. The radiological data for the Site Dam (BSA 04-01) will be included in the survey area release record for LSA 02-03, the survey unit in which it resides.

2.9.3 Site Pond Post FSS Event

FSS had been completed in Site Pond survey unit LSA 02-01 in late July, 2015, with isolation and control of the survey unit being maintained in accordance with site procedures. In summary, on or about August 30, 2015, a significant un-forecasted rain event occurred and moved 15 radiologically contaminated items from LSA 05-04, into adjacent LSA 02-01. As remediation in LSA 05-04 had not yet been completed and is upstream of LSA 02-01, it was determined that the contaminated items originated in LSA 05-04. The contaminated items were removed from LSA 02-01 and the affected area resurveyed. All downstream LSAs were resurveyed to verify the absence of radiologically contaminated items in early September, 2015.

On November 27, 2015, the NRC issued a Notice of Violation in regards to the event (Reference 8.31). In the Notice of Violation was a request for Westinghouse to complete an extent of condition in regards to the possibility of other instances in which radiologically contaminated items could have been transferred into a survey unit in which FSS had been completed. Westinghouse letter HEM-15-131, (Reference 8.32) which was a response to the Notice of Violation, also included the extent of condition investigation results. The extent of condition assessment concluded that there were no radiologically contaminated items residing in any previously FSS completed survey units. The NRC reviewed the extent of condition and in letter dated January 19, 2016 (Reference 8.33), acknowledged receipt of the assessment, review of the assessment and that there were no further questions at that time. Additional information in regards to the event will be provided in the survey area release record for LSA 02-01.

2.10 Tc-99 Areas

Technetium-99 was identified as a radionuclide of concern at Hematite. A historical review of site operations determined that the only source of the Tc-99 was as a contaminant in Department of Energy supplied UF6 originating from reprocessed/recycled spent nuclear fuels and not part of site operations. In 1967, five dry scrubber columns were installed for removal of hydrogen fluoride from the off-gas associated with the conversion of uranium hexafluoride (UF6) to uranium oxide (UO2). These dry scrubber columns used limestone rock chips as the off-gas scrubber media. The limestone media was periodically replaced and the waste limestone stored outside or utilized as onsite fill material, including locations in the northeast portion of the Burial Pit Area. While the limestone was used as an off-gas scrubber media, it also had an affinity for Tc-99. During Hematite operations, the limestone scrubber media became contaminated with Tc-99. Therefore, the primary areas impacted by Tc-99 were those locations where the waste limestone was disposed following removal from the scrubbers.

During excavation activities the identification of Tc-99 contamination was challenging and slowed the overall excavation and remediation process since field instrumentation did not exist for the measurement of Tc-99. As a result, the identification and quantification of Tc-99 had to be based on a laboratory analysis of soil samples. The typical turnaround time from collection to receipt of laboratory results took seven days.

Because the limestone waste from the scrubbers was the source of the Tc-99, all the limestone was removed prior to completion of excavation, RASS, and FSS activities. Limestone was identified to have been stored in four areas onsite: in the Documented Burial Pit Area, as fill under the Process Building before the building was constructed, west of the Red Roof Barn Burial Area, and in a pile just northeast of the central tract area of the site. (See Figure 2-23)

2.10.1 Tc-99 in Documented Burial Pit Area

As discussed in Section 2.1 "Retrospective Review", a review was conducted of the Burial Pit Area surface and sub-surface soil sampling data summarized in the Hematite Radiological Characterization Report. The maximum Tc-99 sample result collected in the Burial Pit Area was 68.3 pCi/g collected within the surface stratum. None of the results were in excess of the Tc-99 Excavation DCGL value of 74.0 pCi/g, and regardless, many of the areas where these samples were collected were also remediated.

Between 2004, and 2008, various core boring characterizations were conducted throughout the Burial Pit Area. A review of this data identified 106 samples that were characterized as collected at a depth of 16 feet or greater and sampled for Tc-99. Out of these samples the highest value was only 9.34 pCi/g. The next highest was 4.99 pCi/g. From over 300 samples collected at all depths, the highest value of Tc-99 was 68.3 pCi/g. Only five samples exceeded the Uniform $DCGL_W$ of 25.1 pCi/g, and four of these were at the surface while the fifth was at 6.5 feet. In other words, there was no expectation that Tc-99 had leached off the limestone and migrated throughout the Burial Pit Area and into the groundwater.

In spite of the expectation that Tc-99 contamination was not a significant problem, thousands of soil samples were collected and analyzed for Tc-99 during the excavation and remediation of the Burial Pit Area. In looking at the "as left condition", a total of 246 FSS samples were collected in the Burial Pit Area and analyzed for Tc-99. Every sample result was less than their applicable $DCGL_W$, with an average Tc-99 activity of 0.33 pCi/g, and a maximum result of 19.1 pCi/g (collected in LSA 10-12).

In summary, based on sample results, it has been determined that there is very little probability that Tc-99 could exist at levels that exceed the excavation DCGL at a depth below where excavation activities ceased because there was very little Tc-99 that existed in the majority of the Burial Pit Area to begin with. The Tc-99 that was identified, as expected, was predominately in the areas where limestone had existed. In addition, all areas where spent limestone was encountered were ultimately excavated.

2.10.2 Tc-99 Used As Fill Under the Process Building Before the Building Was Constructed

Based on historical records and interviews, non-native material was introduced under the Process Building 253. As part of preparing for the construction of Building 253, native soil was excavated due to the presence of gross alpha radioactivity exceeding 30 pCi/g and spent limestone was used as backfill. Because of concerns about undermining Building 240, the excavation was stopped before all soil exceeding this limit was removed. The average alpha concentration in soil of the surface at the conclusion of this excavation was 17 pCi/g. The maximum alpha concentration was 82 pCi/g.

A May 14, 1990, letter to the NRC, *Building 253 Construction Site Soil* (Reference 8.23), includes a figure that provides the locations within the excavation that were selected for soil sampling and subsequent analysis for gross alpha activity concentration. Per this letter, the excavation area reached a maximum depth of approximately six feet below the planned floor level. Based on this description, the maximum depth of the excavation was likely to have been approximately eight to ten feet. However, later interviews with personnel that were present when Building 253 was constructed indicated that the excavation depth may have extended to a depth of 10-15 feet.

Prior to placing backfill, Combustion Engineering requested that NRC allow spent limestone that was stored on-site to be used as backfill material within the excavation area under Building 253. In an NRC letter to ASEA Brown Boveri/Combustion Engineering, Authorizes Backfill of Area of Bldg. 253 Construction Site, dated July 2, 1990, (Reference 8.24), the NRC authorized spent limestone from two piles meeting a 30 pCi/g limit (alpha) to be used as fill below Building 253 with the understanding that the fill may have to be removed upon facility decommissioning. Figure 2-63 shows the location of the limestone backfill. Per a Combustion Engineering letter to the NRC, Spent Limestone Results, (Reference 8.25), dated April 7, 1989, the two limestone piles had average gross alpha concentrations of 7 pCi/g and 8 pCi/g.

Due to the presence of residual radioactivity in soil that could not previously be removed, and due to the presence of spent limestone as backfill, these areas were excavated to remove the limestone and the underlying soil until the approved DCGLs were met.

2.10.3 Tc-99 West of the Red Roof Barn Burial Area

A deposit of spent limestone was located west of the Barns, near State Road P. This area was remediated and the limestone sent to US Ecology for disposal. The limestone encroached upon State Road P, and during portions of the remediation, one lane of State Road P had to be closed (see Section 2.7 discussion). The overall depth of the limestone was approximately 4 – 6 feet below original grade. The limestone was removed to the maximum extent practical; however the extent of excavation activities was limited due to Missouri Department of Transportation (MDOT) restrictions which were put in place to prevent potentially undermining the integrity of State Road P. Soil sampling was performed both vertically downward from the surface and horizontally into the bank of the roadway in order to accurately measure the "as left" condition of the area. This evaluation will be discussed in detail further in the LSA 05-01 Survey Area Release Record.

2.11 Class 1, Class 2 and Class 3 Survey Areas

To allow a more concentrated survey effort in the areas likely to be contaminated, impacted survey areas are subdivided into Class 1, Class 2, or Class 3 survey units (SU). At Hematite, survey units were contiguous areas with similar characteristics and contamination potential. Survey units were 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 was consistent with the exposure pathway modeling used to convert residual radioactivity into dose.

The suggested maximum survey unit sizes by classification are recommended by MARSSIM (Ref. 8.5). This Guidance has been taken into consideration when delineating survey units; and as appropriate, were 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 up to 2,200 m^2 versus the MARSSIM recommended size of 2,000 m^2, the area was considered only one survey unit.

Open Land survey units were designed to have compact shapes rather than highly irregular (gerrymandered) shapes unless unusual shapes were practical given appropriate site operational history or site topography. Flexibility was also maintained to modify survey units based changing site conditions and information. Although these boundaries were altered in several instances, the classification for the purpose of final status survey was never reduced from the original classification.

Survey results were converted to appropriate units of measure and compared to investigation levels to determine if the action levels for investigation had been exceeded. Measurements exceeding the investigation action levels were investigated. If confirmed within a Class 1 survey unit, the location of an elevated concentration was evaluated using an elevated measurement comparison, or the location was remediated and re-surveyed. However, in a few instances survey results were confirmed to exceed the investigation level within a Class 2 or Class 3 survey unit, which required reclassification and a re-survey to be performed consistent with the change in classification. Although this occurred infrequently, it did occur on several occasions. Those occurrences are discussed in the FSSFR chapters specific to those survey units.

Survey units were 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 DCGLw. 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 DCGLw. 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 DCGLw. 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 DCGLw, 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.

2.12 Waste Disposal

The HDP radioactive waste management program was designed to safely control the handling, packaging, transport, and disposal of solid wastes generated during decommissioning activities. Site activities were performed in accordance with the Waste Management and Transportation Plan (Reference 8.22). The WMTP is an integrated plan for management of radioactive and non-radioactive waste, and is designed to protect the personnel, public, and environment during the generation, handling, and transportation of waste. As waste was exposed during remediation activities, additional waste characterization and initial segregation steps were implemented. After removal of the waste, final characterization and segregation of waste into the appropriate waste type was completed. Based upon waste type, waste was either directly loaded into containers or stockpiled awaiting packaging, treatment, or transportation. To prevent the spread of contamination during storage, piles were maintained utilizing Best Management Practices (BMPs). Packaged waste was transported using approved, qualified, and permitted carriers.

HDP used both rail service and truck/trailer as means of transportation of radioactive waste. For both modes of transportation, verification was performed to ensure carriers were permitted to carry the load, and that appropriate security plans were in effect. Pre-transportation checklists from approved procedures were used to ensure compliance with applicable United States Department of Transportation (DOT) and NRC regulations.

The majority of the solid radioactive waste generated during decommissioning was associated with excavation activities. The two general types of solid radioactive waste generated were:

- Demolition debris such as concrete rubble, building materials, piping, conduit, and exhumed Burial Pit waste; and,
- Volumetrically contaminated material such as soil, sediment, charcoal, resin, and limestone.

Solid radioactive waste generated by the project was Class A waste.

The specific isotopes and activity associated with the solid radioactive waste was dependent on the location where the waste was generated:

- The Burial Pit area was contaminated primarily with uranium isotopes: U-234, U-235 and U-238, including the associated decay daughter products; and to a lesser extent Ra-226, and Th-232.
- Radiologically impacted soil areas included the Barn Area, Red Room Roof Burial Area, Site Pond, Site Creek, and Leach Field Areas, and were contaminated primarily with the uranium isotopes. A portion of these areas was also contaminated to a lesser extent with Tc-99.
- The area southeast of and under the processing buildings was contaminated primarily with uranium isotopes, their associated decay daughter products, and Tc-99.

Radioactive waste handling was accomplished primarily with mechanical equipment such as excavators, front-end loaders, and trucks. However, there were occasions when smaller equipment and/or hand shoveling was employed.

For areas where it was determined that there was a reasonable possibility of finding fissile materials based on characterization data and historical knowledge, HDP developed consistent generic screening and handling approaches. Relative to the removal of buried wastes, contaminated soils, and sub-surface structures (e.g., concrete slabs and buried piping), these generic screening and handling approaches were analyzed from a NCS perspective in NCSAs specific to buried waste exhumation and contaminated soil remediation, and sub-surface structure decommissioning. Screening for fissile materials during remediation was the initial goal. Screening typically involved duplicate and independent performance of radiological surveys, using sodium iodide scintillation detectors, and defined volumes of material to ensure that NCS limits had not been exceeded. The objective of the in-situ radiological surveys was to identify any item or region of soil/waste with a fissile concentration exceeding 1 gram U-235 in any contiguous 10 liter volume. The 1 gram U-235/10L threshold provided a high degree of assurance that any items with elevated (i.e., nontrivial) levels of U-235 contamination would be identified. The in-situ radiological surveys were complemented by visual inspection of the survey area with the aim of identifying: 1) Items with the potential to contain fissile material (e.g., a process filter); 2) Items that resemble intact containers; 3) Bulky objects with linear dimensions exceeding the permitted excavation '*cut depth*'; and, 4) Metallic items.

During fissile material screening, the secondary remedial action objectives of identifying hazardous materials (e.g., VOCs), and verifying radioactivity concentrations of potential backfill soils were also completed. Excavation continued in areas of suspected fissile materials to a depth where historical knowledge, and/or visible and radiological evidence indicated that suspect materials had been removed. Once fissile material screening determined that fissile materials were not present in the remediation area in excess of the NCS Exempt Material Limit, NCS controls were then curtailed. By making this the initial goal, continued excavation, Health Physics, and Final Status Survey activities could proceed unencumbered by NCS controls. Identified items that exceeded NCS limits were segregated into designated Field Containers, which were placed in transport containers such as Collared Drums (CDs). Individual designated containers were handled and stored in accordance with NCSAs.

To assess and manage low level radioactive waste (LLRW), field screening was performed to establish an initial material classification, followed by sampling and analysis to validate the field screening classification. The general sequence for excavation, removal, and handling of LLRW was as follows:

- Excavated LLRW was loaded directly into haul trucks for transfer to the WCA or stockpiled until a sufficient quantity was available for transport to the WCA for final visual inspection and assay by High Resolution Gamma Spectroscopy (HRGS); and,
- LLRW was sent to the WHA for stockpiling, loading, verification of compliance with WAC, and subsequent transportation to off-site disposal facilities.

Transportation for off-site disposal was generally by gondola cars; however, alternate conveyances which met the requirements of the Waste Management and Transportation Plan were also utilized. Solid radioactive waste was shipped for processing and/or disposal to an appropriately licensed or permitted facility. The following facilities were used for solid radioactive waste disposition: EnergySolutions, Inc., Oak Ridge, TN; EnergySolutions, Inc., Clive, UT; U S Ecology Idaho, Grandview, ID; and Waste Control Specialists, Andrews TX.

Regardless of the facility selected for processing and disposal, waste was prepared for transport in accordance with the receiving facility's WAC, facility license, or NRC approved exemption, and the DOT regulations.

2.13 Backfill Operations

Site restoration included backfilling excavated areas, compacting to a standard proctor, spreading topsoil, reseeding, and removing temporary features that impeded final site restoration. Excavated soil determined to be below the appropriate DCGLs, and meeting other regulatory requirements for re-use, was used as backfill material. However, additional off-site backfill material was imported from off-site source(s) to provide sufficient soil to meet site cover requirements for radiological and chemical constituents. Grading was performed to achieve pre-remediation contours to the maximum extent practical. Adjustments were made to the grade to mitigate the potential for surface water to pool over the remediated site. Reseeding of backfilled areas was performed with a MDNR approved seed mixture to limit the potential for erosion. Topsoil was placed above backfill material in areas to be seeded, and was cultivated and graded to ensure a smooth, uniform grade with positive drainage towards wetland areas. Winter rye seed and/or other MDNR approved cover were utilized.

Due to the significantly large volume of off-site borrow material required to backfill excavations, Westinghouse identified multiple sources of off-site borrow. In regards to off-site borrow material the DP states:

- DP Section 8.8 - "*Additional off-site backfill material will be imported from an approved off-site source(s), as needed, and tested to ensure it meets site cover requirements for radiological and chemical constituents.*"
- DP Section 14.4.4.1.6.2 - "*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.....*"

Based upon the above stated DP requirements, HDP performed a lengthy process of radiological testing, statistical analyses, and documentation of the off-site borrow material prior to NRC approval. That effort is documented in FSSFR Volume 2 Chapter 1 (HDP-RPT-FSS-100). Subsequently, in NRC (Norato) letter to Westinghouse (Fussell) dated October 8, 2015, *U.S. Nuclear Regulatory Commission Conclusions Associated with the Utilization of Off-site Borrow Material at the Westinghouse Hematite Site*, (Reference 8.34) the NRC provided approval of the off-site borrow referenced in HEM-15-39 (the Horine Road site) (Reference 8.35). The NRC also stated that "*The conclusions presented in this letter are also applicable to any other source of off-site borrow material. Westinghouse informed the NRC on September 24, 2015, that they have procured access to two other sources of off-site borrow. Westinghouse should consider the conclusions contained in this letter as they may apply to these two resources.*"

In conducting the radiological remediation of the HDP site, approximately 180,000 cubic yards of soil was excavated and shipped offsite for disposal or retained for reuse. To replace this soil, several locations near the HDP site were approved where offsite soil could be obtained and used to supplement onsite stockpiled soil that had been segregated for reuse. Each of these locations represented native Missouri soil from farm fields. To complete backfilling the excavated areas onsite, approximately 150,000 cubic yards of offsite borrow soil was brought to the site from these locations by truck and used as backfill.

2.13.1 Management of Soil Used as Backfill

There are two categories of soil used to backfill excavations and complete final grade contouring of the site. The primary type and source of soil used is Off-site Borrow as discussed above. The second type and source is reuse soil which is soil that has been determined to have met the unrestricted release criteria. FSSFR Volume 2, Chapter 1, Reuse Soil and Off-site Borrow Material Overview, provides a detailed discussion on the methodology utilized to determine that both off-site borrow material and reuse soil is acceptable for use as backfill.

As off-site borrow material has been determined to contain no radiological contamination there is no restriction on placement of this material.

Although the reuse soil generated on the site has been determined to be acceptable for unrestricted release, the fact that it does contain residual radioactive contamination necessitates managing the dose associated with each individual reuse stockpile in relation to the total dose for the survey unit. In addition, placement of the reuse soil within a survey unit and also in which stratum the soil is placed must be considered.

Placement of reuse soil and off-site borrow is managed by the use of Work Package HDP-WP-ENG-802, Backfill & Restoration. In summary, the process for placement of reuse soil is as follows; the Radiation Safety Officer (RSO) evaluates the FSS data and the expected dose contribution from all sources for all survey unit(s) to be backfilled; based upon the evaluation the RSO selects the appropriate survey unit and if required the stratum in which the reuse soil is to be placed ensuring the total dose for the survey unit(s) will meet the release criteria; the onsite backfill placement tracking forms are completed which provide the survey unit(s) and depth of excavation in which the reuse soil will be placed; and, in addition to the directions provided by the tracking form to ensure proper placement of the reuse soil, the placement of the reuse soil and the associated elevations are verified by topographical measurements by HDP Engineering.

3.0 RELEASE CRITERIA

In order to demonstrate that the site meets requirements for unrestricted site release, site-specific release criteria or Derived Concentration Guideline Levels (DCGLs) were developed using dose modeling. The DCGLs represented isotope-specific release criteria. Because multiple radionuclides were present at the same time in varying quantities, the dose contribution from each radionuclide was considered such that the total dose from all radionuclides did not exceed the dose based limit. Section 5.1 of FSSFR Volume 1, Chapter 1 provides a discussion of the radionuclides of significance at Hematite.

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.

Surface or volumetric concentrations that correspond to the maximum annual dose criterion are established for the average residual radioactivity in a survey unit and are called the $DCGL_W$. 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.

Conceptual Site Models (CSMs) were developed for soil and the surfaces of remaining buildings. The critical groups and exposure pathways were identified and described. Dose model parameters were selected and sensitivity analyses performed. DCGLs were then calculated for soil and building surfaces. The soil DCGLs are specific to a given CSM and will result in a Total Effective Dose Equivalent (TEDE) of 25 mrem/yr to the average member of the critical group for that CSM. For the building surface DCGLs, two room sizes were considered for the DCGL calculations, representing a small office and an open warehouse. The Small Office CSM resulted in the most limiting DCGLs. Considering the very low levels of residual surface contamination present in the buildings and the limited effort that would be required to reduce surface contamination to acceptable levels, the DCGLs based on the Small Office CSM were used for all building surfaces.

As open land area and building remediation activities were completed, FSSs were conducted to demonstrate that the dose from residual radioactivity at the HDP Site did not exceed the annual dose criterion for license termination for unrestricted use specified in 10 CFR 20.1402 (Reference 8.26), and that the levels of residual radioactivity were ALARA. FSS was performed on all impacted open land areas and structures, systems, and components (SSCs) that will remain at the time of license termination. 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 were to: select/verify survey unit classification; demonstrate that the potential dose from residual radioactivity was below the release criterion for each survey unit; and, demonstrate that the potential dose from small areas of elevated radioactivity were below the release criterion for each survey unit.

Documentation of the FSS provides a FSS Survey Area Release Record for each survey unit. A FSS Final Report (FSSFR Volume 7, Chapter 1) will be prepared that includes the survey area release records (Chapters), and provides a summary of the survey results and the overall conclusions that demonstrate the site meets the radiological criteria for unrestricted use.

3.1 LSA Release Criteria

The soil contamination, including surface and subsurface, was modeled using two different source term geometries; 1) a soil column with three distinct layers that represent different exposure pathways and depths, and 2) a soil column with uniform contamination over the entire depth of the Contaminated Zone. Therefore, the surface and subsurface contamination is represented by four different CSMs (RESRAD models). The first subsurface geometry assumes a soil column that is comprised of three stratums as follows:

Surface - surface soil to a depth of 15 cm below the ground surface;

Root - subsurface soil starting at 15 cm and extending to 1.5 m below the ground surface to include the entire root stratum; and

Deep - subsurface soil located below 1.5 m (i.e., below the root stratum) and extending to the bottom of the Contaminated Zone which was conservatively estimated to be 6.7 m below the ground surface.

The Surface stratum represents the typical surface contamination configuration, i.e., the top 15 cm of soil. The Root stratum represents soil in the root zone (15 cm to 1.5 meters) and accounts for the potential removal of soil due to erosion over a 1,000-year period. The root depth is assumed to be 0.9 m and potential erosion over 100 years is estimated to be 0.6 m. Using the combined 1.5 m depth ensures that the thickness of the root stratum will equal or exceed the 0.9 m root depth for an entire 1,000-year period. The Deep stratum represents soil below the root stratum starting at a depth of 1.5 meters and extending to the bottom of the Contaminated Zone, which was conservatively estimated to be 6.7 m deep.

The second subsurface geometry, Uniform, is comprised of one stratum of uniform soil contamination from the ground surface to the bottom of the Contaminated Zone (6.7 m).

DCGLs were calculated for each of the four CSMs discussed above including Shallow, Root, Deep and Uniform. The four CSMs were designed to address the various configurations that may be present during remediation and at the time of the FSS. DCGLs were also calculated for an excavation scenario CSM to evaluate the effect of changing the in-situ soil configuration after license termination.

The Hematite Radiological Characterization Report (HRCR) (Reference 8.30) discusses the characterization efforts from the several characterization survey campaigns and compiles the data into one report. The data were reviewed, separated into logical areas, and evaluated against the proposed soil and building surface DCGLs. The data sets were tabulated and basic statistics calculated. The data sets were evaluated to determine preliminary classification (Class 1, Class 2, or Class 3) for the impacted areas in accordance with guidance from MARSSIM (NUREG 1575). The soil data were also evaluated to estimate the lateral and vertical depth of the contamination.

