FINAL STATUS SURVEY REPORT

CE WINDSOR SITE
2000 DAY HILL ROAD
WINDSOR, CONNECTICUT

Volume I

Text, Figures and Tables

Prepared for:

ABB INC.

2000 Day Hill Road
Windsor, Connecticut 06095

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LIST OF ACRONYMS AND ABBREVIATIONS

ABB	ABB Inc.
AD	Anderson - Darling
AEC	Atomic Energy Commission
ALARA	As Low as Reasonably Achievable
ANSI	American National Standards Institute
bgs	below ground surface
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
cm	centimeter
Co-60	cobalt 60
cpm	counts per minute
CT	Connecticut
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
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
FUSRAP	Formally Utilized Sites Remedial Action Program
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
HSA	Historical Site Assessment
HEU	highly-enriched uranium
HPGe	high purity germanium detector
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 - Continued

m	meter(s)
m^2	meter(s) 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
$MDCR_{SURVEYOR}$	minimum detectable count rate for the surveyor
N	sample size
NAD	North American Datum
NaI	sodium iodide detector
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 computer program
RSO	Radiation Safety Officer
Site	2000 Day Hill Rd., Windsor, Connecticut
U-234	uranium 234
U-235	uranium 235
U-238	uranium 238
UCL	upper confidence level
USACE	United States Army Corps of Engineers
VSP	Visual Sample Plan computer program
WWTP	wastewater treatment plant

EXECUTIVE SUMMARY

ABB Inc. (ABB) has completed remediation of areas associated with commercial licensed activities (Building Complexes 2, 5, 6A, and 17) at their facility located at 2000 Day Hill Road, in Windsor, Connecticut (Site). As described in the Decommissioning Plan (DP), other portions of the Site are potentially contaminated (impacted), unaffected, or have been designated as part of the Formally Utilized Sites Remedial Action Program (FUSRAP) and are being addressed by the U.S. Army Corps of Engineers (USACE). Final Status Surveys (FSS) have been performed and reported in separate documents in areas associated with commercial licensed activities (Building Complexes 2, 5, 6A, and 17). FUSRAP areas are being evaluated by USACE and no remediation activities have yet occurred in these areas. No further actions are needed for unaffected portions. This report documents the final radiological status of portions of the Site outside of the Building Complex areas that have been identified as having a potential to contain residual radioactivity (impacted).

From the early 1960s to 2000, the Site was involved in the research, development, engineering, production, and servicing of nuclear systems and fuel. ABB has contracted MACTEC Development Corporation (MACTEC) to perform decontamination and decommissioning (D&D) of the Building Complexes 2, 5, 6A, and 17. This included decontamination and dismantlement of structures, removal of concrete slabs, footers, and foundations to four feet below ground surface, removal of pavement areas, removal of buried utilities, and transportation of radioactive waste to appropriate off-site facilities.

It is likely that remediation will be needed in the FUSRAP portions of the Site, but no other radiological remediation has been necessary in these potentially impacted areas.

The FSS did not identify residual radioactivity in excess of the applicable soil radioactivity release criteria. For the potentially contaminated portions of the Site, twelve survey units were created in support of the FSS, including one Class 2 survey unit and eleven Class 3 survey units.

The design and interpretation of the final radiological status survey of the soil 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 cobalt 60 (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 to confidently reject the null hypothesis. Concentrations of residual radioactivity were found to be very minimal and essentially indistinguishable from background. In all of the survey units under consideration, the DCGL was met with greater than 95% confidence.

Quality control (QC) measures were taken during the survey process to assess the accuracy and precision of the measured results. 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 final radiological status survey of the soils at the Combustion Engineering, Inc. (CE) Windsor site concludes that in each survey unit all of the conditions and requirements for unrestricted radiological release have been met. This FSS supports the regulatory decision to terminate the license.

1.0 INTRODUCTION

This radiological Final Status Survey (FSS) report documents the radiological status of the Combustion Engineering (CE) Windsor Site in Windsor, Connecticut. Presently, 2000 Day Hill Rd., Windsor, Connecticut (Site) is subject to U.S. Nuclear Regulatory Commission (NRC) Radioactive Materials License No. 06-00217-06 (NRC, 2002) due to its historical use involving licensable quantities of radioactive materials. The long-term objective of the licensee, ABB Inc. (ABB), is to decommission the Site such that it will meet the criteria for unrestricted use as specified in Title 10 of the Code of Federal Regulations (CFR), part 20.1402 and to terminate NRC license No. 06-00217-06. ABB contracted MACTEC Development Corporation (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. The buildings within those areas have been decontaminated and demolished, building slabs and pavement have been removed to 4 foot below ground surface (bgs), 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. FSSs in areas associated with commercial licensed activities have been performed and were previously reported (MACTEC, 2005; MACTEC, 2006a; MACTEC, 2006b). This report documents the final radiological status of the portions of the Site outside of the Building Complex areas that have been identified as having a potential to contain residual radioactivity. This FSS 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 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 levels (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 across the Site relative to the DCGLs.

1.1 METHODOLOGY AND GUIDANCE USED

The FSS report follows methods outlined in the Multi-Agency Radiation Survey and Site Investigation Manual (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 Quality Assurance (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 evaluates survey data for compliance against the decision criteria. Section 5 includes quality control and data quality assessment evaluations and discussions. Section 6 summarizes the FSS and concludes the outcome of the FSS and Section 7 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 of radioactive materials through licensure. Federal regulations require that 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.

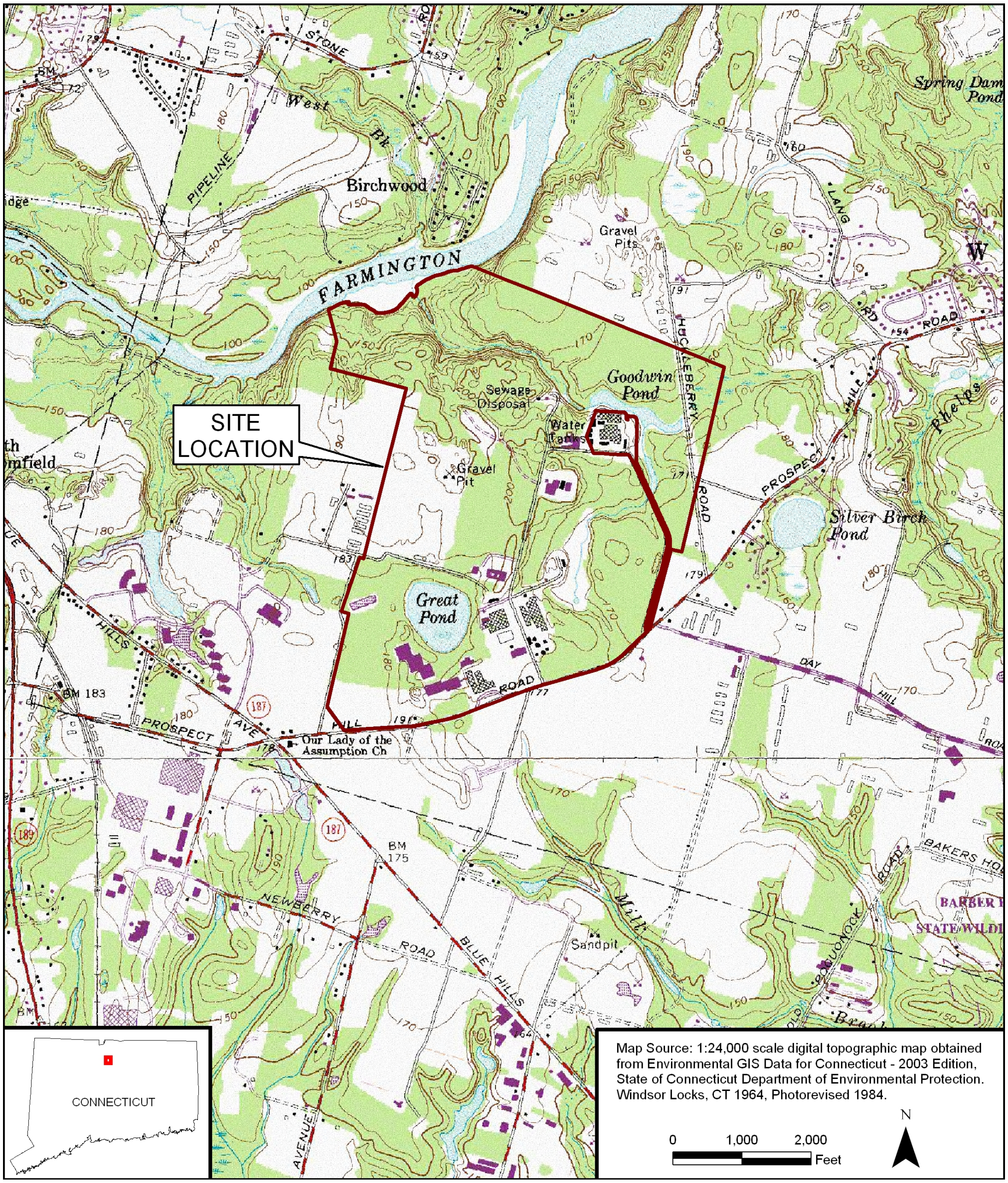

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Figure 1.1: Site Location Map

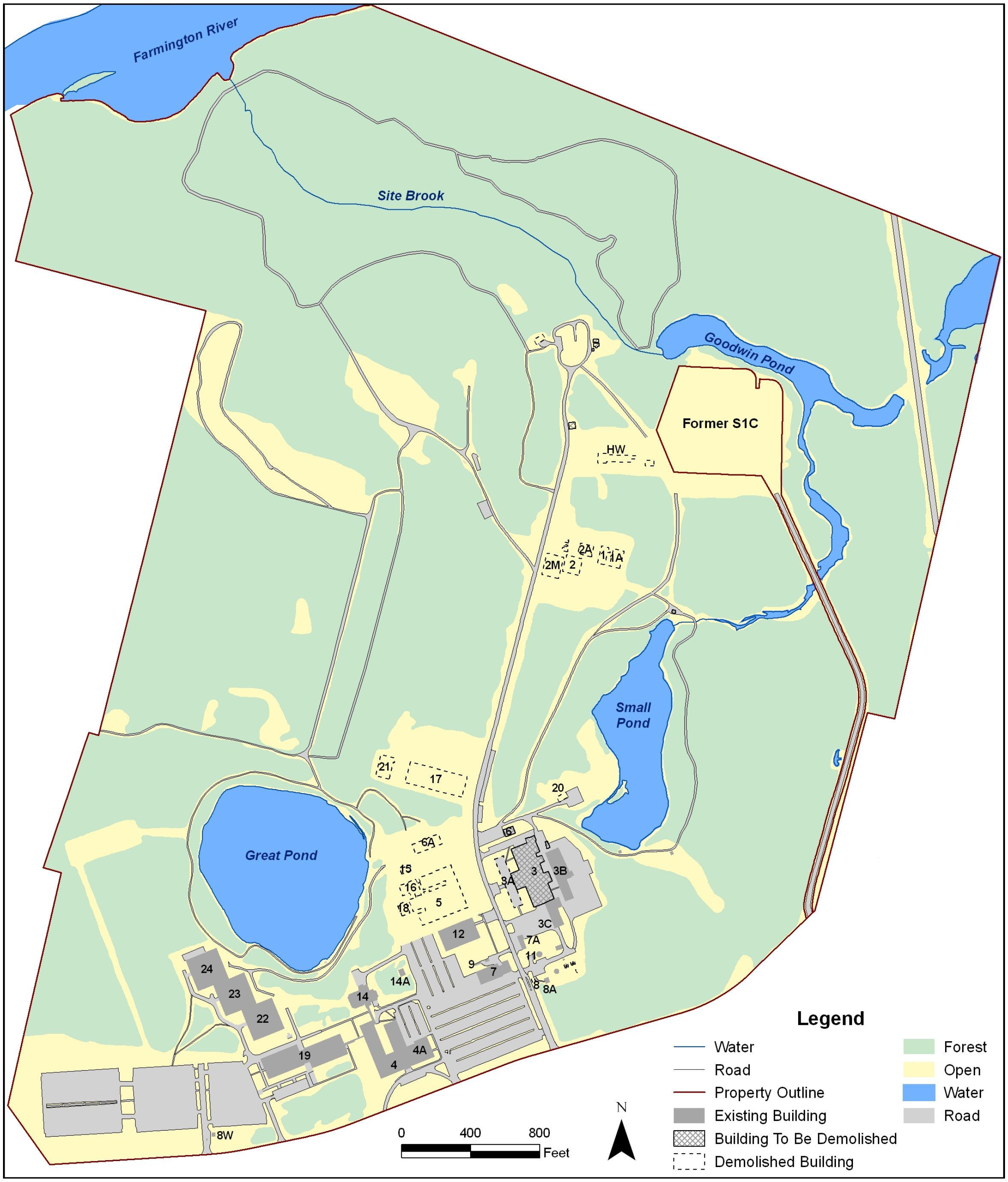

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Figure 1.2: Site Overview

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 the 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 (HSA) (Harding, 2002).

1.4 CURRENT SITE-WIDE CONDITIONS

Commercial licensed activities were conducted in Building Complexes 2, 5/6A, and 17. All areas of the Site where residual radioactivity could be present (impacted areas), based on the HSA, were investigated. For Commercial D&D building complexes, remediation was conducted under the Site 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 soils impacted by residual radioactivity above the DCGLs. All remediation in the Commercial D&D building complexes is complete and FSS results for each of the Commercial D&D building complexes are documented in separate reports (MACTEC 2005, 2006a, 2006b). Commercial D&D areas of the Site are indicated on Figure 1.3

Portions of the Site have been designated as radiologically impacted but under the responsibility of the Formally Utilized Sites Remedial Action Plan (FUSRAP). The FUSRAP areas of the Site are being evaluated by the United States Army Corps of Engineers (USACE). A remedial investigation (RI) has been performed of the FUSRAP areas (ENSR, 2004), and a feasibility study is currently being completed. More specific details about the FUSRAP potions of the site can be found in the DP and RI report. The FUSRAP portions of the site are indicated on Figure 1.3.

Potentially impacted portions of the Site consist of land and surface water bodies adjacent to commercial licensed areas or FUSRAP areas on the Site. This portion of the Site is the primary focus for this FSS report. The potentially impacted portions of the site are indicated on Figure 1.3.

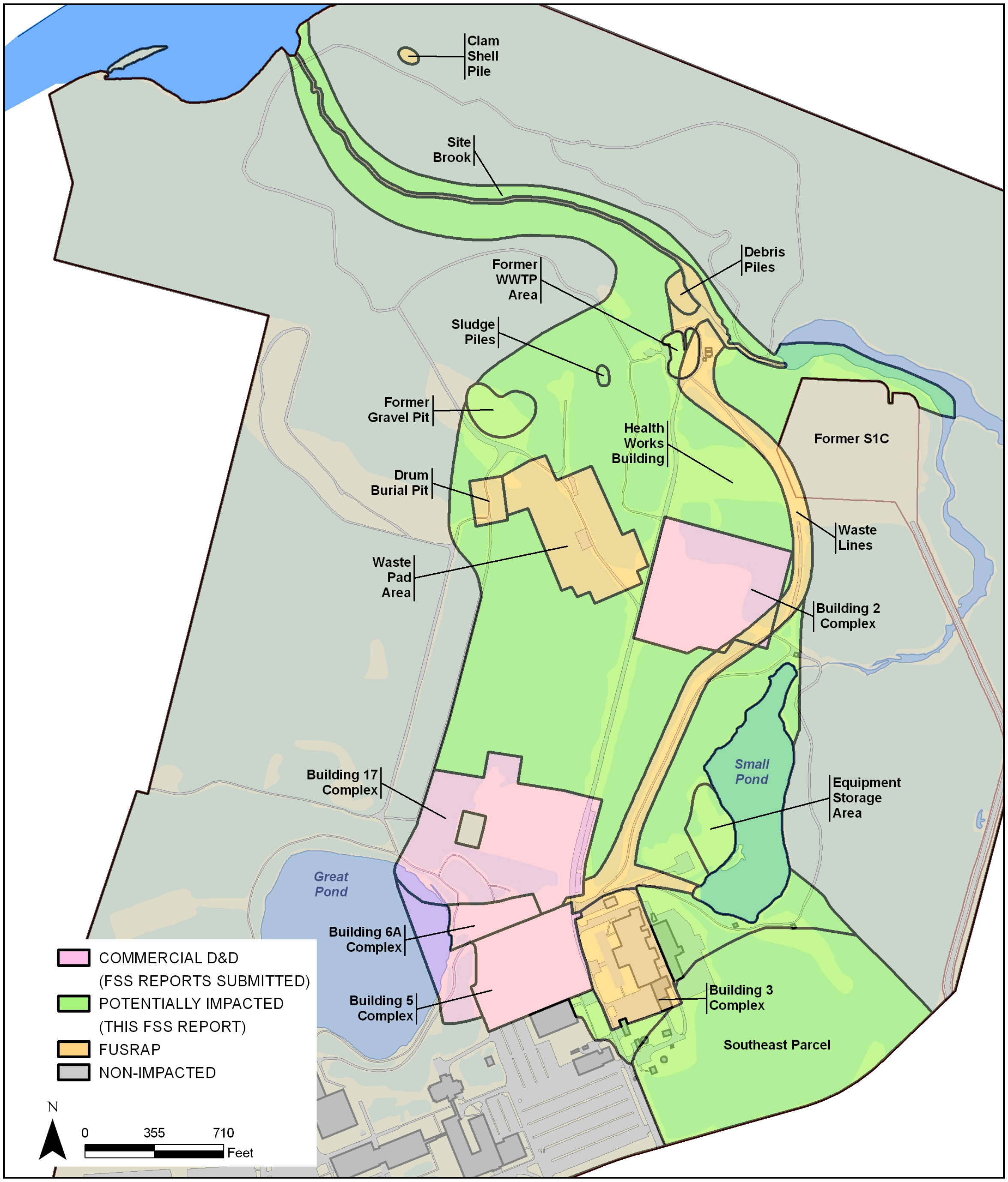

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Figure 1.3: Site Areas

The potentially impacted areas contain a few specific locations know to be impacted by site operations. These include the former waste water treatment plant, digester sludge piles, former gravel pit and demolition debris area, and equipment storage yard. Radiological characterization of these areas did not identify any elevated results (comparable to background concentrations). The remaining portions of the potentially impacted areas include open lands, woods and surface water bodies adjacent to areas known to be impacted by site radiological activities (Commercial D&D, FUSRAP, and S1C). Characterization surveys of these areas did not identify any significantly elevated concentrations of radionuclides, so no remediation was necessary in these areas[1]. These potentially impacted areas are identified in Figure 1.4 (aerial photo) and Figure 1.5.

