FINAL STATUS SURVEY REPORT

BUILDING 5/6A COMPLEX

CE WINDSOR SITE
2000 DAY HILL ROAD
WINDSOR, CONNECTICUT

VOLUME I
TEXT, FIGURES AND TABLES

Prepared for:

ABB PROSPECTS INC.
2000 Day Hill Road
Windsor, Connecticut 06095

April 27, 2006

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LIST OF ACRONYMS AND ABBREVIATIONS

ABB	ABB Prospects Inc.
AD	Anderson - Darling
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
A_m	area factor
ANSI	American National Standards Institute
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
cm	centimeter
Co-60	cobalt 60
cpm	counts per minute
CTDEP	Connecticut Department of Environmental Protection
DCGL	derived concentration guideline level
$DCGL_{EMC}$	derived concentration guideline level, elevated measurement comparison
$DCGL_W$	derived concentration guideline level, survey unit average (median) concentration corresponding to the permissible limit
D&D	decontamination and decommissioning
DP	decommissioning plan
DQA	data quality analysis
DQI	data quality indicator
EMC	elevated measurement comparison
EPA	Environmental Protection Agency
FSS	Final Status Survey (radiological)
FSSP	Final Status Survey Plan
FWHM	full width at half-maximum
g/cm^3	grams per cubic centimeter
GEL	General Environmental Laboratories, Inc.
GIS	geographic information system
GPS	global positioning satellite
HEU	highly-enriched uranium
HPGe	high purity germanium detector
HRR	historical review report
HSA	historical site assessment
keV	kilo-electron volts
LEU	low-enriched uranium
LBGR	lower bound of the gray region
LLRW	low-level radioactive waste

LIST OF ACRONYMS AND ABBREVIATIONS

m	meter(s)
m^2	meters squared
MACTEC	MACTEC Development Corporation
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCA	multi-channel analyzer
MDC	minimum detectable concentration
MDC_{SCAN}	minimum detectable concentration for scan surveys
$MDC_{surveyor}$	minimum detectable count rate for the surveyor
MeV	million electron volts
μR/h	micro-Roentgen per hour
mrem	milli-Roentgen equivalent man
N	sample size
NAD	North American Datum
NaI	sodium iodide detector
NFM	nuclear fuel manufacturing facility
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
pCi/g	picocuries per gram
QA	quality assurance
QC	quality control
RCRA	Resource Conservation and Recovery Act
RESRAD	Residual Radioactivity code
RSO	radiation safety officer
RSR	Remediation Standards Regulations
SITE	2000 Day Hill Rd., Windsor, Connecticut
U-234	uranium 234
U-235	uranium 235
U-238	uranium 238
UCL	upper confidence level
VSP	Visual Sample Plan code
WWTP	wastewater treatment plant

EXECUTIVE SUMMARY

ABB Prospects Inc. (ABB) has contracted MACTEC Development Corporation (MACTEC) to perform decontamination and decommissioning (D&D) of the Building Complexes 2, 5, 6A, and 17 at their facility located at 2000 Day Hill Road, in Windsor, Connecticut (Site). This report is to document the final radiological status of the soils in the Building Complexes 5 and 6A areas post-remediation.

The Building 5 Complex was built in the late 1950s and late 1960s as research and development facilities in support of nuclear fuel manufacturing. More recently, the Building 5 Complex was also utilized for nuclear plant outage and field operation support. Building 6A was built in the mid 1950s as a liquid radiological waste collection and processing facility for Building 5. Later, the liquid radiological waste from the Building 5 Complex was re-routed to Building 6 and Building 6A was converted to a maintenance service facility.

No residual radioactivity in excess of the applicable soil radioactivity release criteria was identified during Final Status Surveys (FSS). For the Building 5 Complex, some soil remediation was necessary in the former hot waste trench areas; however most samples did not have detectable concentrations of uranium or cobalt 60 (Co-60). Nine survey units were created in support of the FSS, including two Class 2 survey unit and seven Class 3 survey units.

Quality control (QC) measures were taken during the survey process. Review and analysis of the QC measures indicates that the data collected meet the data quality objectives and are acceptable for their intended use. In addition, no unexpected results or trends are evident in the data.

The design and interpretation of the final radiological status survey of the soil in support of the D&D project is based on the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) approach using the site-specific soil derived concentration guideline levels (DCGLs). The DCGLs established for soil are 557 picocuries per gram (pCi/g) for total uranium and 5 pCi/g for Co-60.

The null hypothesis for these surveys is that the residual radioactivity in the survey unit exceeds the established DCGLs. The survey data was compared to the DCGLs both statistically and with non-statistical comparisons. The radiological survey data demonstrate that the soils are sufficiently below the DCGLs and therefore reject the null hypothesis using the Sign Test (statistical test). Concentrations of residual radioactivity were found to be very minimal and essentially indistinguishable from background. In all of the survey units under consideration, each residual radioactivity DCGL was met with greater than 95% confidence.

The final radiological status survey of the soils in the Building Complexes 5 and 6A concludes that in each survey unit all of the conditions and requirements for unrestricted radiological release have been met and supports the regulatory decision to terminate the license.

1.0 INTRODUCTION

This FSS report documents the radiological status of the Building Complexes 5 and 6A areas at the Combustion Engineering (CE) Windsor Site in Windsor, Connecticut. Presently, this portion of the Site is subject to U.S. Nuclear Regulatory Commission (NRC) Radioactive Materials License No. 06-00217-06 (NRC, 2002) due to historical use involving licensable quantities of radioactive materials. The long-term objective of the licensee, ABB Prospects, Inc. (ABB), is to decommission the Site such that it will meet the criteria for unrestricted use as specified by Title 10 Code of Federal Regulations (CFR) 20.1402 and to terminate NRC license No. 06-00217-06. ABB contracted MACTEC to decontaminate and dismantle the buildings and remediate the areas in the Buildings 2, 5, 6A, and 17 Complexes in accordance with applicable requirements and regulations. As part of the scope of this work for the Building Complexes 5 and 6A, the buildings within the two Complexes have been decontaminated and demolished, building slabs, pavement, and foundations to 4 foot below ground surface have been removed, all underground utilities have been removed, and residual radioactivity in the soil has been reduced to concentrations less than those specified in the license for unrestricted release. This report documents the final radiological status of the Building Complexes 5 and 6A areas, demonstrates that the criteria for unrestricted use have been met, and serves to support the regulatory decision to terminate the license.

The radiological survey data evaluated in this report was designed to assess the residual radioactivity associated with surface soil (soil) and utility trench excavation areas for compliance with the requirements for unrestricted release specified in the license. This includes the Decommissioning Plan (DP) (MACTEC, 2003b), and site-specific derived concentration guideline level (DCGLs) (MACTEC, 2003a) amended to the NRC license in June 2004. (NRC, 2004) Thus, the data evaluation results present a clear picture to the risk managers and stakeholders of the radiological condition of the soils within the Building Complexes 5 and 6A areas relative to the DCGLs.

1.1 METHODOLOGY AND GUIDANCE USED

The FSS report follows methods outlined in the MARSSIM (NRC, 2000). The data evaluated in this report is presented in the context of the MARSSIM data quality assessment methods. Where appropriate, conventional guidance from the NRC, U.S. Environmental Protection Agency (EPA) and accepted practice and methods used in radiological site assessment and characterization are utilized. Principal guidance documents referenced include:

- NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (NRC, 2000);
- EPA QA/G-4, "Guidance for the Data Quality Objectives Process" (EPA, 2000);
- NUREG-1757 Vol. 2, "Consolidated NMSS Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria" (NRC, 2003); and
- NRC Radioactive Material License No. 06-00217-06 (NRC, 2002).

1.2 SAMPLING AND SURVEY REPORT ROAD MAP

Section 1 of this report provides a brief introduction and discusses the CE Windsor Site history and current Site conditions including radionuclides of concern. Section 2 discusses survey unit designation, survey instrumentation, and methods. FSS survey and sampling results and data evaluations are presented in Section 3. Section 4 presents trench soil survey and sampling data. Section 5 evaluates survey data for compliance against the decision criteria. Section 6 includes quality control and data quality assessment evaluations and discussions. Section 7 summarizes the FSS and concludes the outcome of the FSS and Section 8 offers the references. Appendices are included to provide additional detail where appropriate.

1.3 GENERAL SITE DESCRIPTION

Between 1956 and 2001, the CE Windsor Site was used (at various times) to conduct and support research and development as well as manufacturing of nuclear fuels. Such activities make the Site subject to regulatory requirements governing the use and termination of such use of radioactive materials.

The CE Windsor property is located in the Town of Windsor, eight miles north of Hartford, Connecticut (Figure 1.1). The entire property consists of approximately 600 acres and is located at 2000 Day Hill Road, in Windsor, Connecticut. An overview of the site layout is shown on Figure 1.2.

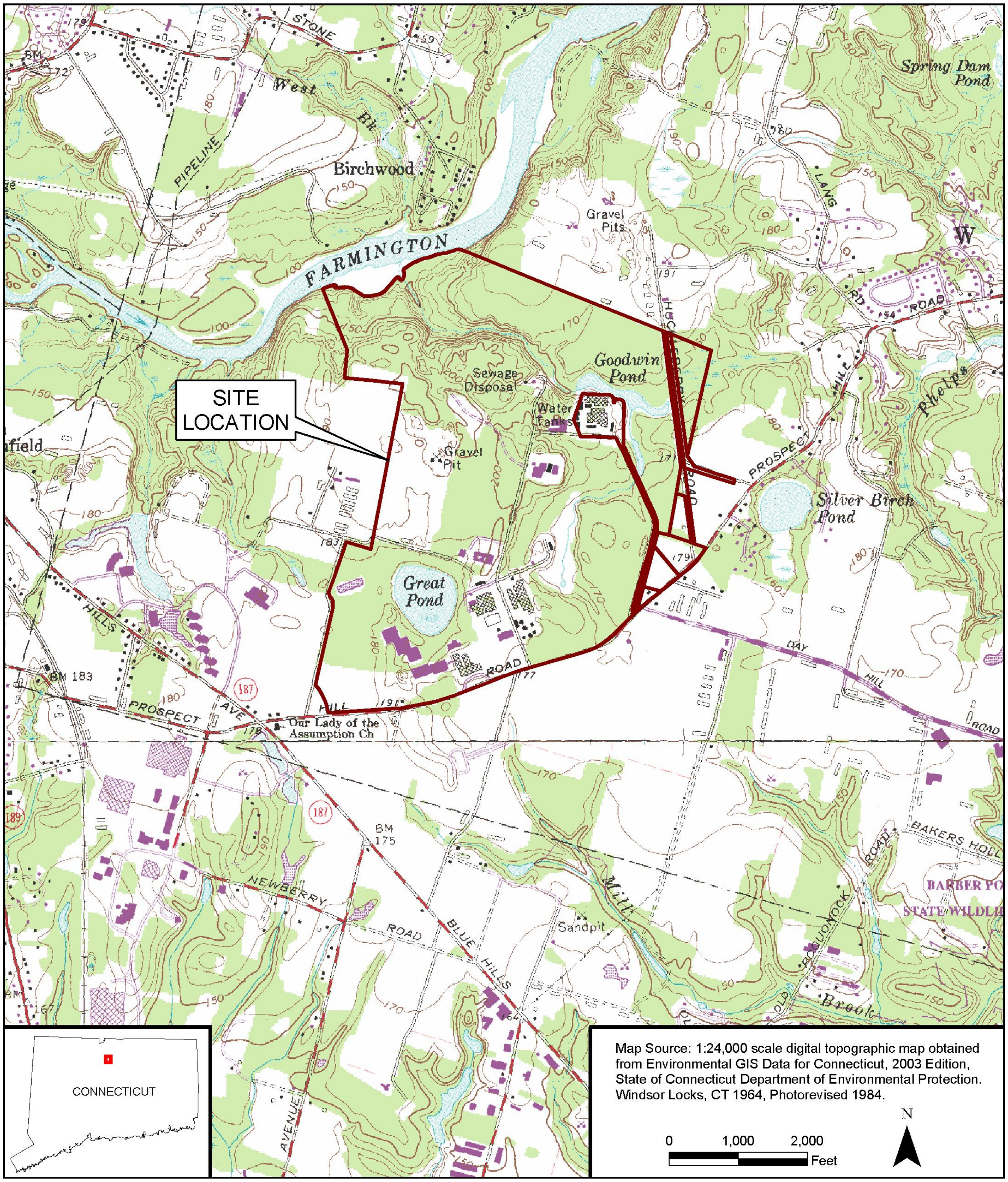

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Figure 1.1: Site Location Map

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Figure 1.2: Site Overview

The Site is industrially zoned by the Town of Windsor, and is located in a Mixed Land Use area of Hartford County. Nearby land uses are primarily commercial, commercial agricultural, industrial, and residential. Much of the northern and western portions of the property are wooded.

The Site is bordered by Day Hill Road to the south; tobacco fields and a sand and gravel quarry to the west; the Windsor/Bloomfield Sanitary Landfill and Recycling Center (Landfill) and the Rainbow Reservoir portion of the Farmington River to the north; and forested land with some residential and commercial development to the east. Within the Site boundary (but excluded as part of the Site) is a 10.6-acre enclave known as S1C. This area is currently owned by the United States Government.

ABB's activities at the Site started in 1955 with an Atomic Energy Commission (AEC) contract to begin research, development, and manufacturing of nuclear fuels for the United States Navy. Activities also included the construction, testing, and operation of the S1C facility, a U.S. Naval test reactor. Contracts with the AEC led to the construction of facilities in 1956 for the development, design, and fabrication of fuel element subassemblies for U.S. Navy submarine reactors. The sanitary wastewater treatment plant (WWTP), power plant, and support buildings were also constructed at that time to support AEC activities. AEC non-licensed manufacturing and research and development activities were terminated by AEC by 1962.

From 1956 to 2001, ABB was involved in the research, development, engineering, production, and servicing of nuclear and fossil fuel systems. These activities were performed under both commercial and federal contracts. Projects included nuclear and combustion research for commercial use, as well as large-scale boiler test facilities and coal gasification. Nuclear fuel research and development and reactor outage servicing was conducted in Buildings 2 and 5, and 17 and components were manufactured in Building 17. The large-scale fossil fuel boiler tests were conducted in Building 3. Wastewater pumping and dilution was conducted in Building 6.

In 2000, ABB's nuclear businesses were sold to Westinghouse, and the fossil fuel businesses were sold to ALSTOM Power. ABB retained ownership of Combustion Engineering, Inc., which owns the CE Windsor site.

The historical processes at the Site generated both low-level radioactive wastes (LLRW) as well as Resource Conservation and Recovery Act (RCRA) hazardous chemical wastes. The most common, in fact virtually all, radioactive waste residues are non-soluble forms of uranium of various enrichments. A more detailed description of the Site history is presented in the Historical Site Assessment (Harding, 2002).

1.3.1 Description of the Building 5 Complex

The Building 5 Complex was constructed in the late 1950s through 1960s and is located in the central portion of the Site, west of the main road and south of the former Building 17 Complex (Figure 1.3 and Figure 1.4). The Building 5 Complex was built as research and development laboratories and nuclear fuel manufacturing. It was comprised of Buildings 5, 15, 16, and 18. Building 5 was built as laboratories and nuclear fuel manufacturing support and later supported various nuclear fuel outage and field operation support. Building 15 was built as a nuclear storage building but has always been used as a carpentry shop. Building 16 was originally used as a tool crib and stock cage to support Buildings 5 and 18 and later to test boronometers. Building 18 housed a scale model reactor test loop for testing of actual (commercial) reactor components. Additional information regarding the Building 5 Complex can be found in the Historical Site Assessment (HSA) (Harding, 2002).

1.3.2 Description of the Building 6A Complex

The Building 6A Complex was constructed in the mid 1950s and is located in the central portion of the Site, west of the main road and in between the former Building Complexes 5 and 17 (Figure 1.3 and Figure 1.4). The Building 6A Complex was built as a liquid radiological waste collection and dilution facility for Building 5. In 1960, the liquid radiological waste from the Building 5 Complex was re-routed to Building 6. Building 6A was decontaminated and renovated for use as maintenance services facility. Additional information regarding the Building 6A Complex can be found in the HSA (Harding, 2002).

1.4 CURRENT SITE-WIDE CONDITIONS

Commercial licensed activities were conducted in Building Complexes 2, 5, and 17. All areas of the Site where radioactive contamination could be present, based on the HSA, were investigated. For Commercial decontamination and decommissioning (D&D) building complexes, remediation was conducted under the DP. Remediation included decontamination of buildings, demolition of all structures within the complexes to ground surface, removal of floor slabs and footings to four feet below ground surface, and the removal of underground utilities and any soils impacted by residual radioactivity above the DCGLs.

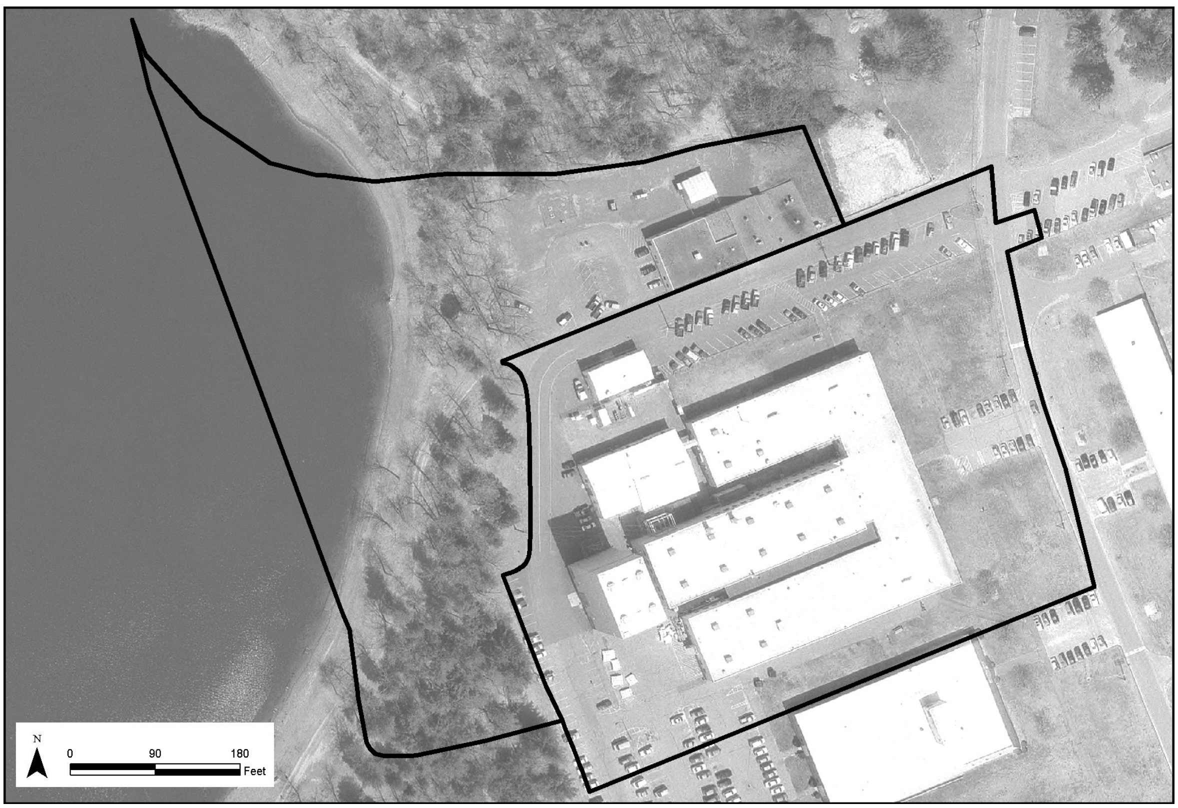

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Figure 1.3: Former Building Complexes 5 and 6A (Aerial Photo)

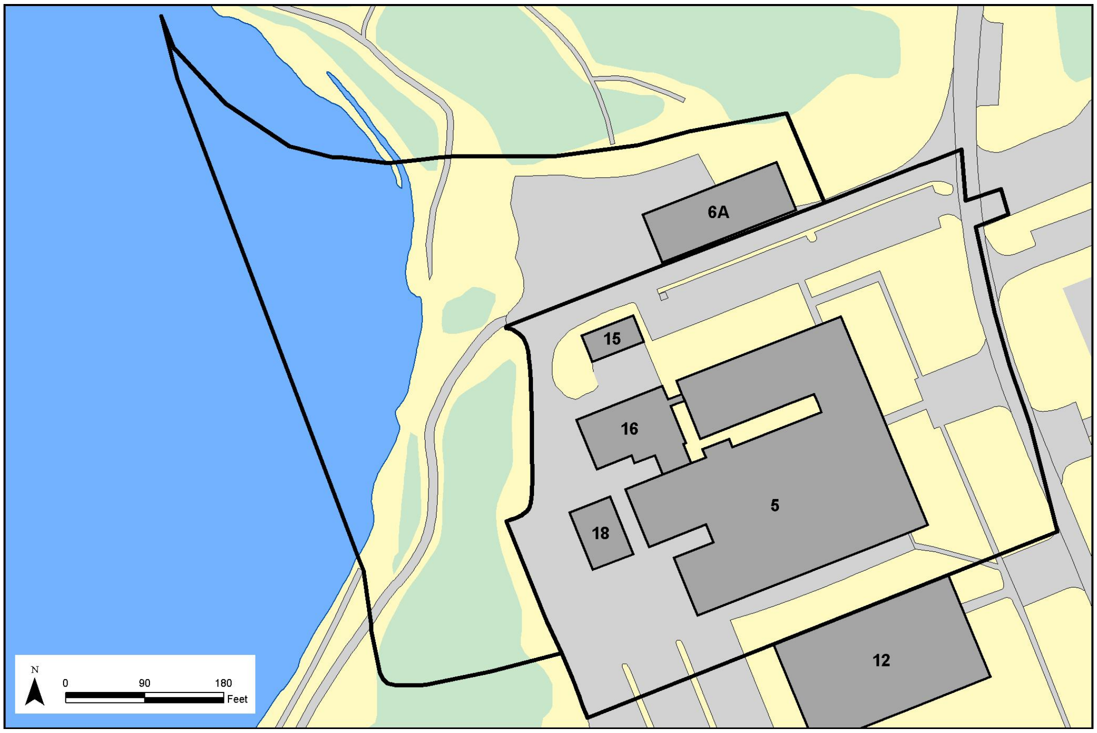

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Figure 1.4: Former Building Complexes 5 and 6A

1.4.1 Building Complexes 5 and 6A Decommissioning Activities

The scope of decommissioning activities for the Building Complexes 5 and 6A included decontamination and dismantlement of structures, removal of concrete slabs, footers, and foundations to 4 feet below ground surface, removal of pavement areas, removal of buried utilities, and the transportation of radioactive waste to appropriate off-site disposal facilities.

Decontamination and remediation of the structures in the Building 5 Complex was performed during 2002. A pre-dismantlement radiological survey report was submitted to the NRC in February 2003 and approved in August 2003. Building dismantlement began in spring 2004 and was completed fall 2004. Exterior asbestos abatement (e.g., transite and roofing materials) was performed prior to dismantlement and all buildings were taken down to the slabs. All materials were disposed of or scrapped to appropriate facilities. Building demolition was performed as described in the DP.

Unlike the Building 5 Complex, Building 6A is not considered contaminated since decontamination and remediation occurred in the 1960s when the building was reconfigured as a maintenance services facility. Some minor decontamination was needed in a sump located in the basement of the building due to historical backflow into the sump from the waste line. A pre-dismantlement radiological survey report was provided to the NRC in May 2005 and building dismantlement was performed during summer 2005.

After building demolition was complete, removal of floor slabs and footings, to a minimum of 4 feet below ground surface, was performed. In the Building Complexes 5 and 6A, there were several footings and foundation supports that were deeper than four feet and these were left in place (Figure 1.5). Any deep slabs remaining after building demolition had penetrations created in order to prevent entrapment of groundwater. Where removal of floor slabs, footings, and pits was not performed (at greater than 4 feet below ground surface), both biased and unbiased soil samples were collected prior to backfill operations and submitted for analysis. These samples represented post-demolition conditions for both FSS release criteria evaluations and comparison to Connecticut Department of Environmental Protection (CTDEP) Remediation Standard Regulation (RSR) limits for chemical contaminants.

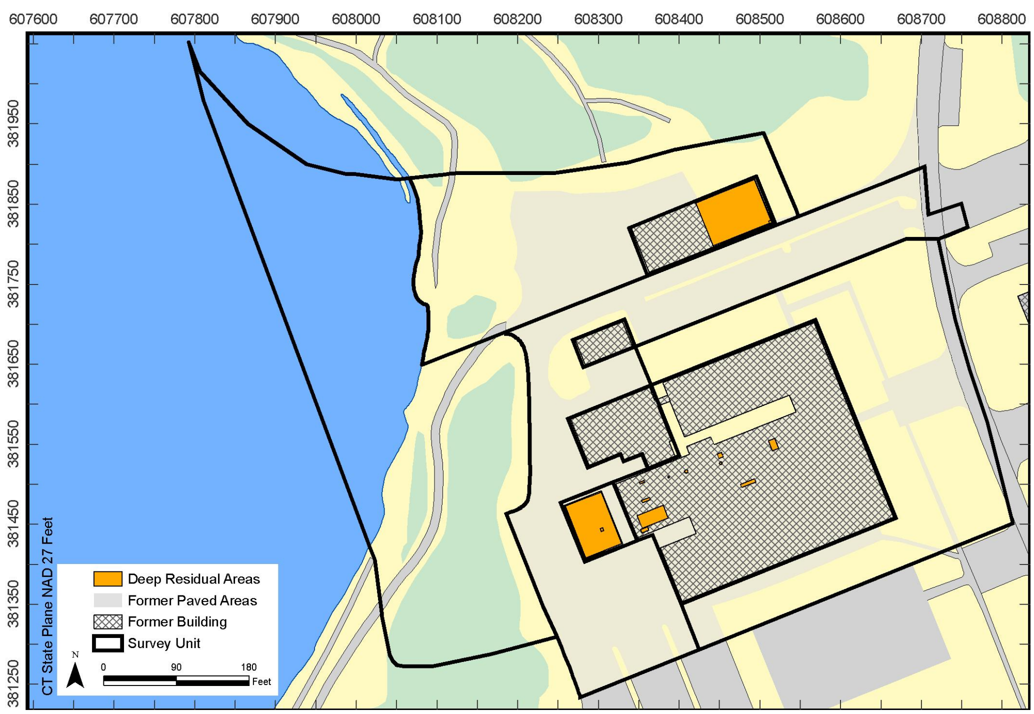

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Figure 1.5: Deep Residual Areas in Building Complexes 5 and 6A

1.4.2 Underground Utility Removal

After removal of the building foundations, Building Complexes 5 and 6A underground utilities were removed from December 2004 through December 2005. Underground utilities in the Building Complexes 5 and 6A area consisted of normal service utilities such as potable water, electrical service, sewer service, industrial waste services and other standard utility lines. Building 5 Complex also contained hot (radiological) waste underground utility services. Detailed trench location and radiological survey results information is presented in Section 4.

1.5 RESIDUAL RADIOACTIVITY PROFILE

Based on the review of historical records, process knowledge, and the results of radiological surveys at the Site, the residual radioactivity potential for the Site soils can be isolated to two credible source terms. The first is uranium series radionuclides associated with nuclear fuel manufacturing and research (depleted, natural, and enriched). The second potential source term is that associated with nuclear power plant outage support services (reactor byproduct series). Radionuclides in this category consist almost exclusively of the longer-lived isotopes of reactor activation products dominated by the radioactivity associated with cobalt 60 (Co-60). Based upon the results of soil sampling and analysis, it is evident that enriched uranium is the predominant radioisotopes found in soils at the Site.

A great deal of radiological data has been collected by CE Site Remediation Services Group in support of the ongoing Radiation Protection Program, and by MACTEC in support of the characterization, decontamination, and dismantling of the buildings as part of decommissioning and license termination for the CE Windsor Site. This data is important because it was used to:

- Identify the radionuclides that were expected to be present in each survey unit;
- Establish the survey unit breakdown and boundaries;
- Determine the classification of impacted survey units;
- Determine the analytical methods needed to detect and quantify residual radioactivity present; and
- Estimate the minimum sample size needed to achieve sufficient statistical power to either accept or reject the null hypothesis within the bounds of the accepted decision errors.

More specific information and details regarding the radiological characteristics of uranium and byproduct materials at the Site were provided as part of the DCGL Derivation Report (MACTEC, 2003a). Results from dose modeling were used to select an enrichment of 3.5% to represent the uranium series and Co-60 to represent the reactor byproduct series.

1.6 DECISION FRAMEWORK

Since remediation is complete for the Building Complexes 5 and 6A areas, the results of the FSS performed outside of trench excavations or spoils piles demonstrate that the potential dose from any residual radioactivity is below the release criterion for each survey unit. Results of the trench release surveys (inside trench excavations and the spoils piles generated during excavation activities) demonstrate that the potential dose from any residual radioactivity is below the release criterion for the trench areas.

1.6.1 Compliance Testing

The Sign Test was used to evaluate compliance with the derived concentration guideline level, survey unit average (median) concentration corresponding to the permissible limit ($DCGL_W$) for FSS and trench volumetric sampling. If the largest measurement of the sample population is below the $DCGL_W$, then the Sign test will always show that the survey unit meets release criteria (NRC, 2000). This was the case for the volumetric samples taken for the Building 5 and 6A Complexes soils.