Table 3-1 presents the site-specific DCGLs as developed for soil for all four CSMs.

3.1.1 Uniform DCGLs

The subsurface geometry for Uniform DCGLs is comprised of one stratum of uniform soil contamination from the ground surface to the bottom of the Contaminated Zone (6.7 m). Demonstration of compliance with the Uniform DCGL is simply a comparison of the DCGL to the average concentration of residual contamination regardless of the depth of the contamination.

DCGL values were initially derived for building surfaces in 2005 and for site soil in 2006. The modeling and resulting DCGL values for building surfaces and site soil were reviewed and revised in Chapter 5 of the HDP. The DCGLs were approved by the NRC with issuance of License Amendment 57 on October 13, 2011.

To help expedite the speed and efficiency of the FSSR process, Westinghouse decided that the Uniform DCGL would be used when evaluating the results of the LSA final status surveys unless otherwise approved by the RSO. It was believed that as a result of the overall quality of the soil remediation that was conducted the vast majority of the LSAs would meet the Uniform DCGL criteria. In the few instances where the three stratum approach would be necessary (surface, root, deep), the RSO would approve the deviation from using the Uniform DCGLs.

3.1.2 Three Stratum DCGLs

Compliance with the "three layer" geometry requires consideration of the Surface, Root, and Deep layers independently. After the original DP submittal and approval, Westinghouse agreed with the NRC that the Deep DCGLs should not be used as they were not protective of the intruder scenario, and the Excavation DCGLs were developed as a replacement. Only the Excavation DCGLs will be used when evaluating the Deep layer as part of the "three layer" approach. Because each of the three DCGLs (Surface, Root, Excavation) represent 25 mrem/yr from each layer independently, the unity rule was used to demonstrate compliance when contamination was present in more than one soil layer. In some cases, less than three layers were present, for example, when there was no contamination below the depth of 1.5 m. In this case, compliance was demonstrated using the unity rule for the Surface and Root DCGLs only.

Overall, the Uniform DCGLs were applied in 60 of the 69 LSAs while the "Three Stratum" DCGLs were applied in nine of the 69 LSAs. Of those nine LSAs, LSAs 09-02, 09-03, 10-12, 08-09, 08-11, 08-12 and 08-13 received offsite soil as backfill. As a result the Surface Stratum would have no dose contribution in those LSAs. In the remaining two LSAs, 08-01 and 08-02, reuse soil from Stockpile 4-7 was placed (only) in the Deep Stratum.

The following is a hypothetical example (see diagram below) of the "3 layer" approach which includes showing how dose from on-site reuse soil will be added into the survey unit that is being evaluated. This hypothetical example would assume that one stockpile was used as backfill within the survey unit, the stockpile was evaluated against the Uniform $DCGL_W$, and the stockpile exceeded the Tc-99 Uniform MIL therefore placement was restricted to the Deep stratum

To simplify the discussion this hypothetical SU is assumed to have no elevated areas that exceed a SOF of 1.0, so no EMC evaluation is necessary, and also assumed to have no remaining subterranean piping. Therefore the dose contributions for these are each set to zero.

As a conservative measure, and to greatly simplify the calculation, area weighting factors will not be used for on-site reuse soil. When an on-site reuse soil stockpile is used in a survey unit, the *entire dose* for that stockpile will be attributed to that survey unit, even if the entire stockpile is not used.

The attached diagram presents a cross sectional view of this hypothetical survey unit, and the following calculations sum the total potential dose for each layer. Please note that the drawing is not to scale, and is only intended as a visual aid for this hypothetical discussion. In the diagram, the Surface stratum has been entirely removed, and only the Root stratum and Deep stratum remain. Using the typical minimum of 8 systematic soil sample locations that would fall in a Class 1 Survey Unit, we assume that 2 of the 8 samples fall in the remaining Root stratum, and all 8 samples are also collected in the remaining Deep stratum.

1. Remaining Surface SOF is 0.0 (layer totally removed)

   Offsite borrow soil is used to backfill the surface stratum.

   Surface stratum dose contribution to SU is 0.00 SOF

2. Remaining Root stratum contribution is (2/8)x(0.20)=0.05

   Offsite borrow soil is used to backfill the Root stratum.

   Root stratum dose contribution to SU is 0.05 SOF

3. Remaining Deep stratum contribution to SU is (8/8) x (0.15) = 0.15

   The onsite reuse soil stockpile with a 0.15 Uniform SOF is used, all dose from the stockpile is conservatively assigned to SU, even if entire stockpile is not used. Use of the Uniform DCGL when restricting the stockpile to the Deep stratum only is also conservative.

   Deep stratum dose contribution to SU is 0.15 + 0.15 = 0.30 SOF

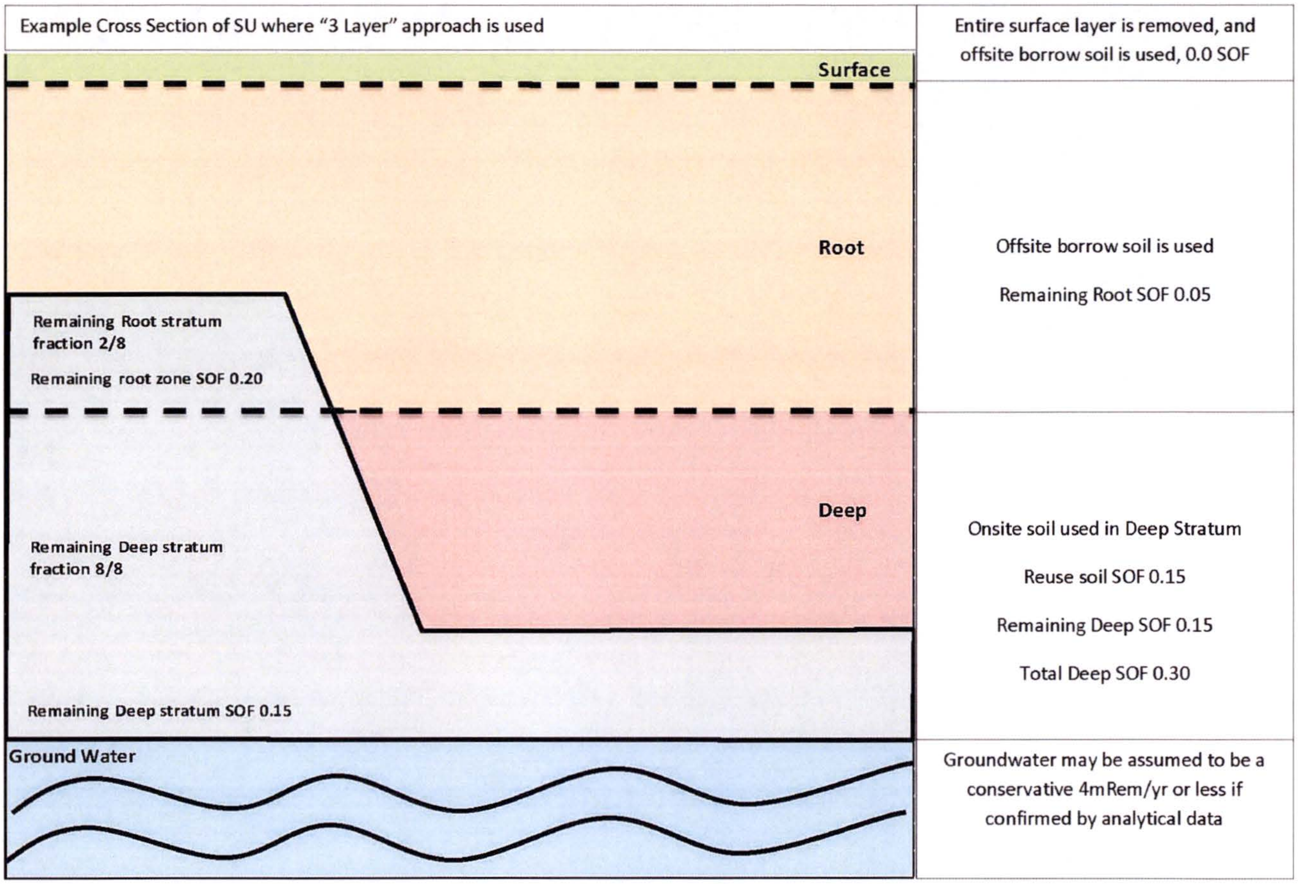
4. Groundwater is assumed to be a conservative 4 mrem per year (0.16 SOF) unless post remediation analytical sampling is available and the results have proven that the groundwater contribution can be accurately assigned a lower value.

5. Total SU dose is determined by SUM of all SOF's

SU Total SOF = Total Surface SOF + Total Root SOF + Total Deep SOF + Ground Water + EMC Dose +Piping

And in this hypothetical case:

SU Total SOF = 0.00 (total surface) + 0.05 (total root) + 0.30 (total deep) + 0.16 (groundwater) + 0.0 (no EMC dose) + 0.0 (no piping dose) = 0.51 SOF or 12.75 mrem/year



3.1.3 Elevated Areas

To address small areas of elevated radioactivity in a survey unit a simple comparison to an investigation level is used to assess the impact of potentially elevated areas rather than using statistical methods. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an Area Factor (AF) to account for the small area of the elevated radioactivity. An area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. (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 soil were developed by using the CSMs and adjusting the size of the contaminated zone. The AFs for the Surface, Root, and Uniform Soil strata are provided in Table 3-2a. The AFs for the Excavation CSM (and corresponding Deep $DCGL_W$) are provided in Table 3-2b and 3-2c.

The areas factors were developed for each CSM by adjusting the size of the Contaminated Zone and dividing the resulting DCGL for the smaller areas of activity ($DCGL_{EMC}$) by the applicable site DCGL. AFs for the Surface, Root, and Uniform CSMs were determined for contaminated zone sizes ranging from 1 to 10,000 m^2, although it is not anticipated that AFs for areas greater than 300 m^2 will be utilized.

AFs for the Excavation CSM were calculated for contaminated zone sizes ranging from 1 to 100 m^2 in accordance with the MARSSIM using the RESRAD parameters which were modified for use with the excavation scenario (as detailed in Section 5.3.6). However, unlike for the development of the $DCGL_W$ values presented in Table 3-1, the elevated activity was not assumed to be mixed with the clean cover. Instead, the elevated activity was assumed to be excavated intact and brought to the surface, with the only modification to "flatten" the material from an assumed depth of 1.5 m to the excavation scenario depth of 0.9 m. The excavation scenario area factors are subject to the following constraints:

1. The excavation scenario for a small area of elevated activity must account for the increase in area after being excavated to the surface. An adjustment factor of 1.67 (1.5/0.9) was applied during modeling for geometrical transformation between the assumed excavation geometry depth (1.5 m) and the geometry modeled in RESRAD (0.9 m). For example, an elevated area of 1 m^2 and a depth of 1.5 m was assumed to be excavated to the surface and cover an area of 1.67 m^2 (1 $m^2 \times 1.67$) and a depth of 0.9 m. The modeled area factors are presented in Table 3-2b with the listed areas representing the post-excavation condition. This constraint limits the concentration within an area of elevated activity which remains contiguous during excavation to less than the $DCGL_{EMC}$.

2. The excavation scenario for a small area of elevated activity must also account for the fact that the scenario's excavation footprint is 200 m^2 and residual activity cannot exceed the $DCGL_W$ for post-excavation configurations that exceed 200 m^2. Therefore the calculated area factor is limited to the quotient 200 m^2 divided by the extent of the elevated area in square meters. This constraint limits the weighted average concentration over each 200 m^2 area to the $DCGL_W$.

The resultant area factors based on the two constraints discussed above are shown in Table 3-2c. The area factor selected for each radionuclide and area combination is the smallest of the values calculated based on each of the two constraints. Table 3-2b is provided for informational purposes only as Table 3-2c will always be used to determine AFs for use with the Excavation $DCGL_W$.

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 $DCGL_{EMC}$. The Elevated Measurement Comparison (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 DCGLw 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. 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.

3.2 Demonstrating Compliance with the Dose Criteria

3.2.1 Average SU Soil Dose

The average radioactivity within the survey unit will be determined from the systematic sampling and measurement results, excluding all biased measurements and any measurements (systematic or biased) within an elevated area, this is to ensure the proper statistical testing of the survey data without skewing the results of the evaluation. Biased sample results are excluded as these were not randomly selected.

$$f_{Avg} = \sum_{j=1}^{x} \frac{\delta_j}{DCGL_{w_j}} \qquad \text{Equation 3-1}$$

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.

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:

$$Average\ SOF_{Weighted} = f_{SZ} \sum_{i=1}^{n} \left(\frac{\overline{C}_{i,SZ}}{D_{i,SZ}} \right) + f_{RZ} \sum_{i=1}^{n} \left(\frac{\overline{C}_{i,RZ}}{D_{i,RZ}} \right) + f_{DZ} \sum_{i=1}^{n} \left(\frac{\overline{C}_{i,DZ}}{D_{i,DZ}} \right) \qquad \text{Equation 3-2}$$

where:

n	=	Number of measured ROCs;
f_{SZ}	=	Fraction of survey unit area at the Surface stratum depth;
$\overline{C}_{i,SZ}$	=	Average concentration of ith measured ROCs in the Surface stratum layer;

$D_{i,\,SZ}$	=	Surface stratum $DCGL_W$ for the *i*th measured ROCs;
f_{RZ}	=	Fraction of survey unit area at the Root stratum depth;
$\overline{C}_{i,\,RZ}$	=	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.

3.2.2 Elevated Area Dose

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 higher and lower AFs will be determined from Table 3-3 and Table 3-2 for building and structural surfaces and soil, respectively, and an exponential interpolation will be performed using the following equation:

$$e^{\left[\frac{(\ln(actual\ area)-\ln(lower\ area))(\ln(higher\ AF)-\ln(lower\ AF))}{\ln(higher\ AF)-\ln(lower\ AF)}+\ln(lower\ AF)\right]}$$

Equation 3-3

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 (Reference 8.5).

The additional dose fraction or contribution from each elevated area will be determined by calculating the average radioactivity within the elevated area (from biased and any systematic samples within the elevated area), subtracting the average radioactivity of the survey unit (from systematic samples not within the elevated area), 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.

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

Equation 3-4

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.

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. However, it is not anticipated that this will be necessary.

3.2.3 Groundwater Dose

Groundwater monitoring was performed prior to and during soil remediation efforts, and provides relevant data in regards to the anticipated groundwater conditions post remediation. FSSFR Volume 6, Chapter 1, Groundwater, {ML16041A340} describes the post remediation groundwater monitoring that is being conducted at HDP.

In summary, to support the NRC review of LSA survey area release records prior to the completion of the post remediation groundwater monitoring, the SOF for groundwater will be set at the conservative value of 0.16 in the survey area release records for each LSA. This value is based upon groundwater not exceeding the EPA drinking water standard of 4 millirem/year. To date, sampling within the sand/gravel, Jefferson City Cotter, and Roubidoux HSUs, which has been ongoing quarterly since June 2007, has not shown any contamination exceeding the EPA drinking water standard thus providing that the use of 0.16 SOF for groundwater is appropriate.

3.2.4 Buried Piping/Structure Dose

Buried piping that will remain at the time of License termination is also subject to FSS for the purpose of determining the additional dose, if any, that is required to be added to a particular survey unit total dose determination.

FSSFR Volume 5, Chapter 1, Piping Survey Areas (PSA) Overview, {ML16076A312} provides the specific details in regards to the survey methodology and evaluation to demonstrate that the piping is acceptable for unrestricted release. The dose determined for a piping survey area will be determined prior to submitting a survey area release record for a LSA in which the piping is located. The dose determined for each piping survey area is applied to each survey unit SOF in which the piping is located.

3.2.5 Reuse Soil Dose

A key component of remedial actions within open land areas of the site was to identify and separate soil that could be used for backfill (reuse soil) from soil that exceeded site cleanup criteria and had to be disposed as waste. FSSFR Volume 2, Chapter 1, Reuse Soil and Off-site Borrow Material Overview provides the specific details in regards to the history, survey methodology and Reuse Soil Release Criteria for each Reuse Stockpile of soil that was generated during remediation. In summary, based upon the FSS data for each individual Reuse Stockpile, an associated dose is derived for each individual Reuse Stockpile. The dose determined for a Reuse Soil Stockpile will be applied to each survey unit SOF in which re-use soil is placed based upon the use of either the Uniform or Three Stratum DCGL and its placement within the stratum.

3.2.6 Off-site Borrow Dose

FSSFR Volume 2, Chapter 1, Reuse Soil and Off-site Borrow Material Overview provides the specific details in regards to the historical sample methodology and evaluation to demonstrate that the soil from the various off-site borrow locations was not radiologically contaminated. The SOF dose contribution for soil from the off-site borrow used as backfill will be the value of 0.00 as offsite borrow soil has been determined to be non-radiologically contaminated soil.

3.2.7 Total Dose

Once all the dose contributions are determined, as described in the above sections, the SOFs are applied as follows:

$$f_{Avg} + f_{EMC} + f_{GROUNDWATER} + f_{SOIL-OFFSITE} + f_{SOIL-REUSE} + f_{PIPING\,/\,STRUCTURE} \leq 1 \qquad \text{Equation 3-5}$$

Provided the SOF is less than or equal to unity (1), the survey unit is acceptable for unrestricted release.

FSSFR Volume 7, Chapter 1, Summary Report, will contain the summary report of all FSS data used to demonstrate compliance with 10 CFR 20.1402. FSSFR Volume 7, Chapter 1 will be compiled upon the completion of the post remediation groundwater monitoring period sample and analysis activity as described in FSSFR Volume 6, Chapter 1, Groundwater. At that time the groundwater SOF, as determined by post remediation groundwater monitoring sample result evaluation and dose determination, will be incorporated into the Total Dose calculation by replacing the 0.16 value.

4.0 DATA QUALITY OBJECTIVES (DQO)

The DQO process is thoroughly integrated within the DP and Hematite FSS procedures. The steps of the DQO process are most explicitly detailed in Section 9 of FSS procedure HDP-PO-FSS-700 *Final Status Survey Program* and correspond to the DQO steps described in

Chapter 14, Section 4.2.1 of the DP. The HDP DQO process reflects the recommendations given in MARSSIM, Chapter 2, Figure 2-2.

The FSS process consists of the following principal elements to which the DQO are applied: Planning and Design, Implementation, and Data Assessment. DQO allow for systematic planning, address situations that require a decision to be made, and provide a framework for selecting actions that result in obtaining data of sufficient quantity and quality. The DQO process is iterative and allows for incorporation of newly gained knowledge to enhance the effectiveness of subsequent actions. The seven steps of the DQO process are as follows:

1. State the problem,
2. Identify the decision,
3. Identify inputs to the decision,
4. Define the study boundaries,
5. Develop a decision rule,
6. Specify limits on decision errors, and
7. Optimize the design for obtaining data.

The DQO process is described below as it applies to the HDP.

1. State the Problem

The problem is the presence of residual radioactive material associated with previous licensed activities at HDP. The primary radionuclides of concern (ROC) and the extent of contamination were assessed in the HAS and the HRCR. The primary ROC are uranium-234 (U-234), uranium-235 (U-235+D), uranium-238 (U-238+D), technetium-99 (Tc-99), and thorium-232 (Th-232+C). Additionally, trace amounts of americium-241 (Am-241), neptunium-237 (Np-237+D) and plutonium-239/240 (Pu-239/240) may be present; however the latter are insignificant contributors to potential dose. Although Radium-226 (Ra-226+C) was identified as an ROC site-wide, it was only identified as an ROC due to radium plates found in the Documented Burial Pit Area. NOTE: The nomenclature "+D" above indicates that the dose contribution of short-lived progeny is accounted for by the parent nuclide and "+C" indicates that the dose contribution of the entire decay chain (progeny) in secular equilibrium is accounted for by the parent nuclide.

2. Identify the Decision

For the FSS, the principal study question is "Is residual radioactive contamination in the survey unit present in quantities which exceed the established $DCGL_W$ values?" The FSS Program is used to demonstrate that the HDP site meets the criteria for unrestricted release specified in 10 CFR 20.1402 of 25 millirem/year total effective dose equivalent (TEDE). Compliance with the release criteria is satisfied using the guidance in MARSSIM. The DP Section 14.4.2.1.1 and HDP-PO-FSS-700 Section 9.1 provide a discussion on the DCGLs, Unity Rule, Area Factors, and Background Measurements needed in order to make the decision relevant to the question "Does the survey unit meet the criteria for release?"

3. Identify Inputs to the Decision

Guidance provided in MARSSIM is the basis for this FSS Program. Inputs include sources of historical information, results of field and laboratory measurements, limitations on detectability, and the acceptable risk of a decision error. These inputs will be provided in each survey area release record. Survey Area Release Records are generated in accordance with HDP-PR-FSS-722, *Final Status Survey Reporting.*

4. Define the Study Boundaries

For the HDP FSS, 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. The initial site designations based on the potential for residual contamination, the determination of survey units based on size and similar characteristics, and survey unit classification, all contribute to defining the study boundaries. The boundaries for each individual survey unit are identified and described in the FSS Plan for each survey unit and provided in the Survey Area Release Record.

5. Develop a Decision Rule

The decision rule is the determination of whether residual radioactivity exceeds the established $DCGL_W$ values. If the SOF is less than, or equal to any applicable action limit and unity (1), then no additional investigation is required and the survey unit will be recommended for unrestricted release. If the SOF is greater than unity (1), then the RSO is consulted to determine the appropriate action(s). Potential actions include re-classification, additional data collection, and additional remediation. To implement the decision rule, HDP-PR-FSS-721, *Final Status Survey Data Evaluation* provides the information necessary to calculate the sum-of-fractions. To ensure the DQO process has been properly implemented, a checklist is provided in HDP-PR-FSS-721, Appendix G-1, *Final Status Survey Data Quality Objectives Checklist.* In addition, HDP-PR-FSS-721 states the following: "The purpose of this procedure is to provide guidance to interpret survey results using the Data Quality Assessment (DQA) process during the assessment phase of Final Status Survey (FSS) activities in support of the Hematite Decommissioning Project."

6. Specify Limits on 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 (alternate hypothesis) as defined and shown below:

- Null Hypothesis (H_0) – the survey unit does not meet the release criterion; or,
- Alternate Hypothesis (H_a) – the survey unit does meet the release criterion.

A Type 1 decision error (α) 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 α value will always be set at 0.05 unless prior NRC approval is granted for using a less restrictive value.

A Type II decision error (β) 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 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.

7. Optimize the Design for Obtaining Data

The results of characterization and/or Remedial Action Support Surveys (RASS) are evaluated and used to optimize the FSS design and ensure the DQOs are met. The RASS data and/or characterization data may include gamma scans, surface scanning surveys (alpha + beta), soil sampling, and surface activity measurements in impacted soil areas and SSCs. This data is evaluated and used to refine the scope of field activities to optimize implementation of the FSS design and ensure the DQOs are met. HDP-PR-FSS-701, *Final Status Survey Plan Development* provides the instructions to evaluate the data and refine the scope of field activities. This is accomplished in concert with HDP-PR-HP-601, *Remedial Action Support Surveys*.

5.0 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, 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 DP Chapter 7. 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 was 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;
- NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM);
- NUREG-1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions; and,
- NUREG-1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys.

Guidance for conducting an FSS on piping internals is outside the scope of MARSSIM. These special situations will be evaluated by judgment sampling and measurements. 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. FSSFR Volume 5, Chapter 1, Piping Areas Surveys contains further discussion regarding FSS of piping.

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;
- Design;
- Implementation; and,
- Data Assessment

The DQO and 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). For additional discussion on the DQA process see section 6.3.

Survey planning includes review of the Historical Site Assessment, 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 10 CFR 20.1402 (Reference 8.26) 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.