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1 A few small localized spots of elevated radioactivity have been identified in this portion of the Site. Biased sampling and analysis of these local anomalies spots has clearly identified the residual radioactivity as NORM, not uranium and Co-60 as addressed under the DP. This area will be addressed with the FUSRAP areas.

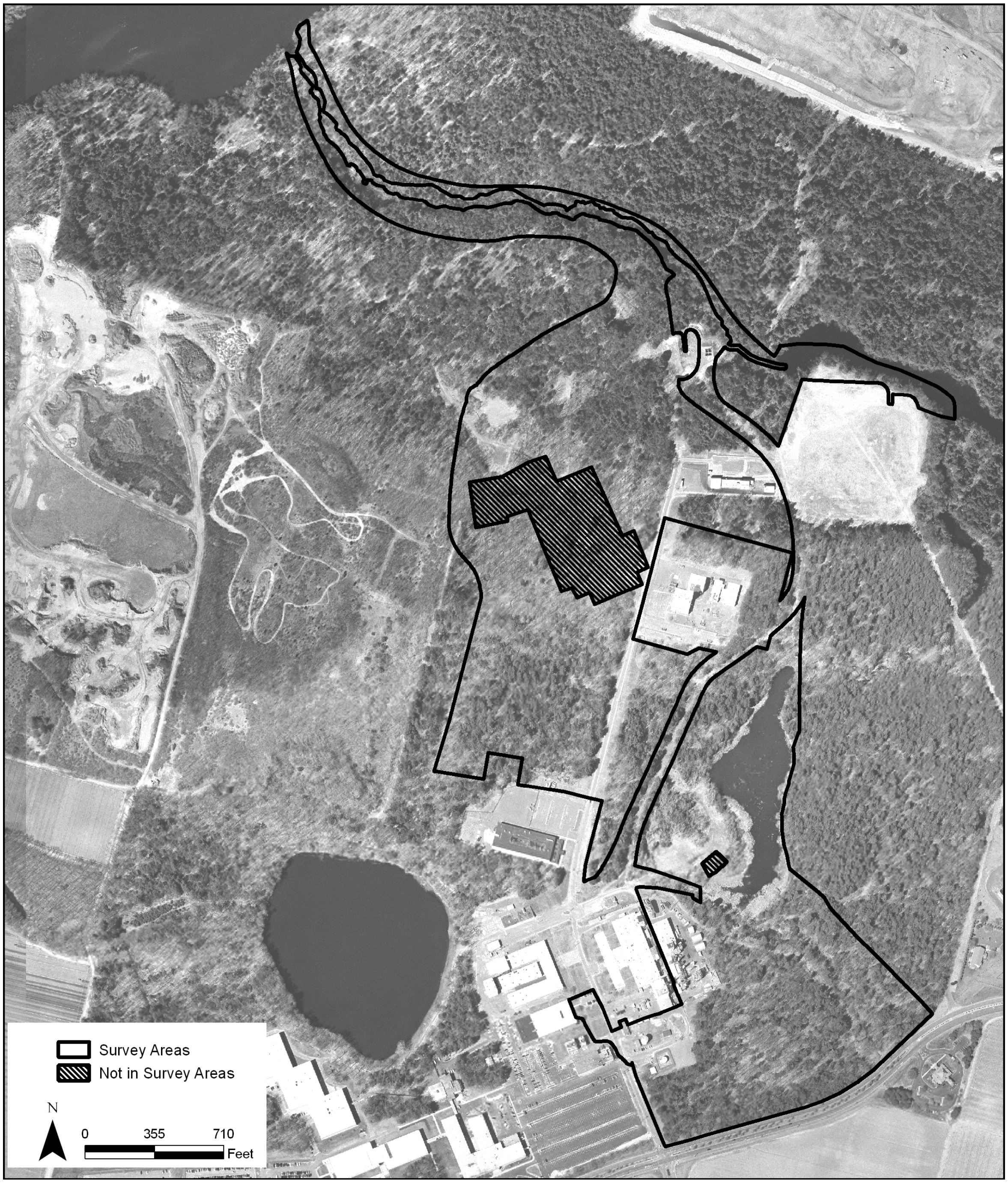

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Figure 1.4: Potentially Impacted Areas (Aerial Photo)

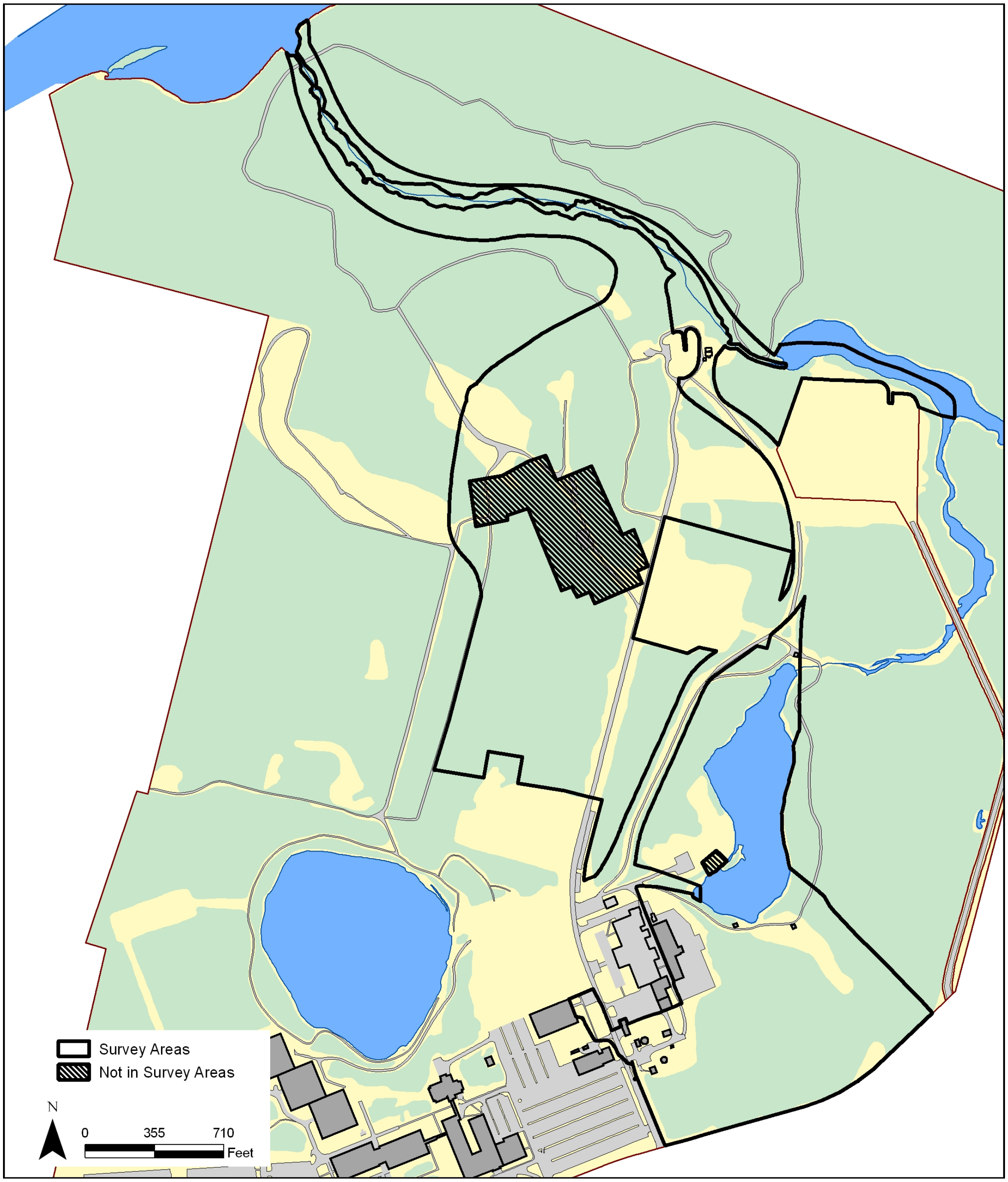

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Figure 1.5: Potentially Impacted Areas

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 radionuclides associated with enriched uranium are 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 are provided as part of the DCGLs (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

As no remediation was necessary for the remaining impacted portions of the site, results of the FSS performed demonstrate that the potential dose from any residual radioactivity is below the release criterion for each survey unit.

1.6.1 Compliance Testing

The Sign Test was used to evaluate compliance with derived concentration guideline level, survey unit average (median) concentration corresponding to the permissible limit ($DCGL_W$) for FSS and 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 potentially impacted areas soils.

The Sign Test is a one-sample, non-parametric test that can be used to evaluate compliance with 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.

The combination of FSS 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 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)

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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.

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

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 derived concentration guideline level for the EMC is shown in Equation 1-2.

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

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.

(Equation 1-3)

$$\frac{\delta_U}{DCGL_U} + \frac{\delta_B}{DCGL_B} + \frac{\overline{\chi}_U - \delta_U}{A_m * DCGL_U} + \frac{\overline{\chi}_B - \delta_B}{A_m * DCGL_B} < 1$$

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.

Site-specific DCGLs were derived for soil and accepted by the NRC as part of the DP. The approved Site-specific $DCGL_W$ for total uranium is 557 pCi/g and the $DCGL_W$ for cobalt 60 is 5 pCi/g. Additional information can be found in the report *Derivation of the Site-Specific Soil DCGL*s (MACTEC, 2003a). Calculations were performed using RESRAD to develop area factors used to assess compliance with the $DCGL_{EMC}$.criteria. Table 1.2 displays the $DCGL_{EMC}$ 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 (Am)	Total uranium $DCGL_{EMC}$ (pCi/g)	Co-60 Area Factor (Am)	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

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2.0 FIELD IMPLEMENTATION

This section of the report documents the FSS in the remaining potentially impacted portions of the Site. FSS reports of the commercial D&D areas have been previously submitted to the NRC (MACTEC, 2005; MACTEC, 2006a; MACTEC, 2006b). FUSRAP portions of the site are being evaluated by USACE.

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 geographic information system (GIS) maps were created that provided survey units and sample locations that were used in conjunction with global positioning satellite (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 (A through L).

Gamma walkover and direct static surveys were performed on soils using a 2 inch x 2 inch thallium-activated sodium iodide (NaI) detector coupled to an appropriate scaler/rate meter instrument to form a complete survey instrument package. Soil volumetric samples were collected and then 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 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. Development of survey units for these remaining land areas concluded in the establishment of twelve individual survey units, with distinguishable and independent characteristics.

For the remaining portions of the Site, large generally wooded areas of the Site were divided into survey units by geographical boundaries associated with Commercial D&D, FUSRAP, S1C and non-impacted areas. This resulted in 6 large survey units that were classified as Class 3 areas since no significant concentrations of residual radioactivity were detected during previous characterization survey activities. Four small areas were specifically isolated from the large wooded areas due to historical uses.

One of these areas is the former waste water treatment plant area that processed sanitary waste. A related area in the woods was used for site sanitary treatment system sludge disposal. Radiological characterization of these areas has identified low concentrations of uranium and Co-60 relative to the DCGLs, so both were classified as Class 3 areas.

The former equipment storage yard had low concentrations of uranium and Co-60 identified during characterization and was kept as a separate Class 3 survey unit due to the nature of use in this area as compared to surrounding areas. A small portion of the equipment storage yard has had FUSRAP material identified and this portion has not been included in FSS.

Another small area was a former gravel pit and demolition debris area, which was a waste disposal area for general site debris (asphalt, concrete, etc.). Characterization of this area found a localized spot of elevated residual radioactivity, so it was classified as a Class 2 area. This hot spot was so small that is was completely removed as part of the sampling process to characterize the elevated readings.

Several surface water bodies (sediment) are also included in the FSS of the Site. Small Pond characterization data identified low concentrations of uranium and Co-60 relative to the DCGLs in the sediment so it was classified as a Class 3 area. A small portion of the southwest end of Small Pond had FUSRAP material identified and this portion has not been included in FSS. A portion of Goodwin Pond is included in the potentially impacted portion of the Site due to its proximity to the S1C Site. Characterization of Goodwin Pond sediment did not identify any significant concentrations of uranium or Co-60, similar to the soil in the woods surrounding S1C, so both the woods and portion of Goodwin Pond were combined to create a single Class 3 area.

Site Brook is impacted with FUSRAP material and is being evaluated by USACE. The woods adjacent to the brook are included in FSS of the Site. The woods along Site Brook have been divided into two sections. The first section (eastern) is adjacent to the outfalls and other FUSRAP portions of the Site since this portion of the brook has greater impact from Site activities than portions downstream. The second section (western) covers the rest of the woods along Site Brook all the way to its discharge into the Farmington River. Both sections are classified as Class 3 since no significant residual radioactivity has been identified outside of the brook in this area. The size of the first area was limited to 10,000 m^2 in order to increase sample density as this portion is adjacent to FUSRAP areas with highest concentrations of residual radioactivity in the Site Brook area.

A summary of the survey units for the Site FSS Areas is presented in Table 2.1 and depicted in Figure 2.1.