The Sign Test is a one-sample, non-parametric test that can be used to evaluate compliance with the DCGL. The Sign Test is the recommended compliance evaluation procedure when the contaminant(s) under evaluation are not present at significant levels in background. While uranium series radionuclides clearly exist in nature, it was decided early on to not use uranium series background activity concentrations to derive a "net" sample activity. This decision was made because background activity concentrations at the Site are appreciably lower than the DCGL values used during Site FSS.

In trench areas when survey or sampling results were greater than investigation levels or greater than the established $DCGL_W$ values, immediate remediation of the identified area was performed and post-remedial sampling and analysis conducted. For the Building 5 Complex, soil remediation was necessary in a section of the hot waste line trench located in the former north wing portion of Building 5 and around the base of two hot waste line manholes.

This combination of FSS and trench volumetric sampling and gamma walkover (scan) survey data was used to demonstrate compliance with the release criterion. In addition to single-point comparisons of the measurement against the limit, the Sign Test was conducted. The decision to release a survey unit and the trench area within the Building Complexes 5 and 6A was based upon the outcome of the comparisons made in Table 1.1.

Table 1.1: Summary of Decision Rules

Survey Result	Conclusion
All measurements less than $DCGL_W$	Survey unit meets release criteria if unity rule is met
Average greater than $DCGL_W$	Survey unit does not meet release criteria
Any measurement greater than $DCGL_W$ and the average less than $DCGL_W$	Conduct Sign Test and elevated measurement comparison (EMC)

Prepared/Date: MPM 10/31/05
Checked/Date: HTD 10/31/05

1.6.2 Unity Rule Testing

Given that there are two different source terms that are unrelated, the unity rule was used. The unity rule ensures that the total dose due to the sum of two discrete source terms does not exceed the release criteria. The unity rule for the Site is shown in Equation 1-1. The unity rule was implemented in conjunction with the Sign Test in order to demonstrate that release criteria were met under all circumstances. This was accomplished by using transformed data for the unity rule (uranium concentration divided by the uranium DCGL and byproduct concentration divided by the byproduct DCGL) as the data set for the Sign Test with a decision level of 1 for each survey unit. This approach ensures that there are no situations such that the individual measurement results (uranium and byproduct) are both less than the DCGLs but the sum of the fractions exceeds unity while only performing the Sign Test one time.

$$\frac{C_U}{DCGL_U} + \frac{C_B}{DCGL_B} \leq 1 \qquad \text{(Equation 1-1)}$$

Where:

C_U = uranium concentration

C_B = byproduct (cobalt 60) concentration

$DCGL_U$ = derived concentration guideline level for uranium

$DCGL_B$ = derived concentration guideline level for byproduct

1.6.3 Elevated Measurement Comparison Decision

Another factor in the decision rule is the EMC. Each measurement in the survey unit (systematic and walkover) is compared to the investigation levels. Any measurement that is greater than the investigation level should be investigated. The EMC is intended to flag potential failures in the remediation process, not to demonstrate compliance with the release criterion. The DCGL for the EMC is shown in Equation 1-2.

$$DCGL_{EMC} = A_m * DCGL_W \qquad \textit{(Equation 1-2)}$$

Where:

$DCGL_{EMC}$ = derived concentration guideline level for small areas of elevated activity

A_m = area factor for the area of the systematic grid (*a priori*) or actual area of elevated concentration (*a posteriori*)

$DCGL_W$ = derived concentration guideline level for average concentrations

If an isolated area where elevated residual radioactivity is found, a variation of the unity rule will be used to ensure that the total dose (uniformly distributed and elevated) is within the release criterion. This variation is shown in Equation 1-3.

$$\frac{\delta_U}{DCGL_U} + \frac{\delta_B}{DCGL_B} + \frac{\overline{\chi}_U - \delta_U}{A_m * DCGL_U} + \frac{\overline{\chi}_B - \delta_B}{A_m * DCGL_B} < 1 \qquad \textit{(Equation 1-3)}$$

Where:

δ_U = estimate of average uranium residual radioactivity in the survey unit

δ_B = estimate of average byproduct residual radioactivity in the survey unit

$\overline{\chi}_U$ = average uranium concentration in elevated area

$\overline{\chi}_B$ = average byproduct concentration in elevated area

A_m = area factor for the actual area of elevated concentration

$DCGL_U$ = derived concentration guideline level for total uranium

$DCGL_B$ = derived concentration guideline level for byproduct

If there is more than one area of elevated residual radioactivity in a survey unit then additional terms can be added to Equation 1-3. An alternative is to use the actual results as input into RESRAD and calculate the dose for each area of elevated residual radioactivity in order to show that the total dose is within the release criterion.

Site-specific DCGLs were derived for soil and accepted by the NRC as part of the DP. The $DCGL_W$ for total uranium is 557 pCi/g and the $DCGL_W$ for Co-60 is 5 pCi/g. Additional information can be found in the Derivation of the Site-Specific Soil DCGLs (MACTEC, 2003a). In addition, calculations were performed using Residual Radioactivity Code (RESRAD) for EMC. Table 1.2 displays the DCGL elevated measurement comparison ($DCGL_{EMC}$) values for various sized areas that may be used for EMC. Additional $DCGL_{EMC}$ values may be calculated for localized areas of elevated residual radioactivity if the values in Table 1.2 are not appropriate.

Table 1.2: Calculated $DCGL_{EMC}$ Values

Area (m^2)	Total uranium Area Factor (A_m)	Total uranium $DCGL_{EMC}$ (pCi/g)	Co-60 Area Factor (A_m)	Co-60 $DCGL_{EMC}$ (pCi/g)
1	19.6	10,922	13.4	66.9
2	12	6,698	7.6	37.9
5	6.8	3,807	4.1	20.3
10	4.6	2,562	2.7	13.4
100	2.4	1,311	1.4	6.7
500	1.7	962	1.1	5.7

Prepared/Date: MPM 10/31/05
Checked/Date: HTD 10/31/05

2.0 FIELD IMPLEMENTATION

2.1 MOBILIZATION

Prior to mobilizing the radiological survey team to the Site, the survey team was trained on the field sampling equipment and procedures to be used. A set of GIS maps were created that provided survey units and sample locations that were used in conjunction with GPS units to locate soil sampling and survey locations within the survey units. GPS sample coordinate locations are provided as part of survey unit data in the appendices.

Gamma walkover and direct static surveys were performed on soils using a 2 inch x 2 inch thallium-activated sodium iodide detector (NaI) coupled to an appropriate scaler/rate meter instrument to form a complete survey instrument package. Soil volumetric samples were collected and analyzed on the on-site gamma spectroscopy system using a high purity germanium (HPGe) detector and Canberra's Genie system software. Detailed information regarding gamma spectroscopy analysis is provided later in this Section.

2.2 SURVEY UNIT DESIGNATION

The survey unit represents the fundamental element for compliance demonstration during FSS results evaluation. There are numerous factors that influence the delineation of a survey unit and the design of the survey within the unit.

Design of final status survey units for the Building Complexes 5 and 6A was performed following the Final Status Survey Plan (FSSP) (MACTEC, 2004). Individual survey units were identified and created based upon the potential likelihood of soils containing residual radioactivity. Since the Building 5 Complex area was determined to be likely impacted (containing residual radioactivity) from Site activities, the survey area of the identified survey units was constrained to a maximum value of 10,000 m^2. This imposed surface area limitation conservatively increased the sampling density within the survey units. This constraint was not imposed on the Building 6A Complex and only one survey unit exceeded 10,000 m^2.

The footprint area of former Building 5 was classified as a Class 2 (as low as reasonably achievable [ALARA]) remediation area. The footprint area of former Building 6A was classified as Class 2 due to historical usage. Since former Buildings 15, 16, and 18 had no significant use of radioactive materials, the footprint areas were classified as Class 3 areas. The general areas of the Building Complexes 5 and 6A were classified as Class 3 areas since no significant concentrations of residual radioactivity were detected during characterization surveys activities.

A summary of the survey units for the Building Complexes 5 and 6A areas are presented in Table 2.1 and depicted in Figure 2.1.

Table 2.1: Summary of Building Complexes 5 and 6A Survey Units

Survey Unit ID	Class	Area (m^2)	Description
CE-FSS-05-01	3	8,100	Former parking lot area and road
CE-FSS-05-02	3	8,100	Area surrounding former building 5
CE-FSS-05-03	3	200	Former building 15 footprint
CE-FSS-05-04	3	800	Former building 16 footprint
CE-FSS-05-05	3	500	Former building 18 footprint
CE-FSS-05-06	2	6,700	Former building 5 footprint
CE-FSS-06-01	3	6,000	Area surrounding former building 6A
CE-FSS-06-02	2	1,000	Former building 6A footprint
CE-FSS-06-03	3	12,600	General area adjacent to building complexes 5, 6A, and 17

Prepared/Date: ECS 3/07/06
Checked/Date: HTD 3/07/06

Once the survey units were identified, the sample size for final status survey was determined. Characterization data was used to provide an estimate of the expected residual radioactivity in these areas. The existing characterization data for the soils in the Building Complex Areas (2, 5, and 17) of the Site is statistically summarized for comparison in Table 2.2. Review of this data indicates there is no significant difference within the Building Complex Areas as compared to the DCGLs of 557 pCi/g for total uranium or 5 pCi/g for Co-60.

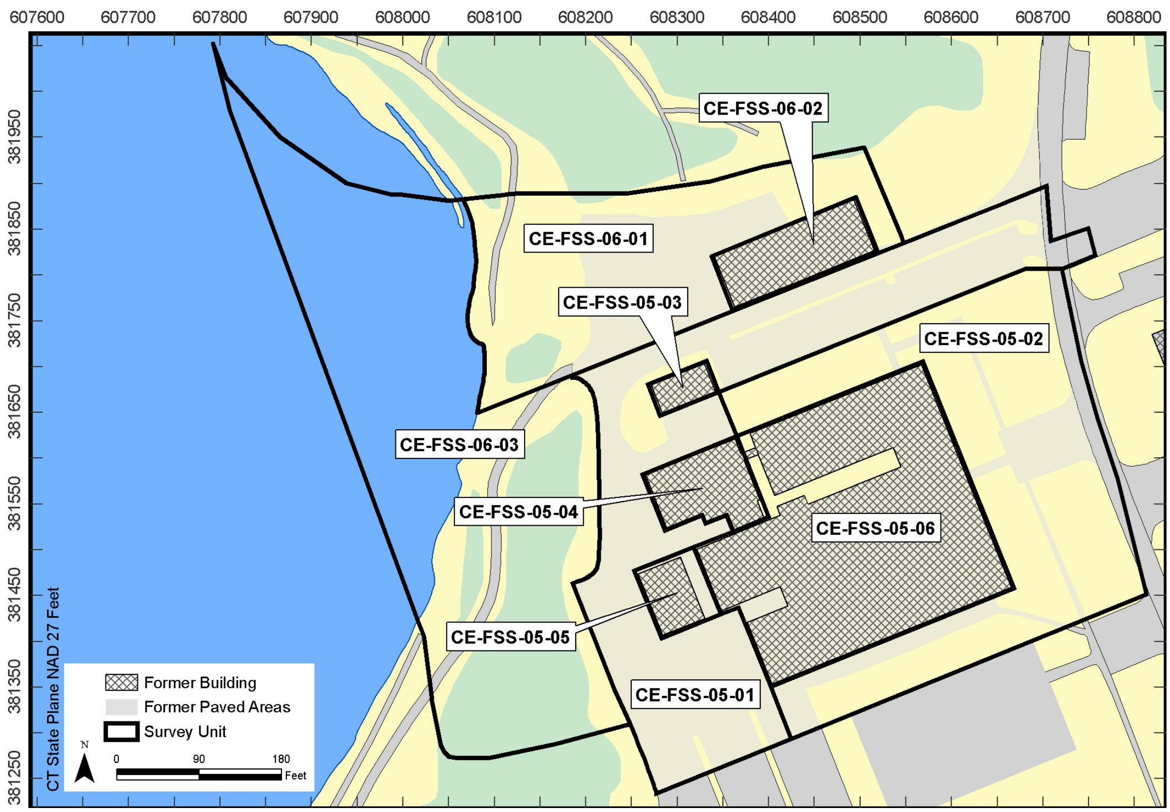

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Figure 2.1: Overview of Final Status Survey Units

Table 2.2: Summary of Building Complex Soil Characterization Data

Complex	Total Uranium (pCi/g)			Cobalt 60 (pCi/g)		
	Mean	Standard Deviation	Max	Mean	Standard Deviation	Max
Building 2 Complex	5.1	3.6	42	0.1	0.1	1.1
Building 5 Complex	5.6	2.5	9	0.2	0.09	0.3
Building 17 Complex	4.3	5.4	64	0.1	0.02	0.1

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2.3 SURVEY UNIT SAMPLE SIZE DETERMINATION

The minimum sample size (N) and location of those samples for each survey unit was determined using the statistical sampling software, Visual Sample Plan (VSP) (PNNL, 2004). VSP uses the statistical approach and algorithms referenced in MARSSIM to calculate the required minimum sample size for a given survey unit. In order to account and compensate for uncertainty in the computations of minimum sample size as well as the possibility that some sample data may be lost or deemed unusable due to analytical and sampling error, minimum sample size computations were increased by twenty percent and rounded up to obtain sufficient data points to yield the desired power. VSP presents a sample distribution on scale drawings of the area(s) to be sampled within the survey unit.

Since the Site has two independent DCGLs, N for each survey unit was determined for each of the DCGLs. The number of samples determined for each DCGL was compared, and the larger of the two values was used to determine the number of samples collected from each survey unit.

2.3.1 Class 1 Survey Unit Sample Size

Class 1 survey units have the potential for residual radioactivity at a large fraction of the DCGL or even greater than the DCGLs, so the lower bound of the gray region (LBGR) was selected to be around 70% of the DCGL. The standard deviation is conservatively approximated high as a safety margin to reduce the chance of failing the decision criteria. The survey design parameters used to calculate the minimum required sample size for Class 1 Survey Units are shown in Table 2.3. For this scenario, VSP calculated one additional sample when compared to the Sign Test table in MARSSIM which yielded a total of 34 samples using the same parameters in Table 2.3. Since having an additional sample is conservative, the VSP calculated sample size was used.

Table 2.3: Class 1 Survey Unit Sample Size

Parameter	Total Uranium	Co-60
α decision error	0.05	0.05
β decision error	0.05	0.05
$DCGL_W$ (pCi/g)	557	5
LBGR (maximum estimated mean/median) (pCi/g)	400	3.5
Standard Deviation (σ) (pCi/g)	180	1.5
Relative Shift (Δ/σ)	0.9	1.0
Sample Size (N)	29	24
Additional 20%	6	5
FSS Sample Size	35	

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2.3.2 Class 2 Survey Unit Sample Size

Class 2 survey units have the potential for residual radioactivity, but are not expected to exceed the DCGLs, so the LBGR was selected to be around 50% of the DCGL. The same standard deviation was used for Class 2 areas, as this should also provide a margin of safety for minimizing the chance of failing the decision rule. The survey design parameters used to calculate the minimum required sample size for Class 2 Survey Units are shown in Table 2.4.

Table 2.4: Class 2 Survey Unit Sample Size

Parameter	Total Uranium	Co-60
α decision error	0.05	0.05
β decision error	0.05	0.05
$DCGL_W$ (pCi/g)	557	5
LBGR (maximum estimated mean/median) (pCi/g)	300	2.5
Standard Deviation (σ) (pCi/g)	180	1.5
Relative Shift (Δ/σ)	1.4	1.7
Sample Size (N)	16	14
Additional 20%	4	3
FSS Sample Size	20	

Prepared/Date: MPM 10/31/05
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2.3.3 Class 3 Survey Unit Sample Size

Since Class 3 survey units are not expected to have residual radioactivity or are expected to have only a small fraction of the DCGLs, the LBGR was selected to be around 10% of the DCGL. The same standard deviation was used for Class 3 areas and this should also provide a margin of safety for minimizing the chance of failing the decision rule. The survey design parameters used to calculate the minimum required sample size for Class 3 Survey Units are shown in Table 2.5.

Table 2.5: Class 3 Survey Unit Sample Size

Parameter	Total Uranium	Co-60
α decision error	0.05	0.05
β decision error	0.05	0.05
$DCGL_W$ (pCi/g)	557	5
LBGR (maximum estimated mean/median) (pCi/g)	60	1
Standard Deviation (σ) (pCi/g)	180	1.5
Relative Shift (Δ/σ)	2.8	2.7
Sample Size (N)	11	11
Additional 20%	3	3
FSS Sample Size	14	

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The total number of samples obtained and the number of samples per survey unit is presented in Table 2.6.

Table 2.6: Number of FSS Volumetric Samples Obtained per Survey Unit

Survey Unit ID	Class	Number of Samples Planned	Number of Samples Obtained
CE-FSS-05-01	3	14	14
CE-FSS-05-02	3	14	14
CE-FSS-05-03	3	14	14
CE-FSS-05-04	3	14	14
CE-FSS-05-05	3	14	14
CE-FSS-05-06	2	20	20
CE-FSS-06-01	3	14	14
CE-FSS-06-02	2	20	20
CE-FSS-06-03	3	14	14
Total Number of Samples		**138**	**138**

Prepared/Date: ECS 3/07/06
Checked/Date: HTD 3/07/06

2.4 SURVEY AND SAMPLE LOCATIONS

Survey and sample locations within a survey unit may be randomly placed, or placed using a systematic grid with a random start location. During FSS activities for Building Complexes 5 and 6A areas, randomly chosen sampling and survey locations were used to place Class 3 survey locations within those survey units and systematic grid patterns were used to place Class 2 survey locations within those survey units. For each Class 2 survey unit, a random start location was selected and used to provide an unbiased set of measurement locations for the FSS.

A geographic information system (GIS) was created for the Site and the survey units and sample locations were integrated into the GIS data. The Site GIS used the Connecticut State Plane North American Datum (NAD) 27 (units of feet) as its reference datum. Sample locations were identified and marked within the survey units using a Trimble Pro XR Sub-meter global positioning system (GPS). Maps of the survey units and sample locations were generated for use during sample marking and survey activities. Survey and sampling locations, in Connecticut State Plane NAD 27 coordinates with units of feet, are provided for each survey unit in the appropriate appendix.

2.4.1 Soil FSS Sample Locations

Surface volumetric soil samples were collected for Building 5 Complex FSS evaluation during May 2005 and for Building 6A Complex during September and October 2005. Soil sample locations for the Building Complexes 5 and 6A areas were randomly chosen for Class 3 survey unit locations and were chosen using a systematic grid placement methodology for Class 2 survey unit locations. Figures of sample locations for each survey unit are provided in the appendices. For each Class 2 survey unit, a random start location was selected and used to provide an unbiased set of measurement locations for the survey unit. Sample collection locations were placed such that a sample would be representative of the sample media, sample volume was large enough to provide sufficient material to achieve the desired detection limit, and sampling density was consistent with assumptions used to develop the conceptual site model and DCGLs.

The soil sample process was designed to collect a surface layer sample of the soil at the designated sample location (for Class 2 Survey Units) or at randomly selected sample locations (for Class 3 Survey Unit). The samples were collected from the top 3 inches of the soil at the sample location. Various sampling methods were used to collect the soil samples in the survey units. However, in most instances, hand collection techniques were used to collect soil samples. Where there was vegetation growing, the vegetative layer was removed prior to sample collection. In areas where a surface cover existed (i.e., pavement, concrete, fabric, etc.), the cover was removed prior to sample collection.

During soil sample collection, a scan survey of the area was performed with a NaI detector (1 meter radius area from the sample location). This survey was used to identify the presence of elevated residual radioactivity within the 1 meter radius area. If elevated activity was identified, a static one-minute measurement was taken at that location. If elevated activity was not identified, then a static one-minute measurement was taken only at the sample location.

Once scanning and static measurements were completed, a 1 square foot area was demarcated and the top 3 inches of soil was collected from that area. Common garden hand rakes were used to scarify and loosen the surface of the soil as necessary. Loosened soil was sieved through a number 3 mesh (0.25 inch) sieve to remove root materials and other foreign debris. Volumetric soil samples were placed in zip-lock type plastic bag and labeled in accordance with the FSSP. To minimize the potential for sample handling error, volumetric samples were homogenized and placed in sample containers in the Health Works building rather than in the field during sampling activities.

Volumetric soil sampling in excavated areas (trenches and excavated foundations) was performed in a similar manner, except that sampling in the trenches was performed at both biased sample locations and non-biased locations (see Section 4 for greater detail).

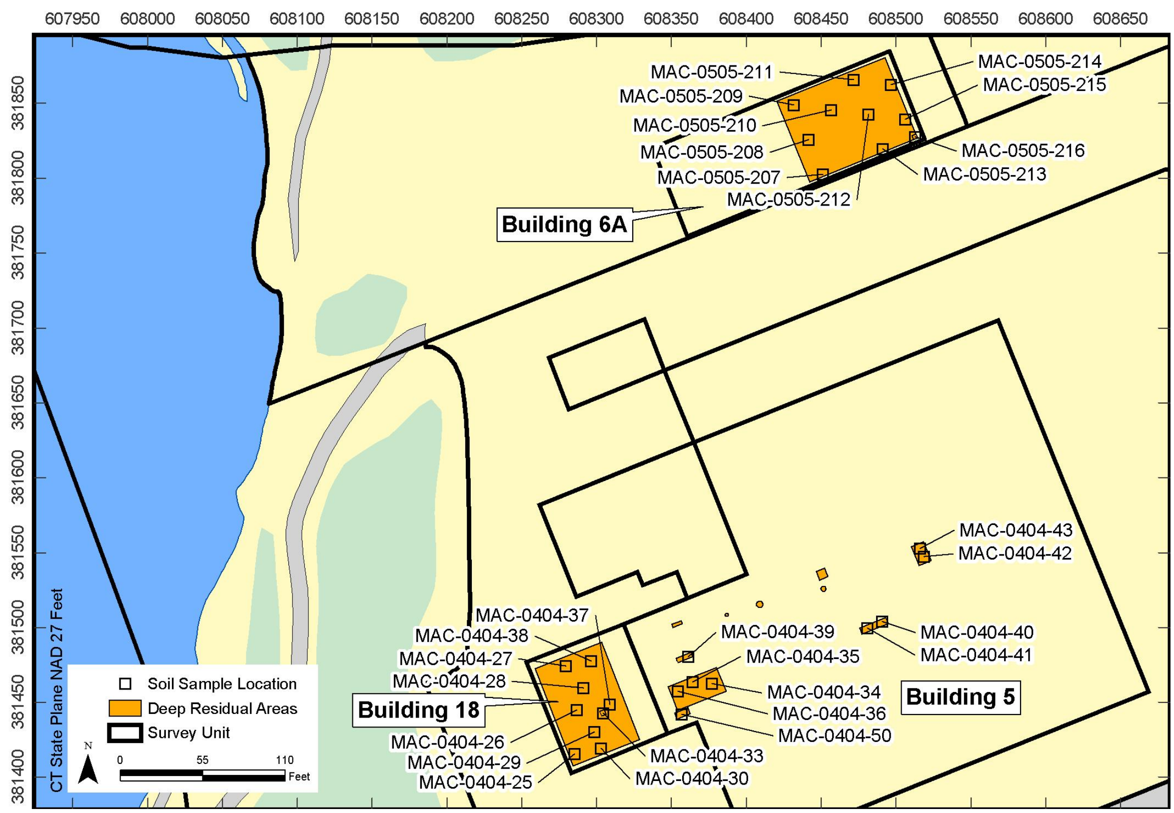
2.4.2 Sub-Slab Soil FSS Sample Locations

As planned, Building Complexes 5 and 6A structural materials (mostly concrete) deeper than four feet below the finished grade were left in place during the dismantlement process of the buildings. Although soils beneath these foundations and structures appeared to be native or clean fill, samples were still collected to demonstrate that site criteria for residual radioactivity in soils underneath the concrete were being met. Pre-backfill sampling was performed at that time because it was decided that it would have been extremely difficult to sample below the foundation material layer after the area had been backfilled with soil. To accomplish this, soil sampling beneath the floors/foundations was performed by coring through the concrete and collecting soil samples below the concrete surface.

A total of 28 samples were collected from the deep basements and pits within Buildings 5, 6A, and 18 (Figure 2.2). These samples consisted of 9 collected from Building 5, 10 collected from Building 6A, and 9 samples from Building 18. Sub-slab soil samples contain a total of 11 biased samples – 8 from Building 5, 1 from Building 6A, and 2 from Building 18.

In Building 5, there were nine deep pits and one sump considered for sub-slab soil sampling. Four deep pits and the sump had nine sub-soil samples collected (eight biased and one non-biased). The other five deep pits were very small or too difficult to collect samples (characterization indicated no residual radioactivity). Two biased samples were collected from the eastern most pit (MAC-0404-042 and -043) about 10 feet below grade, two biased samples from the southeastern pit (MAC-0404-040 and -041) about 8 feet below grade, one biased sample from the western pit (MAC-0404-039) about 7 feet below grade, southwestern pit had two biased samples (MAC-0404-034 and -036) and one non-biased sample (MAC-0404-035) about 10 feet below grade, and the sump had one biased sample (MAC-0404-050) about 15 feet below grade.

Building 6A and 18 each contained a deep basement with a sump considered for sub-slab soil sampling. For Building 6A, a total of 10 samples were collected with 9 non-biased samples and 1 biased sample from the sump (MAC-0505-216) about 15 feet below grade. For Building 18, a total of 9 samples were collected with 7 non-biased samples and 2 biased samples – 1 relocated sample (MAC-0404-038) and 1 from the sump (MAC-0404-033) about 20 feet below grade.



Prepared/Date: BRP 3/09/06
Checked/Date: HTD 3/09/06

Figure 2.2: Sub-Slab Sampling Locations Building Complexes 5 and 6A

2.5 INVESTIGATION LEVELS

Investigation levels (Table 2.7) for the volumetric sample results were developed in accordance with the guidance found in MARSSIM. Any sample result greater than the investigation level would be identified, marked, and further investigation performed to determine the extent of contamination at greater than the $DCGL_W$. After review of the volumetric sample activity results, no sample result was reported at greater than the investigation level.

Table 2.7: Final Status Survey Volumetric Investigation Levels

Survey Unit Classification	Volumetric Analysis Investigation Level (most conservative)
Class 1	> $DCGL_W$
Class 2	> $DCGL_W$
Class 3	> 80% $DCGL_W$

Prepared/Date: MPM 10/31/05
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Investigation levels for the walkover survey were derived using the most conservative assumption basis: the least sensitive instrument of the inventory being used for the survey, the lowest DCGL value of the two DCGLs (Co-60 at 5 pCi/g), and not taking into account any of the area factor correction factors normally included in the development of limits or investigation levels. Using conservative assumption of data and the most conservative soil concentration exposure rate factor developed, a counts per minute (cpm) value was generated at the stated $DCGL_W$ value for the scanning measurement investigation level (Table 2.8).

Table 2.8: Final Status Survey Scanning Investigation Levels

Survey Unit Classification	Scanning Measurement Investigation Level (most conservative)
Class 1	> 4,064 cpm
Class 2	> 4,064 cpm
Class 3	> 4,064 cpm

Prepared/Date: MPM 10/31/05
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2.6 ON-SITE GAMMA SPECTROSCOPY INSTRUMENTATION

Soil volumetric samples analyzed on-site were analyzed by either a 30 percent efficient (detector serial # 9882108) or 11 percent efficient (detector serial # 380394) HPGe gamma spectroscopy system throughout the entire FSS sampling campaign (2003 through 2005) in accordance with the Genie-2000 Spectroscopy System Operations Instructions (Canberra, 2002a).

The gamma spectroscopy system identifies and quantifies the concentrations of gamma-emitting radionuclides in soil with minimum sample preparation. The system consists of a high-purity germanium detector connected to a dewar of liquid nitrogen, high voltage power supply, spectroscopy grade amplifier, analog to digital converter, and a multichannel analyzer (MCA) as shown in Figure 2.3. The system is energy calibrated so the MCA data channels are given an energy equivalence and displays counts versus energy. An efficiency calibration is performed for each geometry so that a curve of gamma ray energy versus counting efficiency is generated. Each peak is identified manually or by the gamma spectroscopy analysis software used with the detector. The counts in each peak or energy range, the sample weight, the efficiency calibration curve, and the isotope's decay scheme are factored together to give the sample activity in pCi/g.

The gamma spectroscopy system was operated using Canberra's Genie 2000 software loaded on a desktop computer system. Genie 2000 software is a comprehensive set of tools for acquiring and analyzing spectra from MCAs (Canberra, 2002b).

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Figure 2.3: On-Site HPGe 30% Detector Shield and LN_2 Dewar

2.6.1 On-Site Gamma Spectroscopy Instrument Calibration

A calibration check of the gamma spectroscopy system was performed daily, prior to counting operations for both energy and efficiency parameter inputs. This was achieved by using a National Institute of Standards and Technology (NIST) traceable multi-line standard calibration source in the same geometry (with a volumetric equivalent density) as the samples to be counted. The calibration and efficiency curves, calibration source certificates, as well as other documentation relating to the calibration of the on-site gamma spectroscopy systems are presented in Appendix L.