Before the FSS process can proceed to the implementation phase, turnover and control measures will be implemented for an area or survey unit as appropriate. A formal turnover process will ensure that decommissioning activities have been completed and that the area or survey unit is in a suitable physical condition for FSS implementation. Isolation and control measures are primarily used to limit the potential for cross-contamination from other decommissioning activities and to maintain the final configuration of the area or survey unit.

Survey implementation is the process of carrying out the survey plan for a given survey unit. This consists of scan measurements, total surface contamination measurements, and collection and analysis of samples. Quality assurance and control measures are employed throughout the FSS process to ensure that subsequent decisions are made on the basis of data of acceptable quality. Quality assurance and control measures are applied to ensure:

- DQOs are properly defined and derived;
- the plan is correctly implemented as prescribed;
- data and samples are collected by individuals with the proper training using approved procedures;
- instruments are properly calibrated and source checked;
- collected data are validated, recorded, and stored in accordance with approved procedures;
- documents are properly maintained; and,
- corrective actions are prescribed, implemented and followed up, if necessary.

The DQA approach is applied to FSS results to ensure the population of the data are complete, the data are valid, and to determine whether the objectives of the FSS have been met. The data quality assessment includes:

- verify that the measurements were obtained using approved methods;
- verify that the quality requirements for the methods were met;
- verify that the appropriate corrections were made to the gross measurements and the data are expressed in proper reporting units;
- verify that the measurements required by the survey design, and any measurements required to support investigation have been included;

- verify that the classification and associated survey unit design remain appropriate based on a preliminary review of the data;
- subject the measurement results to the appropriate statistical tests;
- determine if the residual radioactivity levels in the unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

In some cases, data evaluation will show that all of the measurements made in a given survey unit were below the applicable $DCGL_W$. If so, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the $DCGL_W$ are observed. In these cases, statistical tests must be performed to determine whether the survey unit meets the release criterion. The statistical tests that may be required to make decisions regarding the residual radioactivity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

The statistical tests will include the Sign test, or the Wilcoxon Rank Sum (WRS) test for instances when the measurement results are corrected for the contribution from background radioactivity. Typically, the use of the WRS test will be limited to the evaluation of results obtained within open land surveys where Ra-226 and Th-232 are identified in soil. The balance of the measurements of soil within open land areas, and 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 be reclassified and a re-survey performed consistent with the change in classification.

5.1 Surrogate Evaluation Areas (SEAs)

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 was performed to determine scaling factors that could be used in the FSS planning process by inferring the concentration of one or more radionuclides by the measurement of a surrogate radionuclide. Using the surrogate relationship during the FSS planning process allows Westinghouse to estimate a reduced instrument scan MDC that accounts for the presence of hard to detect nuclides. This reduced instrument scan MDC is then used to calculate prospective soil sample population size to ensure that a sufficient number of soil samples are collected during the initial FSS if instrument scan sensitivities are not sufficient. This reduces the risk of collecting an insufficient number of samples that would require re-sampling of the area, and helps to ensure that all DQO's for the survey area are met.

The surrogate study 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. The table below provides the distribution ratios for the use of U-235 as a surrogate to infer the Tc-99 concentration in soil within three 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 5-1.

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 5-1 that was adjusted for the contributions from insignificant radionuclides was further modified. This calculation was performed using Equation 4-1 of MARSSIM, the result for the Surface Soil stratum in the Plant Soil SEA using the distribution ratio of 9.24 is illustrated below:

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 8.37

$$DCGL_{U-235,\mathrm{mod}} = \frac{1}{\left(\frac{1}{102.3}\right)+\left(\frac{9.24}{151.0}\right)} = 14.1 \text{ pCi/g} \qquad \text{Equation 5-1}$$

where: 102.3 = adjusted and modified soil $DCGL_W$ (pCi/g) for U-235 (from Table 3-1), and
151.0 = adjusted and modified soil $DCGL_W$ (pCi/g) for Tc-99 (from Table 3-1).

Surrogate Evaluation Areas, and the distribution ratios above are utilized in FSS planning only. Tc-99 is analyzed in all FSS samples to demonstrate compliance with the dose based unrestricted release criteria.

5.2 Tc-99 Side Wall Sampling

During the commencement of FSS activities in early 2015, during a NRC Region III inspection, the NRC Inspector questioned the site staff in regards to the FSS program requirements for excavation sidewall sampling. The NRC Inspector was specifically interested in sampling for Tc-99. The site staff reiterated the requirements as provided in the HDP DP Chapter 14.4.4.1.6.2, Subsurface Soil, and provided an explanation of how the requirements were implemented within the FSS program and procedures. This topic was conveyed to NRC Headquarters and HDP was subsequently provided with three options for addressing the issue.

Westinghouse opted to provide a sampling plan for excavation sidewalls. The following excerpt from the NRC – Westinghouse August 11 and 12, 2015, Publicly Noticed Teleconference Summary {ML15230A324}, summarizes resolution of the issue;

> "*Westinghouse described a plan that it developed for sampling the sidewalls. This plan was documented in an internal memo (HEM-15-MEMO-039). In this plan, the sidewalls were sampled if any sample in the survey unit had a concentration of greater than 2.5 pCi/g of Tc-99, which corresponds to 10% of the DCGL. Westinghouse said that this was based on insignificant radionuclides being defined in the DP as those which contribute less than 10% to the 25 mrem/yr dose criteria. The NRC staff questioned whether there would be a correlation between the Tc-99 concentration at the bottom of an excavation and the sidewall. Westinghouse responded that the concentration of Tc-99 in sidewall samples taken to date agree with the systematic samples and that the sidewalls were no more likely to contain elevated Tc-99.*
>
> *NRC staff concluded that the method described in HEM-15-MEMO-039 was acceptable, with the exception that the NRC staff had concerns with the first bullet (i.e., samples would only be taken if the systematic or biased samples from the survey unit exceeded 10% of the applicable DCGLw). Westinghouse committed to revising the memo to delete this bullet and to revise its procedure to include this information. NRC staff also noted that it would also be clearer if a definition of "vertical or near vertical" were included in these documents.*

Westinghouse stated that this process would be used in the future and had been used in the past. Westinghouse noted, however, that there are some survey units that have already been backfilled for which this process was not followed. These survey units had shallower excavations and had low Tc-99. Westinghouse will provide justification for the characterization of Tc-99 in the FSSRs for these survey units. NRC staff stated that it will evaluate the information available for those survey units when those FSS reports are submitted to determine if they have been characterized adequately."

As a result the FSS procedures were revised to include judgmental sidewall samples for Tc-99 using guidance below:

- If vertical or near vertical sidewalls exist within the survey unit (i.e., surfaces that are greater than a 45 degree angle), and
- If those sidewalls exceed 12 inches in height vertically, and
- If those sidewalls exceed (in aggregate) 5 % of the total survey unit surface area (e.g., greater than 100 m^2 of sidewall in a 2,000 m^2 survey unit), then discretionary sidewall sampling is necessary.

Additional discussion is provided in section 7.2.3.

6.0 FINAL STATUS SURVEY

Near the conclusion of remediation activities and prior to initiating a final status survey, isolation and control measures were implemented. The determination of readiness for controls and the preparation for final status survey were based on the results of characterization and/or a RASS that indicated residual radioactivity was unlikely to exceed the DCGLs. The control measures were implemented to ensure the final radiological conditions were not compromised by the potential for re-contamination as a result of access by personnel or equipment. These measures consisted of both physical and administrative controls. Examples of the physical controls included rope boundaries and postings indicating that access was restricted to only those persons authorized to enter by health physics. Administrative controls included approved procedures and personnel training on the limitations and requirements for access to areas under these controls.

6.1 Gamma Walk Over Surveys

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.

The intent of the 100% GWS of accessible surfaces performed in Class 1 areas was to provide a basis that the survey unit met the remedial goal. The GWS was performed and documented in a manner that met the DQOs of the FSS program.

6.1.1 Instrumentation

The data quality objectives process included the selection of instrumentation appropriate for the type of measurement to be performed (i.e., total surface contamination measurement, scan or both), that were 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 was made to identify the ROC at levels sufficiently below the DCGL. Detector selection was based upon detection sensitivity, operating characteristics, and expected performance in the field. The specific DQOs for instruments were 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.

Instruments and detectors were calibrated for the radiation types and energies of interest or to a conservative energy source. Calibration was performed on-site using HDP procedures or off-site by an approved vendor. Instrument calibrations were documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels were also attached to all portable survey instruments. Prior to using any survey instrument, the current calibration was verified and all operational checks were performed.

Radioactive sources used for calibration were traceable to the National Institute of Standards and Technology (NIST) and were obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector was used, suitable NIST-traceable sources were used for calibration, and the software set up appropriately for the desired geometry.

Prior to use on-site, all project instrument calibrations were verified and initial response data collected. These initial measurements were used to establish performance standards (response ranges) in which the instruments were tested against on a daily basis when in use. An acceptable response for field instrumentation was an instrument reading within ±20 percent of the established check source value. Laboratory instrumentation standards were within ±3-sigma as documented on a control chart.

The DQO process determined the frequency of response checks, typically before issue and after an instrument had been used (typically at the end of the work day but in some cases this was performed during an established break in activity, e.g., lunch). This additional check was to expedite the identification of a potential problem before continued use in the field. Instrumentation was response checked in accordance with HDP Site procedures. If the instrument response did not fall within the established range, the instrument was removed from use until the reason for the deviation could be resolved and acceptable response again demonstrated. If the instrument failed a post-survey source check, all data collected during that time period with the instrument was carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that FSS data were discarded, replacement data would be collected at the original locations. However, during the course of FSS surveys at the site, no instruments failed a post-survey source check, and therefore no FSS data was subject to potential failure.

6.1.2 Scan MDC

One of the most important elements of a scan survey is to 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.

If the survey instrument scan MDC is less than the applicable $DCGL_W$ for the stratum (elevation) in which the soil resides, then scanning is the primary method for guiding the remediation. The average net count rate corresponding to the $DCGL_W$ is 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.

$$MDCR_{surveyor} = \frac{1.38\sqrt{10{,}000 \times \frac{1}{60} \times \left(\frac{60}{1}\right)}}{\sqrt{0.5}} = 1{,}512\ cpm \qquad \text{Equation 6-1}$$

The table below 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 for various Uranium enrichments provided in the table below are sufficient to measure concentrations at the $DCGL_W$ for field instruments used for scanning.

Instrument/Detector Type	**Typical Background**	**Typical MDC 95 Percent Confidence Level**
Ludlum Model 2360/Ludlum 44-10 or equivalent 2 in by 2 in NaI scintillation detector	**10,000 cpm**	**84 pCi/g (3 percent enriched Uranium)** **99 pCi/g (20 percent enriched Uranium)** **122 pCi/g (50 percent enriched Uranium)** **140 pCi/g (75 percent enriched Uranium)**

The scan MDC for open land areas may be reduced further by using field instrumentation coupled with a GPS unit there by enabling the scan data to be logged, downloaded, and mapped. By logging and mapping the data, the scan data can 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, specific areas may be more readily identified for further investigation, survey, and sampling as necessary. This technology reduces the impact of surveyor efficiency (e.g., surveyor efficiency of 0.5 without data logging, surveyor efficiency of 0.75 when data logging is used), thereby reducing the scan MDC by approximately 17 percent when GPS data logging technology is used.

6.1.3 Investigation Action Level (IAL)

Due to the nature of remediation operations it is expected that radioactivity above background levels may be detected during the performance of FSS after all necessary remediation has been completed. It is for this reason that a Scan IAL is provided as part of the FSS plan. The IAL provides the HP Technicians performing field work a count rate above general area background that may indicate soils concentrations approaching the applicable $DCGL_W$ that should be investigated further.

To determine what an appropriate Scan IAL is for Class 1 areas of the site, all Final Remedial Action Support Surveys (RASS) of "Area 1" (i.e., LSA's 10-01, 10-02, 10-03, 10-04, and 10-12) were compiled calculating the mean count rate and standard deviation of the data set. All 5 of these LSA's are Class 1 areas where remediation was necessary and had proceeded until completion indicated that the areas were ready for FSS. Additionally all 5 of these LSA's did at one time contain contamination from all 6 of the contaminants of concern (i.e., U-234, U-235, and U-238, Ra-226, Th-232, and Tc-99), and were representative of a typical Class 1 area on the Hematite Site. The mean count rate of the population was calculated to be 10,698 cpm, with a standard deviation of 1,616 cpm. This is consistent with the average general area background observed on the Hematite site of approximately 10,000 cpm as reported in the HRCR.

MARSSIM Chapter 5.5.2.6 *Determining Investigation Levels* describes the industry standard practice of flagging areas for investigation that exceed background levels by a value of 3 sigma. For "Area 1" this would represent a count rate of 4,848 cpm above general area background. For ease of use in the field, and to provide some inherent conservatism, this value will be rounded down to 4,000 net cpm. Appendix D provides a Westinghouse memorandum HEM-15-MEMO-021 which documents the establishment of the 4,000 ncpm IAL.

Additionally, all GWS data will be "post-processed" and reviewed by the FSS Supervisor to determine if any additional areas require investigation based on the recorded GWS data.

GWS Post Processing is performed on each FSS GWS survey. Every data point logged within the survey unit is used to determine a sample population mean and standard deviation. Next the threshold for standard deviations above the mean is set (e.g., 3 standard deviations above the mean). The data is then presented in the form of a "Z-score map" showing only those locations within the survey unit that exceed 3 standard deviations above the mean or greater. Biased sampling locations are chosen both based on this data and also using the professional judgment of the Health Physics Staff.

The Z-score process aids in the identification of small pockets of elevated radioactivity that could potentially remain within the survey unit by identifying how an individual measurement differs from the rest of the data population. Prior to performing the Z-score analysis, the data population is reviewed by the Health Physics Professional Staff to identify any potential anomalies; a standard soil GWS data set collected at HDP will show a mean of approximately 10,000 cpm with a standard deviation of 1,000 cpm. If the data set appears to be free of any anomalies that may potentially skew the results, then the entire data population is used for the Z-score analysis. However if data anomalies do appear (e.g., multiple data populations within a single data set) performing the Z-score analysis on the entire data population may not be conservative, so a "population of interest" is identified for the purposes of Z-score analysis.

Multiple data populations in a single dataset most frequently occur due to dissimilar background materials, geometries, or instrument response curves. In order to eliminate errors caused by differing instrument response curves the Z Scores are calculated for each instrument separately. To evaluate potential geometry issues, elevated areas are compared to the physical layout of the survey unit (e.g., holes, sidewalls, trenches), and biased samples are collected from the highest activity sample within the physical area. In the case of dissimilar materials (e.g., survey areas containing more than one material type such as soil and gravel), the lower activity materials (usually gravel) are analyzed separately from the main data population (usually soil) to ensure that the Z-score defined by standard deviations do not potentially increase the likelihood of a false negative (e.g., ensuring that the "data bins" are not too wide as to obscure analysis).

While the Z-score is a useful tool in identifying potential biased sampling locations, it is important to note that it is most effective when used in conjunction with the Investigation Action Level (IAL) and the professional judgment of the Health Physics Staff.

6.1.4 Exposed Surfaces versus Accessible Surfaces

In the Westinghouse response to RAI's it was clarified that 100% of exposed surfaces would be subject to scanning, and procedures were implemented to direct a 100% scan of all accessible surfaces. During NRC Region III inspections conducted in 2015, an issue arose regarding exactly what "100% scan" meant and the distinction between "exposed surfaces" versus "accessible surfaces" when conducting scan surveys. This issue initially involved the fact that it has become a common industry standard to conduct land surveys wearing a GPS unit to track where the scan is being performed. While the scan itself is being conducted following accepted industry standards, the GPS unit is not always indicative of where the detector is being placed by the surveyor. Once the survey is completed and the GPS data printed out, it may appear to show an area was "missed", which may or may not have occurred. This evolved to a discussion of exactly what constitutes "100% scan". For example, in small areas where the soil surface may be difficult or impossible to access due to such things as standing water or worker safety concerns, an actual 100% scan may in fact not be feasible. (For large areas that could be construed as initially inaccessible, other means are normally employed such as dewatering the area or using man lifts to aid in getting the surveyors into areas safely.)

During an October 8, 2015, publicly noticed teleconference with the NRC, it was agreed by all parties that 100% GPS coverage was not a requirement or a standard, but, 100% scan of exposed surfaces is the expected goal, recognizing that the goal may be mitigated where accessibility becomes an issue. Westinghouse agreed to modify the procedures to document when the circumstances occurred where an area could not be scanned in its entirety. Procedure HDP-PR-FSS-711; Step 8.4.3, was modified to read:

> "For Class 1 areas, a 100% GWS of the exposed surface is required. For Class 2 and Class 3 areas, scan each survey area as specified in the survey instructions and any additional areas based on professional judgment using HDP-PR-HP-416, *Operation of the Ludlum 2221 for Final Status Survey* (Reference 5.8) (or equivalent instrument authorized by the RSO). If a prescribed survey location or area cannot be scanned in its entirety, indicate this and any other deviation in the Field Log of the FSS Plan and Sampling Instructions."

Subsequently, during an October 29, 2015, publicly noticed teleconference, it was agreed that the FSSFR should reflect that 100% GWS is the expected objective and that Westinghouse should provide a justification when 100% GWS coverage was not achieved, or did not appear to be achieved based on the output of the GPS software. The extent of the justification required may be greater for situations where 100% survey was not performed due to inaccessibility than is required when gaps appear due to artifacts of the GPS system. Where these situations occurred they were discussed within the FSSFR for that specific survey unit.

6.1.4.1 GPS vs. GWS Coverage

As discussed above the intent of the statement "100% GWS coverage" is understood to mean that the field of view of the gamma sensitive probe (e.g. 2x2 NaI probe) had passed over the entire exposed surface of the Class 1 soil survey unit, thereby identifying any potentially elevated areas or anomalies that may require further investigation. The absence of such elevated areas or anomalies indicates that the systematic soils samples will be representative of a homogenous soil survey unit, while the presence of elevated areas or anomalies indicates possible heterogeneous areas that require separate evaluation. Also as described above, it is recognized that the field of view of the survey probe is not always adequately represented by the position of the GPS antenna, or the size of the GPS "dots" that are displayed on the GWS figure. In some cases the field of view of the probe is much larger than the nominal 3ft diameter of the GPS "dots", and in some cases it could be smaller (e.g. due to obstructions such as sidewalls, or structures).

There are two predominant reasons for gaps in GPS and/or GWS coverage. The first reason is that the handheld GPS technology used to perform GWS may experience occasional "drift" or momentarily loose signal. To address this issue, training is provided to all Health Physics Technicians performing FSS to recognize these situations, and to attempt to correct these situations in the field by pausing to regain signal, or resurveying areas where the GPS signal may have drifted. Although pausing to regain the signal or resurveying an area has proven effective to address the issues common with using GPS while performing GWS, it is still extremely difficult to achieve 100% GPS coverage within a survey unit by even the most experienced Technicians.

The second reason for gaps in GWS as indicated by GPS is that there may be areas of a survey unit that are not accessible by traditional GWS means due to issues such as the surface of the survey unit containing steep slopes, or there may be excavation safety concerns for the Technicians performing the survey or other similar situations. In these situations the survey unit is surveyed to the maximum extent that is safe and/or practical. If it is not possible to achieve 100% GWS coverage then an evaluation of the survey unit GWS will be performed to determine if the results of the survey are sufficient in regards to GWS completion. The technical aspects of an assessment will include an analysis of the GWS count rates surrounding the inaccessible area, the size of the inaccessible area, or the amount of GPS coverage within the survey unit in order to determine the potential for the inaccessible area to contain an area of elevated activity that could potentially exceed the DCGL.

6.1.4.2 Post Survey Processing of GPS Data, and Determining GWS Coverage

While performing the GWS, if in the professional opinion of the Health Physics Technician, 100% GWS coverage has not yet been achieved, then the survey is repeated or supplemented with additional scans until the Technician believes that 100% GWS coverage has been achieved. Additionally the Technician listens and observes the 2x2 NaI instruments audio and visual scale response while performing the GWS and areas of elevated activity are first marked with paint or flags by the Technician in the field. Upon completion of the GWS the data from the GPS and GWS undergo post survey processing and the 2x2 NaI instrument log data is reviewed to determine if there are any additional areas of elevated activity within the survey unit that require follow up investigation (there by reducing surveyor error and increasing surveyor efficiency). The purpose of the GWS is to identify potential "hot spots" within the survey unit that exceed the IAL and thereby are believed to have the potential to exceed the DCGL. The justification for the IAL of 4,000 net cpm was documented in HDP-TBD-FSS-003 [Reference 8.38], and as documented in Westinghouse letter HEM-15-85, *Regional Response to U.S. Nuclear Regulatory Commission Review of Westinghouse Hematite Final Status Survey Issues and Associated Technical and Regulatory Bases and Paths Moving Forward*, Westinghouse previously provided to NRC Region III the technical basis document HDP-TBD-FSS-003 for review during NRC inspection activities.

As part of HDP-TBD-FSS-003, "Modeling and Calculation of Investigative Action Levels for Final Status Soil Survey Units" [Reference 8.38], Westinghouse developed the basis to determine when an elevated area or anomaly had the potential to exceed a DCGL. In this TBD Microshield© modeling was performed to support the calculations to determine the count rate that when collected with a 2x2 NaI probe would potentially exceed a DCGL. Area Factors were also applied to determine the maximum size of the potentially elevated area, and the calculated result determined that the IAL_{EMC} for a 300 m^2 area was 7,073 net cpm (Uniform DCGL) with a 2x2 NaI probe. The IAL_{EMC} for a 1 m^2 area was calculated to be 107,225 net cpm (Uniform DCGL) with a 2x2 NaI probe.

The result of the modeling demonstrates that an area of elevated activity would have to be physically very large (300 m^2 or greater) in order to potentially exceed the DCGL, as 300 m^2 is 15% of a typical 2,000 m^2 MARSSIM Class 1 survey unit, or very elevated (far exceeding the scan IAL of 4,000 net cpm) as readings in excess of 100,000 net cpm would skew high results in all surrounding areas much larger than 1 m^2.

When it is considered that the scan minimum detectable concentration (MDC) calculation, as well as the overall MARSSIM survey design, includes a 5% alpha error rate and a 10% beta error rate, (i.e. the expectation is not 100% accuracy, but rather a level of effort achieving the 95% confidence interval, at a minimum), it supports the conclusion that it is not necessary to achieve 100% GPS coverage in order to assess the potential for elevated areas or anomalies over 100% of the surface of the survey unit at the 95% confidence interval.

Next, in order to determine how much of the survey unit was covered by the GPS "dots" an assessment is performed by comparing the number of map pixels covered by GPS readings to the number of map pixels not covered by GPS readings to assign a percent value of GPS coverage.

HDP-TBD-FSS-003, "Modeling and Calculation of Investigative Action Levels for Final Status Soil Survey Units" [Reference 8.38] established that in order to potentially exceed a DCGL the area for a "hot spot" of 7,073 net cpm would cover an area of more than 300 m2 of a survey unit (approximately 15% of the surface area of the survey unit that is 2000 m2). As a conservative measure, HDP-TBD-FSS-003 established an Investigative Action Level of 4,000 net cpm.

Therefore, if 85 % or more of the survey unit undergoes a GWS, and no single un-surveyed area exceeds 300 m2, and there are no readings in the vicinity of any apparent GPS coverage gap approaching or exceeding the IAL of 4,000 net cpm there is assurance that there is not an area that would potentially exceed a DCGL that would go undetected.