Table 2.1: Summary of Remaining Site Survey Units

Survey Unit ID	Class	Area (m^2)	Description
CE-FSS-20-01	3	3,900	Former waste water treatment plant area
CE-FSS-21-01	3	500	Digester sludge pile area
CE-FSS-22-01	2	6,500	Former gravel pit and demolition debris area
CE-FSS-23-01	3	6,300	Former equipment storage yard
CE-FSS-24-01	3	128,500	Southeast parcel (south of Building 3 Complex)
CE-FSS-25-01	3	34,500	Small Pond
CE-FSS-26-01	3	33,100	Woods west of small pond
CE-FSS-26-02	3	90,100	Woods north of Building 17 Complex
CE-FSS-26-03	3	131,500	Woods north and west of Building 2 Complex
CE-FSS-26-04	3	25,600	Woods adjacent to S1C
CE-FSS-26-05	3	9,200	Woods adjacent to Site Brook (near outfalls)
CE-FSS-26-06	3	41,300	Woods adjacent to Site Brook (remainder)

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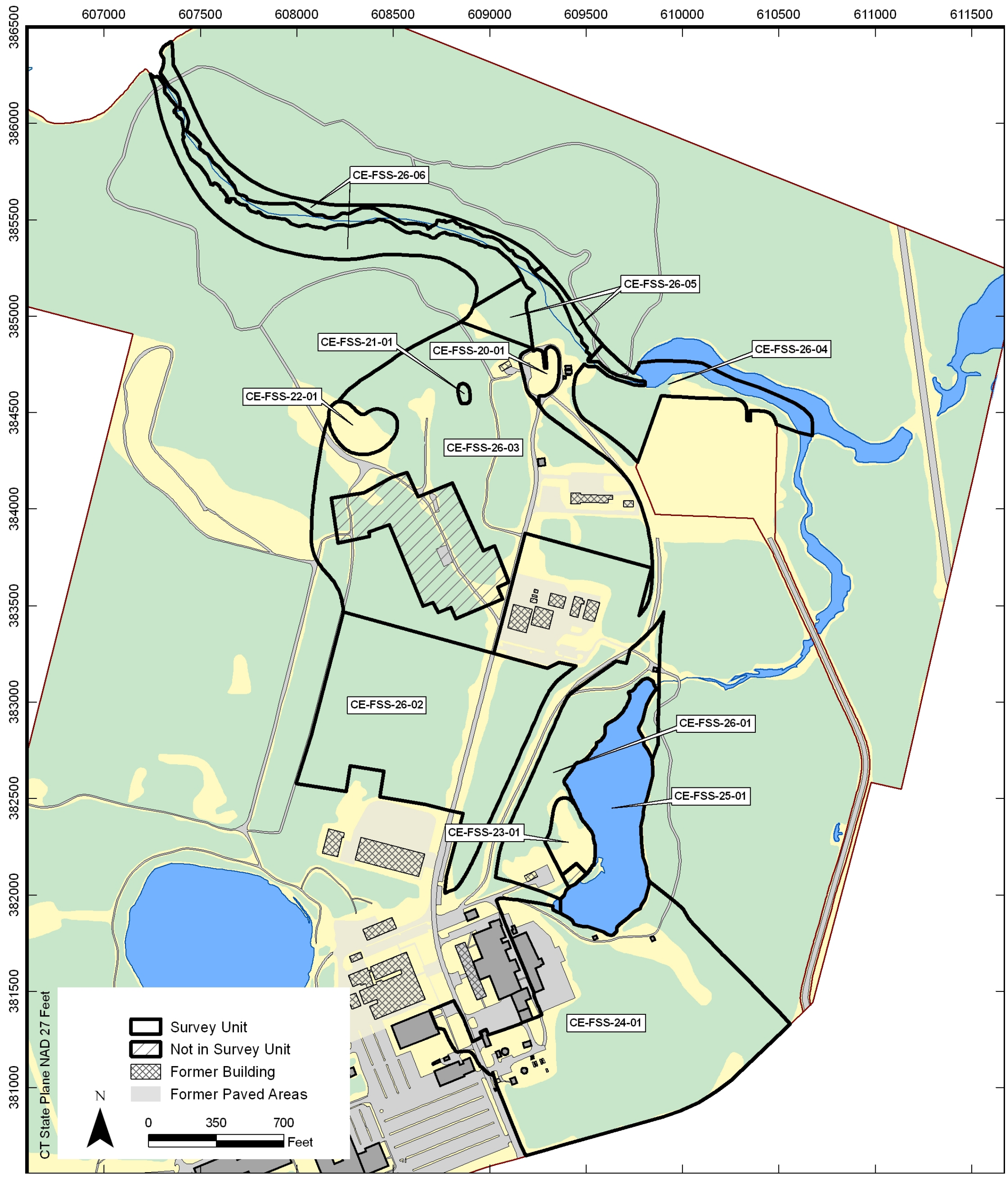

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Figure 2.1: Overview of Remaining Final Status Survey Units

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, 6A, and 17) and the remaining areas of the Site is statistically summarized for comparison in Table 2.2. Review of this data indicates there is no significant difference within these areas as compared to the DCGLs of 557 pCi/g for total uranium or 5 pCi/g for cobalt-60.

Table 2.2: Summary of 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
Remaining Areas	4.2	2.9	26	0.05	0.03	0.2

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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 produces 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. The lower bound of the gray region (LBGR) was conservatively selected to be around 70% of the DCGL. The standard deviation was also 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 2.4. 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. No Class 1 Survey Units were created for the remaining Final Status Survey Units.

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 standard deviation was conservatively approximated high for Class 2 areas to 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. Only one Class 2 Survey Unit was created for the remaining Final Status Survey Units.

Table 2.4: Class 2 Survey Unit Sample

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	

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2.3.3 Class 3 Survey Unit Sample Size

Since Class 3 survey units are not expected to have measurable residual radioactivity in excess of background 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 planned and the number of samples obtained per survey unit is presented in Table 2.6. In every survey unit, the number of samples obtained met or exceeded the number of samples planned.

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-20-01	3	14	14
CE-FSS-21-01	3	14	14
CE-FSS-22-01	2	20	20
CE-FSS-23-01	3	14	14
CE-FSS-24-01	3	14	14
CE-FSS-25-01	3	14	28
CE-FSS-26-01	3	14	14
CE-FSS-26-02	3	14	14
CE-FSS-26-03	3	14	14
CE-FSS-26-04	3	14	14
CE-FSS-26-05	3	14	14
CE-FSS-26-06	3	14	14
Total Number of Samples		**174**	**188**

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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 remaining Site areas, randomly chosen sampling and survey locations were used to place Class 3 survey locations within those survey units. Systematic grid patterns were used to place Class 2 survey locations within those survey units. For the Class 2 survey unit, a random start location was selected and used to provide an unbiased set of measurement locations for the FSS.

A 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 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 FSS evaluation during October and November 2005, and June, July and August 2006. Sediment samples from Small Pond (Survey Unit CE-FSS-25-01) were collected in August 2002. Figures of sample locations for each survey unit are provided in the survey unit data appendices (A through L). 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. 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. The samples were collected from the top 3 inches of the soil at the sample location, consistent with the source term assumptions in the DCGLs. 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. One sample was relocated due to pavement interference in survey unit CE-FSS-20-01. Sediment samples from Small Pond were collected using a dredge sampling tool from a small boat.

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 a 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 Physics Trailer rather than in the field during sampling activities.

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 was to 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 exceeded 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% $DCGL_W$

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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 assumptions of data and the most conservative soil concentration exposure rate factors developed, a counts per minute (cpm) value was generated at the stated $DCGL_W$ value for the scanning measurement investigation level (Table 2.8). No walkover survey result was reported at greater than the investigation level.

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

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2.6 ON-SITE GAMMA SPECTROSCOPY INSTRUMENTATION

Soil and sediment volumetric samples analyzed on-site were analyzed by a 30 percent efficient (detector serial # 9882108) HPGe gamma spectroscopy system throughout the entire FSS sampling campaign and in accordance with the Genie-2000 Spectroscopy System Operations Instructions (Canberra, 2002a). The only exception was that sediment samples from survey unit CE-FSS-25-01 (Small Pond) were sent to General Environmental Laboratories, Inc. (GEL) for analysis since they had been collected prior to the establishment of the on-site gamma spectroscopy system calibration and QA program for FSS.

The gamma spectroscopy system identifies and quantifies the concentrations of multiple 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 2.2. The system is energy calibrated so the MCA data channels are given an energy equivalence and display 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 2.2: 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 for both energy and efficiency parameter inputs was performed daily, prior to counting operations. 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 N.

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 5 for additional information on the QA measurement results). The background data and control charts are provided in Appendix N.

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), along with 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. 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 (Ci/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

Notes:
U-234 = uranium 234
U-235 = uranium 235
U-238 = uranium 238

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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 considered at the Site (uranium and Co-60) 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) was identified by the surveyor, a more focused survey of the area was performed. By slowing or stopping the forward progress and searching for the area of increased activity, a localized area of elevated residual radioactivity could be isolated and a static one-minute count performed. No locations of elevated residual radioactivity that exceeded the investigation level were identified during the surveys. 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 2350-1 coupled with the Ludlum 44-10 NaI detector was used during FSS survey activities of the remaining Site areas.

2.7.2 Gamma Walkover Instrument Calibration

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

2.7.3 Gamma Walkover Measurement Detection Limitations

For any survey instrument, 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 (NRC, 1997) and the MARSSIM.

Derivation of the MDC_{SCAN} for soil is 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 and the gamma 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 expected 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 square meters) and a soil 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 source geometry was input into Microshield™ as the standard material concrete with a density of 1.6 grams per cubic centimeter ($g/cm^{3)}$ (to represent typical soil). The byproduct and uranium source terms were input at the DCGL concentration values and the uranium source was decayed for fifty years in Microshield™ in order to assure that all of the decay products would be present in the modeling. The modeling results established 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} for the instrument 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 2,700 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 786 cpm above background is the minimum value for 95% true positive detection.

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

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: $MDCR_{SURVEYOR}$ Values

Parameter		Value
i	The length of the counting interval (seconds)	1
d'	Index of sensitivity	1.38
C_b	Background count rate (cpm)	2,700
b_i	Number of background counts in counting interval i	45
s_i	Minimum detectable net counts in counting interval i	12.7
MDCR	Minimum detectable count rate (cpm)	555
p	Surveyor efficiency	0.5
$MDCR_{surveyor}$	Surveyor minimum detectable count rate (cpm)	786

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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 centimeters [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.

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

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 Detector

Parameter		Byproduct	Uranium
		Ludlum	Ludlum
$MDCR_{surveyor}$	Surveyor minimum detectable count rate (cpm)	786	786
ε_t	Counting system efficiency (cpm per μR/h)	424	4,582
S_c	Soil concentration exposure rate factor (pCi/g per μR/h)	1.41	309
MDC_{SCAN}	Scan minimum detectable concentration (pCi/g)	2.6	53

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This evaluation shows that the gamma walkover measurement detection limits are acceptable since they are much less than the DCGLs.

2.7.4 Walkover and Static Instrument Background Measurements

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

Background measurements were taken prior to the start of surveying for each survey unit and at the beginning of each workday. Table 2.12 presents the walkover (scan) and static survey background readings for remaining Site area surveys.

Table 2.12: Remaining Site Area Walkover and Static Survey Background Measurements

Walkover and Static Background Measurements	
Survey Unit	**Recorded Background Reading (cpm)**
CE-FSS-20-01	2,200 – 3,000
CE-FSS-21-01	2,700
CE-FSS-22-01	2,700
CE-FSS-23-01	1,500 – 2,500
CE-FSS-24-01	2,700
CE-FSS-26-01	2,700
CE-FSS-26-02	2,700
CE-FSS-26-03	2,700
CE-FSS-26-04	1,500 – 2,000
CE-FSS-26-05	1,500 – 3,500
CE-FSS-26-06	1,500 – 2,500

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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.

The instrument and detector combinations used for the gamma walkover carry the same detection limitations identified in Section 2.7.3. Instrumentation used for the walkover and static surveys is identified Table 2.13. Calibration certificates for the scanning instrumentation are presented in Appendix M.

Table 2.13: Walkover and Static Instrumentation

Instrumentation			
Inst Model	**Serial #**	**Detector Model**	**Serial #**
2350-1	186175	44-10	199144
2350-1	175852	44-10	15203

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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 (A through L) of this report.

3.1 FIELD SURVEY AND VOLUMETRIC SOIL SAMPLING RESULTS OVERVIEW

In all, 188 volumetric soil samples from 12 survey units were collected and analyzed as part of FSS for the remaining impacted Site areas. Sample locations, where a single sample was collected and split into a duplicate sample, are indicated as 'duplicate' samples. Twenty samples were split as part of the overall project QA/QC. For data reduction purposes, the arithmetic mean of the initial sample measurement result and the corresponding duplicate sample measurement result were used as the reported value for the sample location. Further information about duplicate samples and the assurance of precision and variability is presented in Section 6.

3.2 DATA ASSESSMENT

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 statistical basis for comparison of appropriate metrics calculated from the data. The Anderson-Darling (AD) 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 be approximately lognormally distributed; 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 survey unit data appendices (A through L).

Once the survey unit data has been assessed and verified that it is acceptable for comparison to the release criteria, it was evaluated against the $DCGL_w$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 survey unit data appendices (A through L) of this report.

3.2.1 Survey Unit CE-FSS-20-01

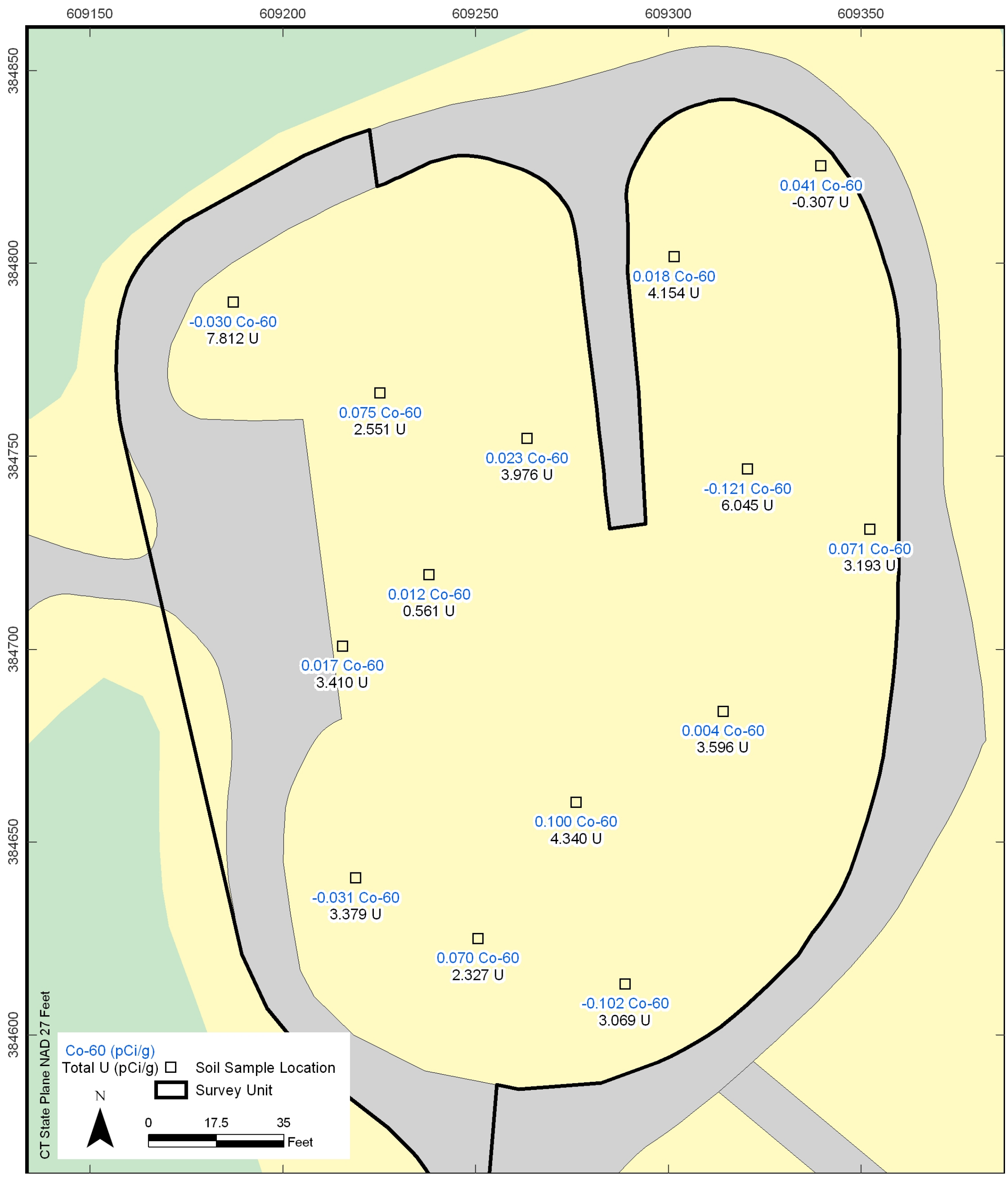
Survey Unit CE-FSS-20-01 encompasses the former WWTP and consists of approximately 3,900 square meters of land area. Figure 3.1 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. 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 CE-FSS-20-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 1,726 cpm to 2,990 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.1.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-20-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.1 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-20-01.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.1: Survey Unit CE-FSS-20-01 Total U and Co-60 Activities (pCi/g)