2.6.2 Gamma Spectroscopy Measurement Detection Limit

The minimum detectable concentration (MDC) for samples analyzed by gamma spectroscopy is calculated by the analysis software. The MDC for gamma spectroscopy is calculated as shown in Equation 2-1. For radionuclides with multiple gamma energies, a separate MDC value is calculated for each energy. The lowest of the values will be assigned as the radionuclide MDC. It is not uncommon for soil sample MDCs to be less than 1 pCi/g by gamma spectroscopy. After sample counting, MDC values were reviewed for acceptable values. If MDC values for the radionuclides of interest were not considered sufficient, then the sample was recounted with a longer count time and reevaluated. Samples were recounted with the adjusted count time duration until an acceptable MDC was reported by the software.

$$MDC = \frac{L_D}{T_1 * \varepsilon * y * V * K_c * K_w * U_f} \qquad \textit{(Equation 2-1)}$$

where:

MDC = minimum detectable concentration
L_D = detection limit
T_1 = collection live time
ε = detection efficiency at peak energy
y = branching ratio of the gamma energy
V = mass of sample
K_c = correction factor for radionuclide decay during counting
K_w = correction factor for the radionuclide decay from the time the sample was collected to the start of counting
U_f=unit conversion factor

2.6.3 Gamma Spectroscopy Instrument Background Measurements

Because the naturally occurring concentrations of background radioactivity in Site soils were expected to be far below the DCGL benchmarks, ABB chose to include soil background radioactivity as part of the residual activity attributable to licensed activities. No attempt was made to measure the concentrations of naturally occurring radioactivity measurable in soils in unaffected areas or "reference survey unit" areas (NRC, 2000). Still, there was the need to measure the Gamma spectroscopy system's response to other ubiquitous sources of background radiation (e.g., cosmic radiation).

A check of the gamma spectroscopy system background data sets (counts and cpm) covering the significant time periods when FSS analysis occurred showed no trends in the data over time. Coupled with the gamma spectroscopy system's QA measurements, the stability in the measured background data presents evidence of the gamma spectroscopy system's stability (see Section 6 for additional information on the QA measurement results). The background data and control charts are provided in Appendix L.

2.6.4 On-Site Gamma Spectroscopy Reporting

The analysis software uses several algorithms to evaluate spectroscopy data – peak locate, peak area, nuclide identification and activity calculation, and reporting. The specific details of these algorithms are provided in software documentation. Another important factor in the analysis of the spectroscopy data is the nuclide library. The nuclide library contains the information about the radionuclide that is needed to calculate the activity – half-life, gamma energy and abundance. The nuclide library was optimized for FSS to only including radionuclides (and necessary progeny) that have been identified at the Site.

Results of gamma spectroscopy analysis are reported by radionuclide as the actual concentration (pCi/g), the uncertainty associated with that result, and the MDC. Statistical evaluations of the data will be performed on the actual results, regardless of its value.

Since only two of the three uranium isotopes are detectable by gamma spectroscopy, a method for calculating total uranium is necessary. Historically, the Site has used a multiplier of 31 to determine the total amount of uranium in a sample from the U-235 result by gamma spectroscopy for low enriched uranium (LEU). Since this value is based on a large amount of samples over a long period of time, it provides an overall representative value. If highly enriched uranium (HEU) is present in a sample, the multiplier of 31 provides a conservative over-calculation of the total uranium in the sample since the multiplier would be a lower number. For very high enriched uranium (>90% enriched), alpha spectroscopy would be necessary in order to determine the total activity of uranium since there can be significant variations in the amount of the three uranium isotopes in this material.

An evaluation of the multiplier of 31 was made by comparing the actual total uranium to the calculated total uranium for variations of the three uranium isotopes in 3.5% enriched uranium. One sample is based on the NRC enrichment formula (specific activity); two additional samples are variations based on typical enrichment results from the gaseous diffusion process. Using the NRC equation produces a multiplier of 23 for total uranium in a sample from the U-235 value. These hypothetical samples and the comparison of the multipliers of 23 and 31 total to the actual total are shown in Table 2.9. The table demonstrates that the multiplier of 31 used to evaluate FSS data overestimates actual total uranium and is therefore conservative.

Table 2.9: Evaluation of Total Uranium Calculation

Parameter	NRC Equation 3.5%	Variation 1 3.5%	Variation 2 3.5%
Specific Activity (pCi/g)	1.8E-6	2.4E-6	2.6E-6
U-234	77.49	83.38	84.66
U-235	4.27	3.15	2.91
U-238	18.24	13.47	12.43
Actual U Total	100	100	100
Calculated U Total (U-235 X 23)	98	72	67
Calculated U Total (U-235 X 31)	132	98	90

Prepared/Date: MPM 10/31/05
Checked/Date: HTD 10/31/05

2.7 GAMMA WALKOVER SURVEY

Volumetric sampling has a low probability of identifying small areas of elevated residual radioactivity. Scanning surveys have a much higher probability of identifying small areas of elevated residual radioactivity and are performed to locate radiation anomalies indicating residual radioactivity that may require further investigation or action. Since both source terms have a gamma radiation decay signature, gamma walkover scan surveys were chosen as the method to investigate for localized areas of elevated radioactivity in soils.

Gamma walkover surveys were performed to locate small areas of elevated residual radioactivity. They were performed by holding the NaI detector close to the ground surface and moving it in a pendulum (back-and-forth) motion while walking forward at a speed that allows the surveyor to detect the desired investigation level. When a discernable increase in the count rate (meter or audible) occurred, a more focused survey of the area was performed. By slowing or stopping the forward progress and searching for the area of increased activity, the localized area of elevated residual radioactivity was identified. Once the location was determined, the surveyor allowed the survey meter response to stabilize and obtained a static reading. Locations of elevated residual radioactivity that exceeded the investigation level were marked for additional investigations. Investigation levels for gamma walkover surveys are presented in Section 2.5.

2.7.1 Gamma Walkover Instruments

Gamma Walkover survey instrumentation consisted of a NaI detector and an appropriate survey meter. The Ludlum 2221 or 2350-1 coupled with the Ludlum 44-10 NaI detector were used during FSS survey activities at the Building Complexes 5 and 6A areas. An inventory of instruments was readily available for use.

2.7.2 Gamma Walkover Instrument Calibration

Calibration of portable survey meters were performed in accordance to the manufacturer's recommendations as well as established standards (American National Standards Institute [ANSI], 1997). All calibration documentation is provided in Appendix L.

2.7.3 Gamma Walkover Measurement Detection Limitations

For any of the survey instruments, the detection sensitivity is affected not only by the factors influencing detector efficiency but also by the detector's residence time over a given area and the uncertainty introduced by the human factors involved in moving the detector and interpreting the instrument response. Another factor is that surveys will be performed on soils and the residual radioactivity will be part of the soil matrix as compared to surface contamination evaluations for building surfaces. The combination of multiple source terms, the energy dependent response rate of the NaI detector, and the residual radioactivity being part of a matrix creates a very complex scenario to determine MDCs. The process follows that established in NUREG-1507 and the MARSSIM.

Derivation of the MDC_{SCAN} for soil was a four step process. First, the relationship between the NaI detectors counting rate to exposure rate (cpm per μR/h) as a function of gamma energy was determined. Second, the relationship between radionuclide concentration in soil and exposure (pCi/g per μR/h) was established. Next, the minimum detectable count rate for the surveyor ($MDCR_{SURVEYOR}$) was calculated, and finally all three parameters were utilized to calculate the MDC_{SCAN}.

Several factors needed to be determined in order to establish the relationship between the detector's count rate to exposure rate. The response of the NaI detector is relative to the gamma energy interacting with the detector. Therefore the cpm produced by the detector is a function of the probability of interaction for a gamma of particular energy. This parameter is determined by taking a known detector response (calibration) and applying it to the relative response of the detector at different gamma energies. For this the manufacturers provided values of 900 cpm per μR/h (Ludlum) or 1,200 cpm per μR/h (Eberline) for Cs-137. The relative response of the detector was calculated by multiplying the probability of interaction by the relative fluence rate for a given gamma energy. The probability of interaction was determined from the mass attenuation coefficients (μ/ρ) for NaI and the fluence rate is determined from the mass energy-absorption coefficients (μ_{en}/ρ) for air.

The second phase of this process is to determine the relationship between the radionuclide concentration in the soil and the exposure rate. To accomplish this, the soil was modeled using Microshield™ to determine the exposure rate. The geometry used for this modeling was input as a cylindrical volume with a radius of 28.2 centimeters (area of 0.25 meters^2) and a thickness of 7.5 centimeters (based on the most likely thickness of the contaminated layer used in RESRAD to derive the DCGLs). The dose point was located 10 centimeters directly above the center of the cylinder to represent the typical height above the surface during scanning. The soil was input into

Microshield™ as the standard material concrete with a density of 1.6 g/cm^3 (to represent typical soil). The byproduct and uranium source terms were input at the DCGL concentration and the uranium source was decayed for fifty years in Microshield™ in order for all the decay products to be present in the modeling. The results were 309 pCi/g per μR/h for total uranium (557 pCi/g divided by 1.801 μR/h) and 1.41 pCi/g per μR/h for Co-60 (5 pCi/g divided by 3.549 μR/h.

The first step in determining the MDC_{SCAN} was to calculate $MDCR_{SURVEYOR}$. $MDCR_{SURVEYOR}$ is a function of the background count rate, the length of the counting interval, surveyor efficiency, and the index of sensitivity (statistical) as shown in Equation 2-2. The mean measured background count rate during walkover surveys for the 2" x 2" NaI detectors was 2,533 cpm (with a high value of 3,200 cpm and a low value of 1,800 cpm) and the index of sensitivity (d'), based upon a 95% true positive rate and a rate of 60% false positive, of 1.38. The surveyor efficiency was selected to be 0.5 and the length of the counting interval was 1 second. The results of this evaluation are shown in Table 2.10 and indicate that 761 cpm above background is the minimum value for 95% true positive detection.

$$MDCR_{surveyor} = \frac{d' * \sqrt{b_i} * (60/i)}{\sqrt{p}} \qquad \textit{(Equation 2-2)}$$

where:

$MDCR_{surveyor}$	=	surveyor minimum detectable count rate (above background)
d'	=	the index of sensitivity (the number of standard deviations between the means of background and radioactivity above background).
b_i	=	the number of background counts in the counting interval, *i*.
i	=	the length of the counting interval in seconds.
p	=	surveyor efficiency

Table 2.10: $\mathbf{MDCR_{SURVEYOR}}$ Values

Parameter		Value
i	The length of the counting interval (seconds)	1
d'	Index of sensitivity	1.38
C_b	Background count rate (cpm)	2,533
b_i	Number of background counts in counting interval i	42.2
s_i	Minimum detectable net counts in counting interval i	12.7
MDCR	Minimum detectable count rate (cpm)	538
p	Surveyor efficiency	0.5
$MDCR_{surveyor}$	Surveyor minimum detectable count rate (cpm)	761

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The minimum detectable exposure rate in µR/h is calculated by dividing the $MDCR_{SURVEYOR}$ by the detector efficiency in cpm per µR/h. Multiplying the minimum detectable exposure rate by the soil concentration exposure rate factor in pCi/g per µR/h will yield the MDC_{SCAN} as shown in Equation 2-3. The parameters for calculating MDC_{SCAN} for a 0.25 m^2 (radius of 28.2 cm) circular hot spot with a depth of 7.5 cm and the dose point located 10 cm directly above the center of the circle are shown in Table 2.11. Since the manufacturers reported different efficiencies for the same size NaI detector, both were used to calculate MDC_{SCAN} values in order to show what range of MDC_{SCAN} might be expected.

$$MDC_{SCAN} = \frac{\mathrm{MDCR}_{\mathrm{surveyor}}}{\varepsilon_t} * S_c \qquad \textit{(Equation 2-3)}$$

where:

MDC_{SCAN}	=	the minimum radioactivity concentration in soil above background radioactivity (in pCi/g) that can be reliably detected.
$MDCR_{surveyor}$	=	surveyor minimum detectable count rate (above background)
ε_t	=	Counting system efficiency in cpm per µR/h.
S_c	=	Soil concentration exposure rate factor in pCi/g per µR/h

Table 2.11: MDC_{SCAN} Values For 2 Inch x 2 Inch NaI Detectors

Parameter		Byproduct		Uranium	
		Ludlum	Eberline	Ludlum	Eberline
$MDCR_{surveyor}$	Surveyor minimum detectable count rate (cpm)	761	761	761	761
ε_t	Counting system efficiency (cpm per µR/h)	424	566	4,582	6,110
S_c	Soil concentration exposure rate factor (pCi/g per µR/h)	1.41	1.41	309	309
MDC_{SCAN}	Scan minimum detectable concentration (pCi/g)	2.5	1.9	52	39

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2.7.4 Walkover/Static Instrument Background Measurements

Because the instrument's response to ubiquitous sources of background radiation (e.g., cosmic radiation) could not be distinguished from the contaminant of concern, instrument background measurements were made periodically over the survey periods.

2.7.4.1 Walkover Survey Background Data

A total of nine background measurement readings or ranges were recorded for the scanning evolution. A single background measurement was taken prior to the start of survey for each survey unit and at the beginning of each workday. Table 2.12 presents the walkover (scan) survey background readings for Building Complexes 5 and 6A walkover surveys.

Table 2.12: Building 5 and 6A Complexes Walkover Survey Background Measurements

Survey Unit	Walkover Background Measurements	
	Recorded Background Reading (cpm)	Average Background Reading (cpm)
05-01	3600-4200	3900
05-02	3600-4200	3900
05-03	3423	3423
05-04	3226	3226
05-05	3323	3323
05-06	3127	3127
06-01	2698	2698
06-02	2725	2725
06-03	2725	2725

Prepared/Date: MPM 03/08/06
Checked/Date: HTD 03/08/06

2.7.4.2 Static Survey Background Data

A total of nine background measurement readings or ranges were recorded for the static survey evolution. A single background measurement was taken prior to the start of survey for each survey unit. Table 2.13 presents the static survey background readings for Building Complexes 5 and 6A static surveys.

Table 2.13: Building 5 and 6A Complexes Static Survey Background Measurements

Survey Unit	Static Background Measurements (cpm)
05-01	3600
05-02	3600
05-03	3423
05-04	3226
05-05	3323
05-06	3127
06-01	2698
06-02	2725
06-03	2725

Prepared/Date: MPM 03/08/06
Checked/Date: HTD 03/08/06

2.7.5 Walkover and Static Instrument Background Adjustment

The instrumentation used in walkover and static surveys to measure the residual radioactivity is influenced by cosmic and terrestrial sources of radiation. In this report, data sets for walkover and direct static measurements are presented with both the gross (uncorrected) measurement and the background-adjusted measurement for evaluation.

Instrument and detector combinations used for scanning of trench bottoms and trench excavation spoil piles were identical to scanning instruments used for the gamma walkover survey and carry the same detection limitations identified in Section 2.7.3. Instrumentation used for scanning of the trenches are identified Table 2.14. Calibration certificates for the scanning instruments are presented in Appendix L.

Table 2.14: Trench Scanning Instrumentation

Scanning Instrumentation			
Inst Model	**Serial #**	**Detector Model**	**Serial #**
2221	97833	44-10	0534
2221	15651	44-10	192589
2221	190224	43-93	215615
2224	183074	43-89	193028
2224	183077	43-93	212501
2350-1	55852	44-10	15203
2350-1	18655	44-10	199144

Prepared/Date: MPM 03/08/06
Checked/Date: HTD 03/08/06

2.7.6 Trench Volumetric Screening Instrumentation

A background-shielded 2 inch by 2 inch NaI detector and instrument system was set up and operated as a screening counter in areas where trenching excavation activities took place. The NaI detector and instrument system was used to screen the volumetric samples collected from both spoils piles and from the trench walls and bottoms during excavation of utilities. The instrument systems used for screening in background-shielded 2 inch by 2 inch NaI setups are listed in Table 2.15 below. Calibration certificates for these instruments are presented in Appendix L.

Table 2.15: Trench Sample Screening Instrumentation

Screening Instrumentation			
Inst Model	**Serial #**	**Detector Model**	**Serial #**
2221	97833	44-10	0534
2221	15651	44-10	192589
2350-1	55852	44-10	15203
2350-1	18655	44-10	199144

Prepared/Date: MPM 03/08/06
Checked/Date: HTD 03/08/06

2.7.6.1 Screening Instrument Maximum Count Rate at DCGL Values

The instrument's detector was placed in a lead-shielded enclosure to minimize background radiation. The lead shield was created by stacking lead bricks, in a cube-like structure, to a height of approximately 1.5 feet. The shield was left open on the upper end to allow for the soil sample container to sit inside the shield with the detector placed on top of the sample.

Prior to use, several factors needed to be determined in order to establish the relationship between the detector's count rate to exposure rate. The response of the NaI detector is relative to the gamma energy interacting with the detector. Therefore, the cpm produced by the detector is a function of the probability of interaction for a gamma of a particular energy. This parameter is determined by taking a known detector response (calibration) and applying it to the relative response of the detector at different gamma energies. For this the manufacturers provided values of 900 cpm per μR/h (Ludlum) or 1,200 cpm per μR/h (Eberline) for Cs-137. The relative response of the detector was calculated by multiplying the probability of interaction by the relative fluence rate for a given gamma energy. The probability of interaction was determined from the mass attenuation coefficients (μ/ρ) for NaI and the fluence rate is determined from the mass energy-absorption coefficients (μ_{en}/ρ) for air.

The second phase was to determine the relationship between the radionuclide concentration in the soil and the exposure rate. To accomplish this, the soil was modeled using Microshield™ to determine the exposure rate. The geometry used for this modeling was input as a cylindrical volume with a radius of five centimeters and a height of 13 centimeters (based on the size of the sample containers used at the site). The dose point was located directly above the center of the cylinder at 0.25 cm from the detector to represent the soil sample sitting on the detector. No sample container shielding was taken into account for the modeling. The soil was input into Microshield™ as the standard material concrete with a density of 1.6 g/cm^3 (to represent typical soil) and a total activity correction was made to equate to activity in 500 grams of soil (the nominal sample weight at the Site). The byproduct and uranium source terms were input at the corrected DCGL concentration and the uranium source was decayed for fifty years in Microshield™ in order for all the decay products to be present in the modeling. The results of the model produced soil concentration exposure rate factors of 866 pCi/g per μR/h for total uranium (557 pCi/g divided by 0.643 μR/h) and 4.7 pCi/g per μR/h for Co-60 (5 pCi/g divided by 1.06 μR/h).

Using the exposure rate factors above and the previously established counting system efficiencies, the expected maximum cpm can be calculated at the DCGL limits (557 pCi/g and 5 pCi/g respectively) for volumetric trench soil samples in the screening configuration (i.e., soil sample container sitting on top of the detector probe) for the specific instruments used (Table 2.16).

Table 2.16: Expected Maximum Screening cpm Values at the DCGL

Parameter	Byproduct		Uranium	
	Ludlum	Eberline	Ludlum	Eberline
Counting system efficiency (cpm per μR/h)	424	566	4,582	6,110
Soil concentration exposure rate factor (pCi/g per μR/h)	4.7	4.7	866	866
pCi/g to cpm conversion (cpm/pCi/g)	90.2	120.4	5.3	7.1
Expected maximum cpm at the DCGL limit	451	602	2,952	3,955

Prepared/Date: MPM 10/31/05
Checked/Date: HTD 10/31/05

By calculating the expected maximum cpm at the DCGL limits for the trench volumetric samples, using the most conservative value calculated of either instrument group and using the byproduct parameters (451 cpm greater than background), a cpm-to-DCGL comparison was performed for all the trench volumetric soil samples screened. None of the trench volumetric sample screening results were greater than 342 cpm after background was subtracted from the gross counting results. This comparison information offers risk managers and decision-makers additional insight regarding the magnitude of compliance for trench soils (trench bottoms and the spoils excavated during utility removal).

2.7.6.2 Screening Instrumentation MDC

The Ludlum 2221 or 2350-1 instrument and the 44-10 (2 inch x 2 inch NaI) detector is a reliable instrument system with adequate detection sensitivity and was readily available for use as a screening counting system (timed static measurements of field volumetric samples). The following formulation is used to derive the MDC, in cpm, for the Ludlum 2221 or 2350-1 instruments and the 44-10 (2 inch x 2 inch NaI) detector probe:

$$MDC = 3 + 4.65\sqrt{C_b} \qquad \textit{(Equation 2-4)}$$

Where:

MDC = the minimum cpm above background radioactivity that can be measured with 95 percent confidence.

C_b = the total number of background counts over the sample count period.

Due to the large number of available instruments and technicians to operate the instruments, and because of the short time frame in which trench and spoil pile samples needed to be collected and screened, a total of four different instrument systems were used. In order to determine the most appropriate MDC value to use, the background readings for each instrument system were averaged and an MDC was calculated using that average background count rate. In addition, the largest background count rate recorded during trench sample screening was also selected and used to calculate the instrument MDC. Table 2.5 presents the results of those calculations and the instrument models and serial numbers. The table shows that any instrument system listed above has a sufficiently low MDC for screening purposes (less than 25 percent of the most conservative cpm at DCGL limit of 451 cpm).

Table 2.17: Trench Screening Instrumentation MDC Values

2x2 NaI Trench Sample Screening Instrument MDCs						
Inst #	**Ser #**	**Dect #**	**Ser #**	**Mean**	**# of Measurements**	**MDC** (cpm)
2221	97833	44-10	0534	307.6	222	84.6
2221	15651	44-10	192589	353.7	103	90.4
2350-1	55852	44-10	15203	253.3	151	77.0
2350-1	18655	44-10	199144	202.0	3	69.1
Highest Observed Background Count Rate				690.0		125.1

Prepared/Date: MPM 03/08/06
Checked/Date: HTD 03/08/06

3.0 FIELD SURVEY AND SAMPLING RESULTS

Field survey and volumetric sampling results are presented by survey unit with a data assessment and comparison to the release criterion. Where anomalies or notable results were identified, additional discussion and data are presented for the specific survey unit. QC data is presented separately in Section 6 of this report. Each survey unit is presented with a summary of the survey results, figures showing the layout of each survey unit and the selected sample locations, data assessment tables, and a preliminary comparison to the decision criteria. All of the data associated with each survey unit and its associated evaluations are provided in the appendices of this report.

3.1 FIELD SURVEY AND VOLUMETRIC SOIL SAMPLING RESULTS OVERVIEW

In all, 138 volumetric soil samples were collected and analyzed from nine survey units as part of FSS for Building Complexes 5 and 6A. In locations where replicate samples were collected, they are indicated on the survey and sampling maps by sample locations where 'duplicate' is indicated. Seven replicate measurements were collected as part of the overall project QA/QC. For data reduction purposes, the arithmetic mean of the replicate measurement and the corresponding initial measurement was used as the reported value for specific sample locations where a replicate measurement was made. Further information about the replicate measurements and the assurance of precision and variability is presented in Section 6.

3.2 SOIL SAMPLING RESULTS

The preliminary data review assesses the FSS data utilizing various numerical and graphical techniques. This includes summary statistics, histograms, probability plots, and box plots. Each technique was run to provide insight that would identify any patterns, relationships, or potential anomalies in the distribution of the data. A key test of the data set is for goodness-of-fit. It is important because it identifies the underlying distribution of the data set and provides a comparison of appropriate metrics calculated from the data. The Anderson-Darling Test was used to measure the relative goodness of the fit of the observed data distribution to the normal and lognormal standard distributions. Distributions other than normal and lognormal were evaluated but were discounted for this data set on the grounds that:

- Based on knowledge of the expected distribution of radioactivity in the environment and in background, the data were expected to have a lognormal distribution; and
- The probability plots and histograms generated (for a host of possible distributions) gave no good evidence that other than normal or lognormal distributions might be present.

Posting plots provide a visual representation of the sampling locations and the activity concentrations at those locations. Posting plots are also used to reveal the heterogeneities in the data, especially possible patches of elevated residual radioactivity. Posting plots are provided in the appendices.

Once the survey unit data has been assessed and verified that it is acceptable for comparison to the release criteria, it will be evaluated against the DCGL survey unit average (median) concentration corresponding to the permissible limit ($DCGL_w$)s. This section of the report provides a summary of the FSS data and statistical data assessment. All of the data associated with each survey unit and its associated evaluations are provided in the appendices of this report.

3.2.1 Survey Unit CE-FSS-05-01

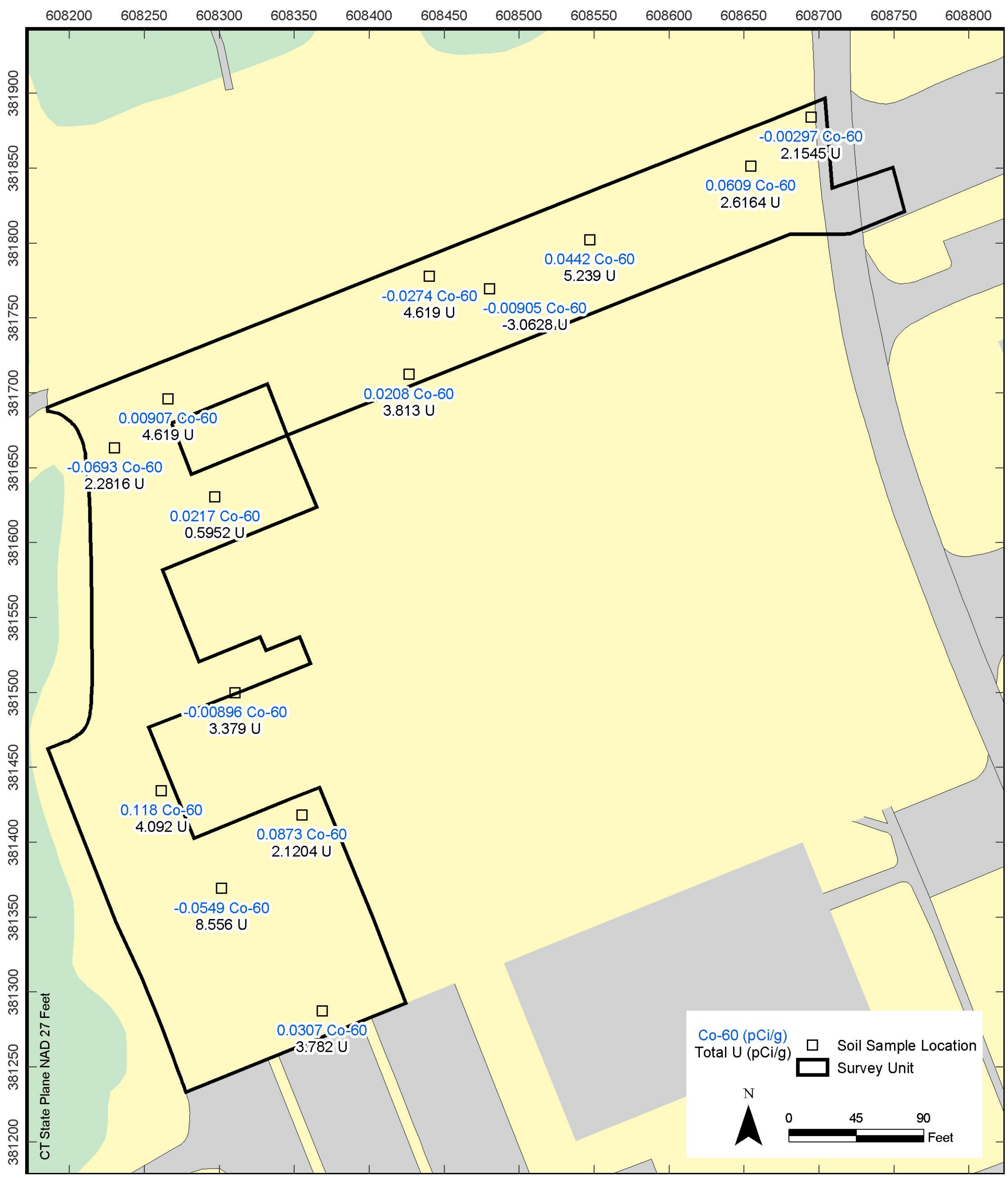
Survey Unit 05-01 is located in the north and west sections of the Building 5 Complex area and consists of approximately 8,100 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building 5 Complex. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix A.

3.2.1.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit 05-01 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. An average reading of 3,600 cpm to 4,200 cpm (background range of variability) was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 4,200 cpm gross, 0 cpm net.

3.2.1.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 05-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.1 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit 05-01.