If the GPS coverage is determined to exceed 95% with no readings approaching or exceeding the IAL of 4,000 net cpm in the vicinity of any apparent GPS coverage gaps, then the survey unit will be determined to meet the intent of the "100% GWS coverage" requirement. If the GPS coverage is less than 95%, or if elevated readings approaching or exceeding the IAL of 4,000 net are present in the vicinity of any apparent GPS coverage gaps then additional investigation will be performed and documented within the specific survey unit release record. These additional investigations will include an assessment of any supplemental manual GWS, investigative soil sampling, or other evaluations performed by the FSS group as necessary.

6.1.4.3 Determining FSS GWS Acceptability

In summary the modeling demonstrates that there is very little concern that small areas accounting for less than 5% of the survey unit would have the potential to contain elevated areas or anomalies that could potentially exceed the DCGL given that the scan Investigation Action Level for FSS surveys has been set at 4,000 net cpm. Therefore, if for all FSS GWS that were completed by a trained and qualified Health Physics Technician, verified by post survey processing to meet the 95% GPS coverage threshold, and not containing any readings approaching or exceeding the IAL of 4,000 net cpm in the vicinity of any apparent GPS coverage gap, then the GWS is adequate to demonstrate acceptability of the survey unit. For survey units in which the 95% GPS coverage threshold is not met, or elevated readings are identified in the vicinity of apparent GPS coverage gaps, then further assessment of GWS coverage will be performed.

6.2 Soil Sampling

6.2.1 Systematic Soil Sampling

All FSS soil sampling performed at HDP is analyzed for Uranium isotopes U-235, U-238, and U-234 (inferred), Technetium isotope Tc-99, Thorium isotope Th-232 (plus decay chain), and Radium isotope Ra-226 (plus decay chain). A Thorium background concentration of 1.0 pCi/g is subtracted from all soil sample results; and a Radium background concentration of 0.9 pCi/g when ingrowth is not used, and 1.07 pCi/g when ingrowth is used is subtracted from all soil sample results as well. No background values are used for Uranium or Technetium.

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 exposed areas 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 is adjusted to ensure that small areas of elevated radioactivity are detected. Special situations are evaluated by judgment sampling and measurements.

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 number of systematic samples collected was dependent on a prospective statistical test (Wilcoxon Rank Sum Test) which results in a minimum of eight samples. Following completion of the systematic sampling and analysis, a retrospective WRS Test was performed to verify the sample size based on the FSS data.~~ Systematic sampling and measurement locations for Class 1 and Class 2 survey units were located in a systematic pattern or grid. The grid spacing, *L*, was determined based on the SU size and the minimum number of sampling or measurement locations determined. Once the grid spacing was established, a random starting point was established for the survey pattern using a random number generator.

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 (excavation surface is within the Surface Stratum):
    - A surface sample from the ground surface to 15 cm bgs (Surface Stratum);
    - A composite sample from 15 cm bgs to 1.5 m bgs (Root stratum); and,

  - If the SOF in the sample obtained from the Root stratum exceeds 0.5, a sample from 1.5 m bgs to 1.65 m bgs (Deep stratum).

- Root Stratum Depth (excavation surface is within the Root Stratum):
  - A composite sample from the excavation surface (e.g., below 15 cm bgs) to 1.5 m bgs (Root stratum); and,
  - A sample from 1.5 m bgs to 1.65 m bgs (Deep stratum).

- Deep Stratum Depth (excavation surface is within the Deep Stratum):
  - Samples will be taken from the top 15 cm of the exposed surface (e.g., below 1.5 m bgs) and analyzed.

6.2.1.1 Performance of the WRS Test

The number of systematic sample locations collected during FSS sampling in a SU is dependent on a prospective statistical test (Wilcoxon Rank Sum Test) performed during the FSS planning phase which typically results in a minimum of eight sample locations. Following completion of the systematic sampling and analysis, a retrospective evaluation is performed to verify the sample size was appropriate based on the FSS data.

While the prospective WRS Test determines the minimum number of systematic sampling locations for the SU, multiple samples may be collected at each location within the SU if more than one stratum of soil remains at that location, and furthermore multiple $DCGL_W$'s may be used within the SU when the "3 Layer" approach is used. The use of multiple $DCGL_W$'s within the SU when the "3 Layer" approach is used prompted discussion between Westinghouse and the NRC regarding how the WRS Test should be applied. This discussion eventually expanded to include the application of the WRS Test when in the Uniform approach is used in a SU as well. During the publicly noticed teleconference help on January 26, 2017, at the request of NRC Headquarters, Westinghouse agreed to implement the WRS Test in the following fashion:

- Regardless of the $DCGL_W$ used, or the depth at which the systematic samples are collected, all systematically collected soil samples within a SU will be used for the performance of the WRS Test (this total number of samples will be equal to or greater than the number of sample locations that was determined by the prospective WRS Test evaluation). While including all systematically collected samples may increase the likelihood that the WRS Test will be successful, it will also increase the likelihood that the WRS Test will be determined to be required, and the NRC believes that this is also the most appropriate way to represent the data within the SU.

- When the "3 Layer" approach is used the most restrictive $DCGL_W$ within the LSA will be used to perform the WRS Test for both the activity concentrations of the background reference area samples, and the systematically collected samples regardless of the depth at which the sample was collected. This will typically mean the Root Stratum DCGLs will be used if there is soil remaining in the Root Stratum within the LSA. However, if the Surface and Root Stratums have been completely removed, then the Excavation Stratum DCGLs will be used.

In accordance with Step 7.8.3 of HDP-PR-FSS-721 *Final Status Survey Data Evaluation*, the WRS statistical test is required for a SU when the difference between the maximum SU data set gross SOF and the minimum background area SOF was greater than one (1.0) using the (most restrictive) $DGGL_W$ for the SU. However the WRS Test is typically performed and provided for illustrative purposes with each SU release record regardless if it is determined to be required or not. The 32 background area soil sample results that are discussed in Volume 1, Chapter 1, Section 5.1.3 are also used for the purpose of the WRS Test. These background soil samples were collected from two separate areas that are considered similar to the site, and therefore representative of background soil activity at the Hematite Site.

Gross SOF values were determined for the 32 background samples, and the systematically collected soil samples from the SU using the appropriate (and most restrictive) $DCGL_W$ for the SU. Next the $DCGL_W$ (i.e. 1.0) is added to the weighted sum of each background sample to obtain adjusted activity concentrations of the 32 samples collected within the Background Reference Area (e.g. adding unity to the gross SOF determination of the background reference area samples). The systematically collected samples in the SU are ranked against the adjusted activity concentrations of the 32 samples collected within the Background Reference Area. The SU passes the WRS Test if the ranked sum of the reference area ranks, or test statistic W_R, exceeds the critical value for the test. As such, when the WRS Test is successful the null hypothesis that the SU average concentration is greater than the $DCGL_W$ is rejected.

6.2.2 Biased Soil Sampling

As discussed above in Section 6.1.3, there are three key methods for identifying areas for biased soil sampling, the IAL, the Z-score of the FSS GWS, and the professional judgment of the HP Staff. These three methods are best when used in conjunction with each other to determine when an area of a survey unit may be non-homogenous, and require separate evaluation (i.e., biased sampling).

When biased soil sample locations are selected, a sample will be taken from the top 15 cm of the exposed surface and analyzed at the offsite laboratory for the same parameters as the systematic FSS samples (e.g., Gamma Spec and Tc-99 by ICP-MS). Biased samples that do not exceed a SOF of 1.0 are not used to calculate final survey unit dose and are not included in the systematic mean. If a biased sample does exceed a SOF of 1.0, then an EMC will be performed to determine if the survey area is still suitable for release as discussed further in Sections 3.1.3 and 3.2.2.

6.2.3 Judgmental/Tc-99 Side Wall Soil Sampling

Although not addressed in the RAI's for Revision 0 of the DP, the NRC raised the concern for the potential lateral movement of Tc-99 (a hard to detect nuclide), and requested that sidewalls within excavated survey areas be evaluated above the requirements for systematic and biased sampling as stated in the DP.

The NRC presented the following as one potential option for addressing these sidewall samples:

"The area of the excavation floor would be compared to the area of the sidewalls of the excavation, and the total number of samples would be increased proportionally. An appropriate number of samples would be determined for the excavation floor per MARSSIM, and a random start sampling grid would be developed for the floor. A random start sampling grid would also be developed for the sidewalls, and the number of samples would be determined as a proportion of the number of samples for the excavation floor. For example, if the floor encompasses 2000 m^2 and the sidewalls represent a 10% areal increase to 2200 m^2, an additional 10% of samples should be taken from the sidewalls. It may also be appropriate for the licensee to set a discretionary number of samples for the sidewalls to ensure that the measurements are representative (i.e., there will always be a minimum of x number of samples from the sidewalls)."

In response to this request from NRC Headquarters, the HDP FSS procedures were revised to include judgmental sidewall samples for Tc-99 using the guidance below:

- If vertical or near vertical sidewalls exist within the survey unit (i.e., surfaces that are greater than a 45 degree angle), and
- If those sidewalls exceed 12 inches in height vertically, and
- If those sidewalls exceed (in aggregate) 5 % of the total survey unit surface area (e.g., greater than 100 m^2 of sidewall in a 2,000 m^2 survey unit), then discretionary (aka. judgmental) sidewall sampling is necessary.

If judgmental sidewall samples are necessary:

- Determine the number of samples to be collected based on the sidewall surface area compared to the two dimensional systematic surface area (e.g., 8 systematic samples were collected over 2,000 m^2, then collect 1 sample per 250 m^2 of sidewall).
- Collect a judgmental sample(s) at sidewall location(s) not based on radiological scans, but selected at the discretion of the Health Physics Technician performing soil sampling.

6.2.4 Quality Control Soil Sampling

During the FSS within an open land survey unit, the laboratory was assessed through the analysis of field and laboratory duplicate samples. Field duplicate samples consisted of splitting a homogenized sample into two or more separate samples for analysis. Field duplicates were obtained from one location, homogenized, divided into separate containers, and treated as separate samples. Laboratory duplicate samples consisted of the re-analysis of the same sample at the laboratory. Both types of quality assurance samples were analyzed at a frequency of one sample per 20 final status survey samples collected (5%). Field duplicate samples were evaluated per guidance in MARLAP. Laboratory duplicates were evaluated internally by the laboratory and were part of the laboratory's QA program.

6.2.5 Off-site Laboratory

Test America in St. Louis, Missouri is the laboratory selected by Westinghouse to conduct the sample analysis for the Hematite decommissioning project. Test America was 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, such as NELAP, NVLAP, and ISO 9001:2000 were also considered during their evaluation. Maintenance of applicable accreditations is imposed as a quality requirement on the purchase order. Methods used were standard industry methods from the U.S. Environmental Protection Agency (EPA) and the Environmental Measurements Laboratory (EML).

Following receipt of the results of laboratory analyses, HDP staff performed a data review to assess the validity of the data for use in the final status survey. This review included an evaluation of the data to ensure that all of the DQOs were met.

As previously discussed with the NRC (and provided to the NRC in Westinghouse letter HEM-11-96 dated July 5, 2011, (Reference 8.20)), the contract laboratory performed data review, verification, and reporting in accordance with approved standard operating procedures (SOPs). In accordance with these SOPs, analytical data was reviewed by the analyst performing the task, followed by a secondary review by a department supervisor/lead analyst or their designee, and then review by the associated project manager. The vendor QA department performed an independent random review as oversight of the process. This review was documented on a data review checklist specific to each analytical method. Following receipt of laboratory data for use in the final status surveys, HDP staff performed an additional data review to assess the validity of the data. This review included an evaluation of the data to ensure that all of the DQOs were met. Essentially, this meant that the program had been structured to place a high degree of responsibility on Test America to verify the validity of the data prior to it being provided to Westinghouse. Therefore, there was no specific basis for HDP personnel to re-validate the analytical results from the backfill soil and off-site borrow locations.

In Westinghouse letter HEM-14-31, dated March 13, 2014, (Reference 8.36), Westinghouse sent the NRC the results of radiological testing of backfill soil from an off-site borrow location. That submittal contained radiological data for Tc-99 in which 2 of the 16 samples showed concentrations that were slightly above their minimum detectable concentrations (MDCs). The data, provided by Test America, was identified as questionable by the NRC, indicating the soil samples shouldn't identify any Tc-99 in excess of the MDC. After conducting an initial review of the data, Westinghouse confirmed that the Tc-99 data was anomalous. As a result of that finding, Westinghouse entered the issue into the Westinghouse Corrective Action Prevention and Learning (CAPAL) Program for follow up.

The CAPALs had two primary areas of focus. The first area focused on assessing the Test America program and its effectiveness in meeting expectations and commitments for data acceptance, quality control, and verification. An investigation of the Test America program included an independent audit by outside experts. The audit identified numerous deficiencies and issues, including: use of a backup counting instrument that laboratory personnel failed to identify as providing biased high results for Tc-99 (for a total of 90 out of 190 samples); a laboratory information management system that contained a Reporting Limit in mg/kg metals concentration that did not meet the HDP Requested Limit of 1 pCi/g Tc-99 activity, and; a failure by laboratory personnel to adequately validate the final sample results that were included in the final report to Westinghouse. All these issues were addressed by Test America. Corrective actions included: performing an Extent of Condition that validated the integrity of historical data produced on the backup instrument and confirmed no impact to off-site borrow soils used for backfill at HDP; the primary instrument which was returned to service after repair and used for Tc-99 analysis results was verified to be providing correct results; conducting intra laboratory comparisons to validate data results: and, conducting numerous meetings with Test America to ensure expectations and contract obligations would be met going forward.

The second area of focus was to understand why HDP personnel did not identify the anomalous sample results before they were transmitted to the NRC. While HDP did have an opportunity to identify the errant data, a number of factors contributed to the failure to do so. In accordance with the contract laboratory SOPs, analytical data is reviewed by the analyst performing the task, followed by a secondary review by a department supervisor/lead analyst or their designee, and then review by the associated project manager. The vendor QA department performs an independent random review as oversight of the process. This review is documented on a data review checklist specific to each analytical method. Following receipt of laboratory data for use in the final status surveys, HDP staff performs an additional data review to assess the validity of the data. This review includes an evaluation of the data to ensure that all of the DQOs have been met. This means that the program was structured to place a high degree of responsibility on Test America to verify the validity of the data prior to it being provided to Westinghouse. Therefore, there was no specific basis for HDP personnel to re-validate the analytical results from the backfill soil and off-site borrow locations.

Regardless, in manipulating the data to perform the statistical analyses, there was an opportunity to identify that the data was questionable. However, that would depend in large part on the experience and knowledge of the individuals conducting the work. Since Tc-99 is normally not found in native soils, and most of that which is present, if detectable, is from atomic bomb testing in the 1950's and 1960's, not everyone who saw Tc-99 in testing of native soil would recognize that it may not be valid. In this case, no one at HDP identified that having any Tc-99 in the soil sample results was suspect. Lastly, the HEM-14-31 letter (Reference 8.36) which contained the questionable data was issued during a turnover of decommissioning contractors at HDP. During that time frame, there was an effort to complete specific tasks before a change in personnel occurred and this may have hampered giving this issue the attention it needed.

Since the beginning of remediation activities in March, 2012, and through 2015, Test America analyzed over 20,000 radiological soil samples for the Westinghouse HDP. The sample results identified as biased high in regard to the use of the backup counting instrument were the only results that were identified as questionable. Since that data represented an extremely small error rate overall by Test America, Westinghouse believes the program as structured was acceptable.

6.3 Data Quality Assessment

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). 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 DCGLw); 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 DCGLw 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.

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 MARSSIM (NUREG-1575), 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.

A detailed statistical review of analytical data, as presented in Chapter 14 of the Hematite Decommissioning Plan, will be provided in each subsequent chapter containing LSA Survey Area Release Records.

7.0 SURVEY AREA RELEASE RECORD ORGANIZATION

In accordance with HDP-PO-FSS-700, *Final Status Survey Program*, documentation of the FSS will transpire in two types of reports, FSS Survey Area Release Records and an FSS Final Report, and will be consistent with Section 8.6 of NUREG-1757, Volume 2, Consolidated Decommissioning Guidance – Characterization, Survey, and Determination of Radiological Criteria.

The FSS Final Report will incorporate multiple Volumes. The first Chapter of Volumes 2 through 5 will include general Site information and an overview of the FSS Program for that subject area. (Volume 2, *Reuse Soil*; Volume 3, *Land Survey Areas*; Volume 4, *Building Survey Areas*; Volume 5, *Piping Survey Areas;* Volume 6, *Groundwater*) Subsequent Chapters within these Volumes will contain FSS Survey Area Release Records.

Survey Area Release Records are prepared to provide a record of the composition and location of the survey area; the measurements obtained during the FSS; the number and location of any small areas of elevated concentration; and a summary of the data that represents the final radiological condition, including a determination that an individual survey area/unit meets the release criteria.

The Survey Area Release Records will be formatted to contain the following information:

a. An Introduction section which will include Survey Unit specific information (e.g., geographical description, summary of historical radiological data).

b. A description of the specifics of FSS Protocol and DQOs, including but not limited to:
   - Survey Unit designation and classification.
   - Background determination.
   - Instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration).
   - Survey methodology and protocols.
   - QC surveys.
   - A discussion of any changes that were made in the FSS from what was proposed in the DP, including its classification.

c. Conclusion, including dose estimate from all pathways and estimated dose contribution from groundwater.

d. Supporting documents (e.g., spreadsheets, statistical analyses, figures, tables).

Appendix C, HDP FSSFR LSA Document Matrix, provides a status of the FSS documents that will be submitted as supporting information to FSSFR Volume 3. This table will be updated as FSS Release Reports are generated.

8.0 REFERENCES

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8.3 License SNM-33 Amendment 57 {ML112101640}

8.4 Hematite Decommissioning Plan Safety Evaluation Report {ML112101630}

8.5 NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual

8.6 NUREG-1757, Volume 2, Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria

8.7 NUREG-1507, Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions

8.8 NUREG-1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys

8.9 Code of Federal Regulations, Title 10, Part 20.304, "Disposal by Burial in Soil," 1964

8.10 Gulf United Nuclear Fuels Corporation, Nuclear And Industrial Safety, Commercial Products Division, Memorandum NIS:DGD-70-332, Peter Loysen, "AEC Inspection Wrap-Up Meeting, November 5, 1970"

8.11 U.S. Atomic Energy Commission, Letter to Combustion Engineering, Inc., J. G. Keppler to H. V. Lichtenberger, Inspection Reports Nos. 070-036/74-08 and 24-16206-01/74-01, October 3, 1974. {ML052510598}

8.12 License SNM-33 Amendment 52 {ML061280324}

8.13 SEA Brown Boveri/Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, J.A. Rode to D.J. Sreniawski, "Building 253 Construction Site Soil," May 14, 1990. {ML052350909}

8.14 U.S. Nuclear Regulatory Commission, Letter to ASEA Brown Boveri/Combustion Engineering, C. J. Haughney to J. A. Rode, "Authorizes Backfill of Area of Bldg. 253 Construction Site," July 2, 1990 {ML052550383}

8.15 Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, "Spent Limestone Results," April 7, 1989 {ML053640233}

8.16 Westinghouse (E. K. Hackmann) letter to NRC (Document Control Desk), HEM-11-56, dated May 5, 2011, "Evaluation of Technetium-99 Under the Process Buildings" {ML111260624}

8.17 Gateway Environmental Associates, Letter to R.W. Sharkey, "Exploratory Probehole Investigation for the Evaporation Ponds at the ABB Combustion Engineering Hematite Facility," April, 17, 1997

8.18 Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, "Evaporation Pond Characterization," October 22, 1997

8.19 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, "Hematite License Amendment Application Dated May 30, 1997," January 26, 1998.

8.20 Westinghouse letter HEM-11-96, Final Supplemental Response to NRC Request for Additional Information on the Hematite Decommissioning Plan and Related Revision to a Pending License Amendment Request, July 5, 2011{ML111880290}

8.21 NRC Inspection Report 07000036/2014005(DNMS) {ML15054A418}

8.22 Westinghouse Electric Company Document No. HDP-PO-WM-900, "Waste Management and Transportation Plan"

8.23 ASEA Brown Boveri/Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, J.A. Rode to D.J. Sreniawski, "Building 253 Construction Site Soil," May 14, 1990 {ML052550381}

8.24 U.S. Nuclear Regulatory Commission, Letter to ASEA Brown Boveri/Combustion Engineering, C. J. Haughney to J. A. Rode, "Authorizes Backfill of Area of Bldg. 253 Construction Site," July 2, 1990 {ML052550383}

8.25 Combustion Engineering, Letter to U.S. Nuclear Regulatory Commission, "Spent Limestone Results," April 7, 1989 {ML053640233}

8.26 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use"

8.27 Westinghouse letter HEM-11-56, *Evaluation of Technetium-99 Under the Process Building {ML111260624}*

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8.29 NRC letter dated April 5, 2013, U.S. Nuclear Regulatory Commission Review of Westinghouse Hematite's Final Status Survey Plan for Piping, HDP-PO-FSS-800 {ML13031A452}

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8.32 Westinghouse letter HEM-15-131, Reply to a Notice of Violation Issued November 27, 2015 {ML15357A074}

8.33 NRC letter dated January Westinghouse Electric Company (Hematite) Acknowledgement of Disputed Violation of NRC Inspection Report 0700036/2015003(DNMS) {ML16020A093}

8.34 NRC letter dated October 8, 2015, U.S. Nuclear Regulatory Commission Conclusions Associated with the Utilization of Off-site Borrow Material at the Westinghouse Hematite Site {ML15275A428}

8.35 Westinghouse letter HEM-15-39, Additional Statistical Analysis for Backfill Soil from an Off-site Borrow Location {ML15117A151}

8.36 Westinghouse letter HEM-14-31, Radiological Testing of Backfill Soil from an Off-site Borrow Location {ML14072A485}

8.37 DO-08-008, Derivation of Surrogates and Scaling Factors for Hard-To-Detect Radionuclides {ML092870492}

8.38 Westinghouse Electric Company Document No. HDP-TBD-FSS-003, "Modeling and Calculation of Investigative Action Levels for Final Status Soil Survey Units"

Table 3-1

Adjusted and Modified Soil DCGLw Values for Demonstrating Compliance, TC-99 Surrogate Evaluation Area (SEA)

Radionuclide	DCGL$_W$ (pCi/g) By Conceptual Site Model									
	Surface Soil		Root Stratum		Deep Volumetric [a]		Uniform [b]		Excavation [a]	
	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99	Measure Tc-99	Infer Tc-99
Plant Soil SEA										
Total Uranium [c]	394.3	191.7	202.4	52.8	2917	2895	170.2	44.1	706.3	202.8
U-234	508.5	508.5	235.6	235.6	2890	2890	195.4	195.4	872.4	872.4
U-235	102.3	14.1	64.1	3.0	3034	2565	51.6	2.5	208.1	11.8
U-238	297.6	297.6	183.3	183.3	3028	3028	168.8	168.8	551.1	551.1
Tc-99	151.0	N/A	30.1	N/A	98649	N/A	25.1	N/A	74.0	N/A
Th-232 + C	4.7	4.7	2.0	2.0	9279	9279	2.0	2.0	5.2	5.2
Ra-226 + C	5.0	5.0	2.1	2.1	13029	13029	1.9	1.9	5.4	5.4

a The distribution ratio for Deep Stratum soil was used to calculate the DCGL$_W$ for Total Uranium and U-235 when inferring Tc-99
b The distribution ratio for Root Stratum soil was used to calculate the DCGL$_W$ for Total Uranium and U-235 when inferring Tc-99
c Total Uranium DCGL$_W$ values were calculated using Equation 4-4 of MARSSIM, adjusted DCGL$_W$ values from Table 14-4 of attachment 4 to HEM-11-96, modified U-235 DCGL$_W$ values from Table 14-9 of attachment 4 to HEM-11-96, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4% in soil.