3.2.2 Survey Unit CE-FSS-21-01

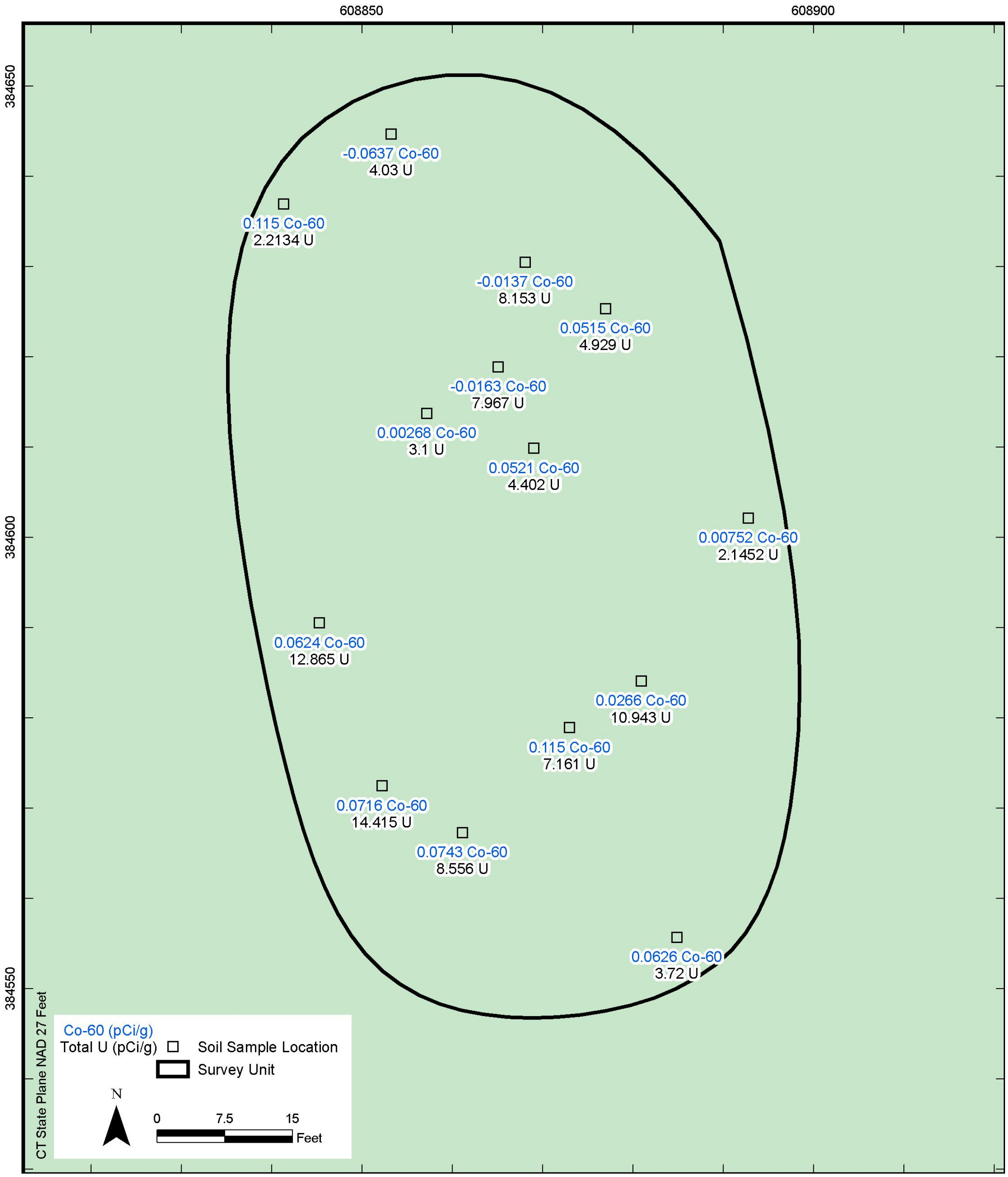
Survey Unit CE-FSS-21-01 covers the digester sludge pile area and consists of approximately 500 square meters of land area. Figure 3.2 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. 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 CE-FSS-21-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 1,096 cpm to 3,139 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.2.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-21-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.2 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-21-01.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.2: Survey Unit CE-FSS-21-01 Total U and Co-60 Activities (pCi/g)

3.2.3 Survey Unit CE-FSS-22-01

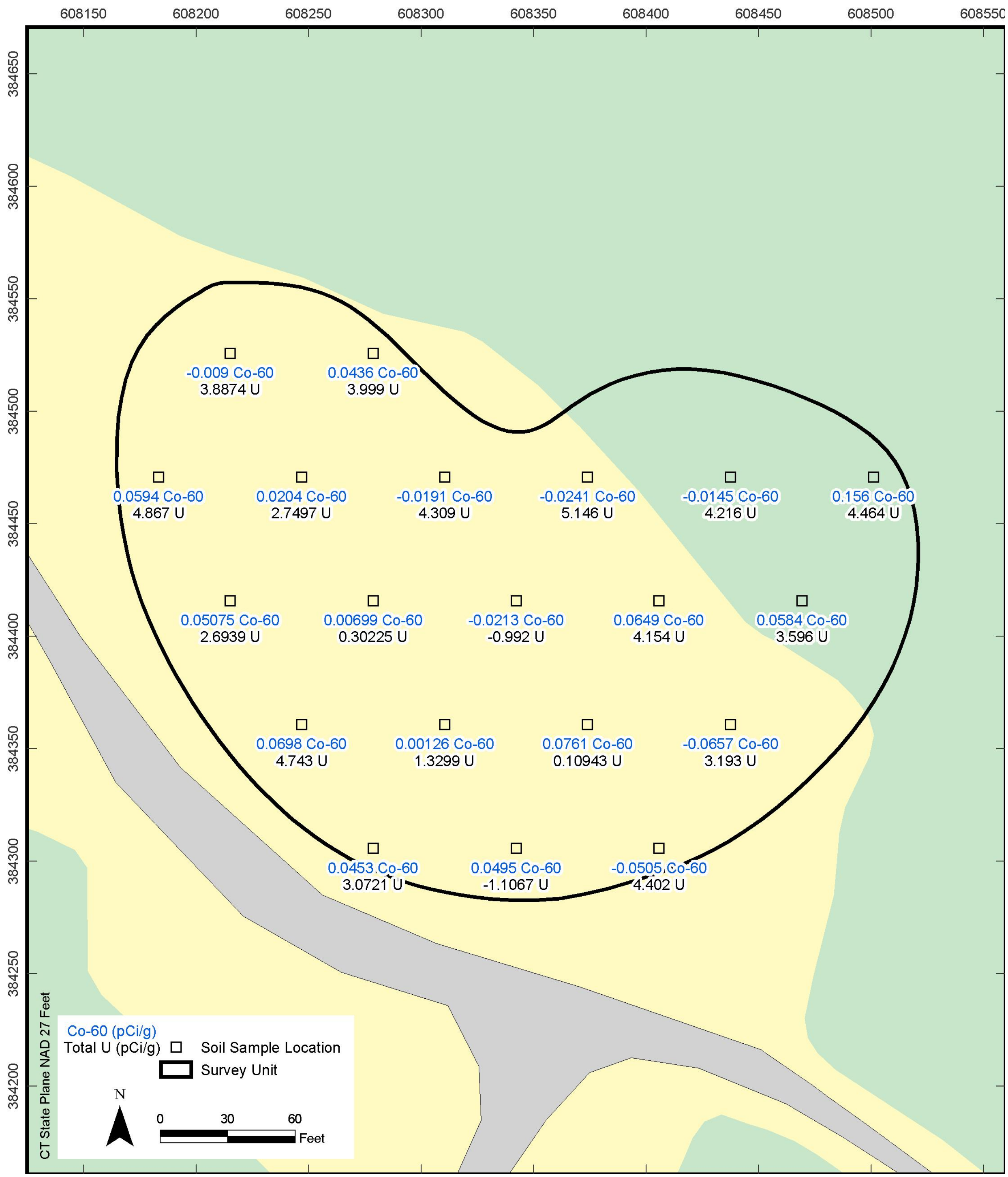
Survey Unit CE-FSS-22-01 covers the former gravel pit and demolition debris area and consists of approximately 6,500 square meters of land area. Figure 3.3 presents an overview of the survey unit. Twenty survey locations were placed on a systematic grid pattern within the Class 2 survey unit to represent the distribution of residual radioactivity for the survey unit. For Survey Unit CE-FSS-22-01, a random start location was selected and used to provide an unbiased set of measurement locations. Data associated with this survey unit are provided in Appendix C.

3.2.3.1 Gamma Walkover Survey Results

Approximately 25 percent of the surface area for Survey Unit CE-FSS-22-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 1,550 cpm to 2,980 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.3.2 Volumetric Soil Sample Results

Twenty systematically placed volumetric soil samples were obtained for FSS in Survey Unit 22-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.3 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-22-01.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.3: Survey Unit CE-FSS-22-01 Total U and Co-60 Activities (pCi/g)

3.2.4 Survey Unit CE-FSS-23-01

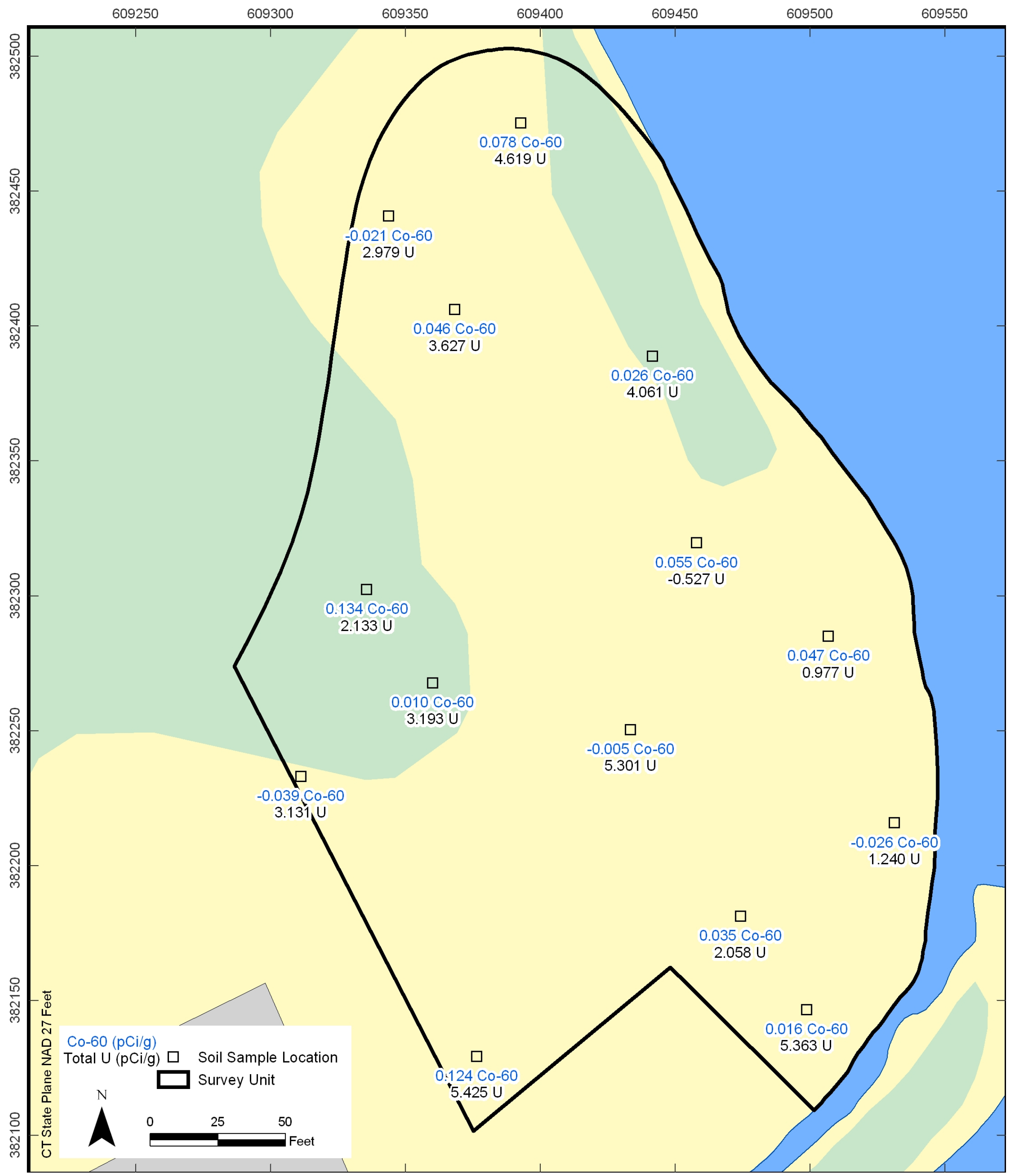
Survey Unit CE-FSS-23-01 includes the former equipment storage yard and consists of approximately 6,300 square meters of land area. Figure 3.4 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix D.

3.2.4.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-23-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 712 cpm to 2,822 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.4.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-23-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.4 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-24-01.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.4: Survey Unit CE-FSS-23-01 Total U and Co-60 Activities (pCi/g)

3.2.5 Survey Unit CE-FSS-24-01

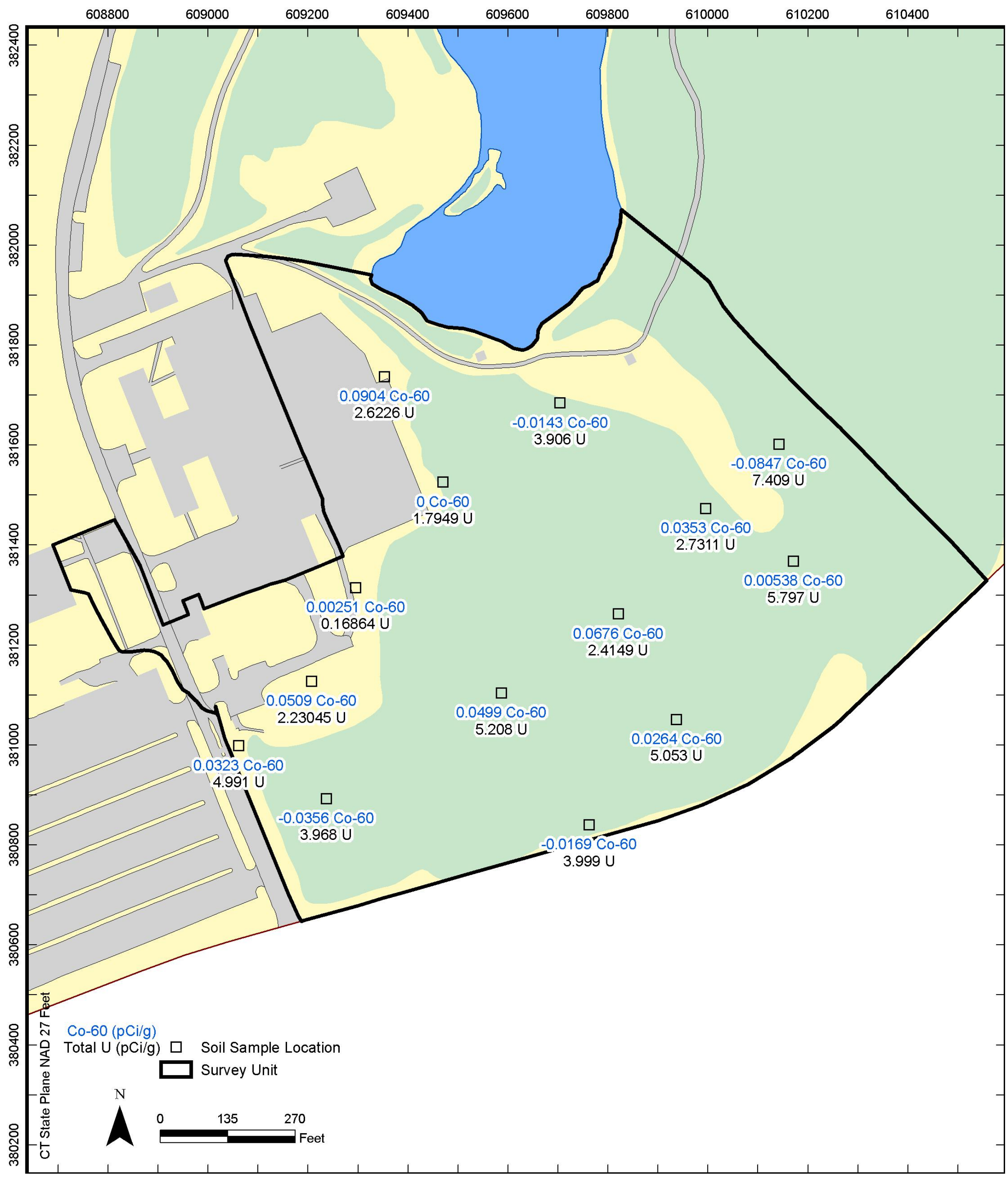
Survey Unit CE-FSS-24-01 is located primarily to the southeast of Building 3 (FUSRAP) and consists of approximately 128,500 square meters of land area. Figure 3.5 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. 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 CE-FSS-24-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 940 cpm to 3,121 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.5.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-24-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.5 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-24-01.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.5: Survey Unit CE-FSS-24-01 Total U and Co-60 Activities (pCi/g)