Prepared/Date: BRP 03/03/06
Checked/Date: HTD 03/03/06

Figure 3.1: Survey Unit 05-01 Total U and Co-60 Activities (pCi/g)

3.2.2 Survey Unit CE-FSS-05-02

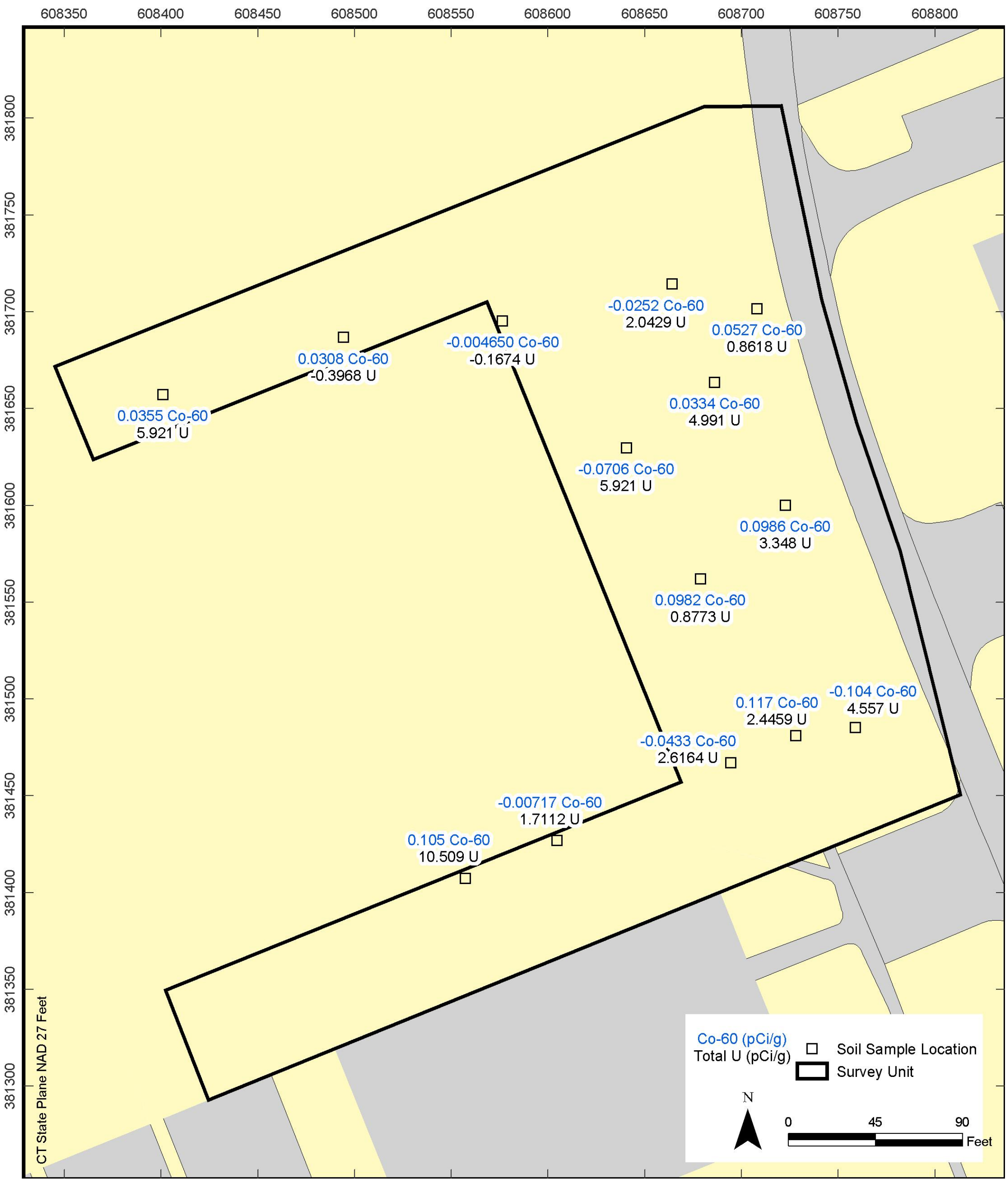
Survey Unit 05-02 is located in the south and east sections of the Building 5 Complex area and consists of approximately 8,100 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building 5 Complex. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix B.

3.2.2.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit 05-02 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. An average reading of 3,600 cpm to 4,200 cpm (background range of variability) was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 4,200 cpm gross, 0 cpm net.

3.2.2.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 05-02 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.2 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit 05-02.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.2: Survey Unit 05-02 Total U and Co-60 Activities (pCi/g)

3.2.3 Survey Unit CE-FSS-05-03

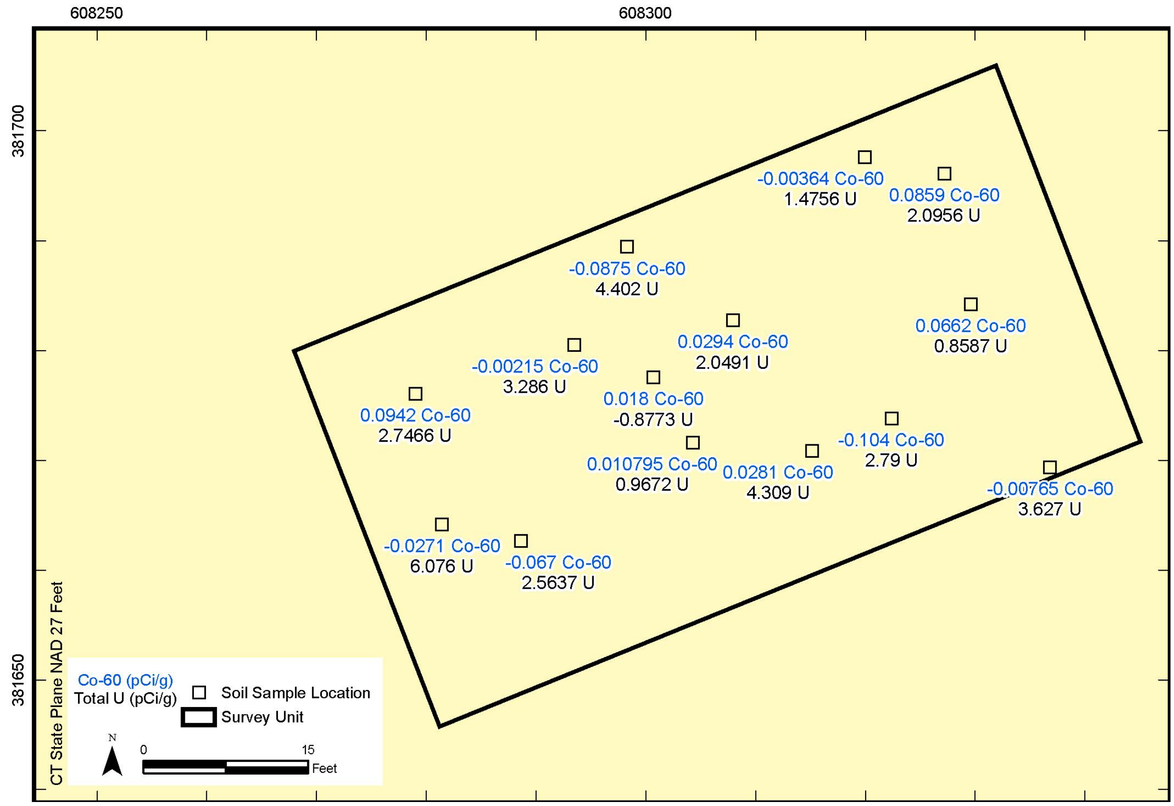
Survey Unit 05-03 is the former Building 15 footprint and consists of approximately 200 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building 5 Complex. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix C.

3.2.3.1 Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit 05-03 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. An average reading of 3423 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 3423 cpm gross, 0 cpm net.

3.2.3.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 05-03 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.3 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 05-03.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.3: Survey Unit 05-03 Total U and Co-60 Activities (pCi/g)

3.2.4 Survey Unit CE-FSS-05-04

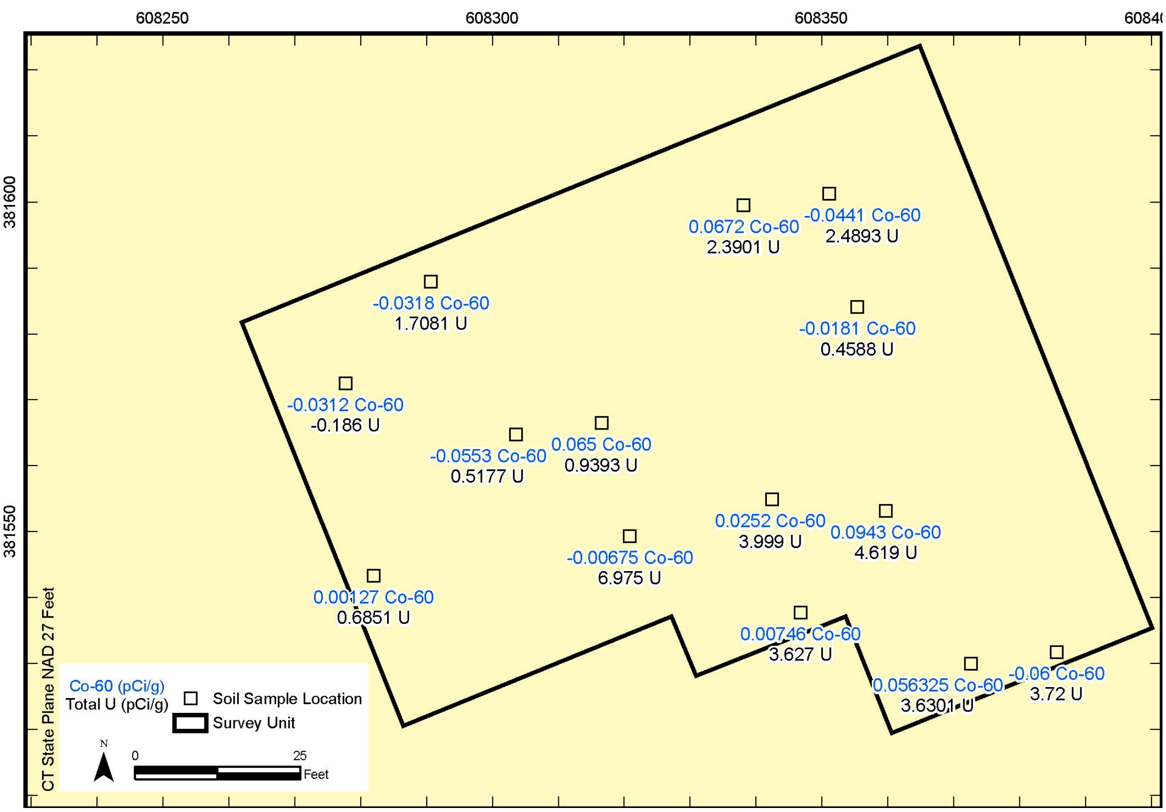
Survey Unit 05-04 is the former Building 16 footprint and consists of approximately 800 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building Complexes 5 and 6A areas. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix D.

3.2.4.1 Gamma Walkover Survey Results

Approximately 50 percent of the surface area for Survey Unit 05-04 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. The surface areas surveyed included a rectangular area through the center of the survey unit from east to west and a smaller rectangular area in the south-east corner of the Survey Unit. An average background reading of 3,226 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 3,226 cpm gross, 0 cpm net.

3.2.4.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 05-04 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.4 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 05-04.



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Checked/Date: HTD 3/03/06

Figure 3.4: Survey Unit 05-04 Total U and Co-60 Activities (pCi/g)

3.2.5 Survey Unit CE-FSS-05-05

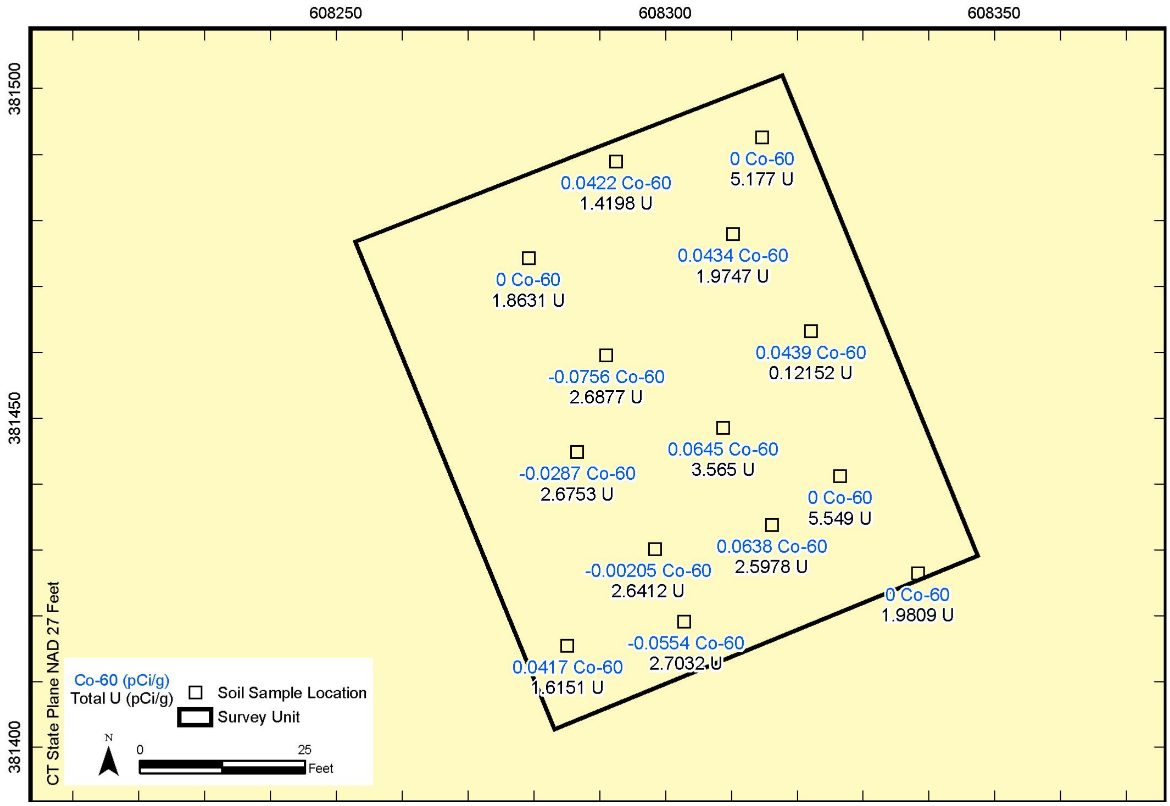
Survey Unit 05-05 is the former Building 18 footprint and consists of approximately 500 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building Complexes 5 and 6A areas. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix E.

3.2.5.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit 05-05 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. An average background reading of 3,323 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 3,323 cpm gross, 0 cpm net.

3.2.5.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 05-05 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.5 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 05-05.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.5: Survey Unit 05-05 Total U and Co-60 Activities (pCi/g)

3.2.6 Survey Unit CE-FSS-05-06

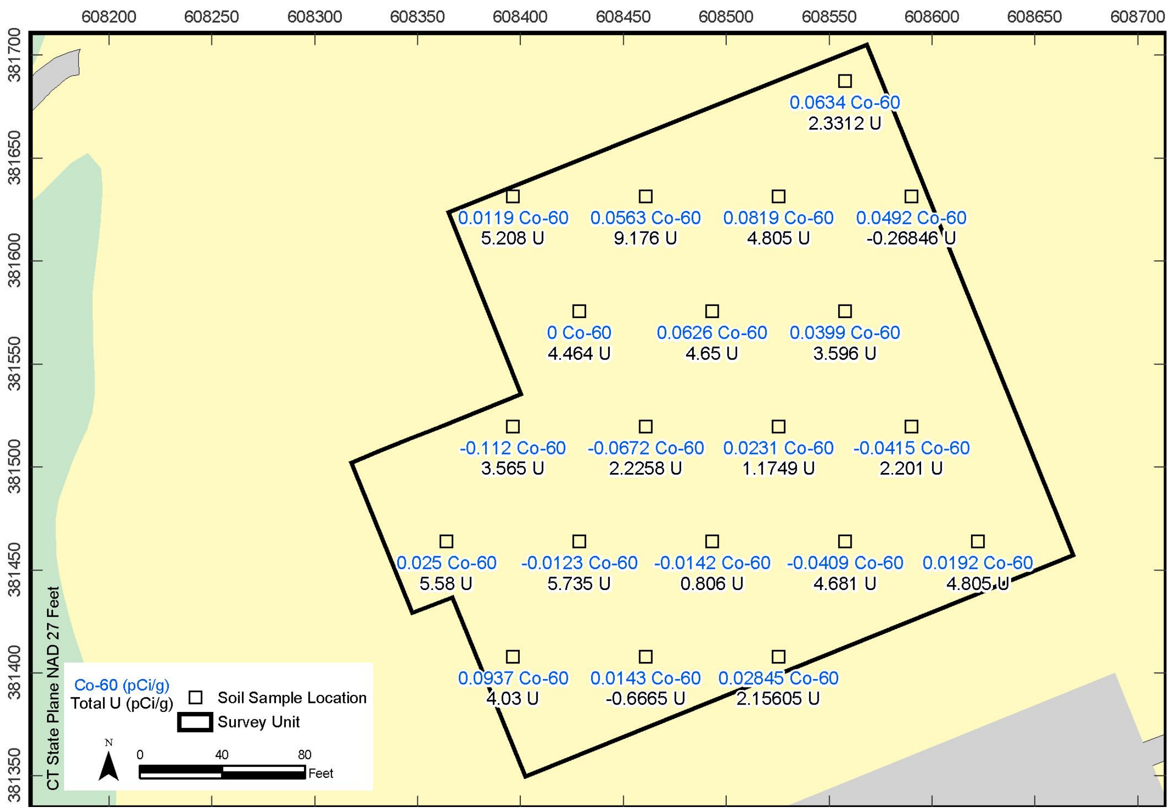
Survey Unit 05-06 is the former Building 5 footprint and consists of approximately 6,700 square meters of land area. Twenty survey locations were placed using a random start, systematic triangular grid pattern within the Class 2 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix F.

3.2.6.1 Gamma Walkover Survey Results

Approximately 50 percent of the surface area for Survey Unit 05-06 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. An average background reading of 3,127 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 3,127 cpm gross, 0 cpm net.

3.2.6.2 Volumetric Soil Sample Results

Twenty volumetric soil samples were obtained for FSS at the triangular grid intersection locations in Survey Unit 05-06 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.6 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 05-06.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.6: Survey Unit 05-06 Total U and Co-60 Activities (pCi/g)

3.2.7 Survey Unit CE-FSS-06-01

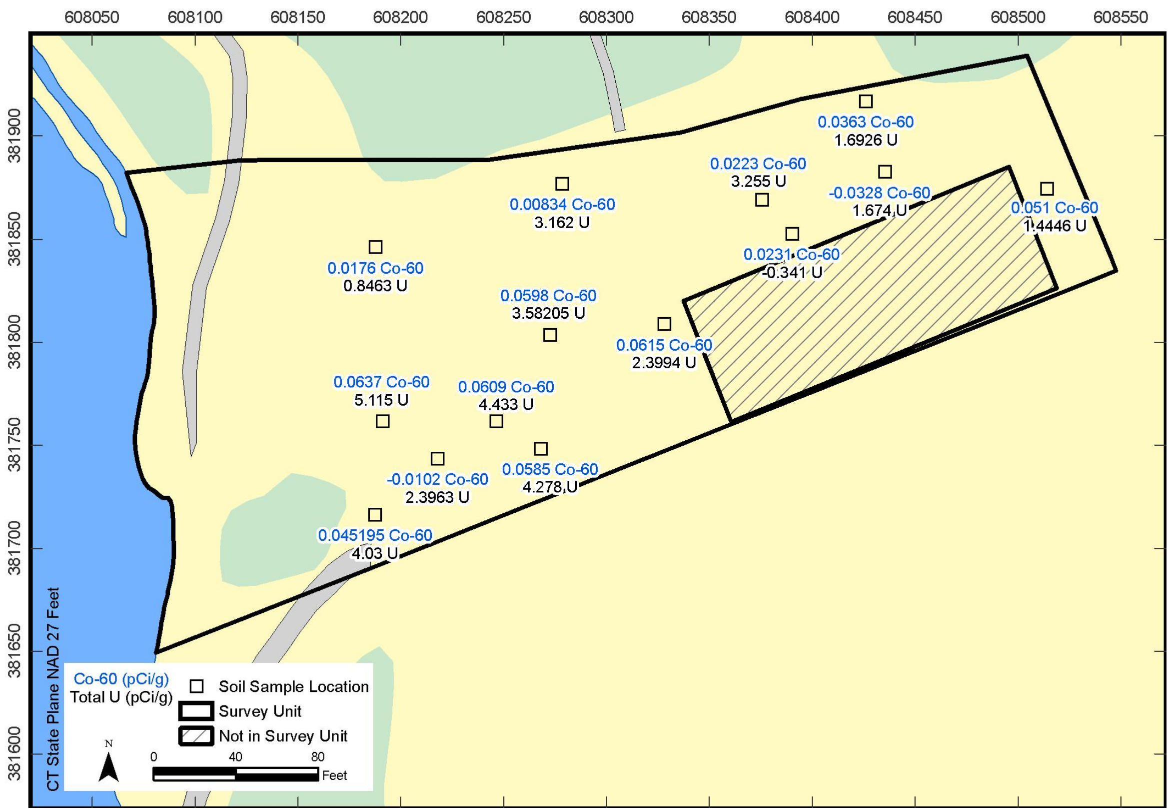
Survey Unit 06-01 includes the area surrounding Building 6A and consists of approximately 6,000 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building 6A Complex. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix G.

3.2.7.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit 06-01 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. The surface areas surveyed included the north, north west, and north east perimeters, and a rectangular area through the south area from east to west of the Survey Unit 06-01. An average background reading of 2,698 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 3,500 cpm gross, 802 cpm net.

3.2.7.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 06-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.7 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 06-01.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.7: Survey Unit 06-01 Total U and Co-60 Activities (pCi/g)

3.2.8 Survey Unit CE-FSS-06-02

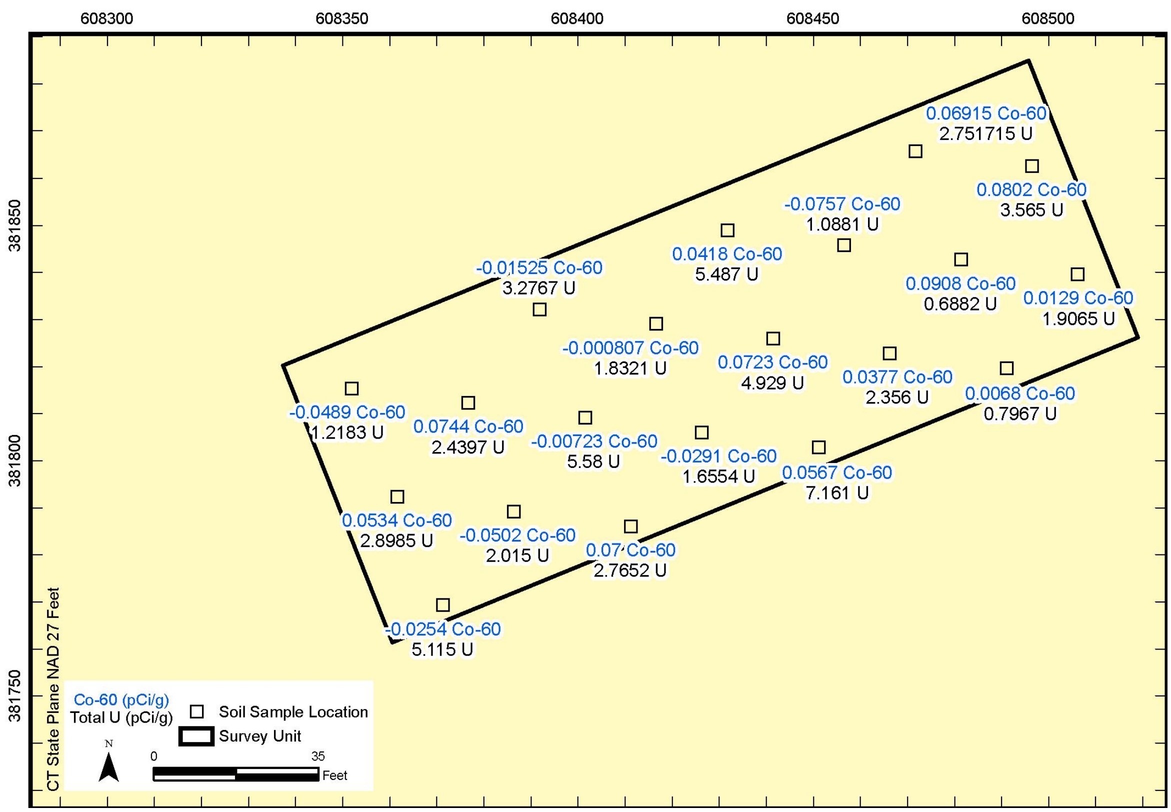
Survey Unit 06-02 is the former Building 6A footprint and consists of approximately 1,000 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building 6A Complex. Twenty survey locations were placed using a random start, systematic triangular grid pattern within the Class 2 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix H.

3.2.8.1 Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit 06-02 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. An average background reading of 2,725 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 3,500 cpm gross, 775 cpm net. The analytical results provide evidence that soil residual radioactivity is appreciably below the $DCGL_W$, as well as the $DCGL_{EMC}$ for the Site.

3.2.8.2 Volumetric Soil Sample Results

Twenty volumetric soil samples were obtained for FSS at the triangular grid intersection locations in Survey Unit 06-02 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.8 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 06-02.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.8: Survey Unit 06-02 Total U and Co-60 Activities (pCi/g)

3.2.9 Survey Unit CE-FSS-06-03

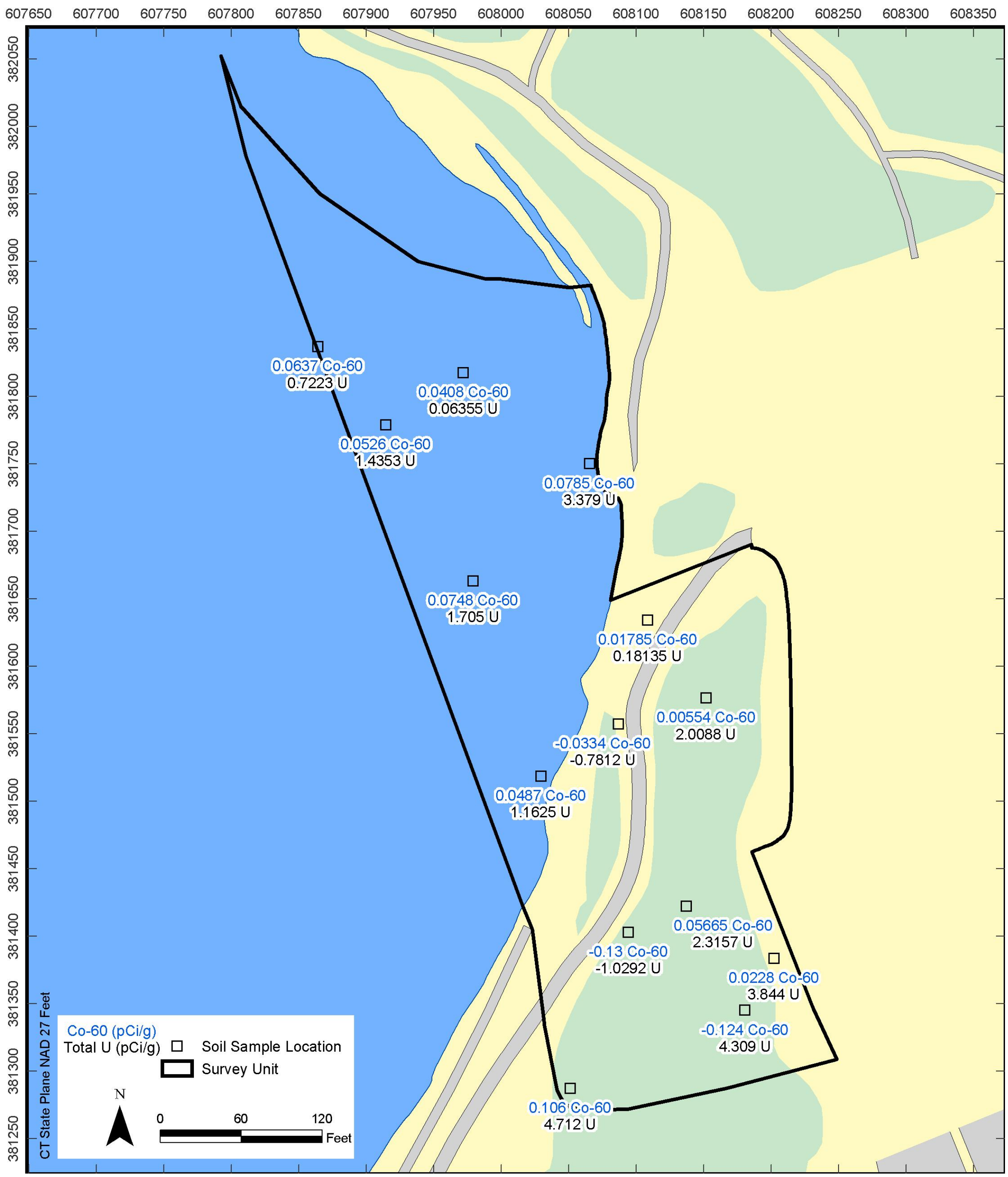
Survey Unit 06-03 is located in the western section of the Building 6A Complex area and consists of approximately 12,600 square meters of land area. Figure 2.1 presents an overview of the Survey Unit within the Building 6A Complex. This survey unit represents generally impacted areas adjacent to Building Complexes 5, 6A, and 17 and is included as part of the Building 6A Complex for completeness for this portion of the Site. This survey unit provides sediment data for Great Pond as previously referenced in the Building 5 Complex FSS report. Fourteen survey locations were randomly selected within the Class 3 Survey Unit to represent the survey/sample population distribution for the Survey Unit. All data associated with this survey unit are provided in Appendix I.

3.2.9.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit 06-03 was surveyed by walking parallel transects across the area moving the detector from side-to-side in a serpentine motion. The surface areas surveyed included the north, north west, and north east perimeters, and a rectangular area through the south area from east to west of the Survey Unit 06-01. An average background reading of 2,725 cpm was recorded during the walkover survey. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils at those locations. The highest scan reading was 2,800 cpm gross, 75 cpm net.