Table 3-1 (continued)

Adjusted and Modified Soil DCGLw Values for Demonstrating Compliance, TC-99 Surrogate Evaluation Area (SEA)

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 of attachment 4 to HEM-11-96, modified U-235 DCGL$_W$ values from Table 14-9 of attachment 4 to HEM-11-96, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4%.

Table 3-1 (continued)

Adjusted and Modified Soil DCGLw Values for Demonstrating Compliance, TC-99 Surrogate Evaluation Area (SEA)

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 of attachment 4 to HEM-11-96, modified U-235 DCGL$_W$ values from Table 14-9 of attachment 4 to HEM-11-96, and radioactivity fractions provided in Table 14-5 corresponding to an average Uranium enrichment of 4%.

Table 3-2a

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 3-2b

Calculated Area Factors Based On Excavation Scenario Constraints 1 And 2

Radionuclide	**Contiguous Elevated Area after Excavation (size of elevated area shown in m^2)***					
	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 3-2c

Effective Area Factor For Use With Excavation DCGLs

Radionuclide	**Size of elevated area shown in m^2**					
	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 3-3

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 3-4

Groundwater DSRs

Radionuclide	Well Water Concentration (pCi/L)	TEDE (mrem/yr) [a] For Water-Dependent Pathways	DSR_{GW} (mrem/yr per pCi/L)
U-234	5.707 E+00	8.744 E-01	0.1532
U-235 + D	5.707 E+00	8.261 E-01	0.1448
U-238 + D	5.707 E+00	8.302 E-01	0.1455
Tc-99	9.415 E+00	8.826 E-03	9.374 E-04
Th-232 + C	3.030 E-01	1.007E+00	3.323
Ra-226 + C	6.346 E-03	4.786E-01	75.42
Np-237 + D	3.966 E+01	1.118 E+02	2.819
Pu-239/240	8.332 E-01	1.744 E+00	2.093
Am-241	1.190 E-01	3.098 E-01	2.603

[a] At t = 0 years

Table 5-1

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 accounting for the dose contribution from insignificant radionuclides (see Equation 14-1 in Section 14.1.3.2 of the Hematite DP).
[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.

Figure 2-1
Area of Documented Burial Pits Based on Historical Information

Figure 2-2
Burial Pits Easily Identified by Change in Soil Color

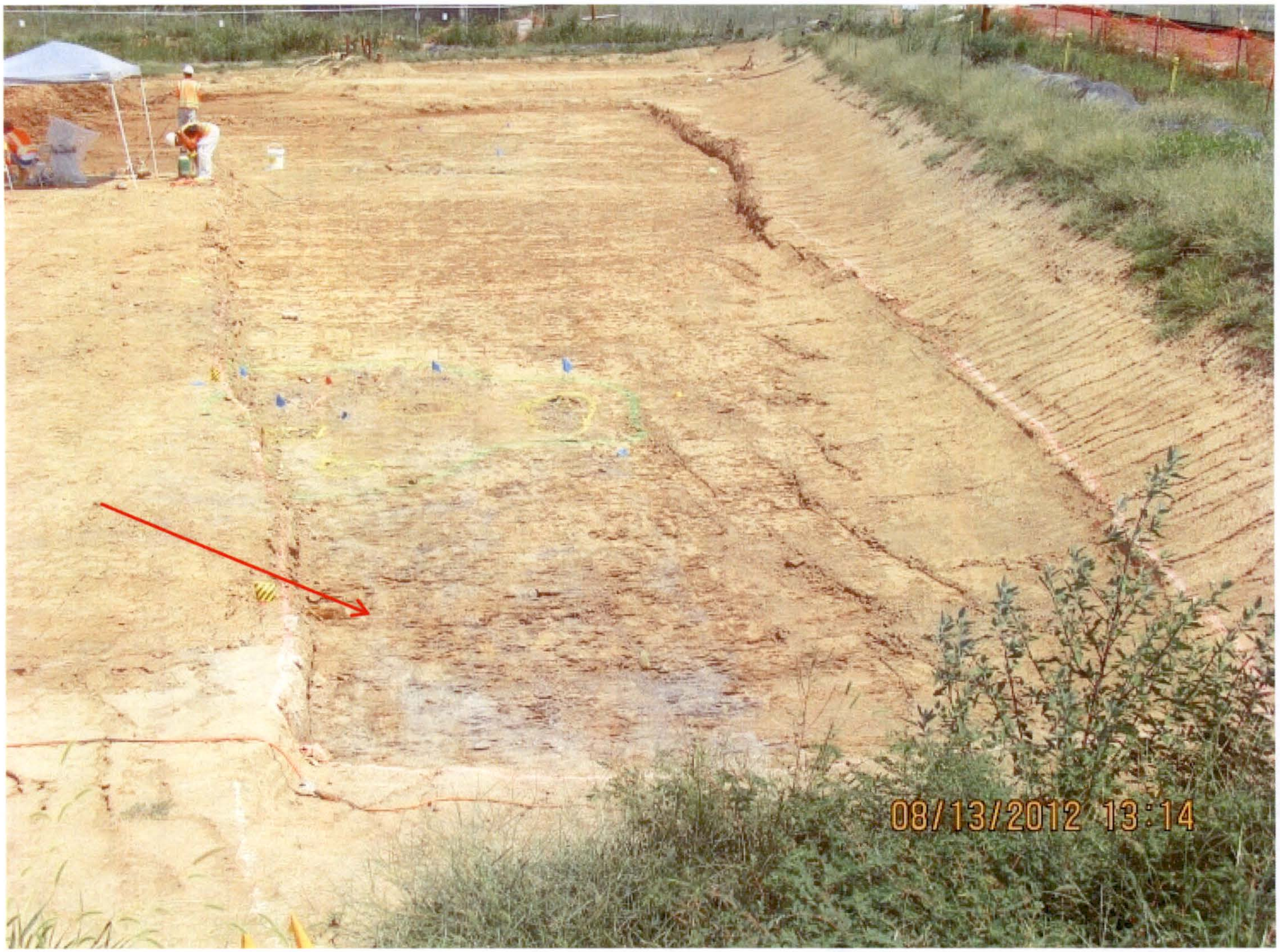


Figure 2-3
Burial Pit Easily Identified by Change in Soil Color

Figures 2-4
Example of Worker Conducting a Gamma Walkover Survey (GWS) in the Burial Pit Area with a NaI 2 x 2 Detector

Figure 2-5
Examples of Radium Contaminated Filter Plates Being Unearthed While Excavating in the Burial Pit Area

Figures 2-6
Examples of Radium Contaminated Filter Plates Being Unearthed While Excavating in the Burial Pit Area

Figure 2-7
Examples of Debris that was Exhumed During Remediation Excavation of the Burial Pit Area

Figure 2-8
Examples of Debris that was Exhumed During Remediation Excavation of the Burial Pit Area

Figure 2-9

Burial Pit Area March 2012 (Looking NW)
The Removal of Overburden Soil Has Commenced

Figure 2-10

Burial Pit Area September 2012 (Looking SE)
The Top Layer of Overburden Which Contained the Sod Has Been Removed and Remediation Is Well Under Way

Figure 2-11

Burial Pit Area March 2013 (Looking SE)
Remediation in Some Areas Is to Significant Depths Much of the Burial Pit Area Elevation Is Below That of the Northeast Site Creek

Figure 2-12

Burial Pit Area September 2013 (Looking SE)
A Significant Portion of Remediation of the Burial Pit Area Proper Has Been Completed: Remediation of Non-Burial Pit Areas Has Been Expanded To the Northeast Site Creek to the Left and Towards the Former Process Building Area to the Right

Figure 2-13

Burial Pit Area November 2013 (Looking SE)
The Geometric Shape of the Excavations Indicate the Ability of the Excavation Process to Identify Burial Pits and Remediate to a Depth that Ensures the Burial Pit Has Been Removed

Figure 2-14

Burial Pit Area September 2014 (Looking North)
Remediation of the Burial Pit Area Proper Is Nearly Complete

Figure 2-15

Burial Pit Area October 2014 (Looking South)
Shortly After This Photograph Was Taken All Visual Inspections and Radiological Surveys Associated With the Burial Pit Area Were Completed and the Data Indicated That the Burial Pit Area Remediation Was Complete

Figure 2-16

Burial Pit Area May 2015 (Looking North)
Photograph of the Burial Pit Area Prepared for Final Status Survey
The Silt Fence Is In Place for Isolation and Control of the Area

Figure 2-17

Burial Pit Area September 2015 (Looking SE)
Backfill Operations in the Burial Pit Area

Figure 2-18

Burial Pit Area September 2015 (Looking South)
NRC Region III Inspector Inspecting the Backfill of the Burial Pit Area

Figure 2-18a
Example of Auger Types Used to Perform NCS Borings

Figure 2-18b

Change in Color of Soil in Bottom of Burial Pit

Figure 2-19

Core Bore Locations Performed for Nuclear Criticality Safety Control Purposes

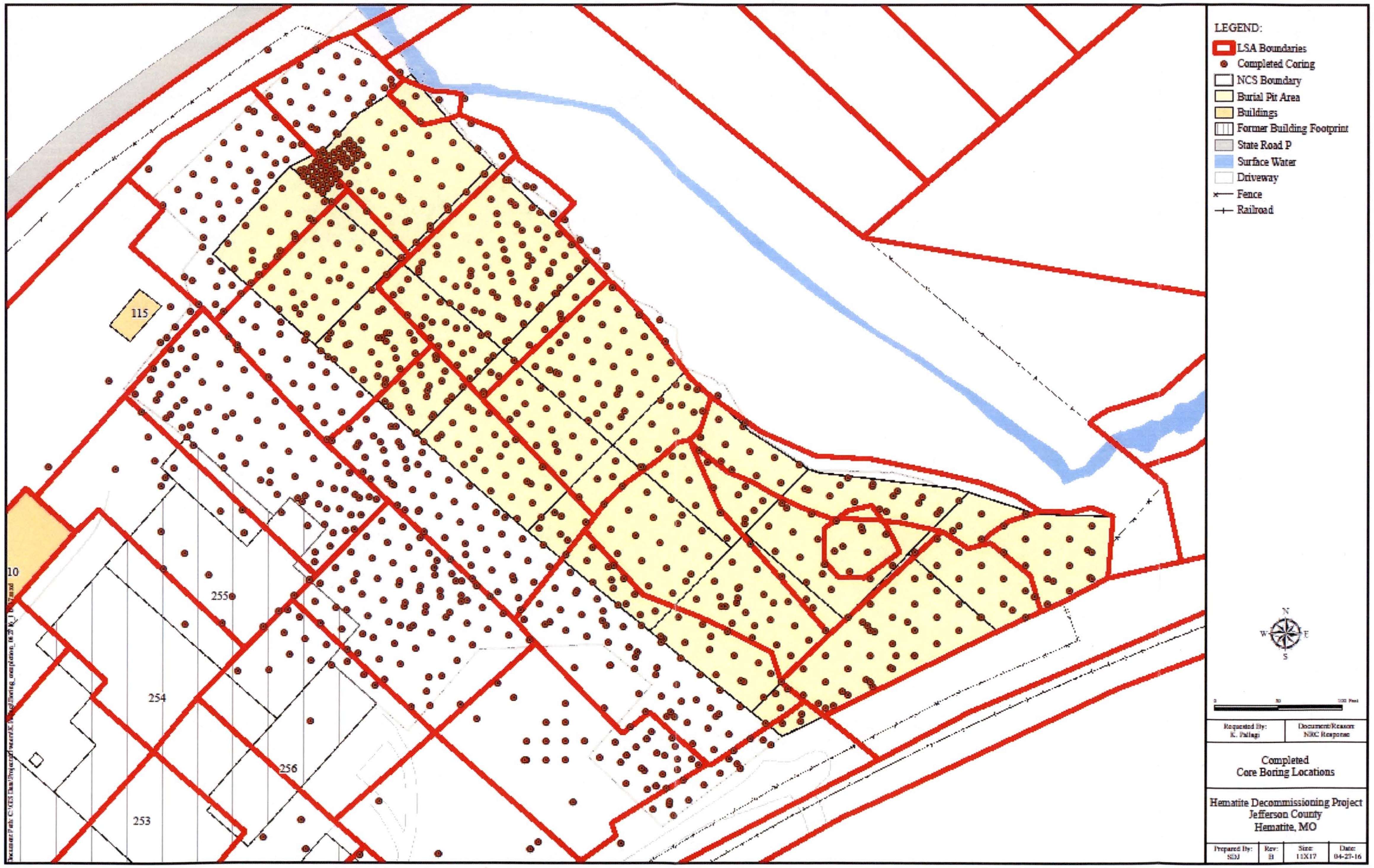


Figure 2-20

Surface and Sub-surface Soil Sampling Locations in the Burial Pit Area

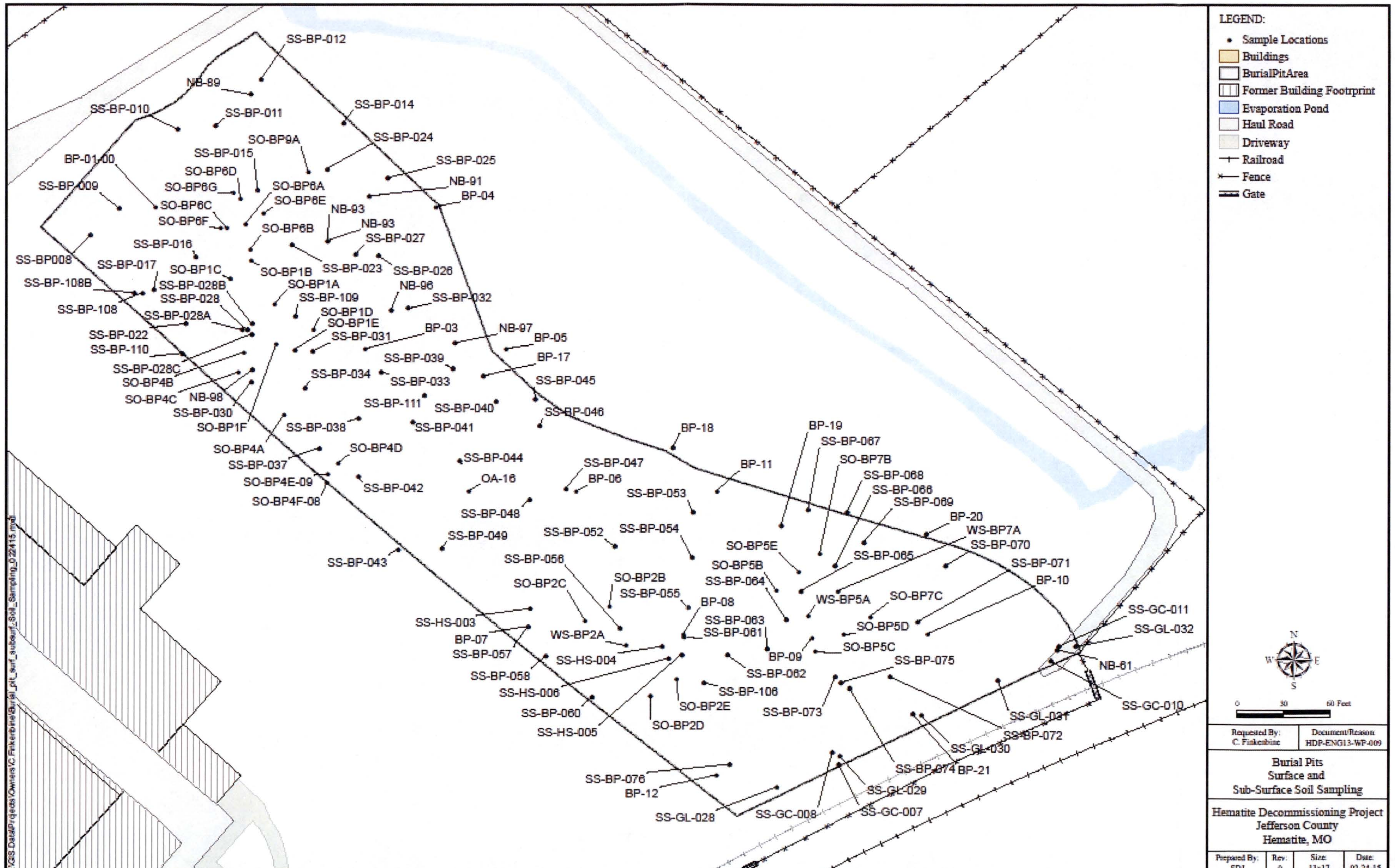


Figure 2-21
Excavated Elevations of the Burial Pit Area Post Remediation

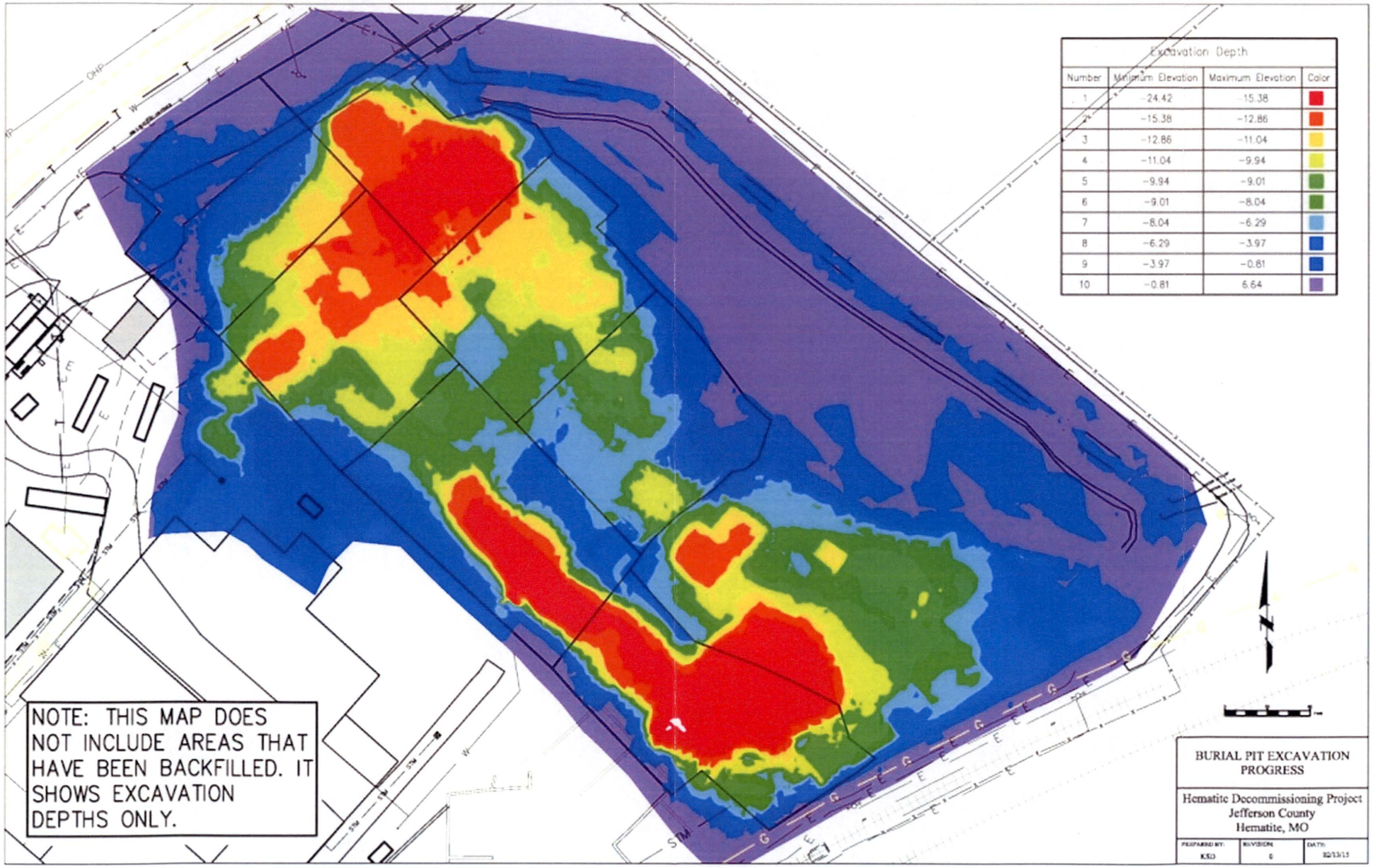


Figure 2-22

Documented Burial Pit Area Boundary and the Outer Boundary of the Area Defined By The Extent of the Undocumented Burials