3.2.6 Survey Unit CE-FSS-25-01

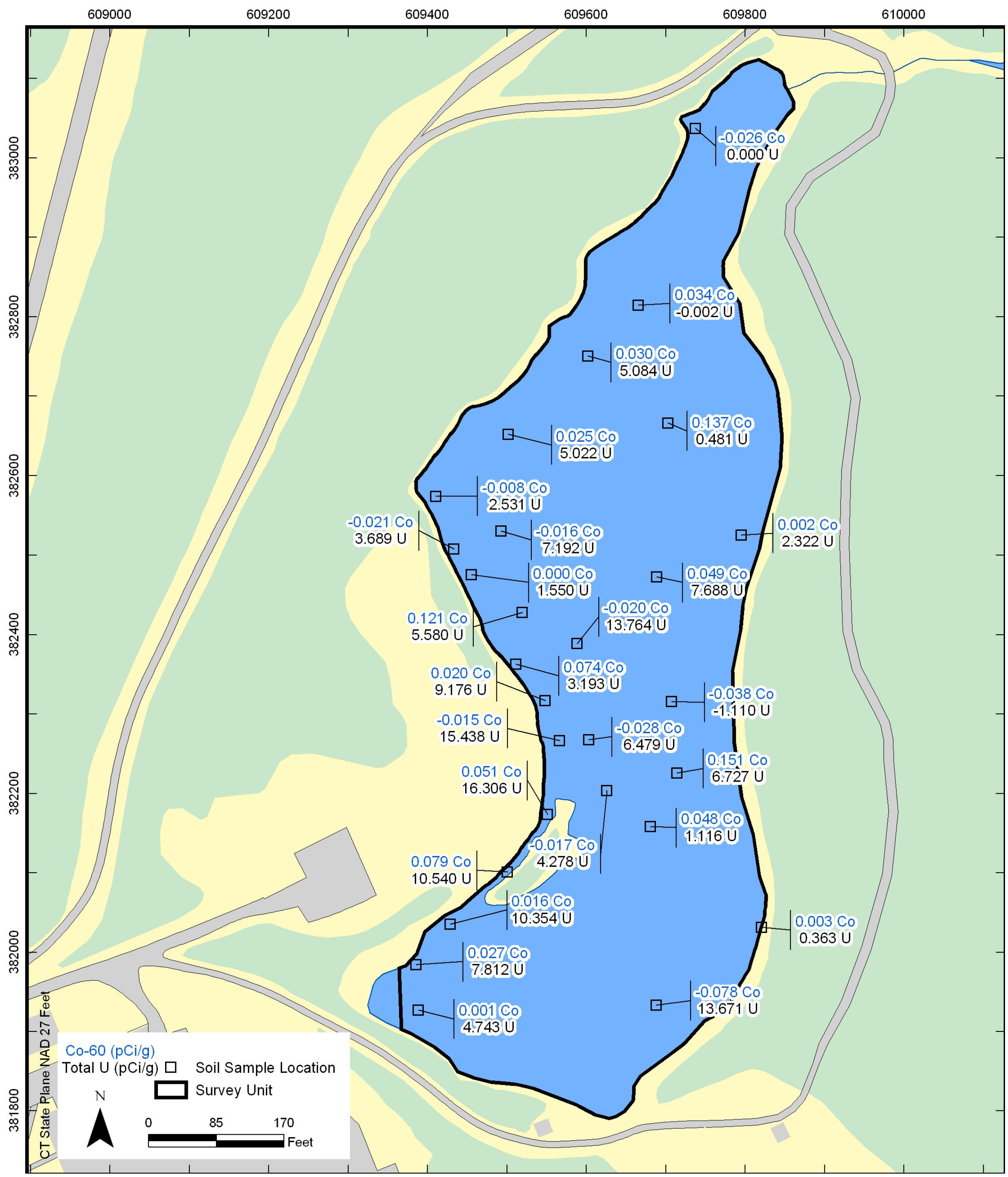
Survey Unit CE-FSS-25-01 is Small Pond and consists of approximately 34,500 square meters of land area. Figure 3.6 presents an overview of the survey unit. Twenty-eight survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix F.

3.2.6.1 Gamma Walkover Survey Results

No walkover survey was performed for this survey unit since it represents a body of surface water at the site. The number of volumetric samples collected from the sediment of Small Pond was increased to provide additional evaluation instead of scan surveys.

3.2.6.2 Volumetric Soil Sample Results

Twenty-eight randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-25-01 and analyzed by GEL. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.6 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-25-01.



Prepared/Date: DBW 03/08/07
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Figure 3.6: Survey Unit CE-FSS-25-01 Total U and Co-60 Activities (pCi/g)

3.2.7 Survey Unit CE-FSS-26-01

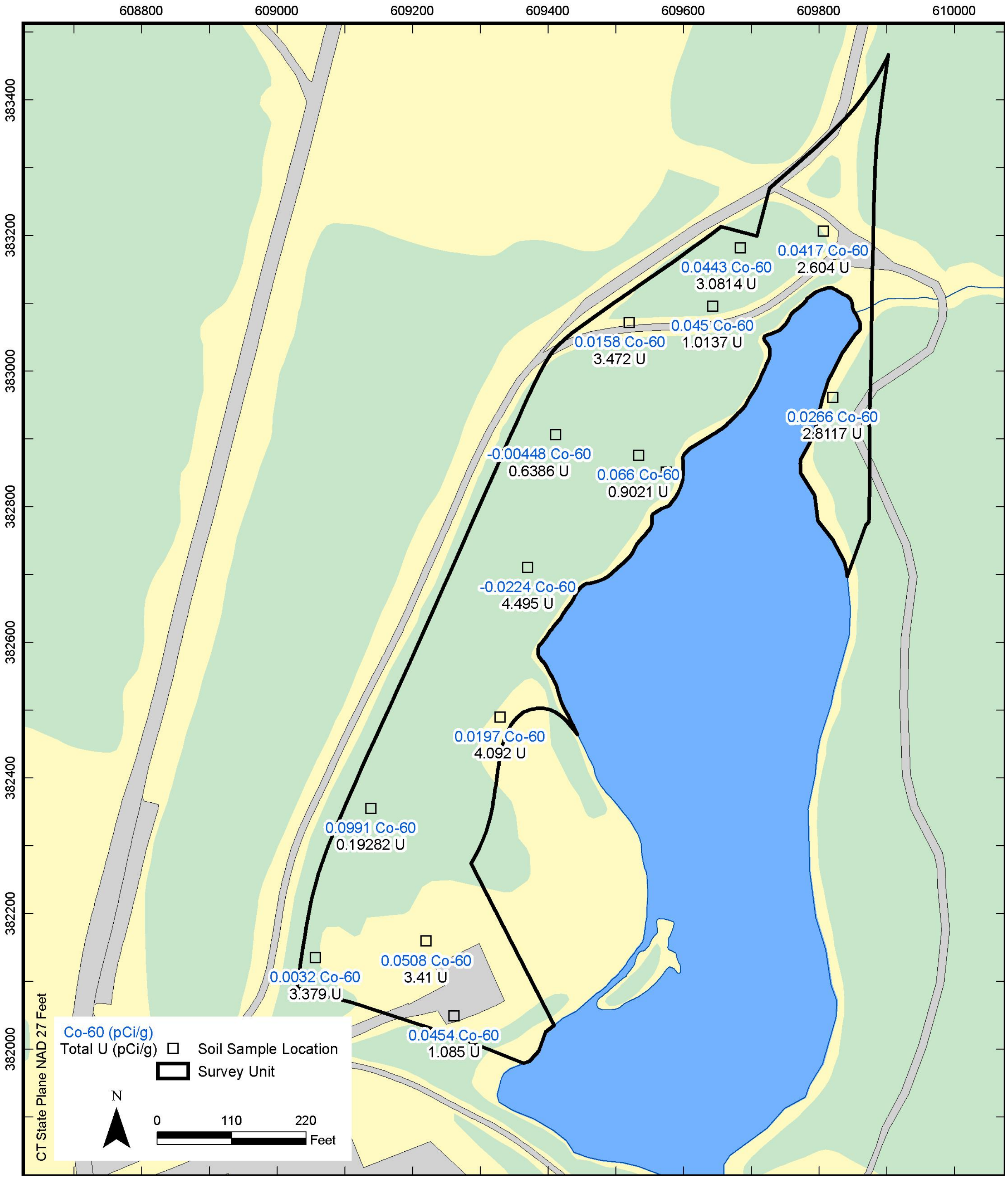
Survey Unit CE-FSS-26-01 is woods primarily to the west of Small Pond and consists of approximately 33,100 square meters of land area. Figure 3.7 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. 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 CE-FSS-26-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 806 cpm to 3,026 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.7.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-26-01 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.7 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-26-01.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.7: Survey Unit CE-FSS-26-01 Total U and Co-60 Activities (pCi/g)

3.2.8 Survey Unit CE-FSS-26-02

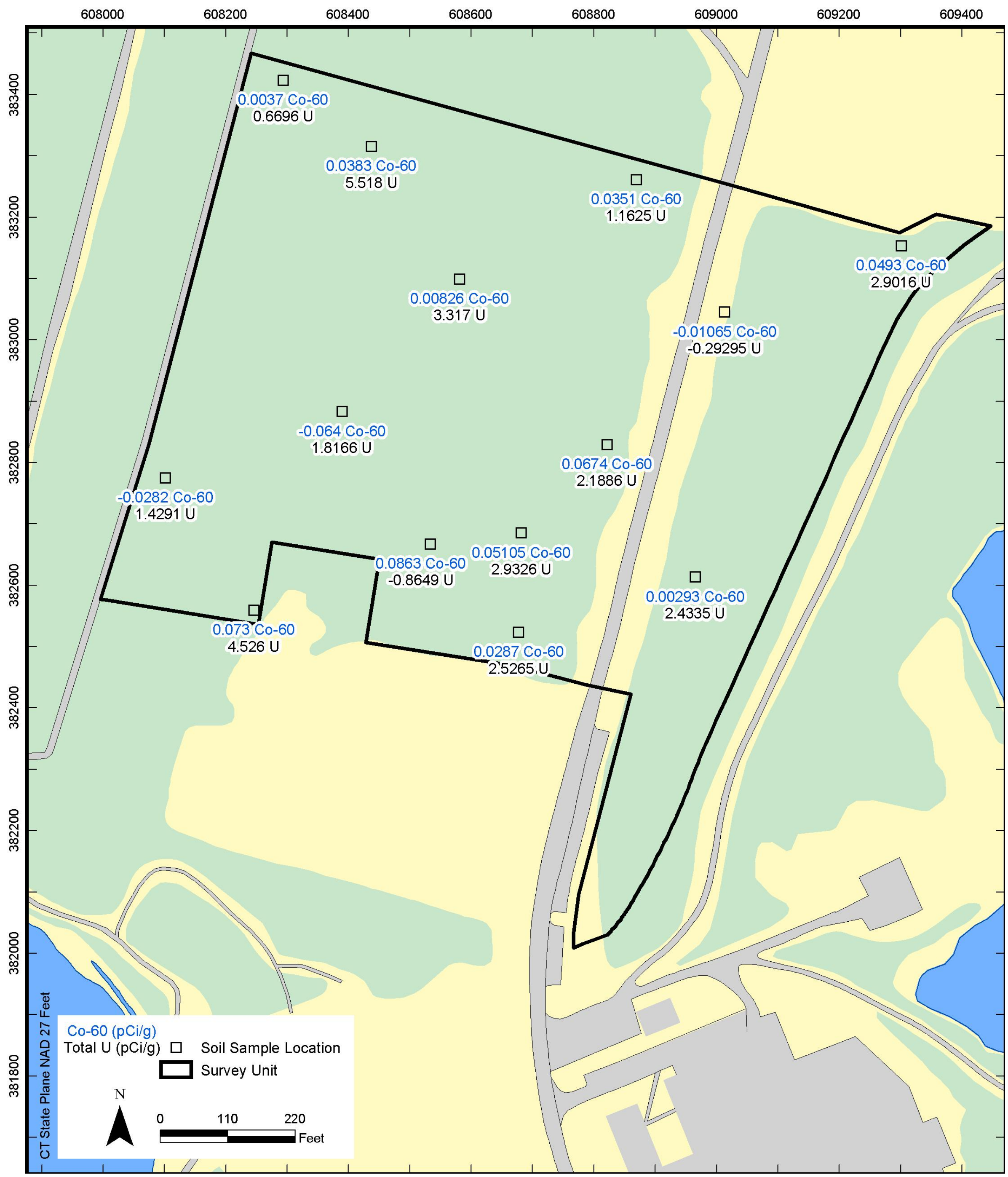
Survey Unit CE-FSS-26-02 is the woods to the north of the Building 17 Complex and to the south of the Building 2 Complex and consists of approximately 90,100 square meters of land area. Figure 3.8 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix H.

3.2.8.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-26-02 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 1,101 cpm to 3,065 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.8.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-26-02 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.8 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-26-02.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.8: Survey Unit CE-FSS-26-02 Total U and Co-60 Activities (pCi/g)

3.2.9 Survey Unit CE-FSS-26-03

Survey Unit CE-FSS-26-03 is the woods to the north and west of the Building 2 Complex and surrounding the drum burial and waste pad (FUSRAP) areas and consists of approximately 131,500 square meters of land area. Figure 3.9 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. 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 CE-FSS-26-03 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 569 cpm to 3,540 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected[2].

3.2.9.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-26-03 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.9 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-26-03.

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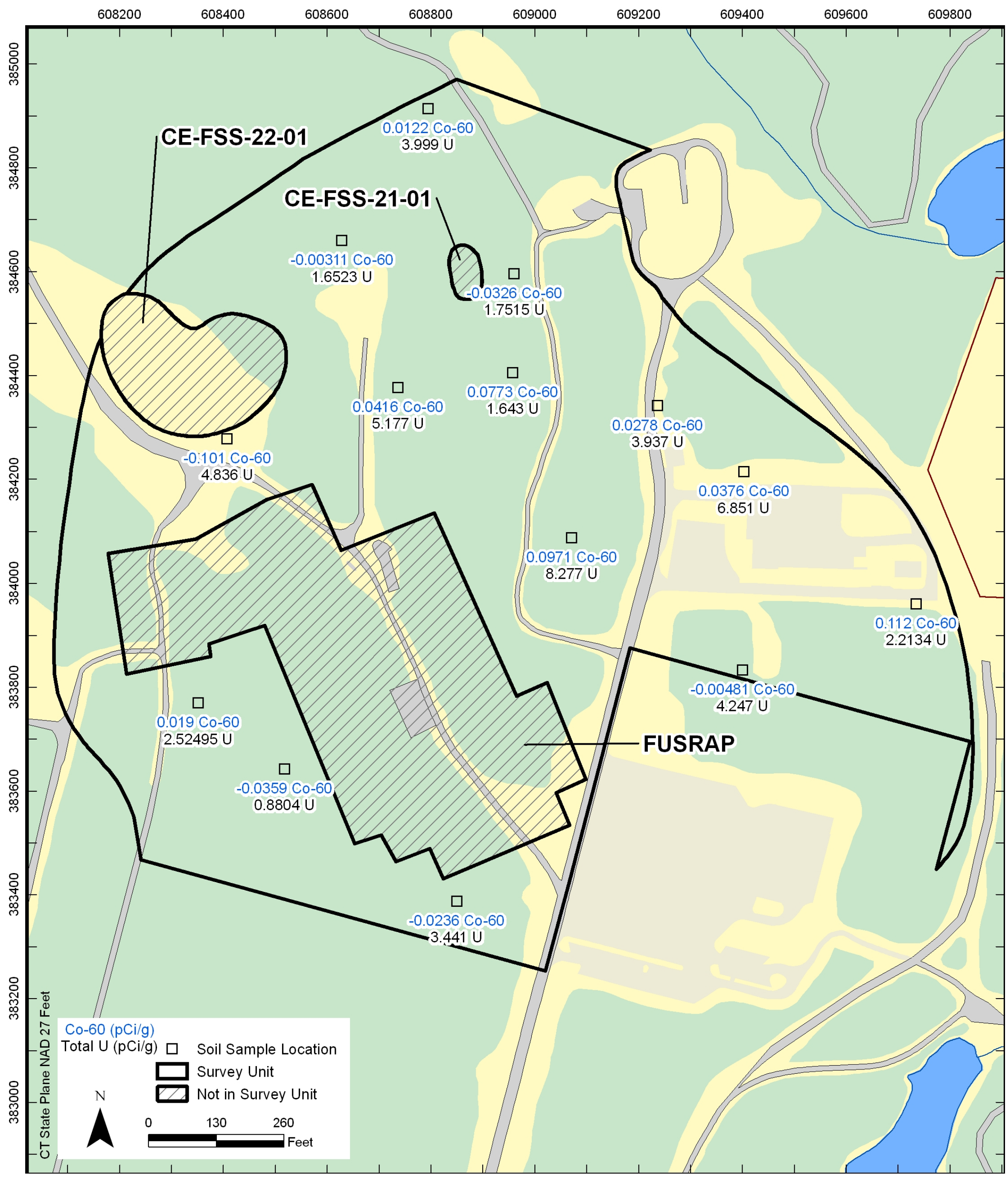
[2] A few small localized spots of elevated radioactivity have been identified in this portion of the Site. Biased sampling and analysis of these local anomalies spots has clearly identified the residual radioactivity as NORM, not uranium and Co-60 as addressed under the DP. This area will be addressed with the FUSRAP areas.



Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.9: Survey Unit CE-FSS-26-03 Total U and Co-60 Activities (pCi/g)

3.2.10 Survey Unit CE-FSS-26-04

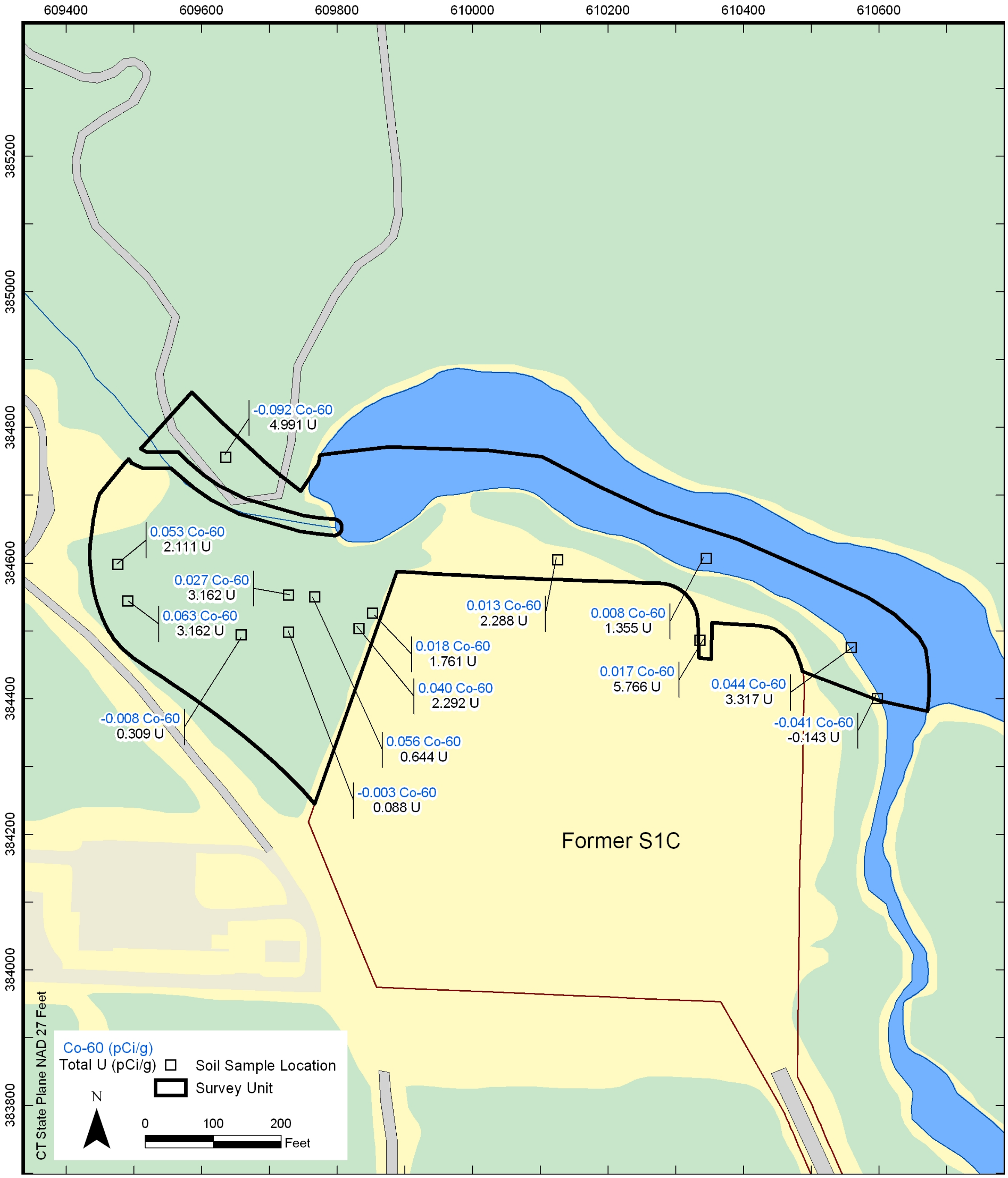
Survey Unit CE-FSS-26-04 is located to the north and west of the former S1C and consists of approximately 25,600 square meters of land area. Figure 3.10 presents an overview of the survey unit. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix J.

3.2.10.1Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-26-04 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 368 cpm to 2,834 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.10.2Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-26-04 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.10 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-26-04.



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Checked/Date: HTD 03/09/07

Figure 3.10: Survey Unit CE-FSS-26-04 Total U and Co-60 Activities (pCi/g)

3.2.11 Survey Unit CE-FSS-26-05

Survey Unit CE-FSS-26-05 is the eastern section of land adjacent to site brook next to the debris piles (FUSRAP) and outfalls into the brook and consists of approximately 9,200 square meters of land area. Figure 3.11 presents an overview of the survey unit. This survey unit is divided into two non-contiguous sections since it represents the land adjacent to site brook, which flows in between them. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix K.

3.2.11.1 Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-26-05 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 1,303 cpm to 5,331 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.11.2 Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-26-05 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.11 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-26-05.

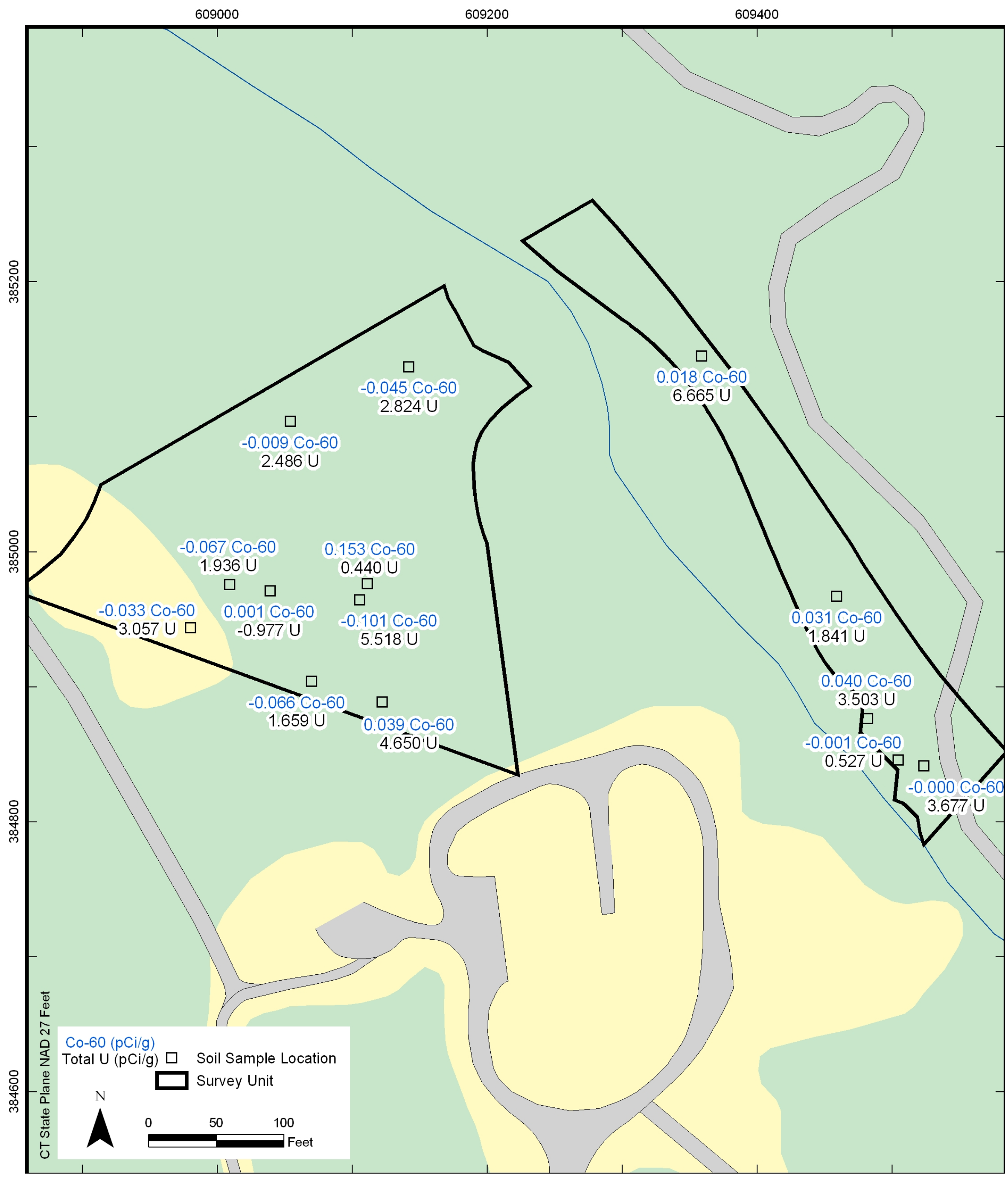

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Figure 3.11: Survey Unit CE-FSS-26-05 Total U and Co-60 Activities (pCi/g)

3.2.12 Survey Unit CE-FSS-26-06

Survey Unit CE-FSS-26-06 is the western section of land adjacent to the site brook and consists of approximately 41,300 square meters of land area. Figure 3.12 presents an overview of the survey unit. This survey unit is divided into two non-contiguous sections since it represents the land adjacent to site brook, which flows in between them. Fourteen survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix L.

3.2.12.1Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-26-06 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 824 cpm to 3,127 cpm (background range of variability) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

3.2.12.2Volumetric Soil Sample Results

Fourteen randomly placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-26-06 and analyzed on Site. The analytical results show that soil residual radioactivity is appreciably below the $DCGL_W$. Data assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.12 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-26-06.

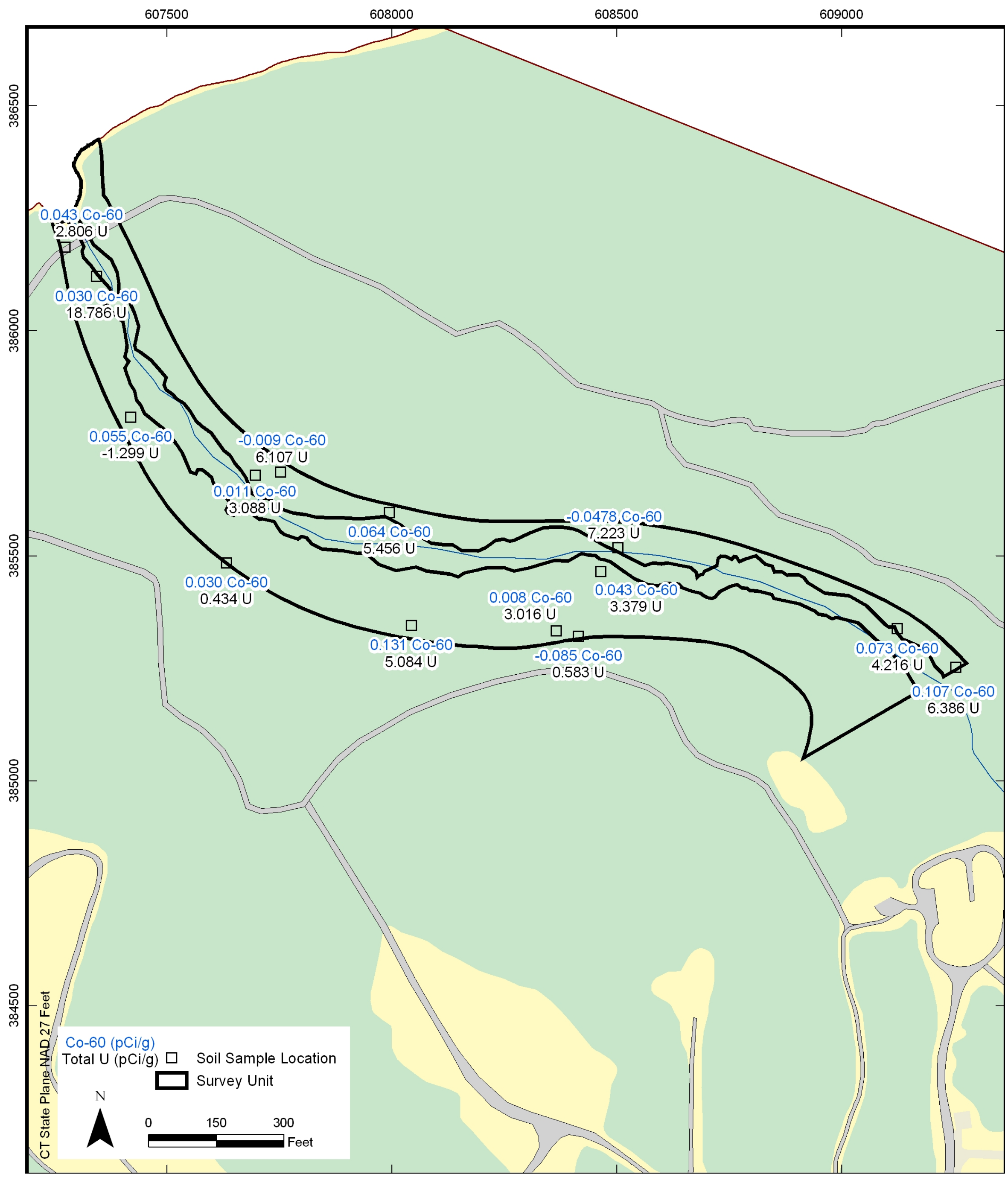

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Figure 3.12: Survey Unit CE-FSS-26-06 Total U and Co-60 Activities (pCi/g)

3.3 SURVEY SUMMARY RESULTS

This section provides a summary of the FSS results by survey unit and includes gamma walkover surveys, direct static measurements, and volumetric sample results.

3.3.1 Gamma Walkover Survey

Table 3.1 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 static counts performed in locations where a discernable increase in the count rate was identified. Gamma walkover surveys paths are identified on the applicable survey unit Radiological Survey Map, located in the survey unit specific appendix. Figure 3.13 indicates areas where gamma walkover surveys were performed.