3.2.9.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit 06-03 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that all the results are acceptable for use. Figure 3.9 presents the resultant activity results for both Co-60 and total uranium concentrations for Survey Unit 06-03.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

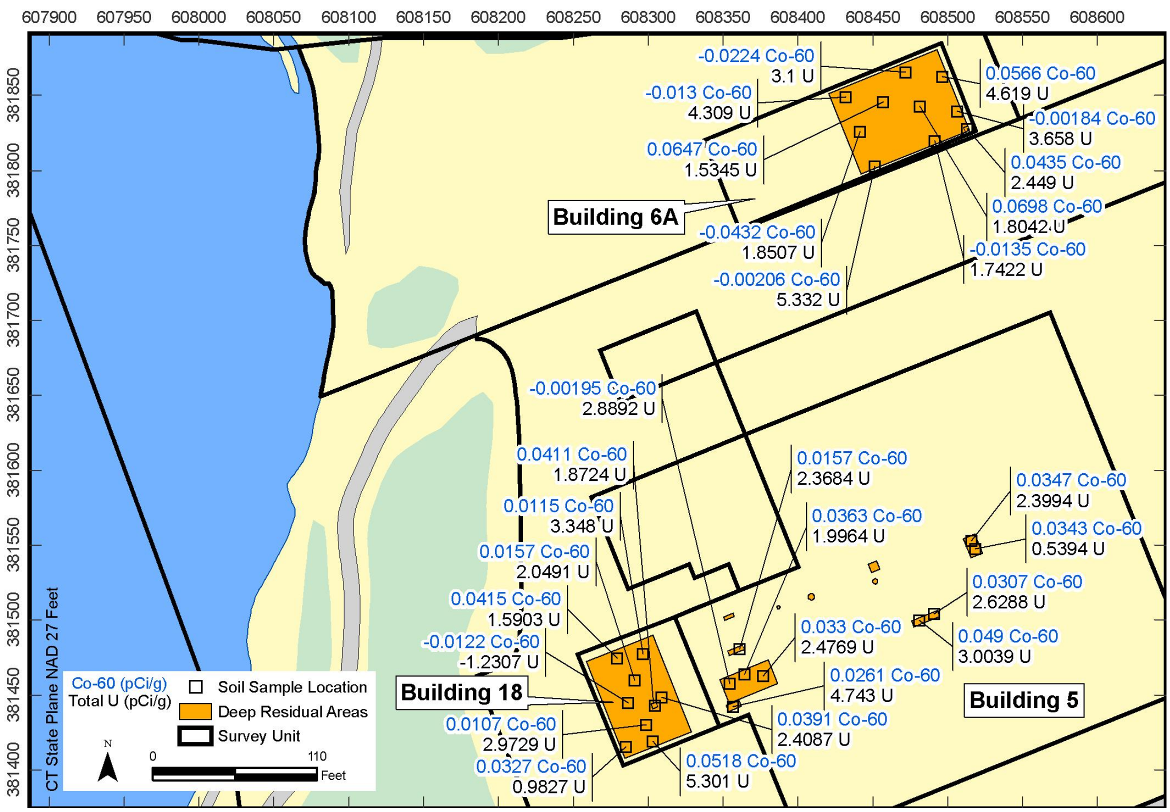
Figure 3.9: Survey Unit 06-03 Total U and Co-60 Activities (pCi/g)

3.3 SUB-SLAB SOIL SAMPLING RESULTS

Twenty-eight sub-slab soil samples were obtained and analyzed on the on-site gamma spectroscopy system for Building Complexes 5 and 6A. The analytical results show that soil residual radioactivity is appreciably below the DCGLws and are presented in Appendix J.

3.3.1 Sub-Slab Population General Description

Survey Units 05-05, 05-06, and 06-02 contained structural material (concrete) deeper than four feet below the finished grade. Survey unit 05-05 had a deep basement with a sump that extended deeper than four feet below finished grade. Eight randomly distributed sample locations fell within the boundary of the deep basement, but only 7 samples could be collected from those locations. One sample was relocated and a sample from below the sump was also collected (both biased). Survey unit 05-06 had nine deep pits and one sump that extended deeper than four feet below finished grade. One systematic triangular grid pattern sample fell within a pit and eight additional biased samples were collected from pits and the sump. Survey unit 06-02 had a deep basement with a sump that extended deeper than four feet below finished grade. Ten randomly distributed sample locations fell within the boundary of the deep basement, but only 9 samples could be collected from those locations. One biased sample was collected from the sump. Figure 3.10 presents the resultant activity results for both Co-60 and total uranium concentrations for sub-slab soil samples.



Prepared/Date: BRP 3/03/06
Checked/Date: HTD 3/03/06

Figure 3.10: Sub-Slab Sample Results

3.3.2 Sub-Slab Population Statistical Summary

None of the sub-slab soil sample results were incorporated into the soil FSS sampling population for Building Complexes 5 and 6A. Instead, they were grouped into a separate "sub-slab" population and ran through the same statistical evaluations as the surface survey unit data. The results of the sub-slab sampling are presented in Table 3.1.

Table 3.1: Summary Statistics, Sub-Slab Isotopes

Statistic	Unbiased		Biased		Combined	
	Total U	Co-60	Total U	Co-60	Total U	Co-60
Number of Measurements	17	17	11	11	28	28
Arithmetic Mean	2.588	0.0183	2.6	0.0289	2.593	0.0224
Standard Deviation	1.677	0.0337	1.009	0.0151	1.43	0.028
Standard Error of the Mean	0.4067	0.00812	0.3042	0.0046	.270	0.0053
Coefficient of Variation	0.6479	1.842	0.3881	0.5235	0.5513	1.248
Geometric Mean	2.515	0.0366	2.34	0.0295	2.442	0.0328
Maximum	5.3	0.0698	4.7	0.049	5.3	0.0698
Median	2	0.0157	2.5	0.033	2.4	0.035
Minimum	-1.2	0.0157	0.5	-0.00195	-1.2	-0.0432
Range	6.5	0.113	4.2	0.05095	6.5	0.113
UCL_{95} (median)	3.7	0.0415	3	0.0411	3	0.0363
LCL_{95} (median)	1.7	-0.0122	1.9	0.0115	1.9	-0.0018

Prepared/Date: ECS 3/07/06
Checked/Date: HTD 3/07/06

3.4 GAMMA WALKOVER SURVEYS

Gamma walkover surveys were performed using a 2 inch by 2 inch NaI detector, held close to the ground surface and moved in a pendulum (back-and-forth) motion while walking forward at a slow and steady speed. When a discernable increase in the count rate (meter face indication or audible tone) was observed, the surveyor slowed or stopped their forward progress and searched for the area of highest activity in the area. Once the elevated location was pin-pointed, the surveyor marked the location. A static 1-minute count was performed with the NaI detector in the marked location and recorded.

3.4.1 Gamma Walkover Survey Results

Table 3.2 presents the summary results of the gamma walkover surveys, the number of volumetric samples obtained as a result of elevated walkover survey readings, and the highest measurements obtained during the static counts performed at the identified elevated locations. No elevated readings were identified during the walkover survey that prompted additional volumetric sampling of soils. Figure 3.11 indicates areas where gamma walkover surveys were performed.

Table 3.2: Gamma Walkover Survey Results Summary

Survey Unit (CE-FSS)	Walkover Field Scan Results						
	Survey Unit Class.	**Percent of Survey Unit Surveyed**	**Number of Elevated Locations Identified and Sampled**	**Recorded Background Reading (cpm)**	**Average Background Reading (cpm)**	**Highest Scan Reading (gross cpm)**	**Highest Scan Reading (net cpm)**
05-01	3	10	0	3,600 - 4,200	3,900	4,200	0
05-02	3	10	0	3,600 - 4,200	3,900	4,200	0
05-03	1	100	0	3,423	3,423	3,423	0
05-04	2	50	0	3,226	3,226	3,226	0
05-05	3	10	0	3,323	3,323	3,323	0
05-06	2	50	0	3,127	3,127	3,127	0
06-01	3	10	0	2,698	2,698	3,500	802
06-02	1	100	0	2,725	2,725	3,500	775
06-03	3	10	0	2,725	2,725	2,800	75

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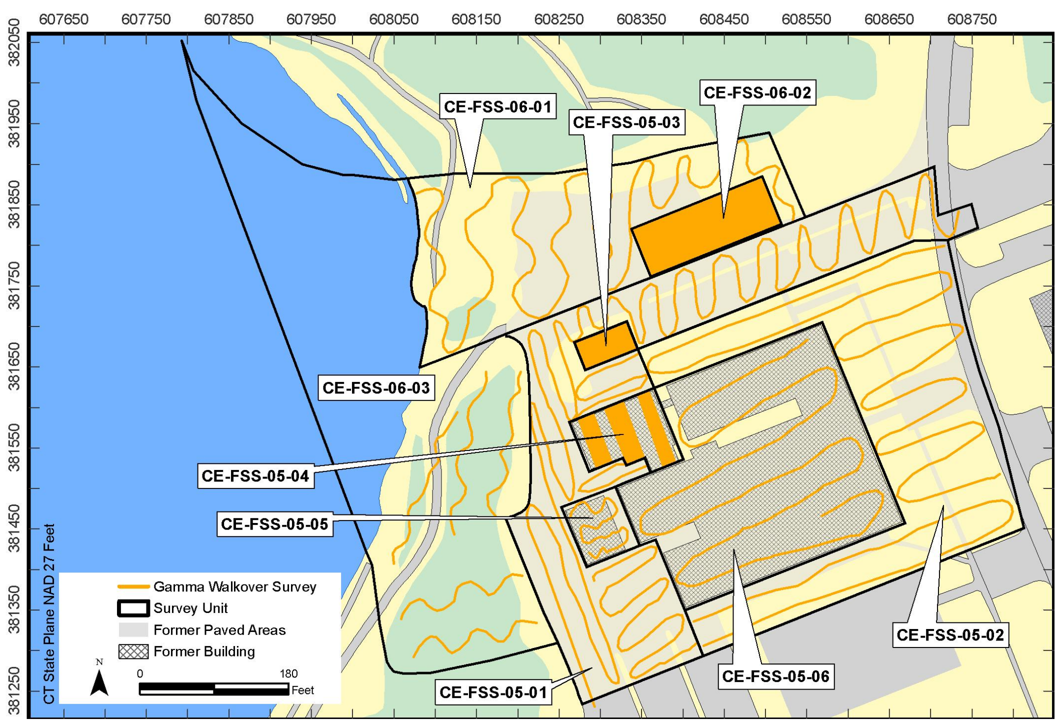

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Checked/Date: HTD 3/03/06

Figure 3.11: Gamma Walkover Surveys

3.4.2 Direct Static Surface Measurements

In addition to gamma walkover surveys and the static surface measurements performed in locations where walkover survey results approached or exceeded the investigation levels, direct static surface 1-minute measurements were performed at all FSS volumetric soil sample locations using the gamma walkover NaI detector. Although not required by the FSS plan, these static 1-minute measurements were used as an additional gauge to help identify areas of elevated residual radioactivity and to support the conclusion that residual radioactive materials are less than the $DCGL_W$ at the Site. Table 3.3 provides a summary of the direct static readings performed at each volumetric sampling location.

Table 3.3: Static Measurement Summary Results

Static Measurement Summary Results				
Sample ID	**Survey Unit**	**Number of Measurements in the Survey Unit**	**Maximum Net (cpm)**	**Avg. Net (cpm)**
SS-FSS-05-052	05-01	13	51	-333.4
SS-FSS-05-068	05-02	14	469	-102.2
SS-FSS-05-029	05-03	14	830	242.6
SS-FSS-05-019	05-04	14	888	387.0
SS-FSS-05-014	05-05	14	469	41.2
SS-FSS-05-072	05-06	20	500	-125.3
SS-FSS-6A-15	6A-01	14	475	25.2
SS-FSS-6A-35	6A-02	20	440	321.2
SS-FSS-6A-54	6A-03	10	-275	-502.2

Prepared/Date: AGS 3/08/06
Checked/Date: HTD 3/08/06

Results of these static measurements are presented in the Appendices. Review of the static measurement data suggests that elevated surface and near-surface residual radioactivity is not present at the survey locations and that results of the static surveys were significantly lower than the established byproduct $DCGL_W$. These static measurement results support the conclusion that residual radioactivity in soils is significantly less than the $DCGL_W$ for the Site.

3.5 FIELD SURVEY SUMMARY

A summary of the FSS results is presented by survey unit in Table 3.4 (for total uranium) and Table 3.5 (for Co-60). These tables provide a statistical summary of the Building Complexes 5 and 6A Soil FSS.

Table 3.4: Summary Statistics, Total Uranium

Statistic	Survey Unit								
	CE-FSS - 05-01	CE-FSS - 05-02	CE-FSS - 05-03	CE-FSS - 05-04	CE-FSS - 05-05	CE-FSS - 05-06	CE-FSS - 06-01	CE-FSS - 06-02	CE-FSS - 06-03
Number of Measurements	14	14	14	14	14	20	14	20	14
Arithmetic Mean	3.2	3.2	2.6	2.5	2.6	3.5	2.7	2.99	1.7
Standard Deviation	2.61	2.94	1.74	2.01	1.42	2.33	1.54	1.81	1.82
Standard Error of the Mean	0.697	0.787	0.466	0.537	0.379	0.521	0.411	0.405	0.487
Coefficient of Variation	0.815	0.915	0.7	0.792	0.542	0.663	0.567	0.606	1.063
Geometric Mean	3.16	2.97	2.5	2.03	2.05	3.43	2.62	2.47	1.38
Maximum	8.6	10.5	6.1	7	5.5	9.2	5.1	7.2	4.7
Median	3.6	2.5	2.65	2.45	2.6	3.8	2.8	2.6	1.55
Minimum	-3.1	-0.4	-0.9	-0.2	0.1	-0.7	-0.3	0.7	-1
Range	11.7	10.9	7	7.2	5.4	9.9	5.4	6.5	5.7
UCL_{95} (median)	4.6	5	3.6	3.7	2.7	4.8	4	3.6	3.4
LCL_{95} (median)	2.1	0.9	1	0.5	1.6	2.2	1.4	1.8	0.1

Prepared/Date: ECS 3/07/06
Checked/Date: HTD 3/07/06

Table 3.5: Summary Statistics, Co-60

Statistic	Survey Unit								
	CE-FSS - 05-01	CE-FSS - 05-02	CE-FSS - 05-03	CE-FSS - 05-04	CE-FSS - 05-05	CE-FSS - 05-06	CE-FSS - 06-01	CE-FSS - 06-02	CE-FSS - 06-03
Number of Measurements	14	14	14	14	14	20	14	20	14
Arithmetic Mean	0.016	0.023	0.0024	0.0050	0.0098	0.014	0.033	0.0207	0.02
Standard Deviation	0.051	0.069	0.060	0.050	0.043	0.051	0.030	0.0505	0.071
Standard Error of the Mean	0.014	0.018	0.016	0.013	0.011	0.011	0.008	0.0113	0.019
Coefficient of Variation	3.273	3.043	24.93	10.01	4.35	3.66	0.900	2.442	3.547
Geometric Mean	0.037	0.062	0.036	0.025	0.049	0.036	0.036	0.0455	0.0409
Maximum	0.118	0.15	0.0942	0.0943	0.0645	0.0937	0.0637	0.0908	0.106
Median	0.015	0.0321	0.0043	-0.003	0	0.0212	0.0408	0.0253	0.0448
Minimum	-0.069	-0.104	-0.104	-0.06	-0.076	-0.112	-0.033	-0.076	-0.13
Range	0.1873	0.221	0.1982	0.1543	0.1401	0.2057	0.0965	0.1665	0.236
UCL_{95} (median)	0.0442	0.0982	0.0294	0.0563	0.0434	0.0492	0.0598	0.0692	0.0637
LCL_{95} (median)	-0.027	-0.043	-0.067	-0.044	-0.029	-0.012	0.0083	-0.015	-0.033

Prepared/Date: ECS 3/07/06
Checked/Date: HTD 3/07/06

4.0 TRENCH SURVEY AND SAMPLING

Field survey and volumetric sampling results are presented by utility trench type with a data assessment and comparison to the release criteria. Where anomalies, or notable results, were identified, additional discussion and data are presented. QC data is presented separately in Section 6 of this report. Each utility trench will be presented with a summary of the survey results, figures showing the layout of each survey unit and the selected sample locations, and a preliminary comparison to the decision criteria. All of the data associated with each trench and its associated evaluations are provided in the appendices of this report.

4.1 UTILITY TRENCH SURVEY AND SAMPLING OVERVIEW

The Building Complexes 5 and 6A industrial waste line and other underground utilities were removed between December 2004 and December 2005. Building structures and foundations were removed to a depth of approximately 4 feet below grade. This greatly minimized the interference during utility removal. As-built drawings, as well as test excavations, were used to locate underground utilities.

Trench volumetric soil sampling was performed mostly along the bottom of the trench floor and from the spoil piles generated as a result of the excavation to reach the utility. Volumetric sampling methods included use of hand trowels and stainless steel spoons to collect soil and sediment samples from the trench sampling areas.

Trench radiological scan surveys were performed using hand-held instruments and appropriate detectors. Trench bottoms and spoil piles were randomly scanned to identify areas of elevated residual radioactivity. If suspect areas (stained or discolored soil) were identified, a biased scan was preformed in that area. In areas where scanning measurements exceeded a predetermined action level, the area was marked and a volumetric soil sample taken.

4.2 TRENCH RADIOLOGICAL SURVEYS

During the excavation of the trench utilities, trench bottoms and the spoils removed from the trenches were radiologically surveyed and sampled. Trench bottoms were scanned using either a 100 cm^2 surface area beta/alpha detector or a 2 inch by 2 inch NaI detector with appropriate instrument. Spoils and trench bottoms were also scanned if there was indication of leakage from the industrial waste line to the surrounding soils. At least one volumetric soil sample was collected at approximately 100 linear foot intervals from the spoil piles and the trench bottoms, and collected in areas where scanning measurements exceeded the predetermined action level. Approximately fifty percent of the volumetric samples collected from the trench bottoms and spoil piles were counted on a background-shielded 2 inch by 2 inch NaI detector and instrument system. Volumetric samples exceeding the predetermined count rate (>1,000 cpm gross) on the 2 inch by 2 inch NaI detector system required on-site gamma spectroscopy analysis. However, none of the volumetric samples exceeded this predetermined count rate. Regardless, about ninety percent of the volumetric soil samples collected from the trench bottoms and spoil piles were analyzed by the on-site gamma spectroscopy system.

4.2.1 Trench Scanning Results

Trench bottoms were scanned using either a 100 cm^2 surface area beta/alpha detector or a 2 inch by 2 inch NaI detector and applicable instrument systems. In areas where scan measurement results exceeded the predetermined action level, the area was marked and a volumetric soil sample taken. Volumetric soil samples were counted on the background-shielded 2 inch by 2 inch NaI detector and instrument system and a subset was analyzed using the on-site HPGe system. Trench scan results are presented in Appendix J. None of the scans of the trench bottoms or spoil piles resulted in readings greater than the action level.

4.2.2 Trench Volumetric Sampling Results

A total of 816 volumetric soil samples were collected from the trench bottoms and spoil piles generated during utility excavation activities for the Building Complexes 5 and 6A. Each volumetric soil sample was analyzed on either the background-shielded, in-field 2 inch by 2 inch NaI detector and instrument system, or by the on-site HPGe gamma spectroscopy system, or both. Of these samples, 447 were counted on the background-shielded 2 inch by 2 inch NaI detector and instrument system, while 719 of the samples collected were measured on the on-site HPGe gamma spectroscopy system. Measurement results of the in-field 2 inch by 2 inch NaI instrument (in units of cpm) and analysis results from the on-site HPGe gamma spectroscopy systems (in units of pCi/g) for the trench volumetric samples are presented in Appendix K.

4.3 UTILITY TRENCH LOCATIONS

Trench excavation and utility removal was performed in sections to minimize the number of trench excavations left open at any one time. After removal of the utility from the trench, the trench was radiologically surveyed and sampled. Once the trench was radiologically checked by the Site radiation safety officer (RSO) and CTDEP, the trench was backfilled and graded to match surrounding grade. In the Building Complexes 5 and 6A areas, trench excavation, utility removal, and radiological survey and sampling were performed from December 2004 through December 2005.

Utility trenches were segregated into five definitive trench groups: hot waste line, industrial waste line, sanitary sewer waste line, storm water drains, and other utilities. Other utilities include those that had minimal potential to contain residual radioactive materials or contribute to the release or spread of residual radioactive materials to the environment. Figure 4.1 shows an overview of the trench excavation locations for the Building 5 and 6A Complexes.

Though not classified as separate survey units, segregation of the utility trenches and the surveys performed during excavation activities provides a reasonable means to present survey and sampling data. The segregation is based on the utility use and characteristics of that utility, from historic Site records and applicable analytical data. Additional and greater detailed information regarding the trench utility segregation is presented in the FSSP (MACTEC, 2004).

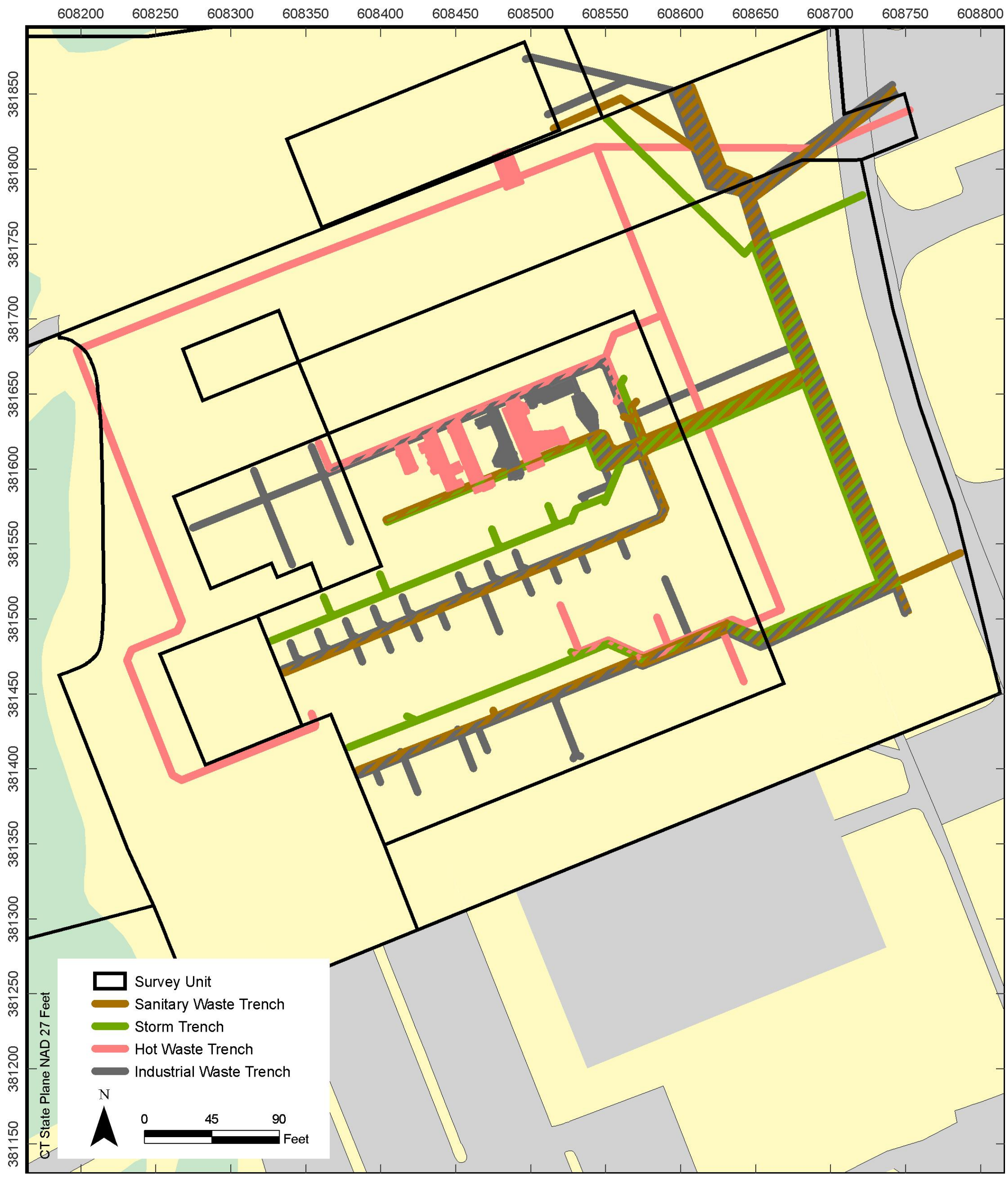

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Figure 4.1: Trench Excavation Overview

4.3.1 Hot Waste Line

There was a large amount of hot waste line removed from the Building Complexes 5 and 6A. The hot waste line trenches have been subdivided into six sections in order to present the volumetric sampling results as shown in Figure 4.2 (next page). Following the overview figure are the hot waste line trench excavation and the volumetric sampling results of the trench and spoils piles for the west section (Figure 4.3), the north section (Figure 4.4), the northeast section (Figure 4.5), the central section (Figure 4.6), the east section (Figure 4.7), and the south section (Figure 4.8). Survey data and additional information is provided in Appendix K.