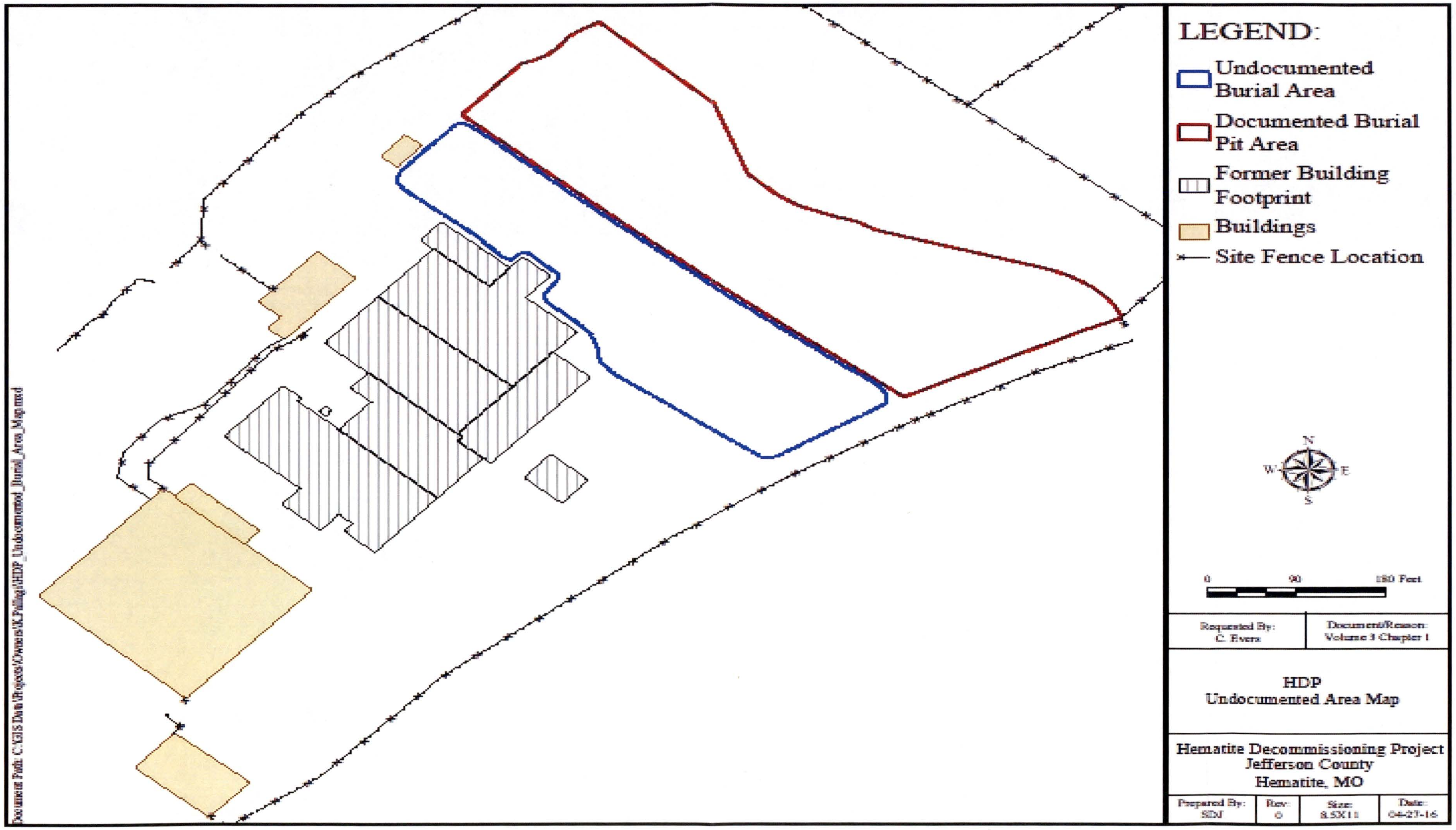


Figure 2-23
Location of Hematite Buildings Prior to Demolition Activities

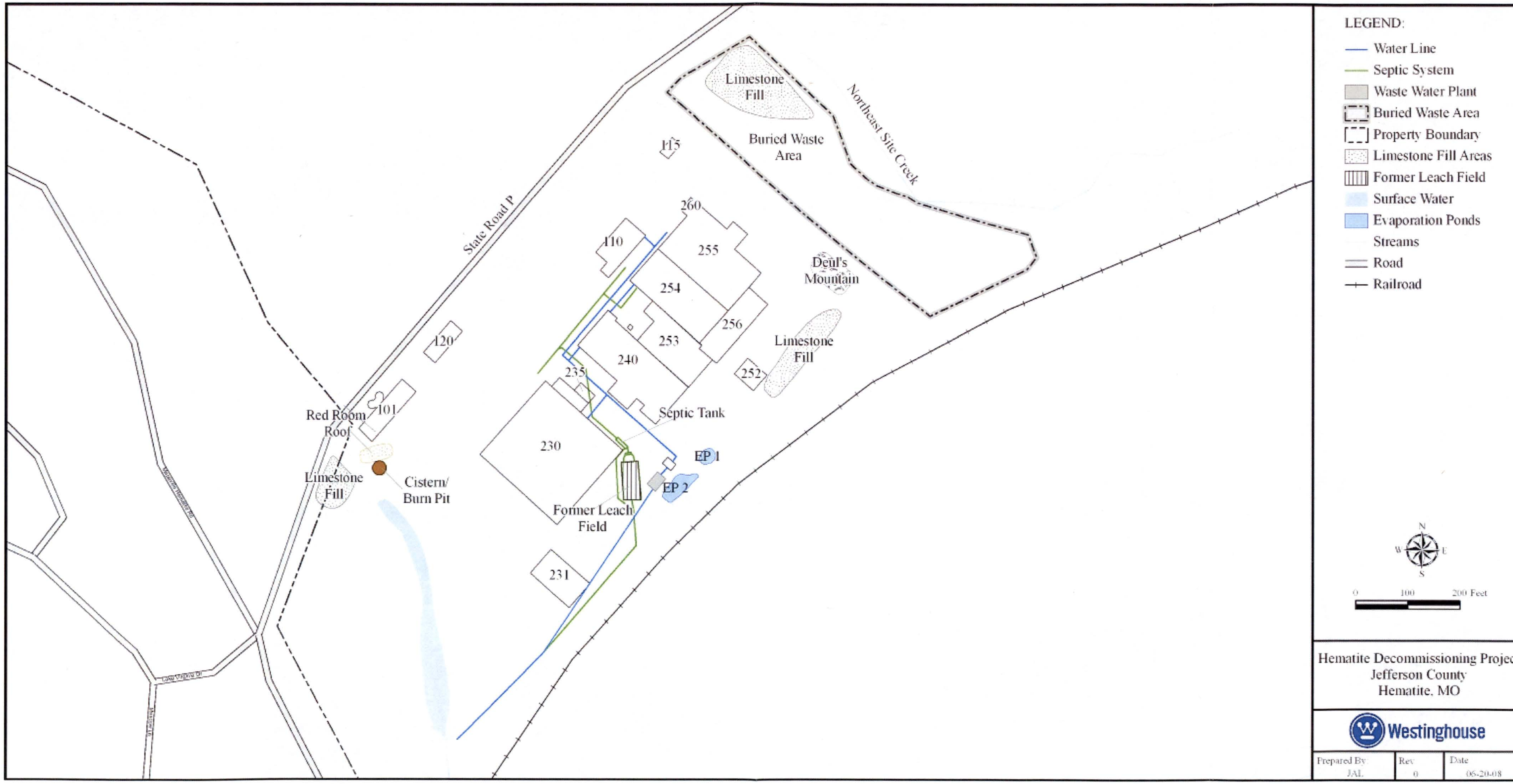


Figure 2-24
Hematite Circa 2006

Figures 2-25 through 2-36
Process Building Demolition Photographs

DELETED

(see FSSFR Volume 4, Chapter 1, Building Survey Areas Overview)

Figure 2-37
Process Building Slab with Process Building Removed - August 2011 (Looking East)

Figure 2-38

Process Building Area Showing the Remediation Removed All of the Subterranean Process Piping Except Those Portions of the Piping That Resided Under the Haul Road (Former Building Slab) (Looking NE)

Figure 2-39

Photograph Taken On November 7, 2015, Showing the Removal and Remediation of the Haul Road (Former Building Slab) Along With Remaining Subterranean Process Piping (Looking NE)

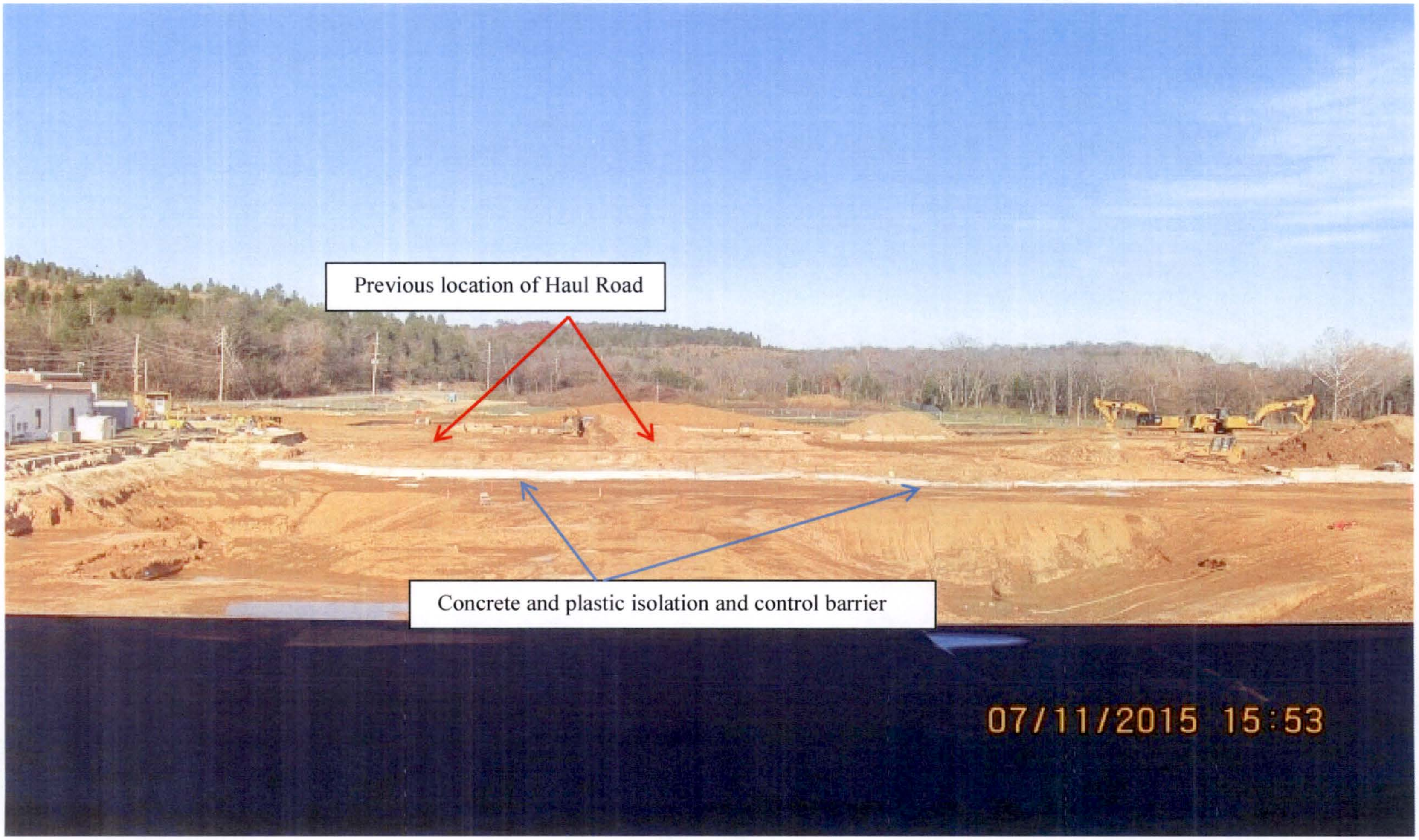


Figure 2-40

Evaporation Ponds (The Second Pond is Below the Pond Shown and Covered with Soil) The Red Line Shows Where an Eight Inch Natural Gas Pipeline is Located

Figure 2-41

Evaporation Ponds (The Second Pond is Below the Pond Shown and Covered with Soil) (Looking NE)

Figure 2-41a
Sampling Initiative for the Evaporation Ponds

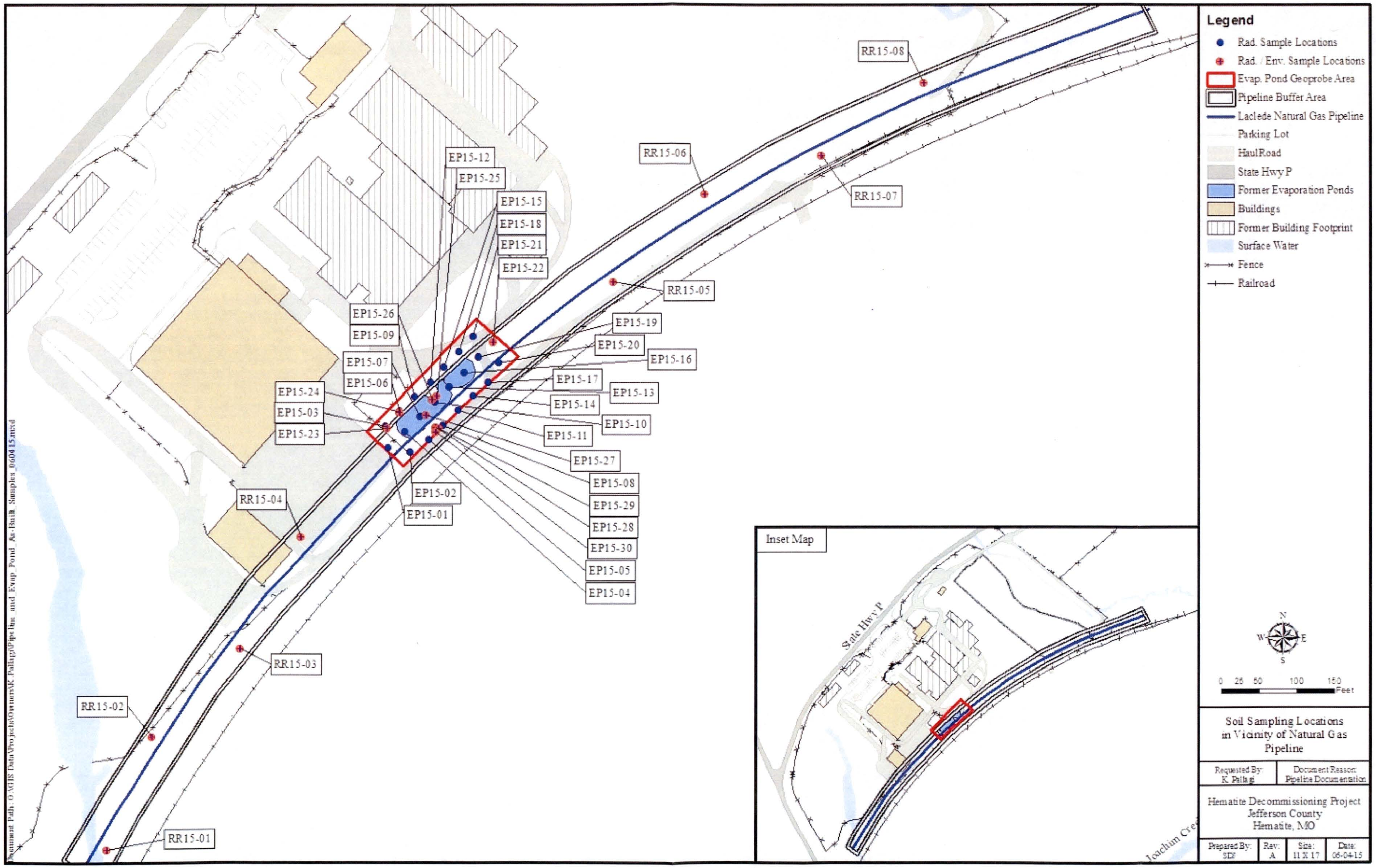


Figure 2-42
Excavation of the Evaporation Ponds (Looking SW)

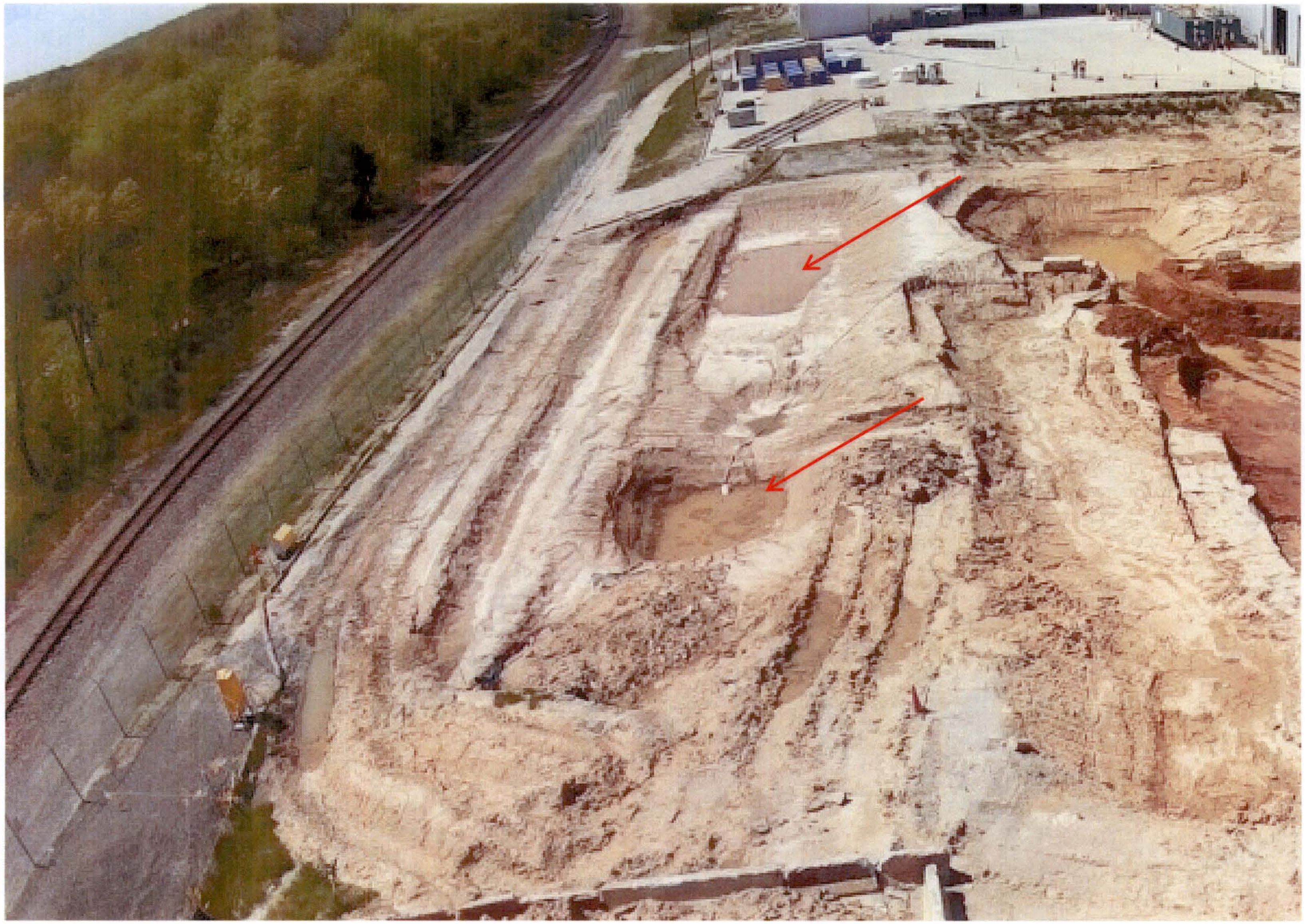


Figure 2-43
Natural Gas Pipline with Vertical Measurements Used to Determine It's Depth

Figure 2. Vertical Measurements Used to Determine the Depth of the NGP

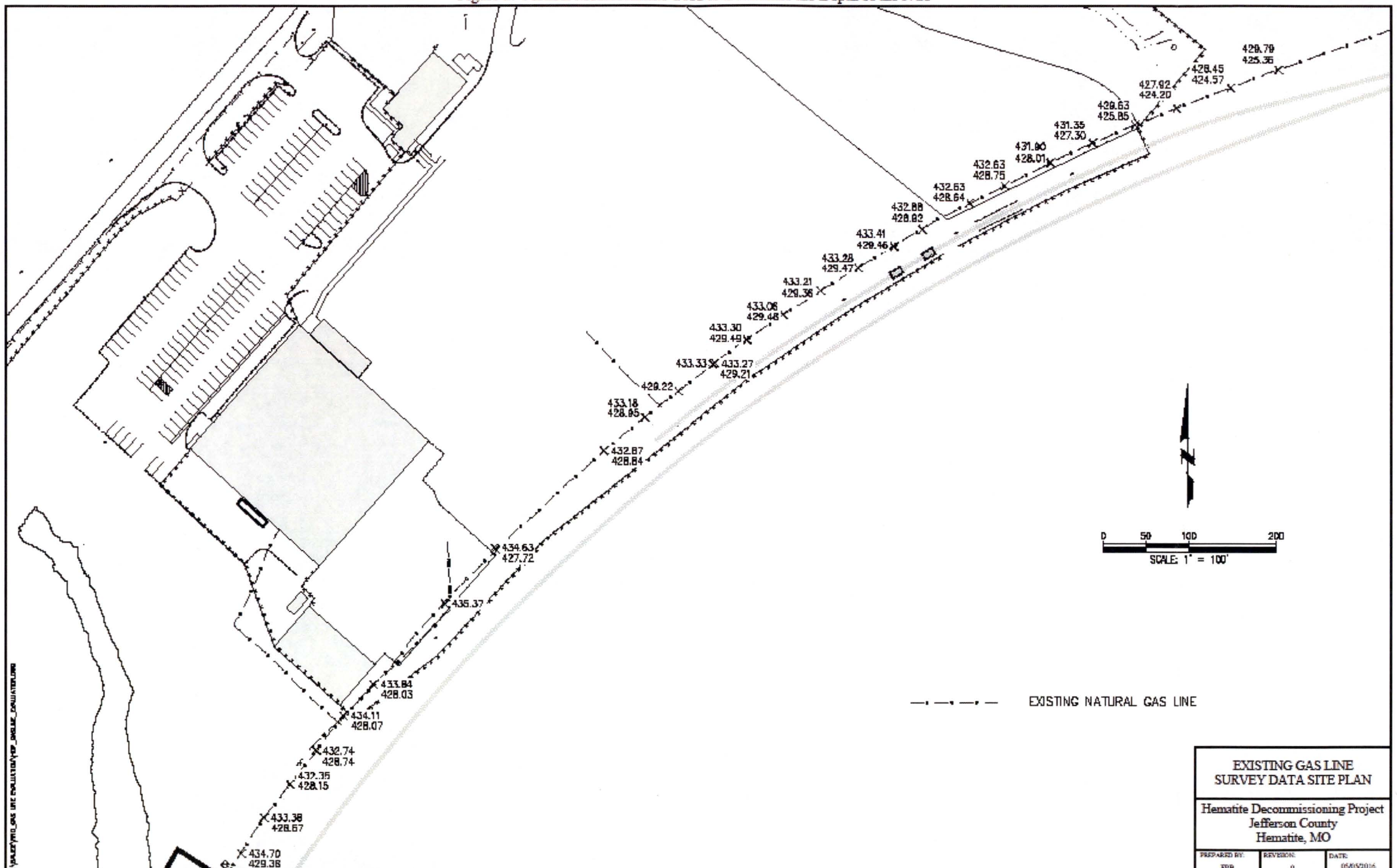


Figure 2-44

Remediation Activities at the Natural Gas Pipeline (Looking NE)

Figure 2-45

The Wood and Tile Barns Prior To Demolition (Looking W)

Figures 2-46 through 2-51
Barns Demolition Photographs

DELETED

(see FSSFR Volume 4, Chapter 1, Building Survey Areas Overview)

Figure 2-52
Barns Area Being Backfilled (Looking N)

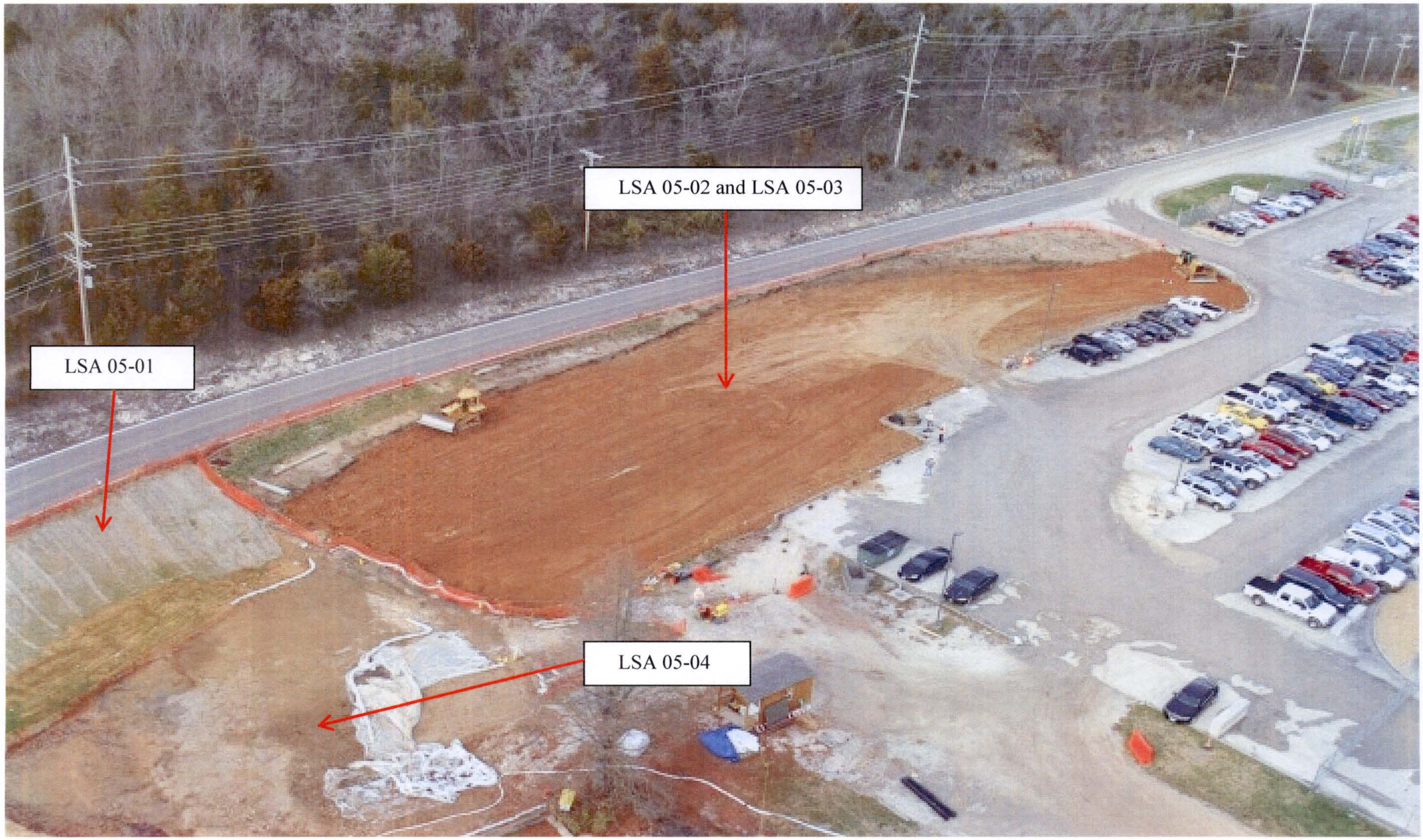


Figure 2-53
February 2013 - Remediation of the Red Room Roof Burial/Cistern Burn Pit Area (Looking NE)