Table 3.1: 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)**
20-01	3	10	0	2,200-3,000	2,600	2,990	390
21-01	3	10	0	2,700	2,700	3,139	439
22-01	2	25	0	2,700	2,700	2,980	280
23-01	3	10	0	1,500-2,500	2,000	2,822	822
24-01	3	10	0	2,700	2,700	3,121	421
25-01	3	0*	0	NA	NA	NA	NA
26-01	3	10	0	2,700	2,700	3,026	326
26-02	3	10	0	2,700	2,700	3,065	365
26-03	3	10	0	2,700	2,700	3,540	840
26-04	3	10	0	1,500-2,000	1,750	2,834	1,084
26-05	3	10	0	1,500-3,500	2,500	5,331	2,831
26-06	3	10	0	1,500-2,500	2,000	3,127	1,127

* 25-01 is Small Pond

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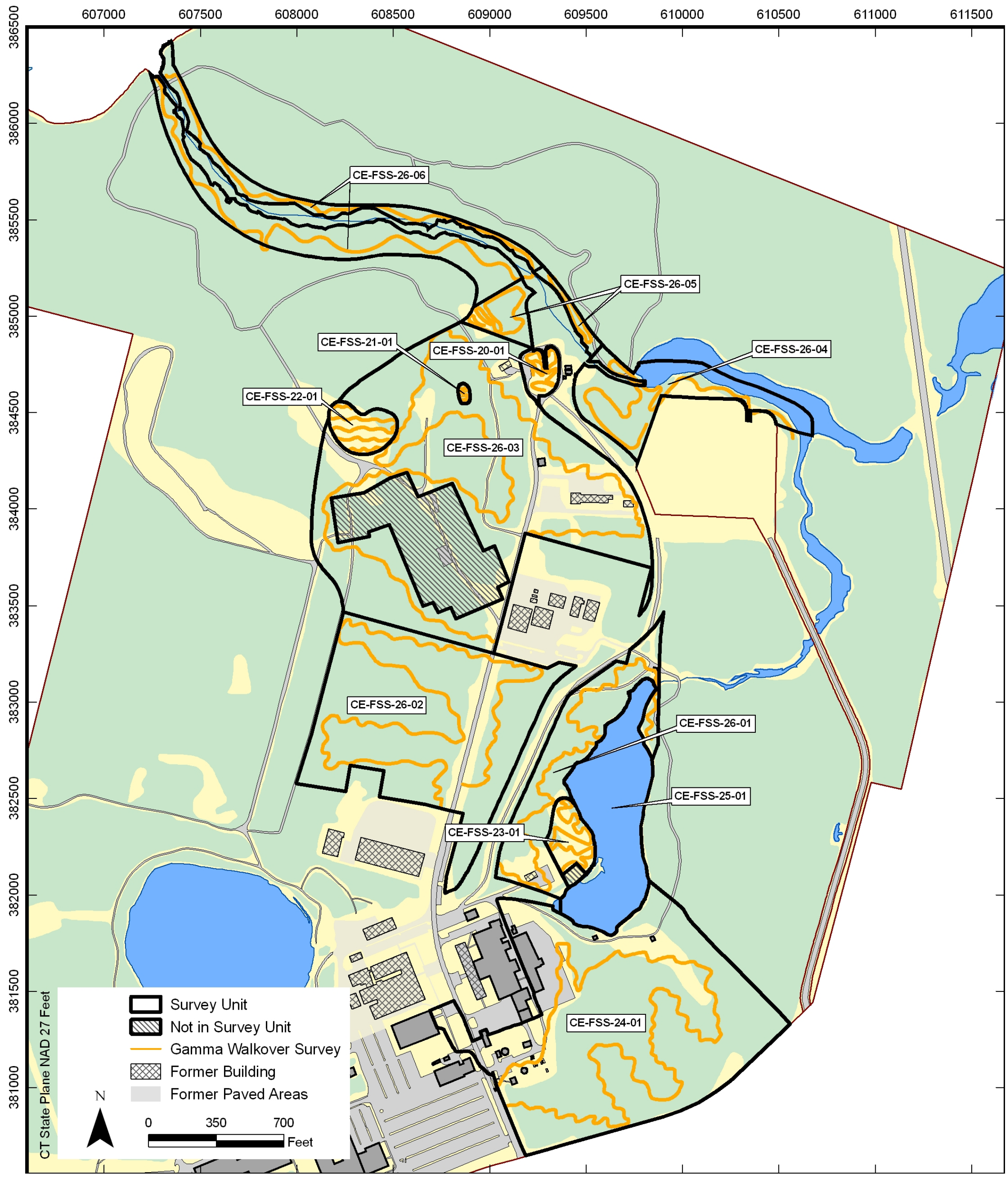

Prepared/Date: DBW 03/08/07
Checked/Date: HTD 03/09/07

Figure 3.13: Gamma Walkover Surveys

3.3.2 Direct Static Surface Measurements

In addition to gamma walkover surveys, 1-minute direct static surface measurements were performed at FSS volumetric soil sample locations using the gamma walkover NaI detector. Although not required by the FSS plan, these 1-minute static measurements were used as an additional gauge to help identify areas of elevated residual radioactivity and to support the conclusion that residual radioactivity in soil is less than the $DCGL_W$ for the survey units. Table 3.2 provides a summary of the direct static readings performed at each volumetric sampling location.

Table 3.2: Static Measurement Summary Results

Static Measurement Summary Results			
Survey Unit (CE-FSS)	**Number of Static Measurements Performed**	**Avg. Static Measurement Result (cpm, gross)**	**Avg. Static Measurement Result (cpm, net)**
20-01	14	2,542	-58
21-01	14	2,171	-529
22-01	20	2,558	-142
23-01	14	2,204	204
24-01	14	2,482	-219
25-01	0*	NA	NA
26-01	14	2,077	-623
26-02	14	2,791	91
26-03	14	2,616	-84
26-04	13	1,938	188
26-05	14	2,509	9
26-06	14	2,249	249

* 25-01 is Small Pond

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Individual static measurement results are presented in the survey unit data appendices (A through L) . 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.

3.2.3 Volumetric Sample Results

A summary of the FSS results is presented by survey unit in Table 3.3 (for total uranium) and Table 3.4 (for Co-60). These tables provide a statistical summary of the potentially impacted survey units.

Table 3.3: Summary Statistics, Total Uranium

Statistic	Survey Unit											
	CE-FSS-20-01	CE-FSS-21-01	CE-FSS-22-01	CE-FSS-23-01	CE-FSS-24-01	CE-FSS-25-01	CE-FSS-26-01	CE-FSS-26-02	CE-FSS-26-03	CE-FSS-26-04	CE-FSS-26-05	CE-FSS-26-06
Number of Measurements	14	14	20	14	14	28	14	14	14	14	14	14
Arithmetic Mean	3.44	6.76	2.96	3.11	3.74	5.86	2.29	2.15	3.67	2.24	2.70	4.66
Standard Deviation	1.99	3.94	1.97	1.80	1.89	4.89	1.46	1.72	2.13	1.77	2.07	4.75
Standard Error of the Mean	0.53	1.05	0.44	0.48	0.50	0.92	0.39	0.46	0.57	0.47	0.55	1.27
Coefficient of Variation	0.58	0.58	0.67	0.58	0.51	0.83	0.64	0.80	0.58	0.79	0.77	1.02
Geometric Mean	3.28	5.69	2.59	3.01	3.00	4.70	1.69	2.27	3.10	1.61	2.32	3.53
Maximum	7.8	14.4	5.1	5.4	7.4	16.3	4.5	5.5	8.3	5.8	6.7	18.8
Median	3.4	6.1	3.8	3.2	4.0	5.1	2.7	2.3	3.7	2.2	2.7	3.8
Minimum	-0.3	2.1	-1.1	-0.5	0.2	-1.1	0.2	-0.9	0.9	-0.1	-1.0	-1.3
Range	8.1	12.3	6.2	5.9	7.2	17.4	4.3	6.4	7.4	5.9	7.7	20.1
UCL_{95} (median)	4.2	8.6	4.3	4.6	5.1	7.2	3.4	2.9	4.8	3.2	3.7	6.1
LCL_{95} (median)	2.3	3.1	2.7	1.2	2.2	2.3	0.8	0.7	1.7	0.3	0.5	0.6

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Table 3.4: Summary Statistics, Co-60

Statistic	Survey Unit											
	CE-FSS-20-01	CE-FSS-21-01	CE-FSS-22-01	CE-FSS-23-01	CE-FSS-24-01	CE-FSS-25-01	CE-FSS-26-01	CE-FSS-26-02	CE-FSS-26-03	CE-FSS-26-04	CE-FSS-26-05	CE-FSS-26-06
Number of Measurements	14	14	20	14	14	28	14	14	14	14	14	14
Arithmetic Mean	0.010	0.039	0.025	0.034	0.015	0.021	0.035	0.024	0.016	0.014	-0.003	0.032
Standard Deviation	0.064	0.051	0.052	0.052	0.045	0.053	0.032	0.042	0.057	0.042	0.062	0.057
Standard Error of the Mean	0.017	0.014	0.012	0.014	0.012	0.010	0.009	0.011	0.015	0.011	0.017	0.015
Coefficient of Variation	6.13	1.31	2.09	1.52	3.03	2.50	0.90	1.71	3.56	3.00	-22.40	1.76
Geometric Mean	0.030	0.040	0.036	0.043	0.026	0.026	0.034	0.026	0.041	0.028	0.023	0.041
Maximum	0.100	0.115	0.156	0.134	0.09	0.151	0.099	0.086	0.112	0.063	0.153	0.131
Median	0.017	0.052	0.032	0.030	0.016	0.009	0.043	0.032	0.016	0.018	-0.001	0.036
Minimum	-0.121	-0.064	-0.066	-0.039	-0.085	-0.078	-0.022	-0.064	-0.101	-0.092	-0.101	-0.085
Range	0.221	0.179	0.222	0.173	0.175	0.229	0.122	0.150	0.213	0.155	0.254	0.216
UCL_{95} (median)	0.070	0.072	0.058	0.055	0.050	0.030	0.051	0.051	0.042	0.044	0.031	0.064
LCL_{95} (median)	-0.031	-0.014	-0.015	-0.021	-0.017	-0.016	0.003	-0.011	-0.033	-0.008	-0.066	-0.009

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4.0 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 a survey unit from radiological controls.

4.1 DECISION RULES

IF the evaluation of the Final Status Survey data from a single survey unit indicates that:

- Each volumetric soil sample measurement results is 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 As Low as Reasonably Achievable (ALARA):

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

An ALARA analysis in agreement with NRC guidance provided in NUREG-1727 was performed as part of the DP. The analysis shows 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, 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.

4.2 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 4.1. Because the DCGL was developed for total uranium (the sum of 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 4.1: Compliance Comparison of Soil Metrics

Metric		Survey Unit											
		CE-FSS-20-01	CE-FSS-21-01	CE-FSS-22-01	CE-FSS-23-01	CE-FSS-24-01	CE-FSS-25-01	CE-FSS-26-01	CE-FSS-26-02	CE-FSS-26-03	CE-FSS-26-04	CE-FSS-26-05	CE-FSS-26-06
Unity	Power of Sign Test	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1
Total U	Median	3.40	6.05	3.75	3.15	3.95	5.05	2.70	2.30	3.65	2.20	2.65	3.80
	UCL_{95} of Median	4.2	8.6	4.3	4.6	5.1	7.2	3.4	2.9	4.8	3.2	3.7	6.1
	Arithmetic Mean	3.44	6.76	2.96	3.11	3.74	5.86	2.29	2.15	3.67	2.24	2.70	4.66
	Geometric Mean	3.28	5.69	2.59	3.01	3.00	4.70	1.69	2.27	3.10	1.61	2.32	3.53
	Maximum	7.8	14.4	5.1	5.4	7.4	16.3	4.5	5.5	8.3	5.8	6.7	18.8
Co-60	Median	0.017	0.052	0.032	0.030	0.016	0.009	0.043	0.032	0.016	0.018	-0.001	0.036
	UCL_{95} of Median	0.070	0.072	0.058	0.055	0.050	0.030	0.051	0.051	0.042	0.044	0.031	0.064
	Arithmetic Mean	0.010	0.039	0.025	0.034	0.015	0.021	0.035	0.024	0.016	0.014	-0.003	0.032
	Geometric Mean	0.030	0.040	0.036	0.043	0.026	0.026	0.034	0.026	0.041	0.028	0.023	0.041
	Maximum	0.100	0.115	0.156	0.134	0.090	0.151	0.099	0.086	0.112	0.063	0.153	0.131

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 concentration do not exceed the $DCGL_W$s. Note in the Compliance Test Statistics Report (survey unit specific appendices) 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, 8.6 pCi/g, is less than the $DCGL_W$ by a factor of more than 64, and the highest Co-60 UCL_{95} estimate of the median, 0.072 pCi/g, is less than the $DCGL_W$ by a factor of more than 69. 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.

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4.3 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.

4.3.1 DCGL Compliance

The average and median uranium and Co-60 concentrations in soils for all survey units 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 18.8 pCi/g and 0.156 pCi/g respectively, significantly below the $DCGL_W$ value of 557 pCi/g for uranium and 5.0 pCi/g for Co-60. Sum of fraction (unity) values were well below 0.1. No locally elevated concentrations of residual radioactivity were identified above the volumetric or walkover (scan) investigation levels.

4.3.2 Sample Size and Statistical Power

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 survey unit data appendices (A through L), 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.

5.0 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 residual radiological conditions in the soils of the various survey units within the potentially impacted areas. The DQA uses guidance from MARSSIM and professional judgment.

5.1 QUALITY ASSURANCE

The goal of quality assurance (QA) 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. QA was achieved through three primary approaches: data management, sample custody, and QC measurements.

5.1.1 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 radiation safety officer (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 survey unit data appendices (A through L).

5.1.2 Sample Custody

Sample quality, related to sample collection, was controlled through the use of trained personnel implementing approved , written 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 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 GEL for off-site analysis are provided in the survey unit data appendices (A through L).

5.1.3 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.

5.1.3.1 Volumetric Duplicate 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 at a frequency of no less than 5 percent (1:20) for the sample population expected, and were submitted to the on-site gamma spectroscopy system for analysis as duplicate samples. Every survey unit was represented with duplicate samples being collected from that survey unit. A total of 20 duplicate samples were collected from an overall sample population of 188 volumetric samples, equating to a sampling frequency significantly greater than the 1:20 minimum requirement. 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 precise (Table 5.1).

Table 5.1: Duplicate Sample Measurement Results

Soil Remaining Site Impacted Areas						
Sample ID	**Co-60**			**U-235**		
	Activity	**Uncert.**	**MDC**	**Activity**	**Uncert.**	**MDC**
FSS-CE-001	-3.96E-02	7.78E-02	1.23E-01	6.48E-02	1.30E-01	2.43E-01
FSS-CE-001DUP	2.16E-02	8.41E-02	1.77E-01	1.86E-01	1.17E-01	2.29E-01
FSS-CE-009	5.32E-02	9.06E-02	2.12E-01	8.55E-02	1.37E-01	2.56E-01
FSS-CE-009DUP	4.83E-02	7.98E-02	1.76E-01	8.83E-02	1.06E-01	2.00E-01
FSS-CE-022	3.89E-02	8.05E-02	1.77E-01	9.15E-02	1.14E-01	2.16E-01
FSS-CE-022DUP	6.29E-02	7.46E-02	1.77E-01	5.24E-02	1.02E-01	1.91E-01
FSS-CE-044	3.96E-02	7.93E-02	2.04E-01	4.27E-01	1.82E-01	3.71E-01
FSS-CE-044DUP	1.36E-02	1.09E-01	2.11E-01	2.79E-01	1.28E-01	2.94E-01
FSS-CE-060	-3.38E-02	1.03E-01	1.95E-01	9.56E-02	1.44E-01	2.71E-01
FSS-CE-060DUP	7.18E-02	9.33E-02	2.20E-01	6.73E-02	1.17E-01	2.23E-01
FSS-CE-070	4.21E-02	9.59E-02	2.09E-01	3.02E-02	1.45E-01	2.64E-01
FSS-CE-070DUP	6.00E-02	1.18E-01	2.51E-01	1.59E-01	1.48E-01	2.85E-01
FSS-CE-074	2.89E-02	8.66E-02	1.83E-01	-4.45E-02	1.09E-01	1.87E-01
FSS-CE-074DUP	-5.02E-02	1.09E-01	1.94E-01	2.56E-02	1.17E-01	2.13E-01
FSS-CE-088	-1.33E-02	8.42E-02	1.60E-01	1.01E-01	1.34E-01	2.50E-01
FSS-CE-088DUP	1.97E-02	1.14E-01	2.25E-01	1.17E-01	1.32E-01	2.53E-01
FSS-CE-123	6.43E-02	1.05E-01	2.53E-01	7.62E-02	1.68E-01	3.13E-01
FSS-CE-123DUP	7.64E-02	1.30E-01	2.83E-01	7.39E-02	1.66E-01	3.09E-01
FSS-CE-134	5.37E-02	1.28E-01	2.63E-01	2.22E-01	1.63E-01	3.19E-01
FSS-CE-134DUP	-7.57E-03	1.00E-01	2.05E-01	3.45E-02	1.45E-01	2.69E-01
FSS-CE-140	-1.17E-02	9.99E-02	1.92E-01	3.06E-02	1.10E-01	2.05E-01
FSS-CE-140DUP	2.84E-02	7.72E-02	1.75E-01	5.68E-02	1.25E-01	2.30E-01
FSS-CE-145	6.88E-02	1.03E-01	2.47E-01	5.58E-02	1.57E-01	2.92E-01
FSS-CE-145DUP	1.03E-02	9.93E-02	2.31E-01	9.21E-02	1.92E-01	3.61E-01
FSS-CE-162	-4.17E-02	1.28E-01	1.88E-01	-6.30E-02	1.40E-01	2.45E-01
FSS-CE-162DUP	1.36E-01	9.70E-02	2.64E-01	1.26E-01	1.49E-01	2.85E-01
FSS-CE-168	8.46E-02	9.74E-02	2.43E-01	1.86E-01	1.44E-01	2.84E-01
FSS-CE-168DUP	7.06E-02	1.05E-01	2.48E-01	1.12E-01	1.46E-01	2.79E-01
FSS-CE-184	-9.68E-02	1.39E-01	2.13E-01	-1.81E-02	1.61E-01	2.87E-01
FSS-CE-184DUP	-3.77E-02	1.03E-01	1.85E-01	1.43E-01	1.46E-01	2.83E-01
FSS-CE-192	9.06E-03	9.24E-02	1.92E-01	2.15E-01	1.38E-01	2.72E-01
FSS-CE-192DUP	-9.87E-03	8.32E-02	1.58E-01	2.22E-02	1.38E-01	2.52E-01
FSS-CE-199	6.05E-02	8.91E-02	2.09E-01	-1.23E-02	1.22E-01	2.18E-01
FSS-CE-199DUP	-8.62E-04	8.65E-02	1.70E-01	4.03E-02	1.27E-01	2.34E-01
FSS-CE-210	1.39E-01	1.41E-01	3.36E-01	2.36E-01	1.66E-01	3.30E-01
FSS-CE-210DUP	7.46E-02	1.22E-01	2.84E-01	1.76E-01	1.63E-01	3.18E-01
SD-194900	-2.16E-02	2.87E-02	4.74E-02	8.72E-02	1.04E-01	2.01E-01
SD-194900DUP	6.01E-03	3.97E-02	6.63E-02	7.61E-02	1.56E-01	2.84E-01

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The overall quality of the volumetric soil sample data is evident in the graphic presentation in Figure 5.1 (U-235) and Figure 5.2 (Co-60).