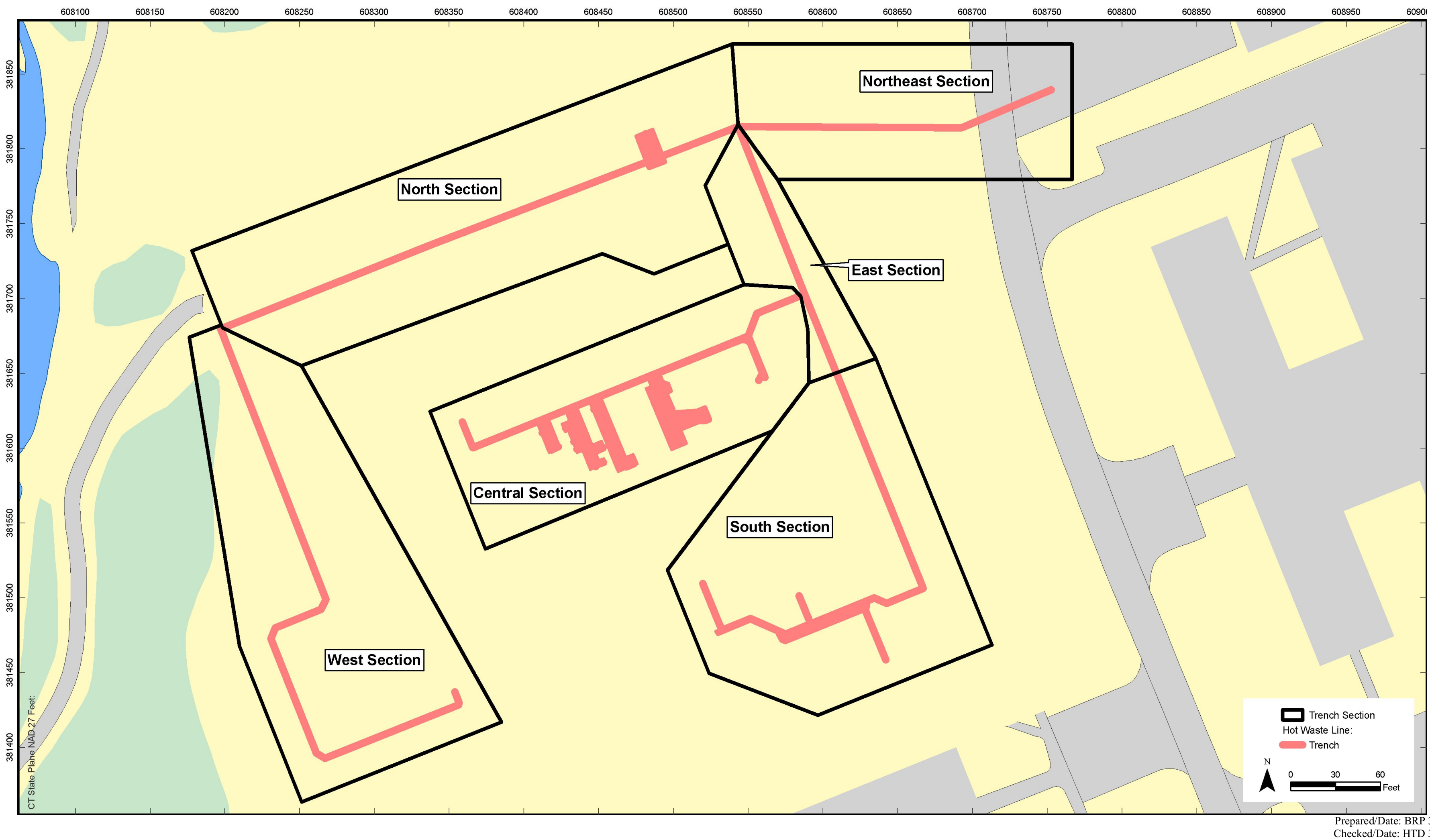

Figure 4.2: Hot Waste Line Trench Excavation Survey - Overview

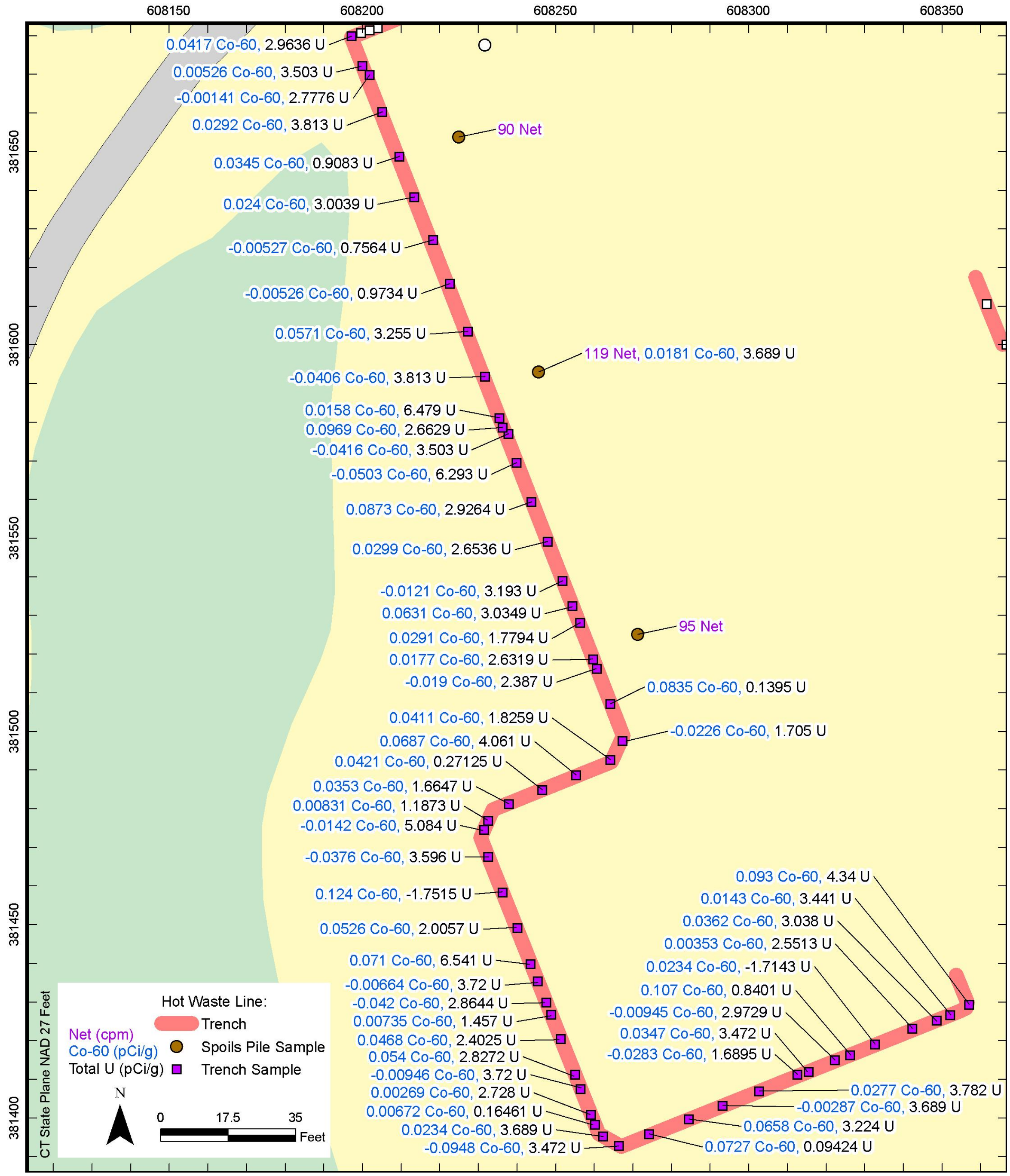

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Figure 4.3: Hot Waste Line Trench Excavation Survey – West Section

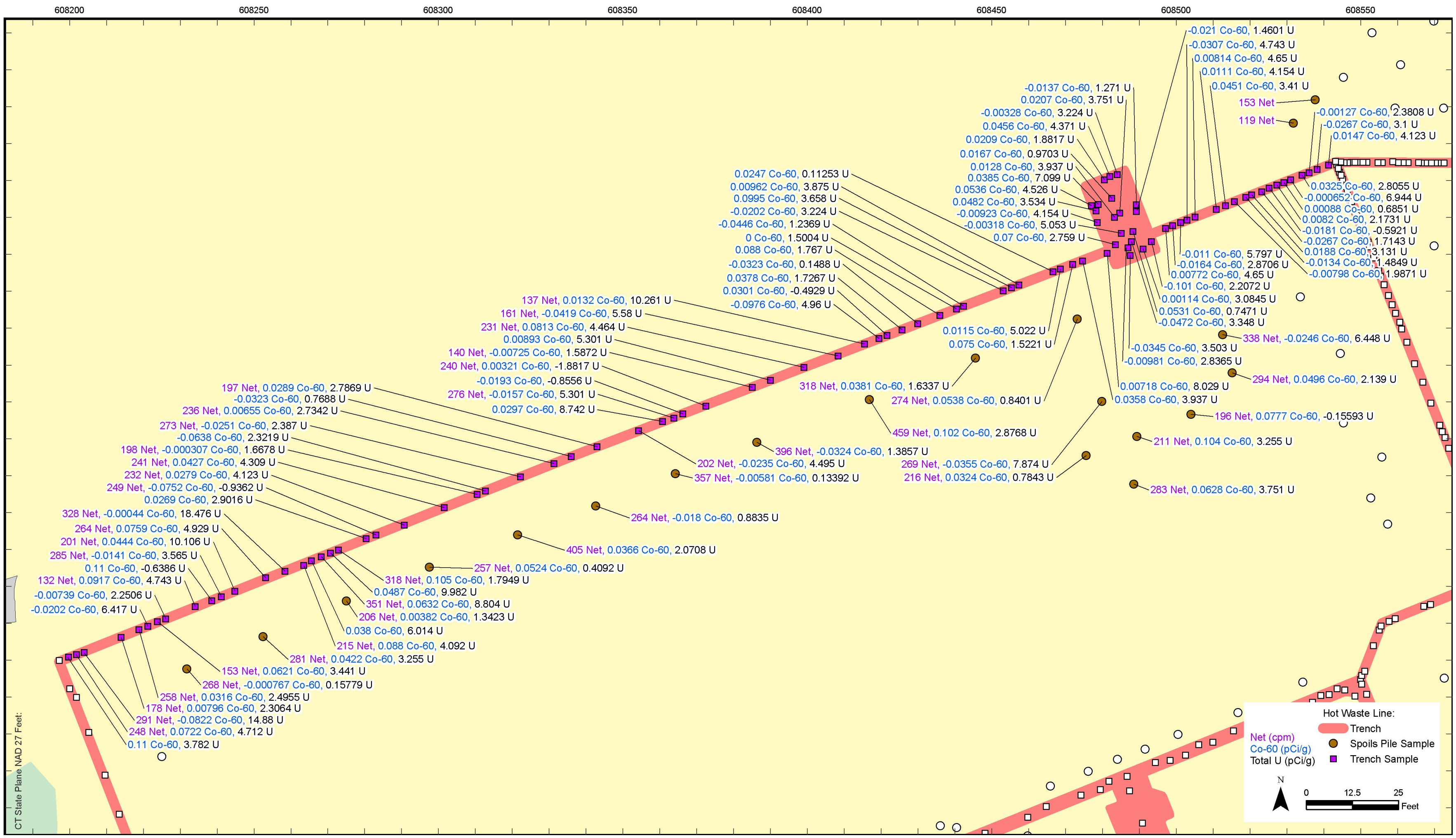

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Figure 4.4: Hot Waste Line Trench Excavation Survey – North Section

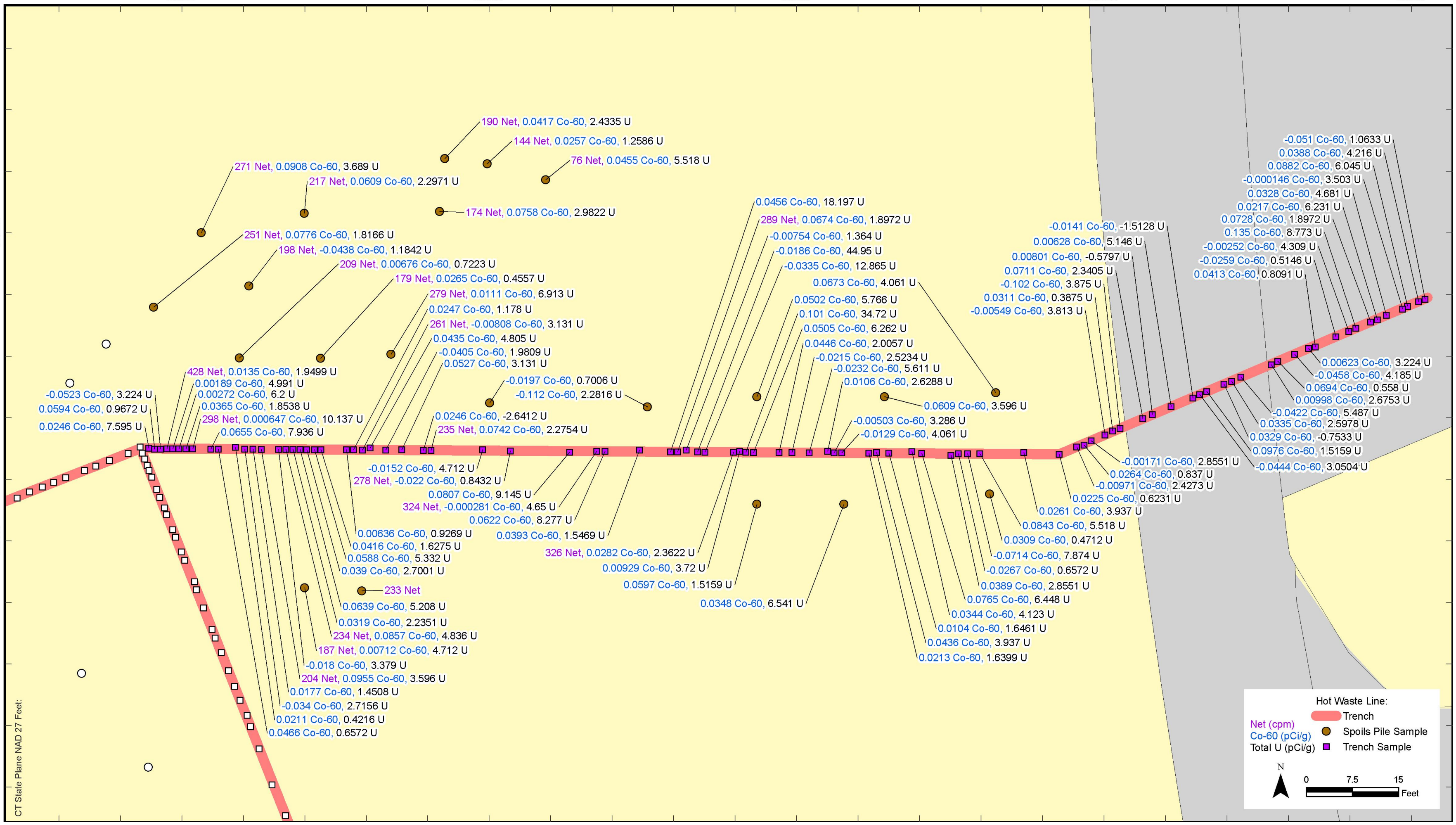

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Figure 4.5: Hot Waste Line Trench Excavation Survey – Northeast Section

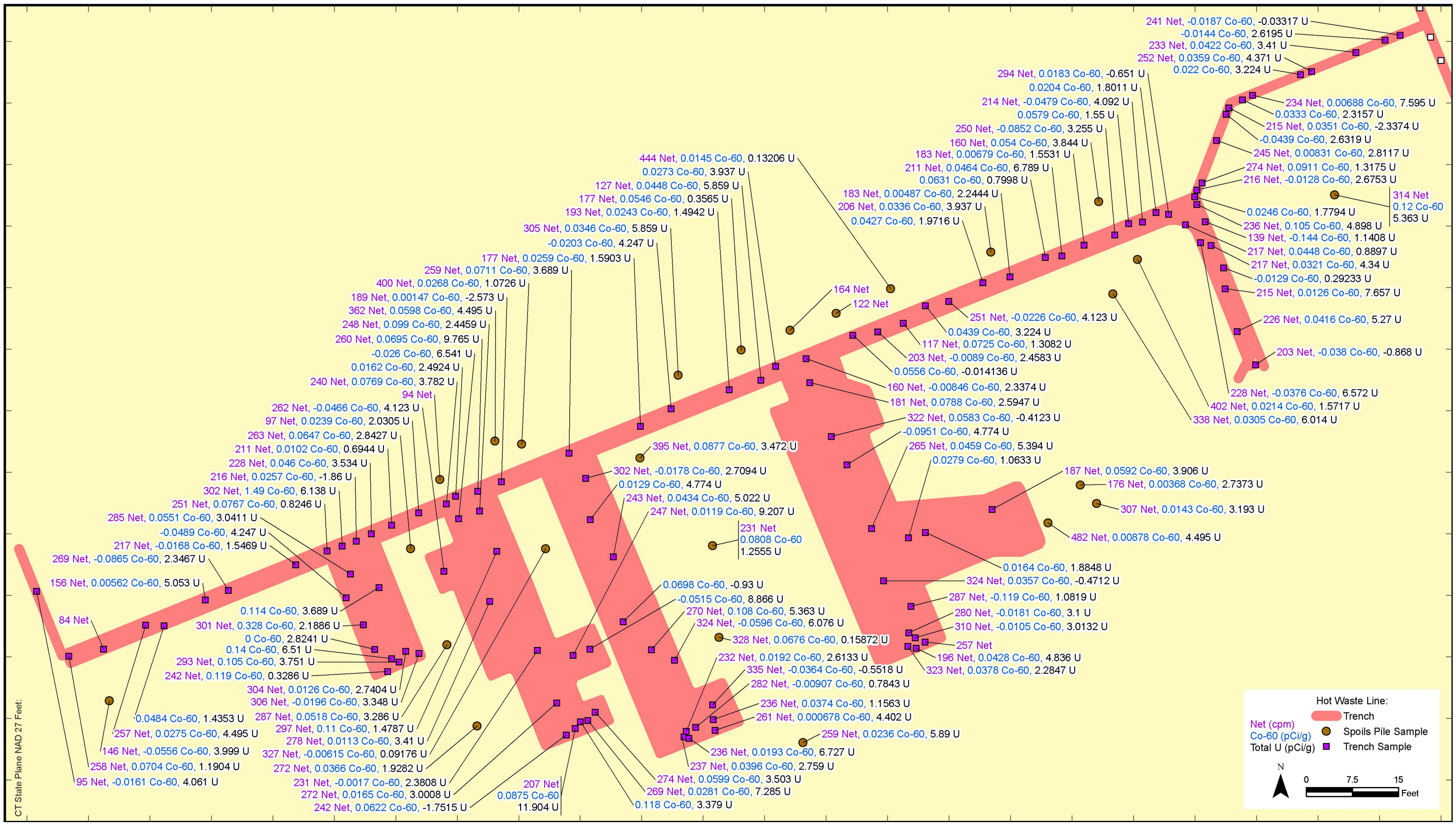

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Figure 4.6: Hot Waste Line Trench Excavation Survey – Central Section

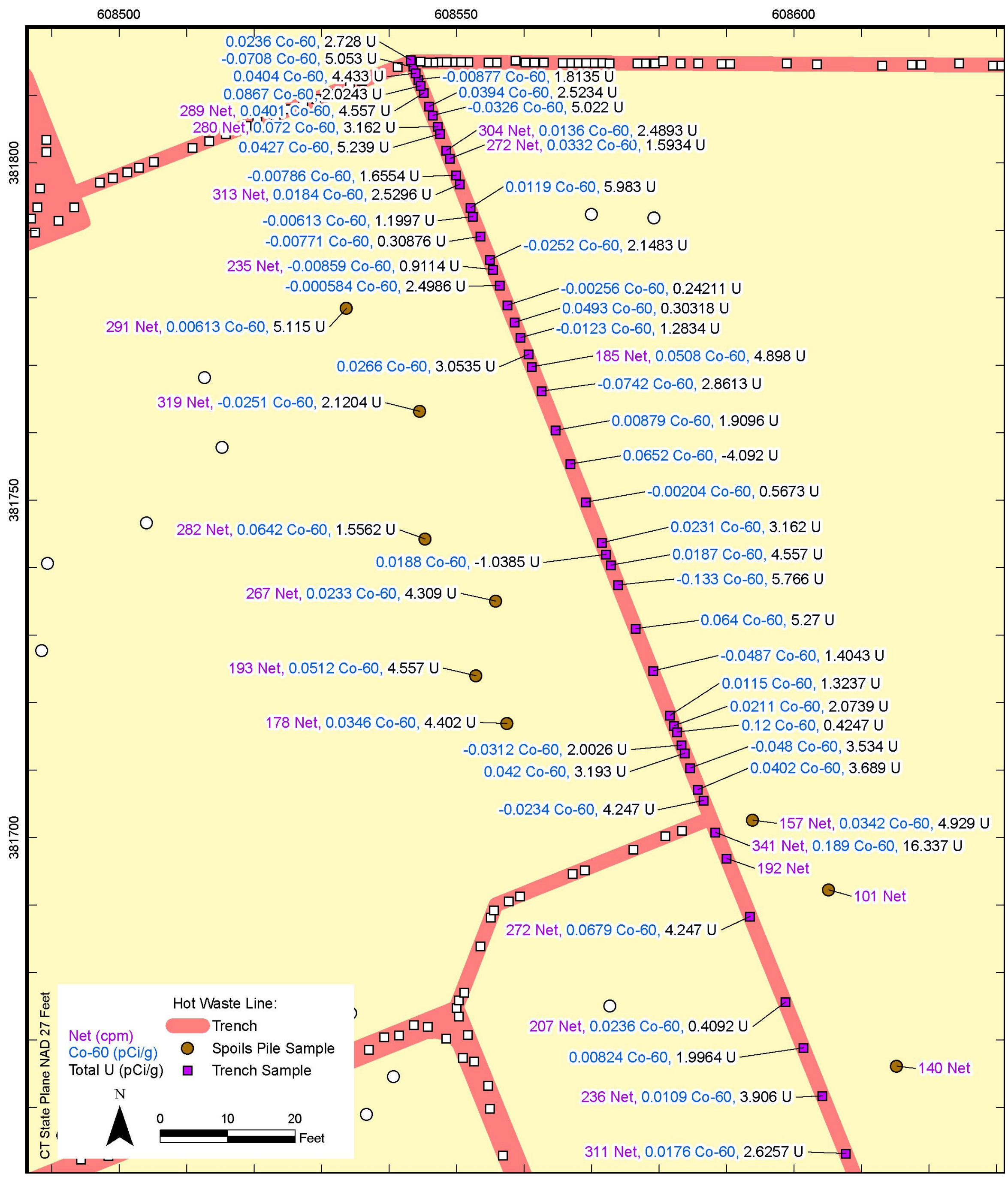

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Figure 4.7: Hot Waste Line Trench Excavation Survey – East Section

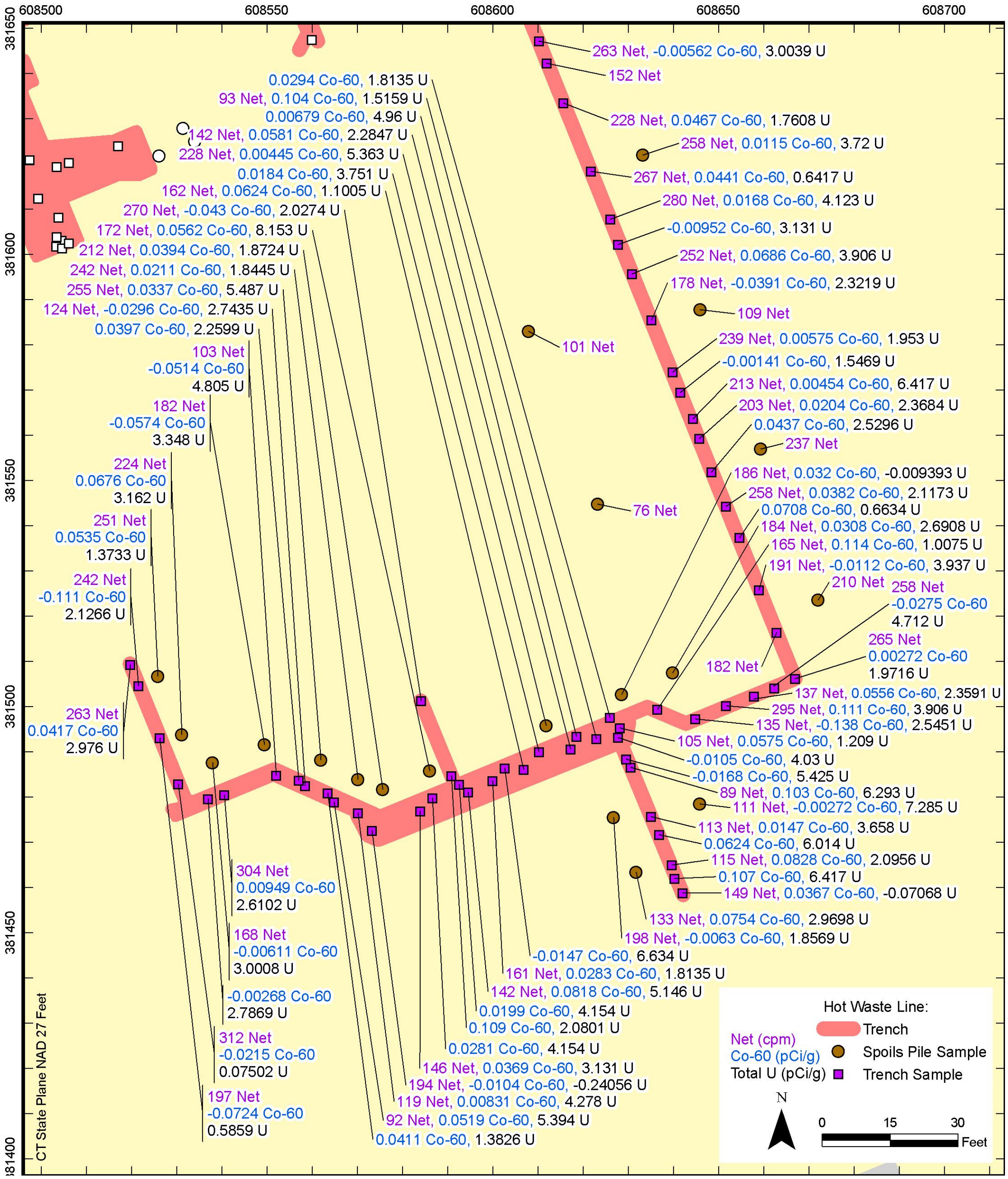

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Figure 4.8: Hot Waste Line Trench Excavation Survey – South Section

4.3.2 Industrial Waste Line

There was a large amount of industrial waste line removed from the Building Complexes 5 and 6A. The industrial waste line trenches have been subdivided into two sections in order to present the volumetric sampling results as shown in Figure 4.9 (next page). Following the overview figure are the industrial waste line trench excavation and the volumetric sampling results of the trench and spoils piles for the west section (Figure 4.10) and the east section (Figure 4.11). Survey data and additional information is provided in Appendix K.

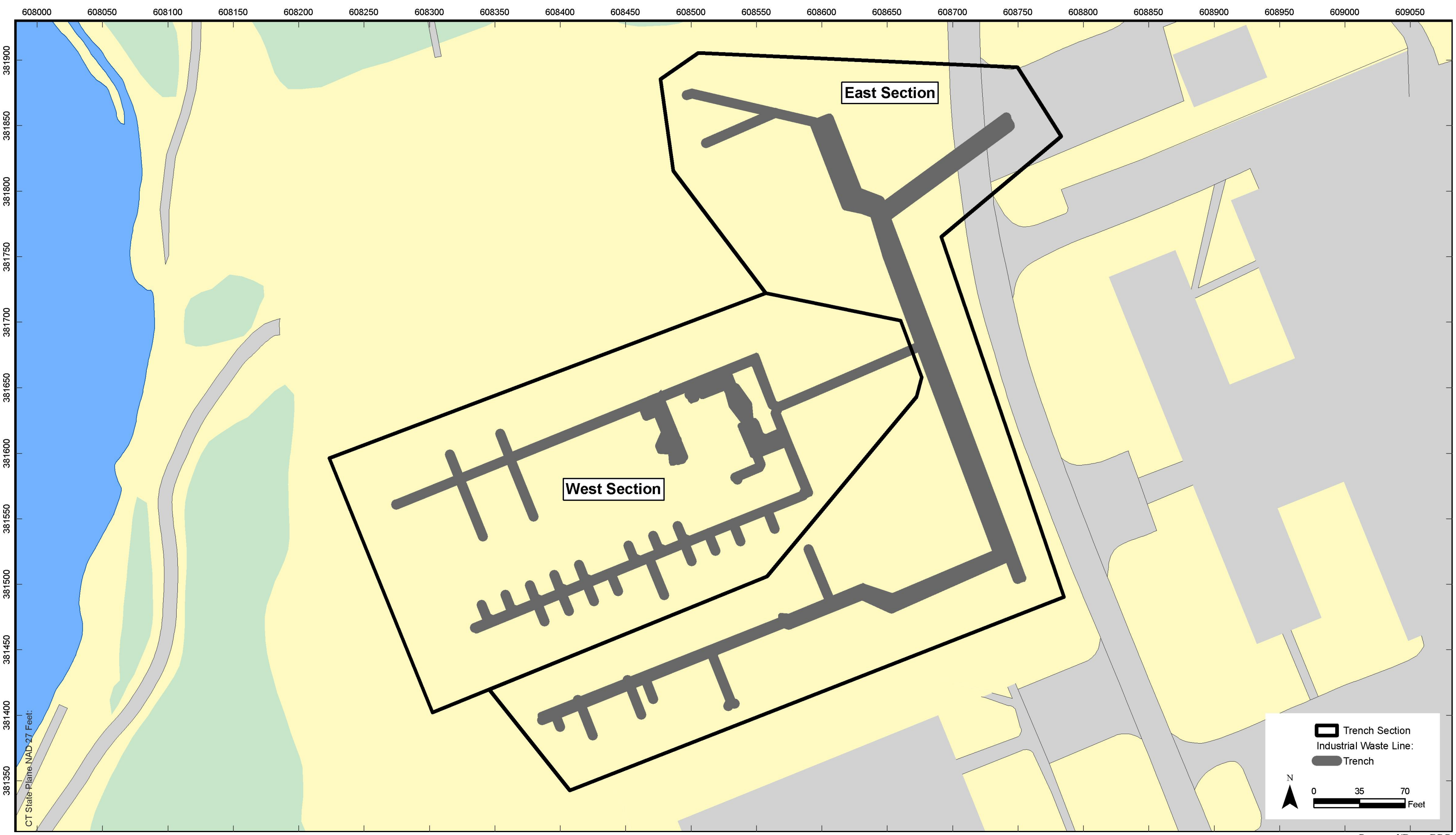

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Figure 4.9: Industrial Waste Line Trench Excavation Survey - Overview

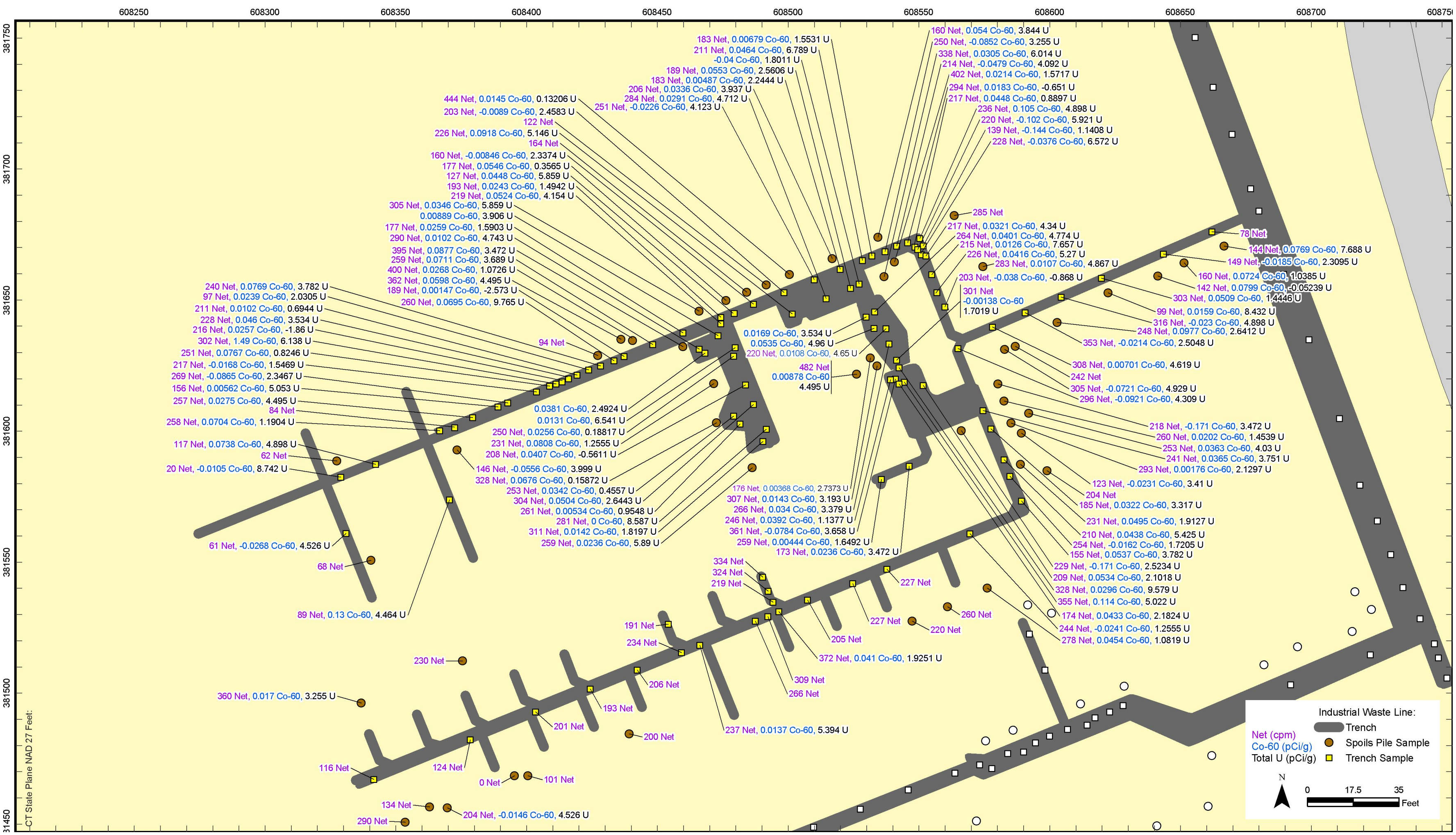

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Figure 4.10: Industrial Waste Line Trench Excavation Survey – West Section