Figure 2-54

Concrete Slabs Covering the Location of the Septic Tank At the Southeast Corner of Building 230

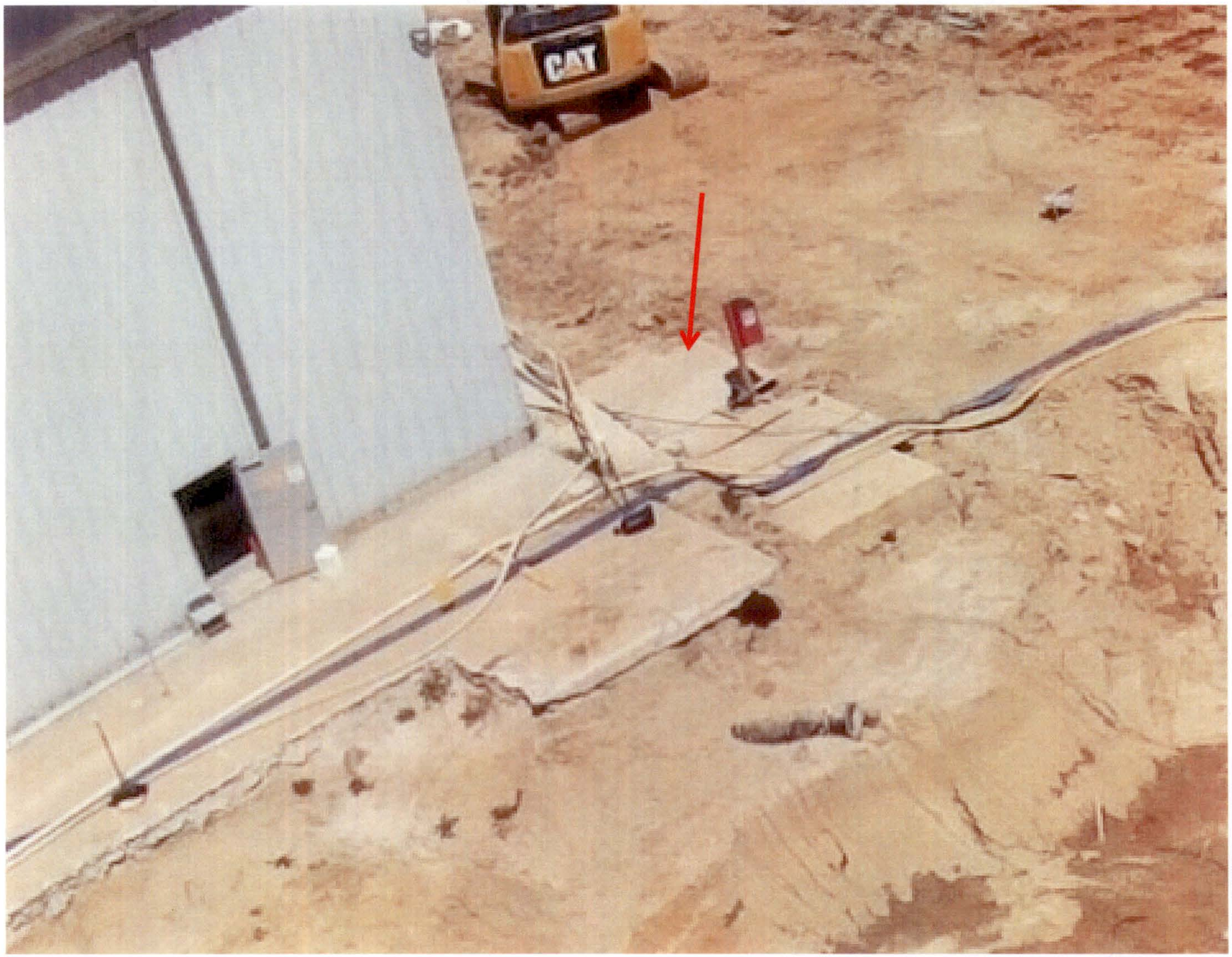


Figure 2-55

Interior of the Septic Tank

Figure 2-56
Excavation of Leach Field (Lower Right) (Looking W)

Figure 2-57
The SWTP Tanks Location Prior To Demolition and Removal (Looking NE)
(Between Building 230 to the Left and the Evaporation Ponds on the Right)

Figure 2-58

Sanitary Wastewater Tanks (Looking SW)

Figure 2-59

Excavation Remaining After Demolition and Removal of the SWTP Tanks

Figure 2-60

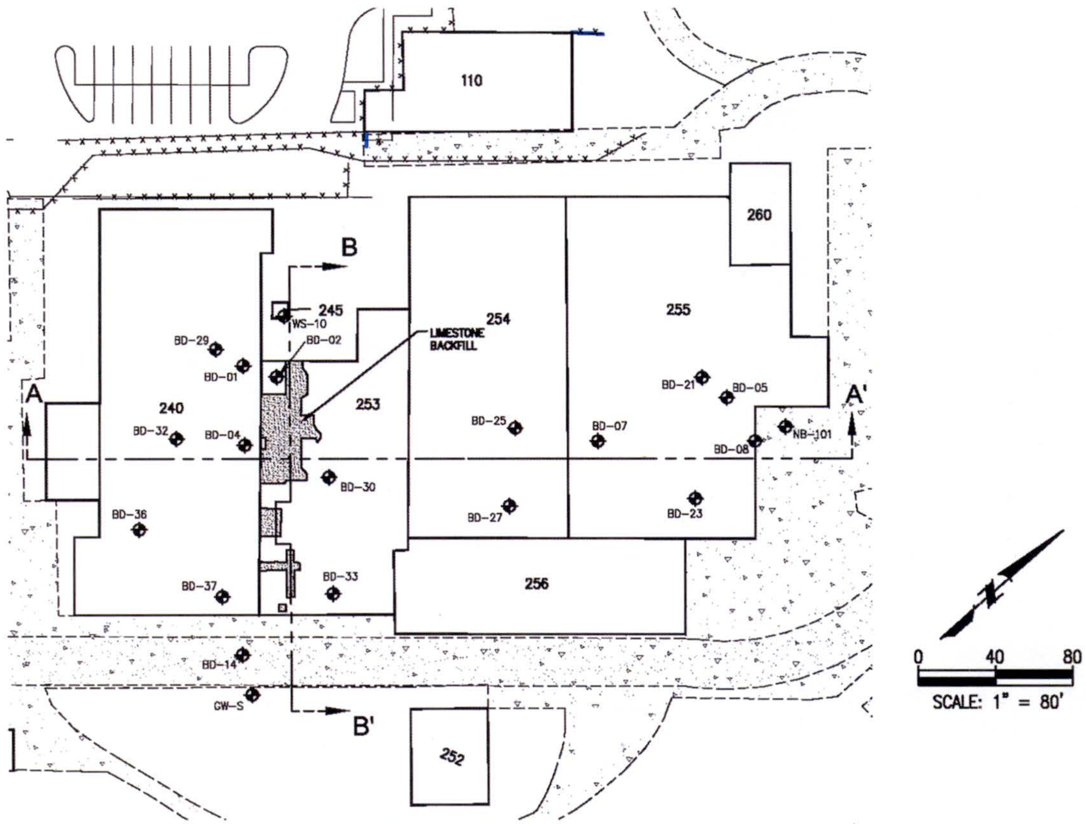
Site Pond Prior to Excavation and Remediation (Looking NE)

Figure 2-61
Installation of the Site Pond Diversion Ditch Which Was Constructed To Divert Water From The Site Spring Around And Downstream Of The Site Pond (Looking N)

Figure 2-62

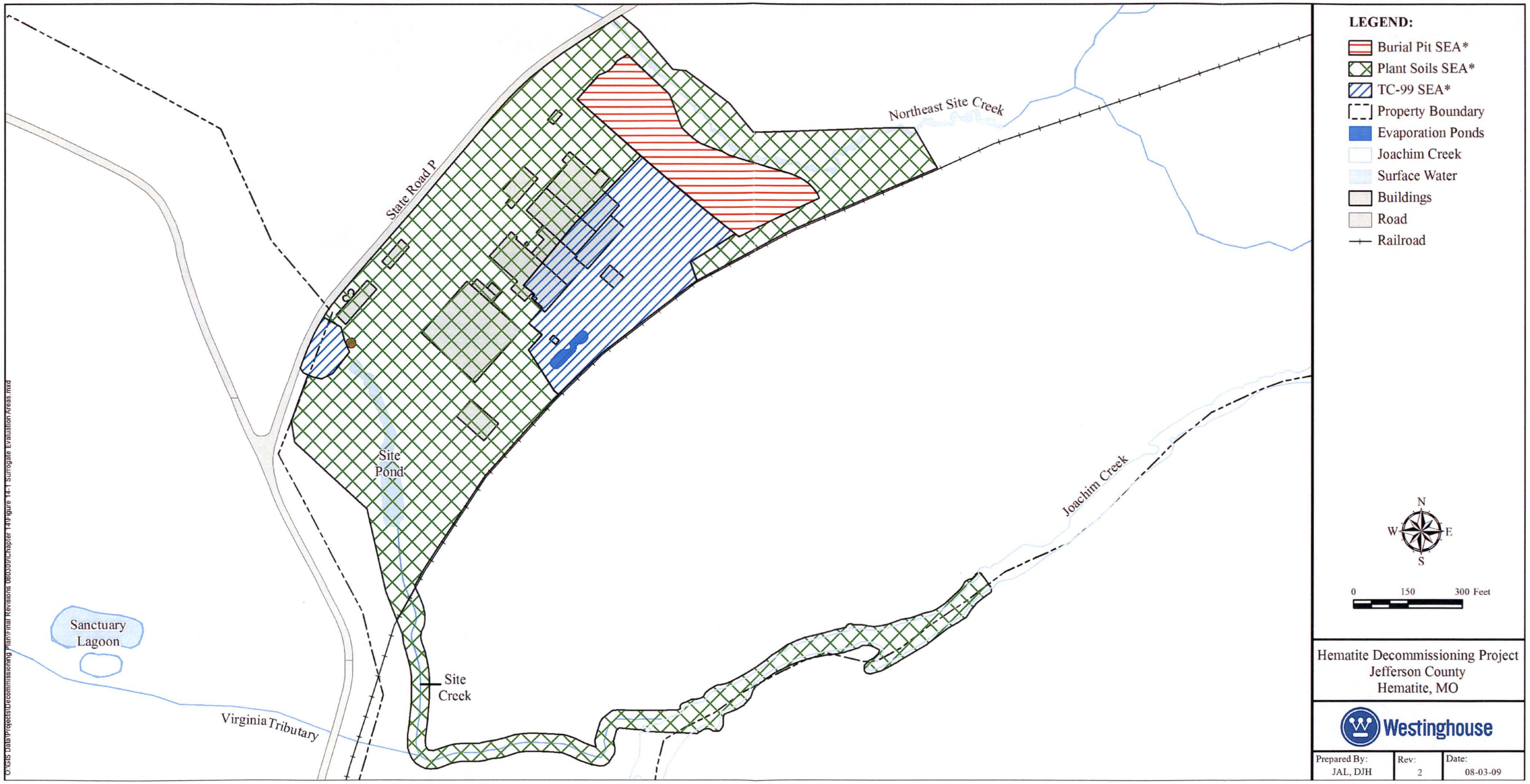
Site Pond Excavation and Remediation Nearing Completion (Looking S)

Figure 2-63
Location of Limestone Fill Under Building 253



SITE PLAN

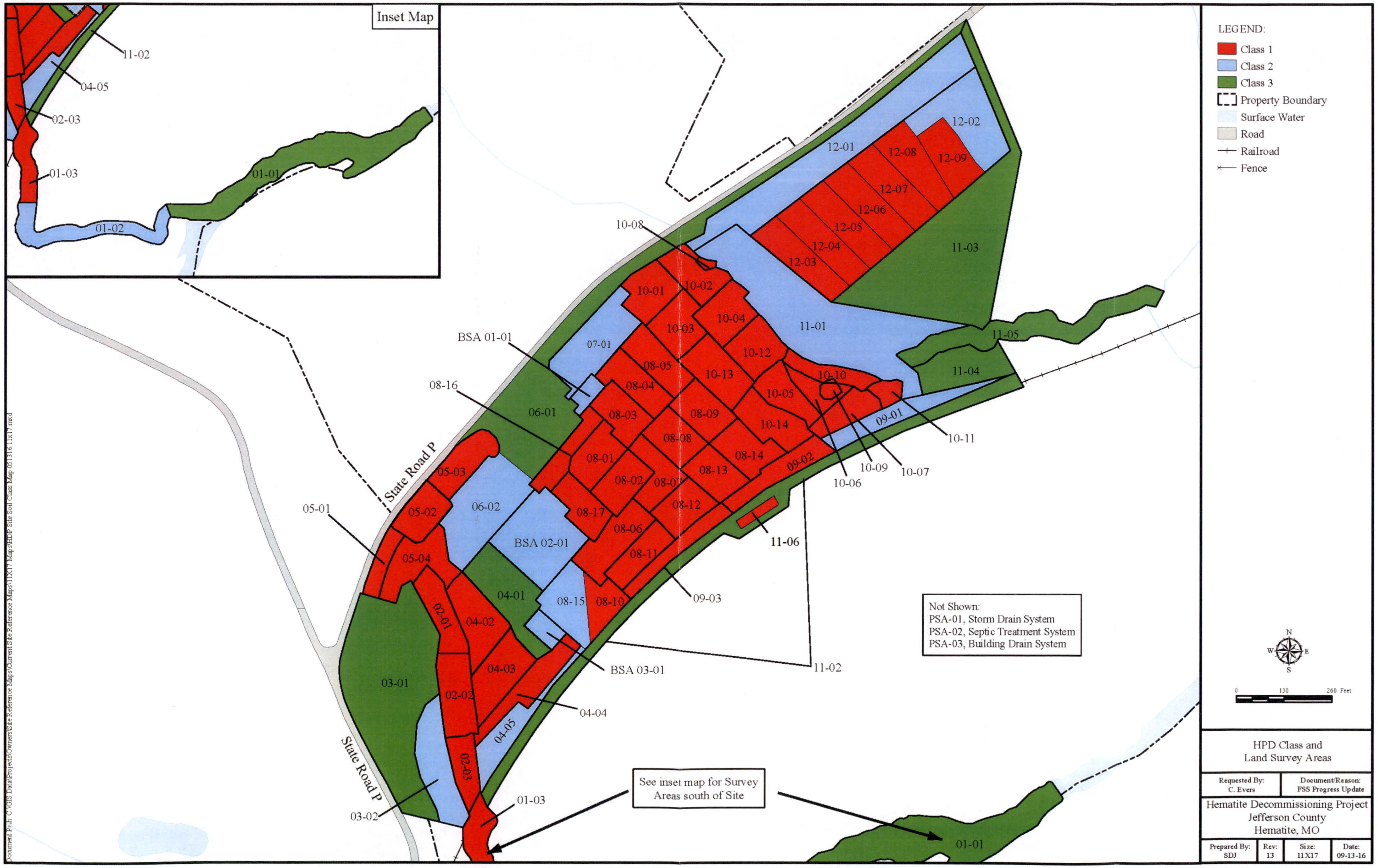
Figure 5-1
HDP Site Surrogate Evaluation Area Map



*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.

Appendix A
Final Configuration of HDP Land Survey Areas



Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PO-FSS-700 Final Status Survey Program		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the policy. Implements the requirements of the Final Status Survey plan contained within the DP.
1	12/04/2012	Updated policy to include soil sampling requirements during the abandonment of hybrid wells, per Section 14.4.3.4.2 of the DP.
2	01/31/2013	Updated Section 13, Final Status Survey Reporting to reflect the changes of Section 14.6 in the DP.
3	11/26/2014	Revision included clarifications and enhancements.
4	02/19/2015	Added information regarding scan MDCs and revised wording to be consistent with Westinghouse letter HEM-11-56.
5	10/28/2015	Changed policy to Westinghouse Proprietary Class 2. No technical changes.
6	04/07/2016	The revision incorporates a new section 15, "Surveillance Following FSS" as a component of a corrective action to a Notice of Violation. Clarifications have been made to section 9.7 regarding isolation and control.

HDP-PR-FSS-701 Final Status Survey Plan Development		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the procedure.
1	02/04/2013	Provided clarification on soil sampling by stratum as indicated in Decommissioning Plan Table 14-24.
2	02/12/2013	Provided additional instructions for creating FSS Plans for Reuse Soil.
3	11/26/2014	Significant revision for clarification and minor corrections.

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-701 Final Status Survey Plan Development		
Revision Number	**Effective Date**	**Summary of the Revision**
4	01/07/2015	Subsequent to comments received by NRC Region III regarding content of this procedure a technical readiness review was performed and the procedure revised accordingly.
5	02/11/2015	Updated the scan MDCs for U, Th-232 and Ra-226.
6	03/25/2015	Clarification of guidance for background ranges including acceptable ranges for use of a 10,000 cpm background for calculations and direction when background values are outside that range or survey parameters differ from those in HDP-TBD-FSS-002.
7	06/15/2015	Added information regarding piping survey plans, updated Ra-226 in-growth background value, clarified mean of SO equation, added direction on adjusting grid spacing to account for potential Tc-99 hotspots.
8	08/21/2015	The revision is initiated upon an agreement between the NRC and Westinghouse HDP in regards to Tc-99 sidewall sampling. The agreement was reached during a NRC Public Teleconference Meetings held on August 12, 2015, and August 19, 2015
9	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.
10	11/19/2015	Resolution and clarification of 100% GWS based on discussions with NRC Headquarters.

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-703 Final Status Survey Quality Control		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the procedure.
1	11/26/2014	Clarification for SSC Survey units.
2	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

HDP-PR-FSS-711 Final Status Surveys and Sampling of Soil and Sediment		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the procedure.
1	02/04/2013	Updated reference to HDP-PR-FSS-701.
2	02/13/2013	Typographical correction. The procedure still had "Draft A, Proposed 1" listed under the revision heading on the title page when it was placed onto SharePoint.
3	04/25/2013	Provided clarification when survey instructions cannot be followed as written.
4	11/26/2014	Added instructions for notifications when working in or near physical security structures and components.
5	02/09/2015	Added steps to provide further details on where to document background readings and how to apply the backgrounds during the survey.
6	04/15/2015	Added detail for the notification and direction in the event that isolation controls are breached in a survey unit.
7	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.
8	11/19/2015	Added clarification to Step 8.4.3 that a 100% GWS is required in Class 1 areas, and that professional judgment may be used to scan additional areas than the minimum requirements for Class 2 and Class 3 areas.

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-720 Final Status Survey Data Integrity and Database Management		
Revision Number	**Effective Date**	**Summary of the Revision**
0	09/13/2011	Initial issuance of the procedure.
1	01/13/2015	Revised section for Data Download to update the process for downloading data to a FSS computer that is password protected. Inclusion of a section to review and store GWS data. Addition of new Data Download Report Form. Changed back up of FSS data from being backed up as part of the routine site media backup process to being backed up on a password protected external hard drive and included a minimum frequency of weekly for hard drive backup.
2	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

HDP-PR-FSS-721 Final Status Survey Data Evaluation		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the procedure.
1	02/06/2013	Administrative revision only, forms changed to appendices.
2	02/20/2013	Administrative revision only, references updated.
3	11/26/2014	Provided clarification SOF, Corrected typographical error in SOF equation and other steps.
4	01/06/2015	Minor revision to insert procedure number in front of procedure title in Section 5.0 References.
5	02/09/2015	Clarifications on the calculation of statistics in Section 8.3 and SOF in Section 8.4, Additional guidance on completion of the WRS test and correct terminology used for the Test Statistic, Clarifications between the requirements for soil and structural survey units, added notes to clarify using the DCGLs used to develop the FSS plan unless instructed otherwise by the RSO, added step to Appendix G-1 to verify Laboratory quality control parameters are within acceptable

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-721 Final Status Survey Data Evaluation		
Revision Number	**Effective Date**	**Summary of the Revision**
		limits.
6	03/25/2015	Clarifications on the calculation of statistics to correct negative values to zero when listing basic statistical data and listing the background values. Replaced Appendix C with table 14-4 from Attachment 4 to HEM-11-96 as this table is more appropriate since inferred Tc-99 is prohibited for use to demonstrate compliance. Added a note to clarify that it is prohibited to use U-235 to infer Tc-99 to demonstrate compliance. Removed references to SEA in the procedure and in the acronym list. Various terminology corrections. Added the calculation to use for obtaining a U-234 value when not measured for the calculation. Changed scan action level for Class 1 to be consistent with change to HDP-PR-FSS-701. Added underground piping.
7	04/15/2015	Added detail in Step 7.5.4 as to notification and direction in the event that isolation controls are breached in a survey unit.
8	06/15/2015	Updated the Ra-226 soil background value. Added instruction for use of other DCGL conceptual site models rather than only the uniform DCGL.
9	08/13/2015	This revision implements clarification on the determination of dose for survey units. The changes are based upon an agreement between the U.S. NRC and Westinghouse in regards to calculating dose. The technical changes were agreed upon during the NRC-Westinghouse teleconference held on August 12, 2015.
10	10/28/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-FSS-722 **Final Status Survey Reporting**		
Revision Number	**Effective Date**	**Summary of the Revision**
0	09/13/2011	Initial issuance of the procedure.
1	02/05/2015	Updated to reflect changes to HDP-PO-FFS -700 regarding FSS reporting.
2	01/31/2014	Removed the formatting guidelines which were contained in Appendix A and B.
3	01/13/2015	Editorial corrections.
4	11/06/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

HDP-PR-HP-416 **Operation of the Ludlum 2221 for Final Status Survey**		
Revision Number	**Effective Date**	**Summary of the Revision**
0	01/16/2012	Initial issuance of the procedure.
1	02/09/2015	Clarified instruction that the probe be kept as close to the ground as possible, nominally 1 inch from the surface and not to exceed 3 inches. Clarified instruction to scan at 1 foot per second and that head phones should be used when it is noisy and interferes with hearing audible count rate. Revised wording consistent with current wording in FSS instructions. Added a step to stop and pause when elevated count rates are identified to determine if further investigation is needed. Added clarification that documentation using HDP-PR-FSS-701 is following directions as specified in the FSS instructions.
2	11/19/2015	Administrative changes. Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