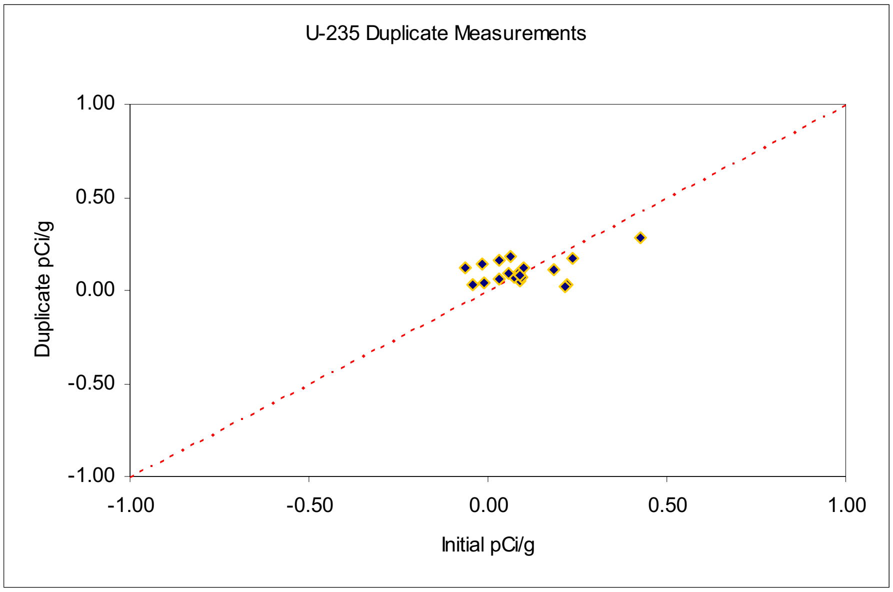

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Figure 5.1: U-235 Duplicate Measurement Result Comparisons

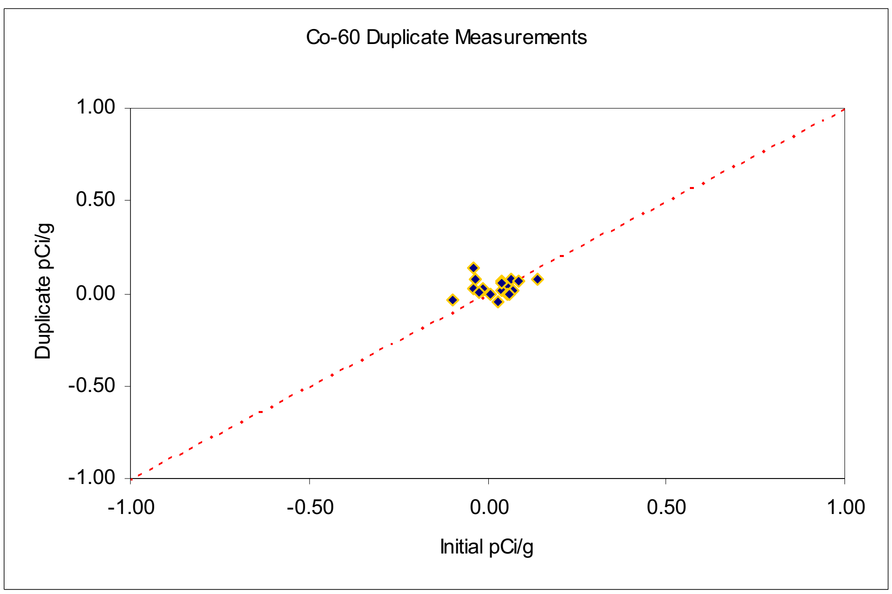

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Figure 5.2: Co-60 Duplicate Measurement Result Comparisons

5.1.3.2 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 instrument was response checked to a known source. The Survey Instrument Response check data sheet is provided in Appendix M.

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

A control chart for the instrument was created to evaluate the instruments' responses to the radioactive source over the sampling period time frame. No degradation or unexplained variability of the instruments' response was observed during the performance of FSS. A control chart and supporting data for the field instrument is provided in Appendix M.

5.1.3.3 Laboratory Instruments

The prescribed QC for laboratory instruments consists of instrument source response checks, energy calibration checks, efficiency calibration checks, background checks, and replicate volumetric measurements performed on a percentage of the samples collected using an off-site system. 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 kilo-electron volts [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 N. 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 replicate measurements of the sample using independent, off-site, analytical equipment. Replicate counting of samples was performed by GEL. A total of 20 volumetric samples obtained from the Site during FSS activities were analyzed by the on-site gamma spectroscopy system and then sent to GEL for isotopic analysis by gamma spectroscopy (HPGe).

To assess the comparability between the initial and replicate measurements, a simple linear regression analysis was performed and is graphically presented in Figure 5.3 (U-235) and Figure 5.4 (Co-60) for sample activities near or at background activity values. Tabular comparison of on-site to laboratory GEL analytical results, along with GEL Certificates of Analysis, are presented in Appendix N.

In addition to the regression analysis of the replicate data sets for the replicate measurements, two-sample comparison density traces of the data set are presented in Figure 5.5 (for U-235) and Figure 5.6 (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 accuracy 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% (UCL_{95}) was specified.

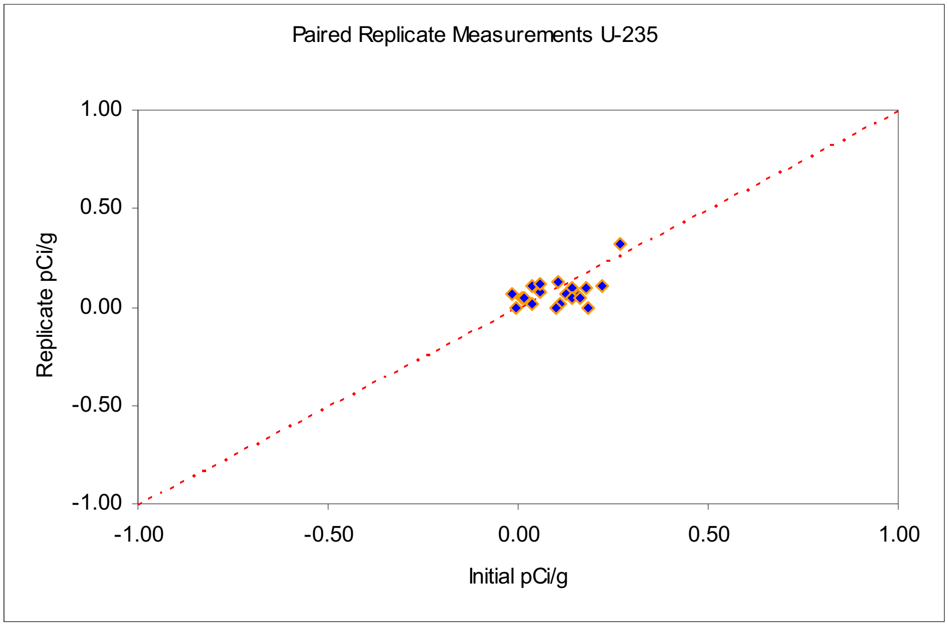

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Figure 5.3: U-235 Comparison Between Replicate Measurements

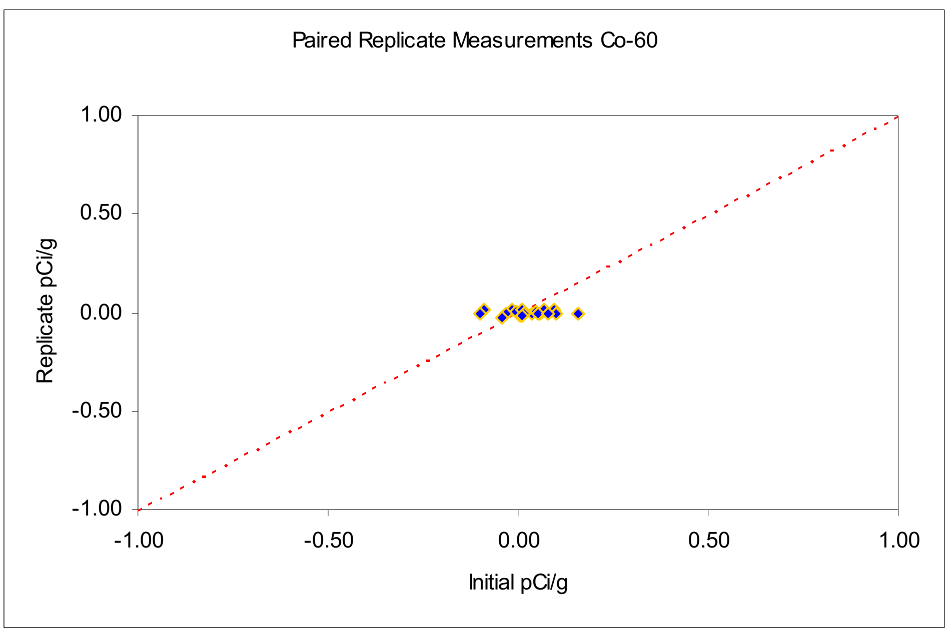

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Figure 5.4: Co-60 Comparison Between Replicate Measurements

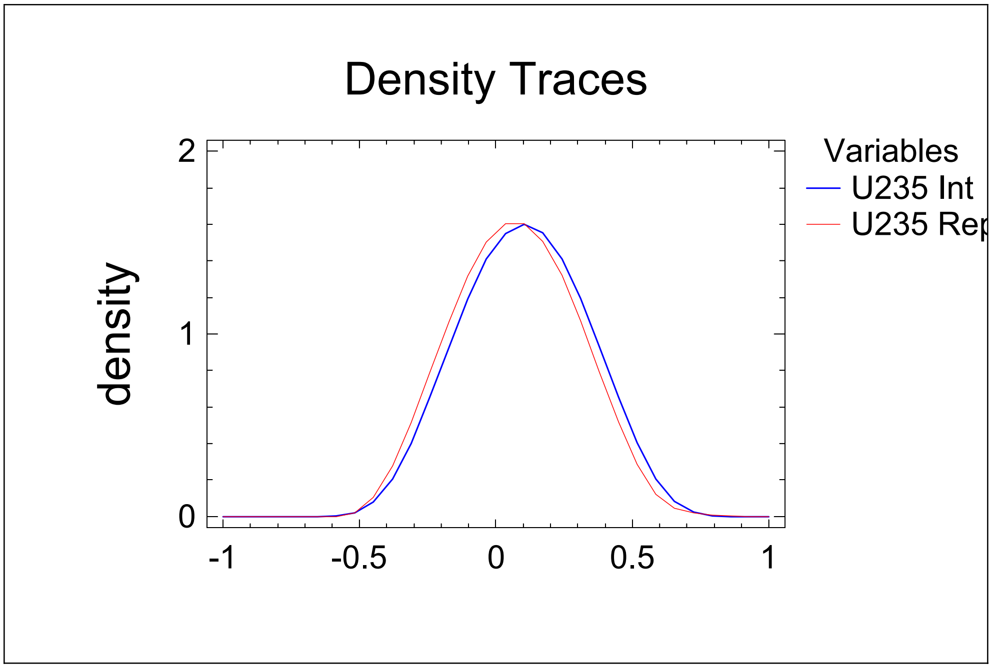

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Figure 5.5: U-235 Two-Sample Comparison of Density for Replicate Measurements

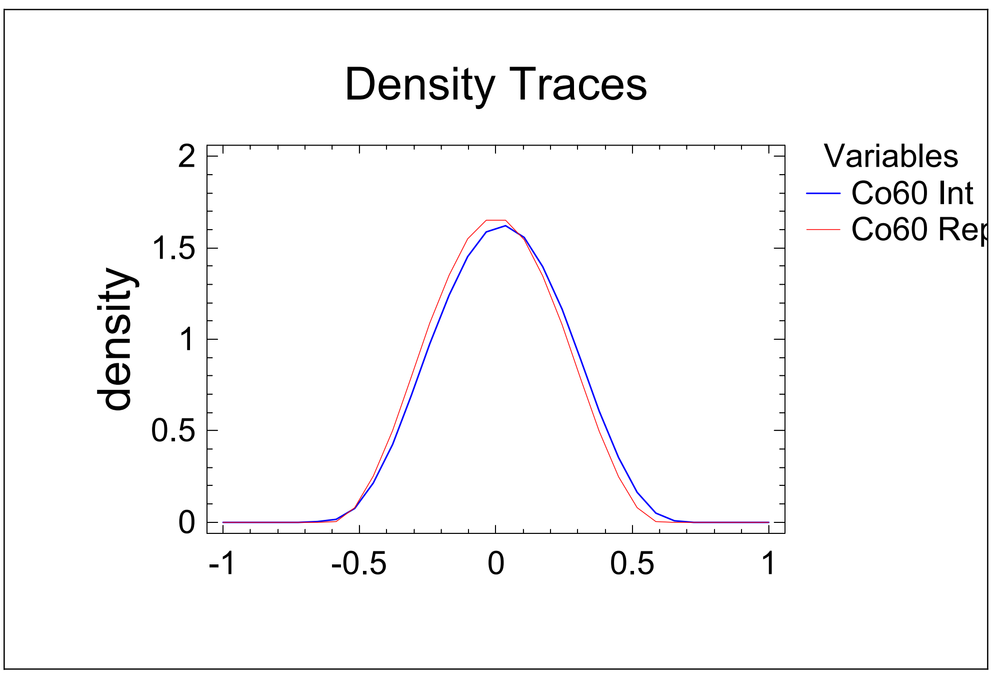

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Figure 5.6: Co-60 Two-Sample Comparison of Density for Replicate Measurements

5.2 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 would 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 potentially impacted areas, FSS activities were evaluated against the project goals developed for project. Table 5.2 presents the target DQIs and summarizes the post-sampling data quality assessment.
Inspection of Table 5.2 indicates that the DQIs were achieved, and thus, 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 $DCGL_W$.

5.3 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 and acceptable for its intended use.

Table 5.2: 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 of 20 volumetric soil samples from Survey Unit CE-FSS-22-01or 14 volumetric soil samples from the rest of the Survey Units was planned. 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 188 volumetric soil samples were actually collected from all survey units and each survey unit had at least its minimum number of samples collected.	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 would 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 remaining impacted Site areas. No critical deviation from these procedures was encountered.	DQI accepted
Representativeness	Non-representativeness increases or decreases Type I error depending on the bias.	Sample allocation included a minimum number of unbiased, randomly distributed sample locations based on survey design.	Sample allocation for 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 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 were governed by procedures. Duplicate volumetric samples, background measurements, and source response check measurements were used to gauge reproducibility.	All sampling and field measurement processes were controlled by approved written procedures. The specified minimum number of duplicate (split) volumetric samples (10) was obtained. Duplicate 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 DCGLs. All procedures were implemented. Duplicate 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 were governed by procedures. Instruments were 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

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6.0 SUMMARY AND CONCLUSIONS

On the basis of the analyses presented in this report, the data demonstrates that each of the survey units associated with the potentially impacted areas has met the decision criteria.

More specifically, the FSS of the potentially impacted areas soils demonstrates that:

- No unexpected results or trends are evident in the data.
- The sampling and survey results demonstrate that soil residual radioactivity in the potentially impacted areas 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 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— should be rejected for each of the survey units in the potentially impacted areas. The areas surveyed and sampled during FSS should be released from further radiological controls.

7.0 REFERENCES

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