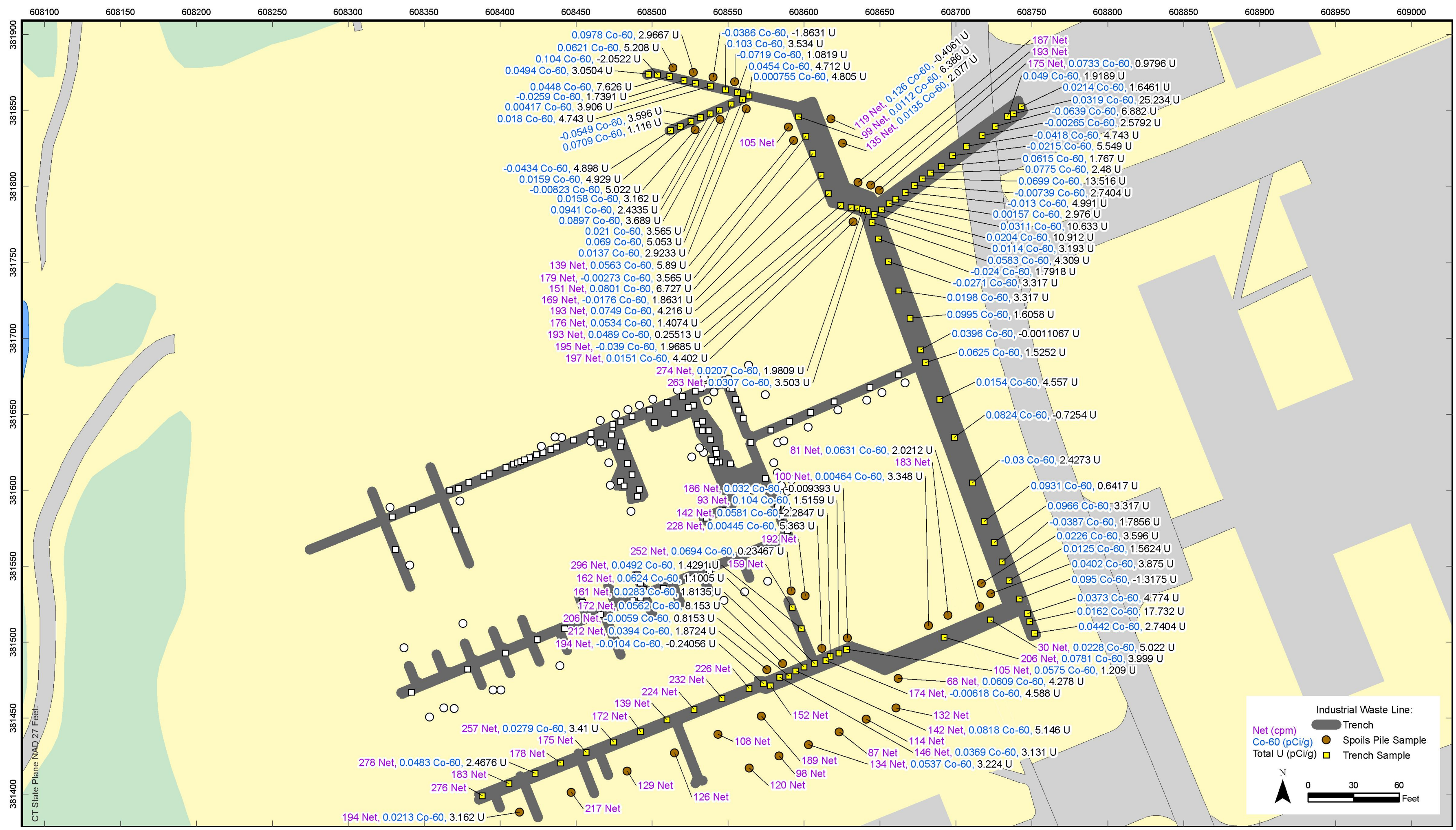

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Figure 4.11: Industrial Waste Line Trench Excavation Survey – East Section

4.3.3 Sanitary Sewer Waste Line

Figure 4.12 (next page) depicts the sanitary sewer waste line trench excavation and the volumetric sampling results of the Building Complexes 5 and 6A trench and spoils piles. Survey data and additional information is provided in Appendix K.

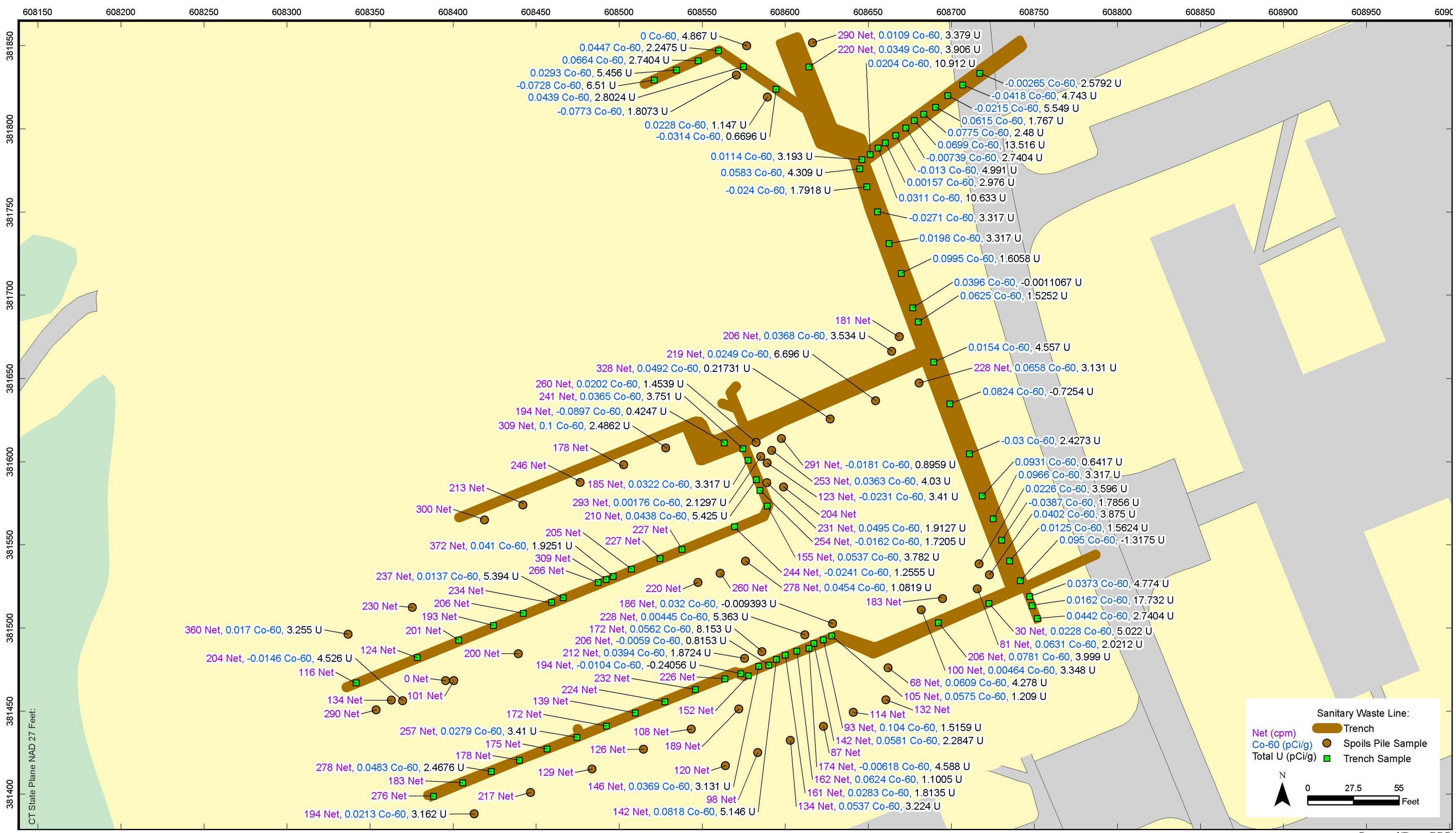

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Figure 4.12: Sanitary Sewer Waste Line Trench Excavation Survey

4.3.4 Storm Water Drain

Figure 4.13 (next page) depicts the storm water drain excavation and the volumetric sampling results of the Building Complexes 5 and 6A trench and spoils piles. Survey data and additional information is provided in Appendix K.

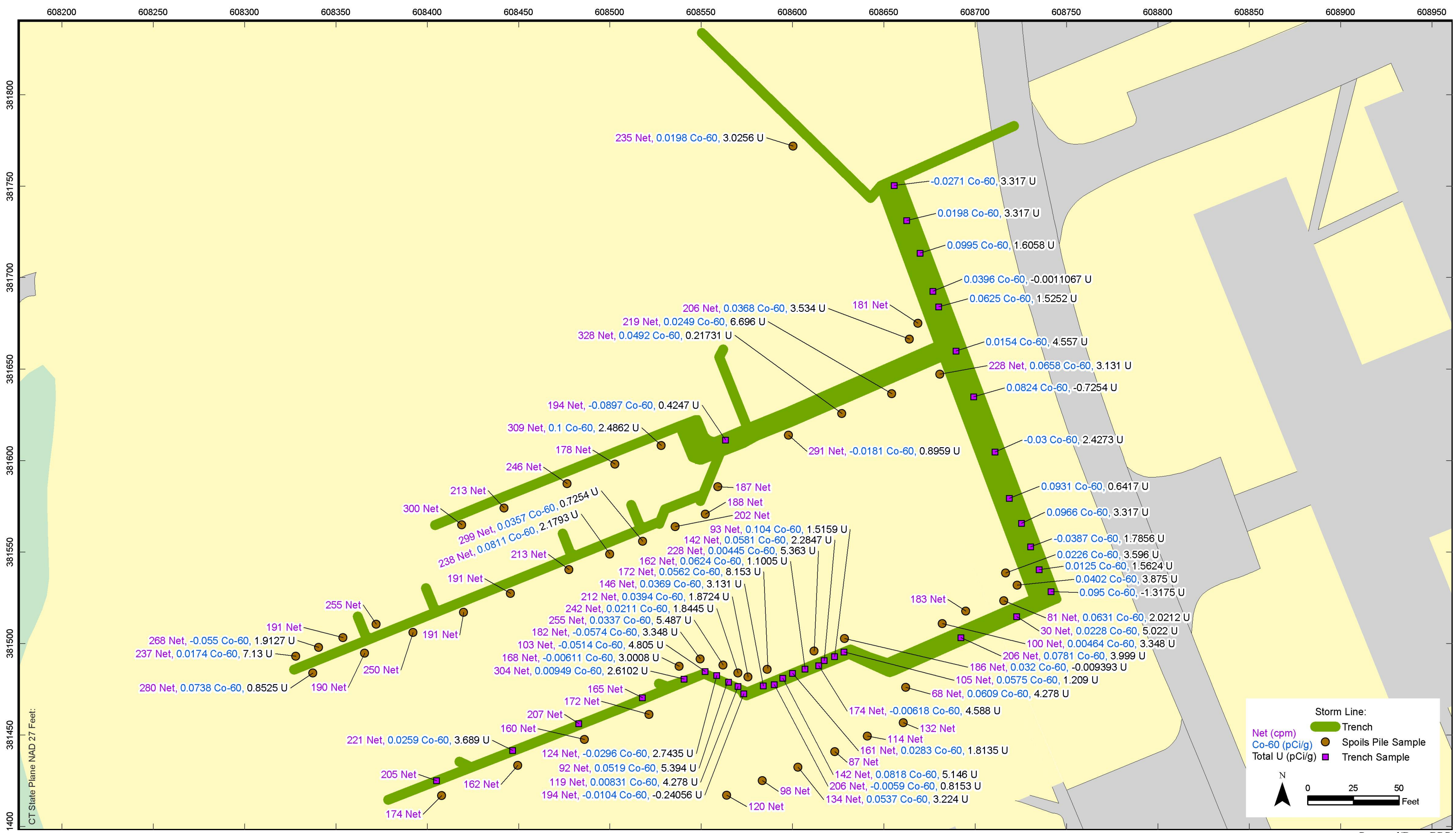

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Figure 4.13: Storm Water Drain Trench Excavation Survey

4.4 TRENCH SURVEY SUMMARY

Statistical testing was performed on trench bottom and spoil pile volumetric soil samples that were analyzed on the Site's HPGe gamma spectroscopy systems. Because none of the volumetric trench sample results exceeded the threshold value, the samples selected for analysis on the on-site HPGe system were chosen randomly. Review of the HPGe trench sample results showed that sample activities are significantly less than the DCGL for either Co-60 or total uranium. Because the sample populations have similar characteristics, regrouping of the segregated sample populations was allowed for statistical testing.

An overview of the statistical summary of the Building Complexes 5 and 6A trench area analytical results data is presented in Table 4.1 for all trench areas. Descriptive statistics, histograms, and probability plots for trench soil sample analytical results for uranium and Co-60 are presented in Appendix K.

Table 4.1: Summary Statistics, Trench Areas

Statistic	Total U	Co-60
Number of Measurements	719	719
Arithmetic Mean	3.346	0.0238
Standard Deviation	3.305	0.0729
Standard Error of the Mean	0.123	0.0027
Coefficient of Variation	0.988	3.06
Geometric Mean	2.69	0.0324
Maximum	45	1.49
Median	3	0.0228
Minimum	-4.1	-0.51
Range	49.1	1.661
UCL_{95} (median)	3.2	0.0265
LCL_{95} (median)	2.7	0.0187

Prepared/Date: ECS 3/13/06
Checked/Date: HTD 3/13/06

5.0 ANALYSIS OF RESULTS FOR COMPLIANCE

As part of the data quality objective process specified in MARSSIM (NRC, 2000) and other environmental remediation and compliance guidance (EPA, 2000), the *"decision rule"* provides the objective basis for determining whether survey units meet the established criteria for release from radiological controls without restriction. The decision rules, identified below, specify conditions, based on final radiological status survey results, which must be met to enable release of the site from radiological controls.

5.1 DECISION RULES

IF the evaluation of the Final Status Survey data indicates that:

- All volumetric soil sample measurement results are less than the $DCGL_W$ (5 pCi/g Co-60 and 557 pCi/g Total U); **AND**
- The unity rule is met if both radionuclides are present in a single sample location; **AND**
- There are no areas having locally elevated concentrations of residual radioactivity in soil greater than the $DCGL_W$; **AND**
- The cost benefit analysis indicates that residual radioactivity in soils at the Site has been reduced to concentrations that are ALARA:

THEN conclude that the soil survey unit meets the criteria for release from radiological controls without restriction.

An ALARA analysis was performed as part of the DP that was in agreement with NRC guidance provided in NUREG-1727. Both analyses show that shipping soil to a low-level waste disposal facility is not cost effective for unrestricted release. Therefore by demonstrating that the rest of the decision criteria have been met also demonstrates that the level of residual radioactivity is ALARA without taking additional remediation action.

These decision rules, having been derived from the dose-based radiological criteria for unrestricted release, will ensure that residual radioactivity in soils on the Site will not pose an unacceptable radiological risk to humans under any reasonable and foreseeable future use or occupancy.

5.2 FIELD SURVEY AND SAMPLING RESULTS COMPARED TO THE DCGLS

The compliance comparisons provide the risk managers and decision-makers with the quantitative information necessary to decide whether the Site can be released from radiological controls without restriction. In addition to the 95% upper confidence limit (UCL_{95}) estimate of the median, several additional metrics (e.g. arithmetic mean, maximum, etc.) are provided to offer risk managers and decision-makers additional insight regarding the magnitude of compliance or non-compliance.

Compliance comparisons for Co-60 and uranium soil survey units are presented in Table 5.1. Because the DCGL was developed for total uranium (U-234, U-235, and U-238) and the laboratory analytical results are reported only for the U-235 isotope, the results were multiplied by a factor of 31 as described previously in Section 2.

Comparisons are made using measurements not corrected for background, providing the risk managers and decision-makers additional depth and insight into the magnitude by which the levels of residual radioactivity compare to the DCGLs.

Table 5.1: Compliance Comparison of Soil Metrics

		Survey Unit								
		CE-FSS-05-01	CE-FSS-05-02	CE-FSS-05-03	CE-FSS-05-04	CE-FSS-05-05	CE-FSS-05-06	CE-FSS-06-01	CE-FSS-06-02	CE-FSS-06-03
Unity	Power of Sign Test	~1	~1	~1	~1	~1	~1	~1	~1	~1
Total U	Median	3.6	2.5	2.65	2.45	2.6	3.8	2.8	2.6	1.55
	UCL_{95} of Median	4.6	5	3.6	3.7	2.7	4.8	4	3.6	3.4
	Arithmetic Mean	3.2	3.2	2.6	2.5	2.6	3.5	2.7	2.99	1.7
	Geometric Mean	3.16	2.97	2.5	2.03	2.05	3.43	2.62	2.47	1.38
	Maximum	8.6	10.5	6.1	7	5.5	9.2	5.1	7.2	4.7
Co-60	Median	0.015	0.0321	0.0043	-0.0027	0	0.0212	0.0408	0.0253	0.0448
	UCL_{95} of Median	0.0442	0.0982	0.0294	0.0563	0.0434	0.0492	0.0598	0.0692	0.0637
	Arithmetic Mean	0.016	0.023	0.0024	0.0050	0.0098	0.014	0.033	0.0207	0.02
	Geometric Mean	0.037	0.062	0.036	0.025	0.049	0.036	0.036	0.0455	0.0409
	Maximum	0.118	0.15	0.0942	0.0943	0.0645	0.0937	0.0637	0.0908	0.106

1) No measure of the soil radioactivity in any survey unit exceeds the applicable criterion.
2) Comparison of the median from each survey unit indicates that in no case were the $DCGL_W$s exceeded. More importantly, the significance of the Sign-Test results are all greater than 95% [(1-'p') *100 = % confidence]. Thus, it is assured, with at least 95% confidence, that the median residual soil radioactivity concentrations do not exceed the $DCGL_W$s. Note in the Compliance Test Statistics Report that the 'p' values for these tests are far below 0.05 and, in many cases, they are reported as 0.0000.
3) Comparison of the UCL_{95} of the median from each survey unit indicates that in no case were the $DCGL_W$s exceeded. The highest total U UCL_{95} estimate of the median, 5 pCi/g, is less than the $DCGL_W$ by a factor of more than 100, and the highest Co-60 UCL_{95} estimate of the median, 0.0982 pCi/g, is less than the $DCGL_W$ by a factor of more than 50. Thus, a wide margin of safety between the acceptable and actual concentration of residual radioactivity exists.
4) Comparison of the maximum total U and Co-60 from each survey unit to 557 pCi/g (Total U DCGL) or 5 pCi/g (Co-60 DCGL) indicates that in no instance was the DCGL exceeded.
5) Comparison of the arithmetic and geometric means from each survey unit indicates that in no case are these central tendency indicators even approaching the $DCGL_W$s.

Prepared/Date: ECS 3/16/06
Checked/Date: HTD 3/16/06

5.3 TRENCH SURVEY AND SAMPLING RESULTS COMPARED TO THE DCGLS

The compliance comparisons provide the risk managers and decision-makers with the quantitative information necessary to decide whether the trenched areas can be released from radiological controls without restriction. In addition to the UCL_{95} estimate of the median, several additional metrics (e.g. arithmetic mean, maximum, etc.) are provided to offer risk managers and decision-makers additional insight regarding the magnitude of compliance or non-compliance.

Compliance comparisons for Co-60 and uranium in trench soil areas are presented in Table 5.2. Because the DCGL was developed for total uranium (U-234, U-235, and U-238) and the laboratory analytical results are reported only for the U-235 isotope, the results were multiplied by a factor of 31 as described previously in Section 2.

Comparisons are made using measurements that are not corrected for background, providing the risk managers and decision-makers additional depth and insight into the magnitude by which the levels of residual radioactivity compare to the DCGLs.

Table 5.2: Compliance Comparison of Trench Soil Metrics

Metric		Trench Results
Unity	Power of Sign Test	~1
Total U	Median	3
	UCL_{95} of Median	3.2
	Arithmetic Mean	3.35
	Geometric Mean	2.69
	Maximum	45
Co-60	Median	0.023
	UCL_{95} of Median	0.027
	Arithmetic Mean	0.024
	Geometric Mean	0.032
	Maximum	1.49

1) No measure of the soil radioactivity in any survey unit exceeds the applicable criterion.
2) Comparison of the median indicates that in no case were the $DCGL_W$s exceeded. More importantly, the significance of the Sign-Test results are all greater than 95% [(1-'p') *100 = % confidence]. Thus, it is assured, with at least 95% confidence, that the median residual soil radioactivity concentrations do not exceed the $DCGL_W$s. Note in the Compliance Test Statistics Report that the 'p' values for these tests are far below 0.05 and, in many cases, they are reported as 0.0000.
3) Comparison of the UCL_{95} of the median indicates that in no case were the $DCGL_W$s exceeded. The total U UCL_{95} estimate of the median, 3.2 pCi/g, is less than the $DCGL_W$ by a factor of more than 50, and the Co-60 UCL_{95} estimate of the median, 0.027 pCi/g, is also less than the $DCGL_W$ by a factor of more than 185. Thus, a wide margin of safety between the acceptable and actual concentration of residual radioactivity exists.
4) Comparison of the maximum total U and Co-60 to 557 pCi/g (Total U DCGL) or 5 pCi/g (Co-60 DCGL) indicates that in no instance was the DCGL exceeded.
5) Comparison of the arithmetic and geometric means indicates that in no case are these central tendency indicators even approaching the $DCGL_W$s.

Prepared/Date: ECS 3/16/06
Checked/Date: HTD 3/16/06

5.4 COMPLIANCE SUMMARY

The radiological final status survey demonstrates that the soils meet all of the quantitative compliance decision rules that must be met to qualify for release from radiological controls, without restriction. This conclusion is summarized below.

5.4.1 Building Complexes 5 and 6A Soils

The average and median uranium and Co-60 concentrations in soils (Survey Units 05-01 through 05-06 and 06-01 through 06-03) are well below the $DCGL_W$ value of 557 pCi/g for total uranium and the $DCGL_W$ of 5.0 pCi/g for Co-60.

The median uranium and Co-60 concentrations in soils have been demonstrated to be less than the $DCGL_W$ value of 557 pCi/g for U-235 and 5.0 pCi/g for Co-60, with at least 95% statistical confidence. The statistical test used to make this comparison was the Sign test, recommended by MARSSIM (NRC, 2000). Observing that in no case did a UCL_{95} of the median closely approach the DCGL further evidences this conclusion.

No single soil sample was identified as having uranium and Co-60 activity greater than 10.5 pCi/g and 0.118 pCi/g respectively, significantly below the $DCGL_W$ value of 557 pCi/g for uranium and 5.0 pCi/g for Co-60. No locally elevated concentrations of residual radioactivity were identified above the volumetric or walkover (scan) investigation levels.

5.4.1.1 Sample Size and Statistical Power – Retrospective Power Curves

The retrospective power curve was calculated using the actual sample size obtained and the sample standard deviation measured for the population. The gray region boundaries represent the concentrations between which there is insufficient power at the prescribed alpha and beta error rate, given the sample size obtained and the variability observed in the data set.

The Retrospective Power Curves for each survey unit are provided in the appendices, and illustrate the power of the Sign Test to conclude that the null hypothesis (that the volumetric radioactivity in soil exceeds the allowable radioactivity concentration) should be rejected for all soils.

5.4.2 Trench Soils

The average and median uranium and Co-60 concentrations in trench soils are well below the $DCGL_W$ value of 557 pCi/g for total uranium and the $DCGL_W$ of 5.0 pCi/g for Co-60.

The median uranium and Co-60 concentrations in trench soils have been demonstrated to be less than the $DCGL_W$ value of 557 pCi/g for U-235 and 5.0 pCi/g for Co-60, with at least 95% statistical confidence. The statistical test used to make this comparison was the Sign test, recommended by MARSSIM (NRC, 2000). Observing that in no case did a UCL_{95} of the median closely approach the DCGL further evidences this conclusion.

No single soil sample was identified as having uranium and Co-60 activity greater than 45 pCi/g and 1.49 pCi/g respectively, significantly below the $DCGL_W$ value of 557 pCi/g for uranium and 5.0 pCi/g for Co-60. No locally elevated concentrations of residual radioactivity were identified above the volumetric or scan investigation levels.

5.4.2.1 Sample Size and Statistical Power – Retrospective Power Curves

The retrospective power curve was calculated using the actual sample size obtained and the sample standard deviation measured for the population. The gray region boundaries represent the concentrations between which there is insufficient power at the prescribed alpha and beta error rate, given the sample size obtained and the variability observed in the data set.

The retrospective power curve for trench soils is presented in Appendix K, and illustrates the power of the Sign Test to conclude that the null hypothesis (that the volumetric radioactivity in trench soil exceeds the allowable radioactivity concentration) should be rejected for trench soils.

6.0 QUALITY CONTROL AND DATA QUALITY ASSESSMENT

An important aspect of any survey or sampling evolution is the effort made to assure the quality of data collected. It is critical to assure the quality of the data through quality checks and controls, calibrations, and training. The purpose of data quality assessment (DQA) is to evaluate the data collected from the field in light of its intended use in decision making. Decision makers should obtain an understanding of the verity of the data used in the FSS from reading this section.

Quality checks and controls were designed into the FSS to ensure adequate data quality. QC measurements were designed to provide a means of assessing the quality of the data set as a whole and demonstrate that measurement results had the required precision and were sufficiently free of errors to accurately represent the Building Complexes 5 and 6A. The DQA uses guidance from MARSSIM and professional judgment.

6.1 QUALITY CONTROL

The goal of QC is to identify and implement sampling and analytical methodologies that limit the introduction of error into analytical data. During sampling and survey activities at the site, controls were implemented to ensure sufficient data of adequate quality and usability was collected for confirming that the project's release levels were met. These controls also ensured that data was verified authentic, was appropriately documented, and is technically defensible. QC was achieved through three primary approaches: data management, sample custody, and QC measurements.

6.1.1 Data Management

Volumetric sample collection and field measurement results were recorded both electronically (GPS logging of sample locations) and through hard copy (radiological survey forms, maps, and chain-of-custody forms). Volumetric sample laboratory analytical result data were recorded electronically by the Genie software program. Records of field-generated data were reviewed by MACTEC supervisory personnel and the ABB Site RSO. Electronic copies of original electronic data sets are preserved on a retrievable data storage device. No data reduction, filtering, or manipulation was performed on the original electronic versions of data sets.

Record copies of surveys, sampling, and analytical data (and supporting data) are provided in the appendices.

6.1.2 Sample Custody

Sample quality, related to sample collection, was controlled through the use of trained personnel implementing approved operating procedures. Methods employed in operating procedures took into account the need to prevent sample contamination through the use of dedicated equipment, decontamination of equipment between sample collection, and isolation of samples in discrete sample containers.

FSS sample custody and control was accomplished by:

- Assigning a unique sample identification number to each sample collected in accordance with the FSSP,
- Recording the date, time, sample type, and location and linking that information with the sample identification number and the required analysis,
- Requiring that sampling personnel, possessing the physical samples, be accountable for and be the Chain-of-Custody for the sample, and
- Implementing a Chain-of-Custody protocol for sample materials processed on-site as well as those samples sent for analysis at an off-site laboratory.

Chain-of-Custody records for both volumetric soil samples staying physically on-site and those samples that were shipped to General Environmental Laboratories, Inc. (GEL) for off-site analysis are provided in the appendices.

6.1.3 Quality Control Measurements

A significant portion of the data comes from in situ field measurements using conventional health physics techniques and practices and from volumetric media samples measured by HPGe measurement methods. Both require additional steps in order to ensure accuracy of the sampling techniques and analysis methodologies.