Appendix B
Volume 3 Chapter 1 - HDP Procedure Revision History

HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
0	03/14/2012	Initial issuance of the procedure.
1	04/09/2012	Updated procedure to accommodate performing 6 inch excavation lifts instead of 12 inch excavation lifts, updated Table 1 Column B screening count rate to 46k cpm for saturated soil, which is more conservative than the 58k cpm and is associated with a lump of material, Updated flow chart in Appendix B to move evaluation of items after the initial survey, and to include pipe surveys, Section 8.1.6.2 updated to identify items for further evaluation and evaluate them after the initial survey to determine if they are NCS Exempt. Clarifications for the count rates are less than or equal to (≤) instead of "less than", consistent with wording in Section 8.7 of this procedure and the applicable NCSA, removed "general area" to clarify that the background count rate used will be obtained from the daily background check. Added reference to NSA-TR-HDP-11-06 for surveys of pipe and added section for surveying piping to determine if it is NCS Exempt. Specified that the screening level for reuse is on Form HDP-PR-HP-100-2.
2	06/11/2012	Administrative changes and clarifications.
3	06/26/2012	Updated procedure to use the general area background count rate when performing surveys or items and containers and Sections 8.4.3 and 8.7 to specify the external surfaces of containers need to be surveyed, not only the bottom of the containers.
4	07/05/2012	Updated Section 8.2 to clarify that the radiological surveys do not need to be formally documented prior to determining if an area meets NCS Exempt criteria and can be excavated. Additionally, 2 independent HP Technicians can verify all hotspots/items have been evaluated and handled required to ensure the surveyed area meets NCS Exempt Material criteria. Updated Section 8.4.3 to remove the requirement to have intact containers in a vertical position when surveying them because this is not required as all external surfaces of

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		the container are surveyed. Updated Section 8.7 to clarify the criteria for Field Containers with net count rates between 300,000 and 1,400,000 cpm.
5	08/14/2012	The requirement to identify areas that are NCS Exempt for a 12 inch cut depth (i.e., areas that are 19k ≤ ncpm < 66k dry soil, marked as blue) was removed because the maximum permitted cut depth will be 6 inches. Blue will no longer be used to mark areas as it is no longer applicable since it is associated with the NCS Exempt criteria for a 12 inch cut depth. This change includes the removal of Column A in Table 1. Updated the count rates in Table 1 for Columns C & D, and in associated locations throughout the procedure. These count rates were previously based on a cut depth of 7 inches (an extra 1 inch greater than the cut depth for conservatism). The count rates were adjusted in this revision to the count rates associated with a cut depth of 6 inches. Based on operational experience and the current maximum permitted cut depth (i.e., 6 inches), an extra 1 inch of conservatism is not necessary. Updated the NCS Field Container Loading Limit from 1,400k ncpm to 1,500k ncpm. 1,400 ncpm was based on 102 g235U, assuming up to 3 field containers in a collared drum (total mass in a collared drum <350 g235U). Since only 2 field containers can fit into a collared drum, the count rate was increased to 1,500k based on 150 g 235U.
6	08/23/2012	Corrected misnumbered step references.
7	09/06/2012	Added note to include the segregation of material with substantially differing characteristics (i.e., radiological, media type) when loading a field container.
8	12/03/2012	NSA-TR-09-15 Revision 4 was issued. Changes to the CSCs in Revision 4 were incorporated throughout the procedure. Sections 7.12 and 8.9 were revised to include instructions on assigning a 235U gram quantity based on close proximity radiological survey results.
9	03/07/2013	This was a significant revision which incorporated nuclear criticality information and requirements based upon NSA-

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		TR-09-08, NCS of Subsurface Structure Decommissioning at Hematite Site and NSA-TR-11-11, NCS Assessment of the USEI Site for Landfill disposal of Additional Decommissioning Waste from Hematite Site.
10	04/08/2013	Added to Step 8.1.7.2 "NOTE: Sections of subterranean pipe less than the NCS Exempt Piping Limit may be crushed and treated as soils, in accordance with HDP-PR-HP-607 (Reference 5.18); however, depending on the size, thickness and material type of the subterranean piping pieces (i.e., Bulky pieces with linear dimensions exceeding the permitted cut depth, and thick metallic items) should be treated as Non-Conforming Items. (NSA-TR-09-08 CSC 11)".
11	05/01/2013	Instructions were added for evaluating hot spots using the LaBr3 probe.
12	05/22/2013	A clarification was made in regards to evaluating intact containers.
13	06/21/2013	Added instruction to ensure Pre and Post Assay QC checks are performed in accordance with HDP-PR-NC-008, Probe, when the LaBr3 probe is used to determine if an area meets NCS Exempt requirements.
14	07/22/2013	Added instruction to allow for Ex-situ scanning of soil in less than 6 inch layers. Added instruction for surveying material underneath the Former Building 240 Slab to allow excavation of material up to 0.8 g235U/L as NCS Exempt.
15	07/25/2013	Instructions added for performing measurements using a NaI 2x2 detector with a "Closed Window" setting of 75 to 250 keV and added option to survey a greater than 5L container to less than or equal to 34k to determine if it meets NCS exempt criteria.
16	08/07/2013	Revised to incorporate the requirements of HEM-13-MEMO-067, Remediation of Soil Under the Concrete Slabs.
17	08/15/2013	Revised to allow the survey of piping from discovered Burial

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		Pits to be performed in one foot increments.
18	09/11/2013	Redundant FSS design requirements were removed from as they are captured in FFS procedure. Introduced the terms "Large Volume Container" and "Small Volume Container" throughout procedure. Removed all instructions for Inspector 1000 with LaBr3 probe as it will no longer be used to evaluate "hot spots" as part of the initial radiological survey. Extensive changes were made to this procedure to remove directions on performing measurements using a NaI 2x2 detector with a "Closed Window" setting of 75 to 250 keV. Steps were added to Section 8.5 to satisfy expanded NCS Controls to perform a visual inspection of Ex-situ excavated material in layers prior to consolidation at the WCA.
19	09/12/2013	Step 8.5.2 was revised to clarify the requirement of NSA-TR-09-15 CSC 25, to "Identify and demarcate an area, using paint, flags, or other appropriate means, adjacent to the excavation area that does not contain any Non-NCS Exempt Material (identified in accordance with Section 8.1)".
20	09/30/2013	The term Inspection Items was introduced. References to the use of GPS in section 8.1 were removed since GPS use is considered optional. References to DCGL were replaced with Reuse Material Screening Level (RML) and a count rate of 12k ncpm is now specified. Section 8.4 was revised to include detailed direction on how intact containers are to be emptied in the field. Section 8.9 title was changed to "Surveys of Non-Conforming Items or Items Potentially Containing Fissile Material"
21	10/01/2013	Minor typographical errors were corrected from the previous revision.
22	10/23/2013	Step added to "Identify the area to be excavated" prior to excavation of an area. Added that the adjacent working surface should be demarcated with pink paint or flags. Step was revised to specify that a GWS with Visual Inspection

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
		will be performed of the ex-situ soil layer regardless if visible Non- Conforming Items were identified.
23	11/22/2013	Added instruction for remediation of soils surrounding radium filter press plates in the Burial Pits.
24	01/10/2014	Revised the scope of Section 8.10 to apply to all pavement and building slabs covering soil under NCS controls. Added saturated soil numbers to Sections 8.11 and 8.12. Added step to address the use of ISOCS measurements performed in the field as part of evaluation of radium contaminated soil areas.
25	05/19/2014	Incorporated the requirements of Memo HEM-13-MEMO-099, Radiological Requirements for the Handling of Re-Use Soils During Development of Stockpile 8. Added the requirements of NSA-TR-09-08 CSC 17, and CSC 20 and NSATR- HDP-11-11 CSC 15 related to septic and sewage treatment tanks. Removed references to remediation contractor.
26	06/02/2014	Updated work package references throughout procedure, as work package numbers have changed.
27	07/10/2014	Added potential NCS Evaluation for Non-Conforming Items which are > 0.1 g U-235/L to Appendix C.
28	08/11/2014	Clarified the requirements for excavation cut depth outside of NCS controlled areas. Clarified how "areas where buried debris has been identified" is defined by NCS.
29	09/18/2014	Added instruction to allow the third visual inspection and processing of potentially NCS exempt material to take place on a working surface that will no longer be required to be located adjacent to the excavation area.
30	11/05/2014	A step was added to provide guidance when performing surveys and visual inspections under artificial lighting.
31	11/06/2014	Administrative changes.

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HDP-PR-HP-601 Remedial Action Support Surveys		
Revision Number	**Effective Date**	**Summary of the Revision**
32	12/05/2014	Added instruction to implement the enhanced work controls developed to ensure compliance with the USEI WAC when excavation in the Site Pond. Added Reference to HEM-14-MEMO-112, NCS Analysis of the HDP Site Pond Remediation Activities.
33	02/04/2015	Added instruction on the direct loading of waste for disposal at a NRC licensed burial facility up to a concentration of 0.8 g235U/L.
34	11/19/2015	Changed procedure to Westinghouse Proprietary Class 2. No technical changes.

Appendix C
HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-01 South Site Waterways					
LSA-01-01	Site Creek/Joachim Creek	3	12/16/15	20	
LSA-01-02	South Section of Site Creek	2	1/14/16	20	
LSA-01-03	North Section of Site Creek	1	1/14/16	20	
LSA-02 Site Pond					
LSA-02-01	North Section of Site Pond	1	9/29/15	20	
LSA-02-02	Central Section of Site Pond	1	9/29/15	20	
LSA-02-03	South Section of Site Pond	1	9/29/15	20	
LSA-03 West Open Land Area					
LSA-03-01	Area West of Site Pond	3	11/6/15	20	
LSA-03-02	Area Southwest of Site Pond	2	11/12/15	20	
LSA-04 Southwest Open Land Area					
LSA-04-01	Area between Buildings 230/231 and Site Pond	3	4/5/16	15	
LSA-04-02	Area East of North Section of Site Pond (west soil laydown area)	1	4/22/16	15	
LSA-04-03	Area East of Central Section of Site Pond (west soil laydown area)	1	4/21/16	15	
LSA-04-04	Area South of Building 231	1	4/12/16	15	
LSA-04-05	Wooded Area South of Building 231	3	6/21/16	15	
LSA-05 Barns and Cistern Open Land Area					
LSA-05-01	Site Spring Area adjacent to State Road P	1	3/9/14	16	
LSA-05-02	Tile Barn and Red Room Roof	1	9/13/13	16	

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HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-05-03	Wood Barn	1	11/7/13	16	
LSA-05-04	Site Spring and Cistern	1	4/27/16	16	
LSA-06 North Open Land Area					
LSA-06-01	Main Parking Lot	3	6/24/16	17	
LSA-06-02	West Parking Lot	2	6/17/16	17	
LSA-07 North Central Open Land Area					
LSA-07-01	Truck Scale Area	2	5/3/16	17	
LSA-08 Central Open Land Area					
LSA-08-01	Process Building Area Section 1	1	3/15/16	12	
LSA-08-02	Process Building Area Section 2	1	2/4/16	12	
LSA-08-03	Process Building Area Section 3	1	3/18/16	11	
LSA-08-04	Process Building Area Section 4	1	4/7/16	10	
LSA-08-05	Process Building Area Section 5	1	4/12/16	10	
LSA-08-06	Process Building Area Section 6	1	1/6/16	11	
LSA-08-07	Process Building Area Section 7	1	1/7/16	11	
LSA-08-08	Process Building Area Section 8	1	4/7/16	10	
LSA-08-09	Process Building Area Section 9	1	4/28/16	12	
LSA-08-10	Process Building Area Section 10	1	7/13/16	14	
LSA-08-11	Process Building Area Section 11	1	12/16/15	13	
LSA-08-12	Process Building Area Section 12	1	4/22/16	12	
LSA-08-13	Process Building Area Section 13	1	4/21/16	12	
LSA-08-14	Process Building Area Section 14	1	5/24/16	10	
LSA-08-15	Process Building Area Section 15	1	7/18/16	14	
LSA-08-16	Process Building Area Section 16	1	3/17/16	11	
LSA-08-17	Process Building Area Section 17	1	1/19/16	11	

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HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-09 Rail Spur open Land Area					
LSA-09-01	East Rail Spur Area	2	5/24/16	18	
LSA-09-02	Central Rail Spur Area	1	7/5/16	18	
LSA-09-03	West Rail Spur Area	1	5/24/16	18	
LSA-10 Burial Pits Open Land Area					
LSA-10-01	Burial Pit Area Section 1	1	6/17/15	2	10/27/2016
LSA-10-02	Burial Pit Area Section 2	1	6/17/15	2	10/27/2016
LSA-10-03	Burial Pit Area Section 3	1	6/17/15	3	11/07/2016
LSA-10-04	Burial Pit Area Section 4	1	6/17/15	3	11/07/2016
LSA-10-05	Burial Pit Area Section 5	1	2/13/14	6	
LSA-10-06	Burial Pit Area Section 6	1	1/10/14	6	
LSA-10-07	Burial Pit Area Section 7	1	1/10/14	6	
LSA-10-08	Burial Pit Area Section 8	1	9/2/15	6	
LSA-10-09	Burial Pit Area Section 9	1	10/21/13	6	
LSA-10-10	Burial Pit Area Section 10	1	2/20/14	6	
LSA-10-11	Burial Pit Area Section 11	1	5/21/15	7	12/12/2016
LSA-10-12	Burial Pit Area Section 12	1	6/17/15	4	11/14/2016
LSA-10-13	Burial Pit Area Section 13	1	6/10/15	5	11/16/2016
LSA-10-14	Burial Pit Area Section 14	1	6/10/15	5	11/16/2016
LSA-11 East Open Land Area					
LSA-11-01	Northeast Site Creek	2	10/29/15	7	12/12/2016
LSA-11-02	Rail Road Line	3	7/5/16	19	
LSA-11-03	East Site Wooded Area	3	6/24/15	19	
LSA-11-04	Small East Site Wooded area	3	4/24/15	19	
LSA-11-05	Northeast Site Creek East Section	3	6/3/15	19	
LSA-11-06	Rail Road Line Elevated Area	1	7/5/16	19	

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HDP FSSFR LSA Document Matrix

Survey Unit	Description	FSS Class	FSS Complete Date	Assigned FSSFR Volume 3 Chapter Number	Date Submitted to NRC
LSA-12 Lay Down Area					
LSA-12-01	Reuse Soil Laydown Area Section 1	2	7/13/16	8	01/05/2016
LSA-12-02	Reuse Soil Laydown Area Section 2	2	7/13/16	8	01/05/2016
LSA-12-03	Reuse Soil Laydown Area Section 3	1	7/12/16	9	01/05/2016
LSA-12-04	Reuse Soil Laydown Area Section 4	1	7/12/16	9	01/05/2016
LSA-12-05	Reuse Soil Laydown Area Section 5	1	7/12/16	9	01/05/2016
LSA-12-06	Reuse Soil Laydown Area Section 6	1	7/12/16	9	01/05/2016
LSA-12-07	Reuse Soil Laydown Area Section 7	1	7/12/16	9	01/05/2016
LSA-12-08	Reuse Soil Laydown Area Section 8	1	7/12/16	9	01/05/2016
LSA-12-09	Reuse Soil Laydown Area Section 9	1	7/12/16	9	01/05/2016

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INTER-OFFICE MEMORANDUM

Date: March 2, 2015 HEM-15-MEMO-021

To: Brian Miller, FSS Manager
Ellen Jakub, FSS Data Manager

From: W. Clark Evers, Radiation Safety Officer

Cc: Steven Grice, FSS Task Manager

Subject: Evaluation of the Scan Investigation Action Level (IAL) for Class 1 areas at the Westinghouse Hematite Site

Class 1 areas at the Hematite site are impacted areas requiring remediation where historic sampling, characterization, or remediation surveys have detected contamination above the applicable DCGLw. Therefore radioactivity above background levels may be detected during the performance of Final Status Surveys (FSS) after all necessary remediation has been completed. It is for this reason that a Scan IAL is provided as part of the FSS plan. The IAL provides the HP Technicians performing field work a count rate above general area background that may indicate soils concentrations approaching the applicable DCGLw that should be investigated further.

Attempting to calculate the Scan IAL using theoretical values poses significant challenges since the Sum of Fractions (SOF) must be calculated based on several different isotopes, all with different physical characteristics, specific activities, and DCGLw values. For this reason a review of empirical data was performed.

To determine what an appropriate Scan IAL is for Class 1 areas of the site, all Final Remedial Action Support Surveys (RASS) of "Area 1" (i.e. LSA's 10-01, 10-02, 10-03, 10-04, and 10-12) were compiled calculating the mean count rate and standard deviation of the data set. All 5 of these LSA's are Class 1 areas where remediation was necessary and had proceeded until completion indicated that the areas were ready for FSS. Additionally all 5 of these areas did at one time contain contamination from all 6 of the contaminants of concern (i.e. U-234, U-235, and U-238, Ra-226, Th-232, and Tc-99), and were representative of a typical Class 1 area on the

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Hematite Site. The mean count rate of the population was calculated to be 10,698 cpm, with a standard deviation of 1616 cpm. This is consistent with the average general area background observed on the Hematite site of approximately 10,000 cpm as reported in the Hematite Radiological Characterization Report (HRCR). The Gamma Walkover Survey (GWS) data for Area 1 is attached in Figure 1.

MARSSIM Chapter 5.5.2.6 Determining Investigation Levels describes the industry standard practice of flagging areas for investigation that exceed background levels by a value of 3 sigma. For "Area 1" this would represent a count rate of 4848 cpm above general area background. For ease of use in the field, and to provide some inherent conservatism, this value will be rounded down to 4,000 net cpm.

While count rates observed in the field above the Scan IAL of 4,000 ncpm could be an indication of soil concentrations approaching the DCGL, it could also be an indication of a change in soil surface features (e.g. sidewall or trench) or fluctuating background. It is for this reason that when the HP Technician identifies areas above or approaching the Scan IAL in the performance of FSS that site procedures instruct the HP Technician to pause and investigate the area further. If the HP Technician determines based on professional judgment that the area does in fact exceed 4,000 net cpm above the general area background then the area should be "flagged" for further investigation. Chapter 14.3.2 of the Hematite Decommissioning Plan states "The average net count rate corresponding to the DCGLw 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". Therefore the Scan IAL of 4,000 ncpm was compared to the biased sample results that were collected during Final RASS of "Area 1".

LSA 10-02 provides a good example of a Land Survey Area that presents a favorable survey geometry (e.g. area is relatively flat, lacking sharp side walls), so soil areas with elevated count rates could be sampled for comparison, and have no concern for a "low bias" (e.g. sampling clean soil area due to unfavorable counting geometry). As can be seen in Figure 1 there are three distinct soil areas that are elevated above 3 sigma from the mean (circled in Figure 1), and one biased soil sample was collected at each of these locations. The results were compared to the Uniform DCGLw using the Tc-99 surrogate value for U-235 since Tc-99 has no gamma emitter and cannot be readily identified using field scanning techniques. To be consistent with FSS data analysis, values less than the minimum detectable activity (MDA) were set to zero, U-234 was calculated based on enrichment, and the 0.9 pCi/g Ra-226 and 1.0 pCi/g Th-232 background values were subtracted. The biased sample results are presented in Table 1, the maximum sample SOF was 0.44 of the Uniform DCGLw using the conservative Tc-99 surrogate value.

Lastly, consideration should be given to the potential presence of discrete material. While the Uniform DCGLw for soil is based on diffuse contamination, discrete contamination (e.g. radioactive debris or pellet fragments) must also be identified and removed to the maximum extent practical. A previous FSS survey was performed in an area where a fragment of a

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Uranium fuel pellet was identified. This event was documented in CAPS Item # 13-155-W008.01, and it was identified in the findings of the CAPS investigation that small discrete items are difficult to detect when buried beyond 3 inches beneath the soils surface. However since at the time FSS is to begin, all remediation activities in the area have been completed it is reasonable to assume that any remaining discrete items will be within the top 3 inched of soil. The small pellet fragment that was identified during FSS previously was evaluated to theoretically result in an additional dose to the Survey Unit (SU) of 0.08 milliRem per year, and exhibited a count rate of 5,445, ncpm with the probe held 1 inch above the soils surface, and the pellet fragment buried at approximately 3 inches. This count rate is significantly greater than the proposed Scan IAL of 4,000 ncpm and therefore would provide a high level of confidence that any potential remaining discrete items would be identified and flagged for further investigation and subsequent removal.

It is the conclusion then that the proposed Scan IAL of 4,000 ncpm is appropriate for use in Class 1 survey units as it represents a value that is conservatively less than the DCGLw and thus will also be significantly less than the DGCLEMC. Additionally, all GWS data will be "post-processed" and reviewed by the FSS Supervisor to determine if any additional areas require investigation based on the recorded GWS data.

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Figure 1 – Area 1 GWS

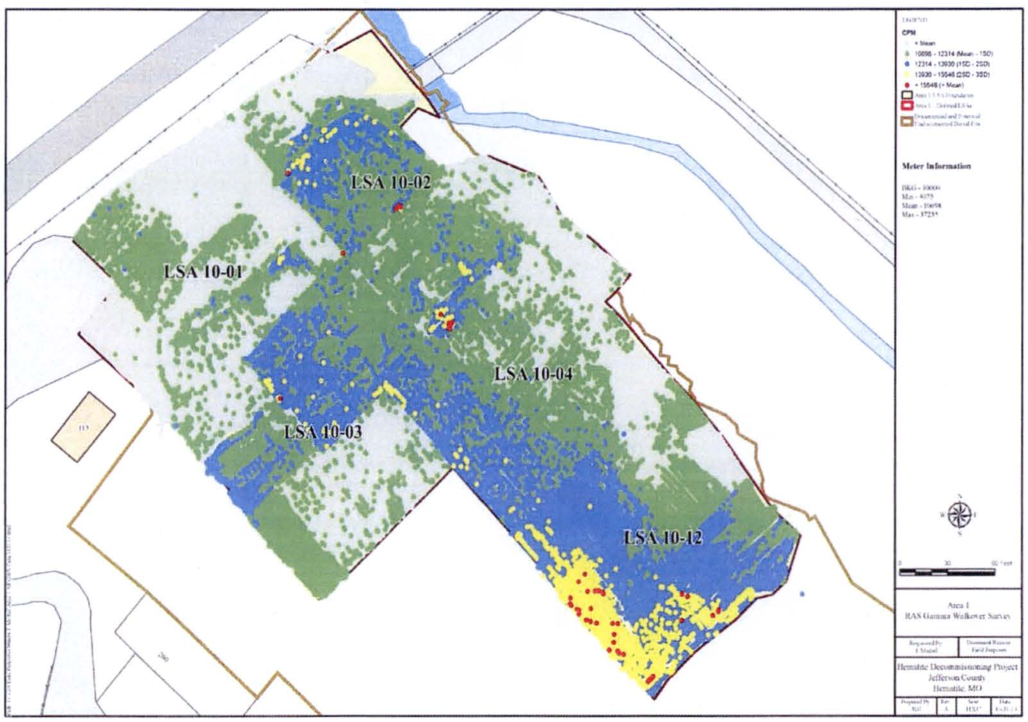

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Table 1 – LSA 10-02 Biased Results

SOF Calculation

	RA226	Tc99	Th232	U234	U235	U238
Avg. Conc	0.27	0.00	0.05	7.70	0.38	1.29
DCGL	1.9	25.1	2	195.4	5.8	168.8
SOF	0.14	0.00	0.02	0.04	0.06	0.01

SOF= **0.28** **MAX_Sample (SOF)** **0.44**

RaBKGD 0.90
ThBKGD 1.00

	Ra-226	Th-232	U-235	U-238	U-238 / U-235	% En.	U-234 Ratio (enrichment) U234 /U-235	U-234	Sample SOF
0183-SS-141114-05-01	1.27	1.16	0.00	0.00	0.00	HEU	32.50	0.00	0.28
0183-SS-141114-05-02	1.02	1.08	0.18	0.00	0.00	HEU	32.50	5.95	0.17
0183-SS-141114-05-03	1.21	0.90	0.95	3.86	4.08	3.7	18.12	17.15	0.44

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Owner/Author:

W. Clark Evers
RSO

3/2/15
Date

Reviewed & Approved By:

Steven A. Grice
FSS Task Manager

3/2/15
Date