6.1.3.1 Volumetric Replicate Samples

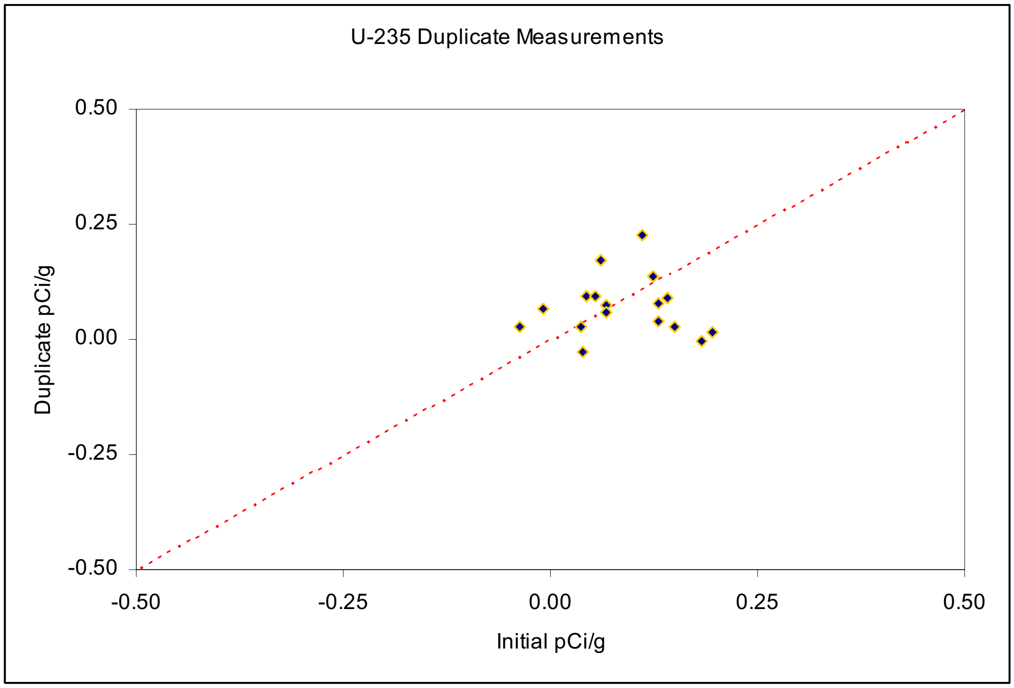
The prescribed QC for volumetric media sampling activities consists of duplicate (split) sampling. Duplicate sampling provides the means to assess the consistency and precision of the overall sampling and analytical system. Field duplicate samples were prepared in the field by the sampling team at a frequency of 5 percent (1:20) for the sample population, and were submitted to the on-site gamma spectroscopy system for analysis as duplicate samples. While not all survey units had duplicate sample collected, seven duplicate samples were collected from an overall sample population of 139 volumetric samples, roughly equating to a 1:20 sampling frequency. The results of the field duplicate sample analyses were evaluated in comparison to the results obtained from the initial sample. Each of the field duplicate sample results was within the expected tolerance for the analysis, providing additional evidence that the sample preparation, extraction, and measurement processes were accurate (Table 6.1).

Table 6.1: Duplicate Sample Measurement Results

Soil Building Complexes 5 and 6A						
Sample ID	**Co-60**			**U-235**		
	Activity	**Uncert**	**MDC**	**Activity**	**Uncert**	**MDC**
SS-FSS-05-001-00	1.57E-01	1.57E-01	4.70E-01	3.76E-02	1.89E-01	3.41E-01
SS-FSS-05-001-00-DUP	-2.94E-02	2.09E-01	4.22E-01	1.30E-01	1.70E-01	3.31E-01
SS-FSS-05-024-00	7.65E-03	1.07E-01	1.98E-01	1.73E-01	1.41E-01	2.88E-01
SS-FSS-05-024-00-DUP	1.05E-01	9.43E-02	2.76E-01	6.12E-02	1.54E-01	2.94E-01
SS-FSS-05-035-00	2.09E-03	1.08E-01	2.26E-01	2.55E-02	1.39E-01	2.59E-01
SS-FSS-05-035-00-DUP	1.95E-02	3.91E-02	1.44E-01	3.69E-02	1.34E-01	2.55E-01
SS-FSS-05-054-00	4.22E-02	1.24E-01	2.82E-01	2.28E-01	1.49E-01	3.08E-01
SS-FSS-05-054-00-DUP	4.62E-02	1.10E-01	2.62E-01	1.10E-01	1.64E-01	3.13E-01
SS-FSS-05-068-00	6.71E-02	1.59E-01	3.36E-01	2.59E-02	1.48E-01	2.79E-01
SS-FSS-05-068-00-DUP	-7.64E-02	1.29E-01	1.45E-01	-3.67E-02	1.42E-01	2.56E-01
SS-FSS-05-090-00	-1.80E-02	1.07E-01	2.5E-01	9.54E-02	1.40E-01	2.67E-01
SS-FSS-05-090-00-DUP	7.49E-02	1.09E-01	2.67E-01	4.37E-02	1.35E-01	2.55E-01
SS-FSS-6A-04	9.35E-02	1.06E-01	2.51E-01	1.36E-01	1.42E-01	2.71E-01
SS-FSS-6A-04 DUP	-3.11E-03	9.97E-02	1.97E-01	1.24E-01	1.33E-01	2.56E-01
SS-FSS-6A-09	5.21E-02	7.87E-02	1.86E-01	8.91E-02	1.31E-01	2.46E-01
SS-FSS-6A-09 DUP	6.75E-02	1.14E-01	2.44E-01	1.42E-01	1.32E-01	2.55E-01
SS-FSS-6A-22	-4.75E-02	9.34E-02	1.63E-01	1.54E-02	1.41E-01	2.56E-01
SS-FSS-6A-22 DUP	1.70E-02	9.42E-02	2.03E-01	1.96E-01	1.25E-01	2.49E-01
SS-FSS-6A-33	8.44E-02	1.03E-01	2.40E-01	-4.47E-03	1.43E-01	2.56E-01
SS-FSS-6A-33 DUP	5.39E-02	9.49E-02	2.22E-01	1.82E-01	1.39E-01	2.72E-01
SS-FSS-6A-43	1.02E-01	1.05E-01	2.61E-01	9.57E-02	1.59E-01	2.99E-01
SS-FSS-6A-43 DUP	1.13E-02	1.19E-01	2.39E-01	5.37E-02	1.59E-01	2.92E-01
SS-FSS-6A-47	-3.50E-02	1.02E-01	1.74E-01	-2.64E-02	1.33E-01	2.37E-01
SS-FSS-6A-47 DUP	7.07E-02	1.04E-01	2.36E-01	3.81E-02	1.38E-01	2.55E-01

Prepared/Date: MPM 3/08/06
Checked/Date: HTD 3/08/06

The overall quality of the volumetric soil sample data is evident in the graphic presentation contained in Figure 6.1 (U-235) and Figure 6.2 (Co-60).



Prepared/Date: MPM 3/09/06
Checked/Date: HTD 3/09/06

Figure 6.1: U-235 Duplicate Measurement Result Comparisons

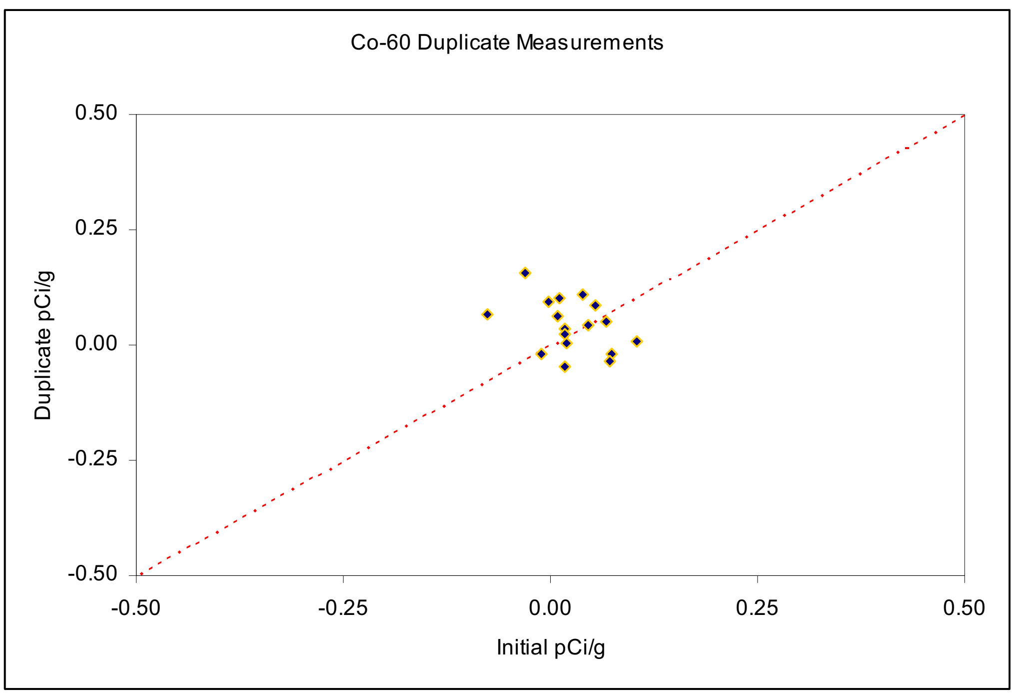


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Figure 6.2: Co-60 Duplicate Measurement Result Comparisons

6.1.3.2 Field Instrument Response Checks

The prescribed QC for radiological surveys (gamma walkover, static, or screening surveys) consists of survey instrument response checks. Daily or prior to initiating the surveys, the survey instruments were response checked to a known surface activity source. Survey Instrument Response check data sheets are provided in Appendix L.

The survey instruments used for the performance of the FSS were also used at the Site for other survey purposes and source response checks were performed for these instruments prior to and following the time during which FSS surveys where conducted.

A control chart for each instrument was created to evaluate the instruments' responses to the radioactive source over the sampling period time frame. No degradation of the instruments' response was observed during the performance of FSS. Control charts and supporting data for field instruments are provided in Appendix L.

6.1.3.3 Laboratory Instruments

The prescribed QC for laboratory instruments consists of instrument source checks, energy calibration checks, efficiency calibration checks, background checks, and replicate volumetric measurements performed on a percentage of the samples collected. The on-site HPGe system used in the analysis of volumetric soil media during FSS was controlled by Canberra's Genie System software. The software was used to perform the energy and efficiency calibration checks.

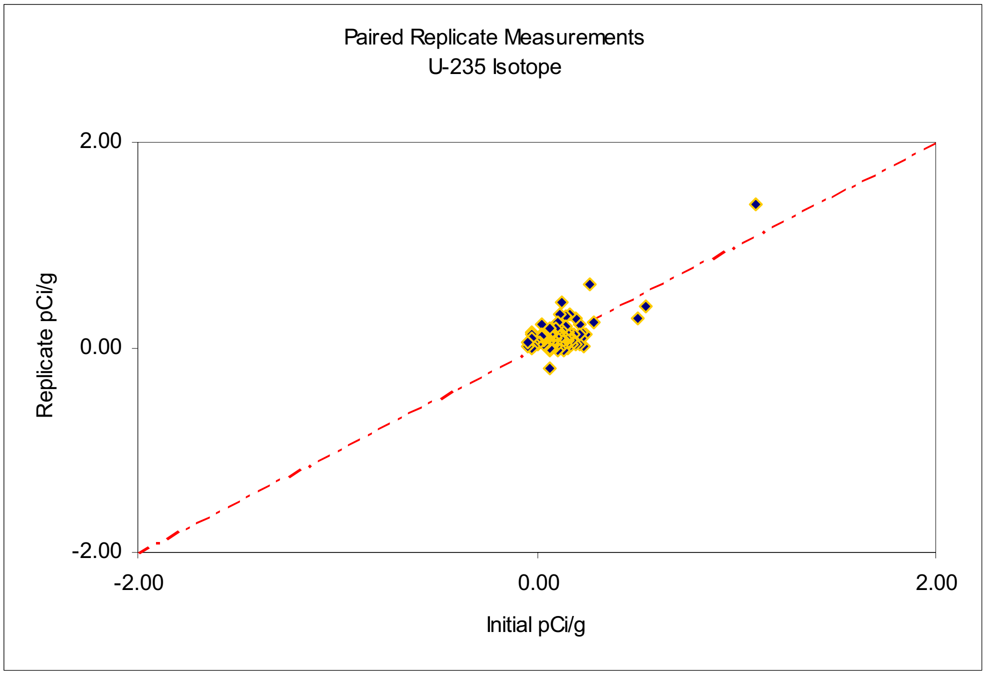
The QA checks preformed on the gamma spectroscopy system verify that the system parameters have not changed such that the energy and efficiency calibrations are still valid. This is accomplished by using a low-energy peak (59 keV) and a high-energy peak (1332 keV) from a calibration source to evaluate a set of three parameters for each peak. These parameters include peak centroid (indicate a problem with energy calibration), peak energy resolution (full width at half maximum [FWHM]) (indicate a problem with the energy shape calibration), and decay corrected activity (indicate a problem with the efficiency calibration). Control charts for these parameters, the energy calibration curve, the efficiency calibration curve, and other associated data are provided in Appendix M. Examination of this data concludes that the gamma spectroscopy system was functioning correctly during FSS.

Another QC method used to assess the potential error that might occur with laboratory measurements of volumetric soil media is to perform secondary measurements of the sample using independent, off-site, analytical equipment. Secondary counting of samples was performed by GEL. A total of 139 volumetric samples obtained from the Site during FSS activities for all commercial decommissioning areas were analyzed by the on-site gamma spectroscopy system and then sent to GEL for isotopic analysis by gamma spectroscopy (HPGe). Paired replicate measurement results for the common analyzed isotopes (Co-60 and U-235) are presented in Appendix M along with GEL Certificates of Analysis.

To assess the comparability between the initial and replicate measurements, a simple linear regression analysis was performed and is graphically presented in Figure 6.3 (U-235) and Figure 6.4 (Co-60) for sample activities near or at background activity values. Samples with activities greater than approximately 3 pCi/g, which were the minority of samples evaluated (12 of 139 total samples), were intentionally ignored in this particular evaluation. The separation of background activity samples from elevated activity samples was done to reduce the magnitude of the graphical X and Y axis scale. Tabular comparison of on-site to laboratory GEL analytical results are presented in Appendix M

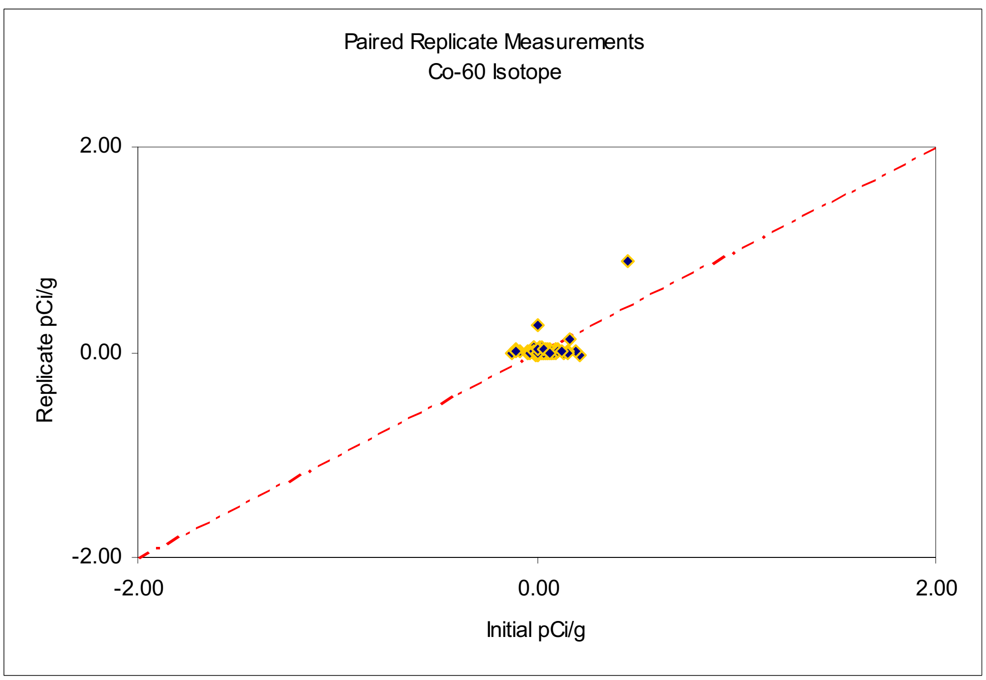
In addition to the regression analysis of the replicate data sets for the replicate measurements with activity < 3 pCi/g, two-sample comparison density traces of the data set (all sample activities) are presented in Figure 6.5 (for U-235) and Figure 6.6 (for Co-60). These figures graphically portray the virtually identical probability density functions of the initial and replicate data set populations and offer solid evidence that the analytical measurements made on the GEL HPGe system and the on-site HPGe system are similar. Thus, the figures serve as a good indicator of the measurement precision of the on-site HPGe analysis system when compared against the off-site laboratory gamma spectroscopy system.

Analytical quality control for samples submitted to GEL for analysis was specified by contractual agreement and were designed to ensure that the detection confidence levels were adequate to demonstrate compliance with the decision criterion for a given sample or sample set. An upper confidence level of 95% was specified (UCL_{95}).



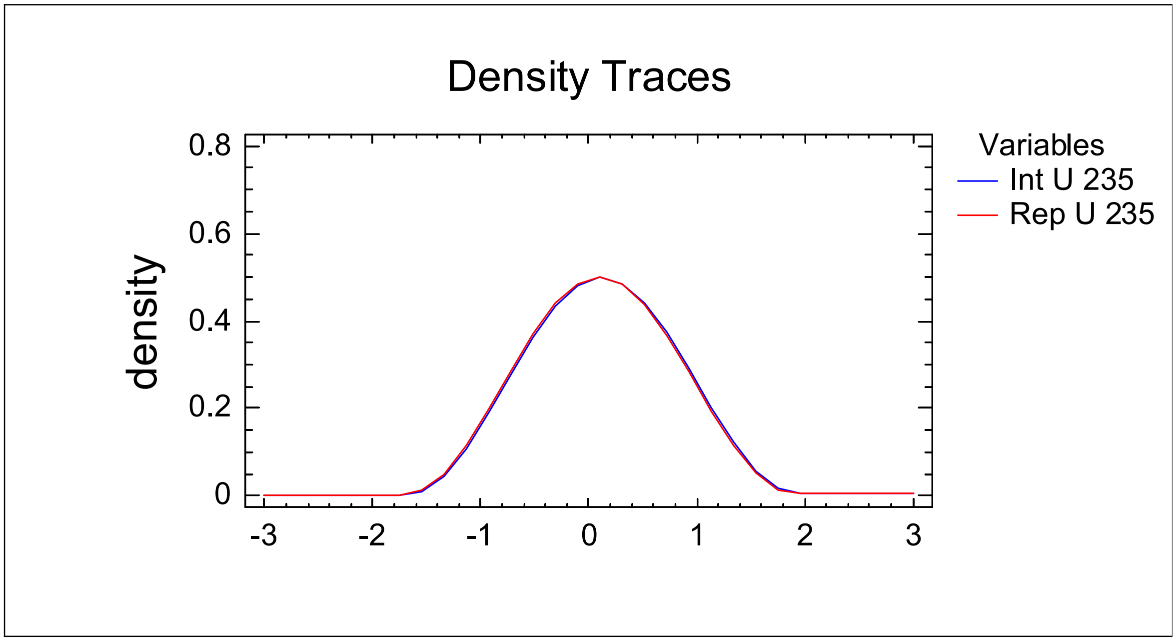
Prepared/Date: MPM 3/09/06
Checked/Date: HTD 3/09/06

Figure 6.3: U-235 Comparison Between Replicate Measurements < 3 pCi/g



Prepared/Date: MPM 3/09/06
Checked/Date: HTD 3/09/06

Figure 6.4: Co-60 Comparison Between Replicate Measurements < 3 pCi/g



Prepared/Date: MPM 3/09/06
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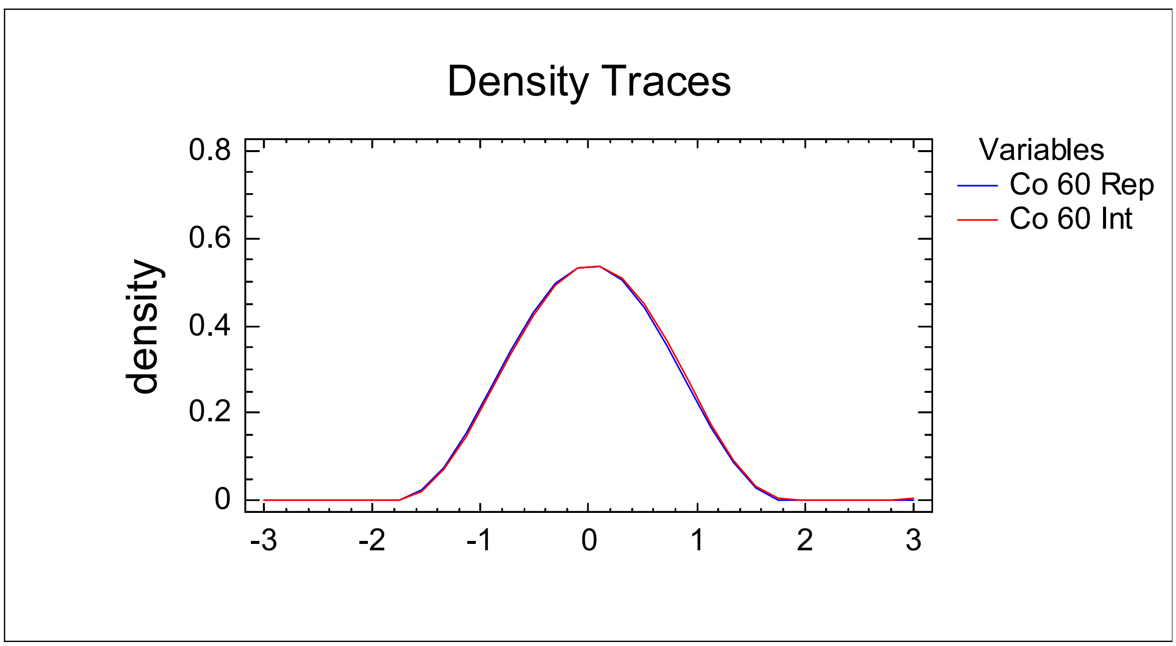
Figure 6.5: U-235 Two-Sample Comparison of Density for Replicate Measurements



Prepared/Date: MPM 3/09/06
Checked/Date: HTD 3/09/06

Figure 6.6: Co-60 Two-Sample Comparison of Density for Replicate Measurements

6.2 MEASUREMENT UNCERTAINTY AND DATA QUALITY INDICATORS

Measurement uncertainty in the techniques prescribed for the final status survey arises from two principal sources: field-sampling variation and instrument measurement variation. Of the two sources, field-sampling variation will likely be the greatest contributor to overall uncertainty because of the inherent logistics of sample collection activities. To minimize the uncertainty contributed by field-sampling variation, field survey and sampling operations were governed by procedures and protocols, and survey personnel were trained on survey instrumentation use and sample collection techniques and procedures. Additionally, individuals who were well versed in the overall survey approach and its data quality objectives provided guidance and refereed when unclear situations arose. The measurement methods, on the other hand, employed standard instrument and laboratory procedures whose aspects and nuances were well understood. Procedures and their associated rigor also governed instrument calibrations, source checks, and operations at the Site.

An important activity in determining the usability of the data obtained during the survey of the site is assessing the effectiveness of the sampling and survey program relative to the design objectives (NRC, 2000; EPA, 2000). Data Quality Indicators (DQIs) were used as a cornerstone for quality comparisons performed against sampling and surveying activities. Identified deficiencies or short-comings were corrected and redirected, increasing the overall data quality and usability. Project goals for measurement uncertainty were developed in line with DQIs and assessed during sampling and survey activities. Upon completion of FSS of the Building Complexes 5 and 6A, FSS activities were evaluated against the project goals developed for project. Table 6.2 presents the target DQIs and summarizes the post-sampling data quality assessment.

Inspection of Table 6.2 indicates that the DQIs were achieved, and the data are regarded as having sufficient quality to be useable for the intended purpose of confidently demonstrating that:

- All volumetric soil sample measurement results are less than the $DCGL_W$ (5 pCi/g Co-60 and 557 pCi/g Total U); **AND**
- The unity rule is met if both radionuclides are present in a single sample location; **AND**
- There are no areas having locally elevated concentrations of residual radioactivity in soil greater than the $DCGL_W$.

6.3 OVERALL QUALITY ASSURANCE AND QUALITY CONTROL

Based on the forgoing analysis and observed practices in the field, the overall project QA/QC goals were obtained. There are no significant data problems or gaps, nor any procedural inadequacies that might compromise the findings of this survey report. The data collected in the final status survey is regarded as high quality data for its intended use.

Table 6.2: Target Data Quality Indicators and Evaluation Results

DQI	Quality Objective	Significance	Action/Remark	Finding
Completeness	90% completeness	Less than complete data set could decrease confidence in supporting information	A minimum 14 volumetric soil samples from Survey Units 05-01 through 05-05, 06-01, and 06-03 while 20 from Survey Units 05-06 and 06-02 was planned in each of the survey units. As a contingency, the minimum sample size specified was increased by 20% to accommodate the possibility that some data might be lost, unusable, or otherwise incomplete. A total of 138 volumetric soil samples were actually collected from all survey units and each survey unit had at least its minimum number of samples collected. 816 volumetric soil samples were collected from the trench areas. Although the exact number of samples collected was never formalized during the planning stage, 816 volumetric sample locations is considered an acceptable quantity of samples for the trench areas at a consideration of 1 sample per 100 linear feet of trench excavation.	DQI accepted
Comparability	Affects ability to combine analytical results	Data collected from randomly selected locations within a survey area are unbiased and comparable by design and can be combined. Combining of other data sets will be subject to appropriate two-sample statistical test methods designed to detect significant differences between samples or populations.	Sampling procedures and protocols were used throughout the FSS process for Building Complexes 5 and 6A. No critical deviation from these procedures was encountered. Although different field survey instrumentation was used during the survey, each instrument was properly calibration and source response checked daily while in use. Sampling and survey data from the trench and sub-slab areas was intentionally kept separated from the surface layer survey units. Although the sample and analysis results were not combined, each separate survey area (trench, surface layer) was evaluated and presented in the report.	DQI accepted
Representativeness	Non-representativeness increases or decreases Type I error depending on the bias.	Sample allocation will include a minimum number of unbiased, randomly distributed sample locations based on survey design.	Sample allocation for Class 1, Class 2 and 3 Survey Units were identified using the computer software program *Visual Sample Plan*. The survey was designed to produce a random sample allocation distribution within each of the Class 3 survey units and a random start for a triangular grid for Class 1 and Class 2 survey units. The sample locations selected meet the intent of the survey design and are considered representative of conditions of the Site soils. There are no analytical or measurement effects (e.g., holding times or compositing effects) affecting representativeness.	DQI accepted
Precision	Measurement variability, due to techniques and/or technology, may increase uncertainty.	Field sampling and instrument operation will be governed by procedures. Replicate volumetric samples, background measurements, and source response check measurements will be used to gauge reproducibility. The number of field replicate measurements specified meets or exceeds MARSSIM guidance (NRC, 2000).	All sampling and field measurement processes were controlled by approved written procedures. The specified minimum number of replicate (duplicate) volumetric samples (7) was obtained. Replicate volumetric sample analysis showed adequate precision even at the low activities encountered (many were below the detection limit for the method). Field instrument response checks also demonstrate the precision of the field survey measurement. Caution must be exercised when attempting to measure precision on replicate measurements with activity near and below the detection limit. Statistical variability at near zero activity limits the likelihood that measurements results will be precise even when sampling and analytical methods are in fact precise and suitable at concentrations approaching the DCGL. All procedures were implemented. Replicate measurements and response check measurements returned expected results. Instruments were calibrated to MACTEC and industry standard specifications and yielded responses to NIST certified calibration sources within ±10% of the known amount of radioactivity. Field responses to a low-activity response check source were consistently within the acceptable range of ±20%. As represented above, precision was acceptable.	DQI accepted
Accuracy	Sampling and data handling can introduce bias and affect Type I and Type II errors.	Sampling and measurements will be governed by procedures. Instruments will be calibrated with NIST traceable sources.	All sampling and field measurement processes were controlled by approved written procedures. Analytical measurements were controlled by approved procedures. Survey and sampling results were recorded in accordance with approved written procedures.	DQI accepted

Prepared/Date: AGS 3/20/06
Checked/Date: HTD 3/20/06

7.0 SUMMARY AND CONCLUSIONS

On the basis of the analyses presented in this report, the data demonstrates that each of the survey units and the trench excavation areas associated with the Building Complexes 5 and 6A has met the decision criteria.

More specifically, the FSS of the Building Complexes 5 and 6A soils demonstrates that:

- No unexpected results or trends are evident in the data.
- The sampling and survey results demonstrate that soil residual radioactivity in Building Complexes 5 and 6A soils is very minimal and, for the most part, indistinguishable from background levels.
- The data quality is judged to be excellent for its intended purpose.
- The amount of data collected from each survey unit and the trench excavation areas is adequate to provide the required statistical confidence needed to decide that the DCGLs are met.
- The retrospective power of the Sign Tests, used to judge compliance, was consistently near 100% and always greater than 95%.

Thus, the null hypothesis—that soil residual radioactivity exists in concentrations above the applicable DCGLs—for each of the survey units and the trench areas from the Building Complexes 5 and 6A should be rejected, and that the areas surveyed and sampled during FSS should be released from further radiological controls.

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