FINAL STATUS SURVEY REPORT

SUBMITTAL NUMBER 1

CE WINDSOR SITE
WINDSOR, CONNECTICUT

VOLUME I
TEXT, FIGURES AND TABLES

US NRC LICENSE NUMBER 06-00217-06
DOCKET NUMBER 030-03754

JULY 2011

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(PROVIDED IN VOLUME II)

μR/h	microrem per hour
ABB	ABB Inc.
AD	Anderson - Darling
AEC	Atomic Energy Commission
ALARA	As Low as Reasonably Achievable
AMEC	AMEC Environment & Infrastructure (formerly MACTEC)
ANSI	American National Standards Institute
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
cm	centimeter
Co-60	cobalt 60
COA	certificate of analysis
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
DP	Decommissioning Plan
DQA	Data Quality Analysis
DQI	Data Quality Indicator
EMC	elevated measurement comparison
EPA	Environmental Protection Agency
FSS	Final Status Survey (radiological)
FSSP	Final Status Survey Plan
FWHM	full width at half-maximum
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
IWL	Industrial Waste Line
keV	kilo-electron volts

LEU	low enriched uranium
LBGR	lower bound of the gray region
LLRW	low-level radioactive waste
LR	laboratory recounts
m^2	meter(s) squared
MACTEC	MACTEC, Inc.
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCA	multi-channel analyzer
MDA	minimum detectable activity
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
NIST	National Institute of Standards and Technology
NORM	naturally occurring radioactive material
NRC	Nuclear Regulatory Commission
pCi/g	picocuries per gram
QA	quality assurance
QC	quality control
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
VSP	Visual Sample Plan computer program
WWTP	wastewater treatment plant

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EXECUTIVE SUMMARY

From the mid-1950s, the Combustion Engineering, Inc. (CE) Site at 2000 Day Hill Road in Windsor Connecticut (Site)was involved in research, development, engineering, production, and servicing of nuclear fuels, systems, and services until 2000. The site is undergoing decommissioning that will lead to license termination and unrestricted release in accordance with the requirements of the License Termination Rule at 10 CFR Part 20, Subpart E. This Final Status Survey (FSS) Report provides the design, field implementation and results of FSSs conducted for a portion of the Site in support of decommissioning activities. It is the first of seven FSS Reports that will cover the remaining 248 acres of the Site under U.S. NRC license 06-00216-06. This report specifically addresses the following areas: Industrial Waste Lines (IWLs), Waste Water Treatment Plant, and East Main Street Sanitary Line.

Remediation is complete in the areas addressed in this FSS Report. In accordance with the Site Decommissioning Plan (DP), underground utilities (industrial waste lines and sanitary waste lines) as well as structures in these areas have been removed. Low-level radioactive waste generated as part of these activities has been properly disposed at a licensed disposal facility. The FSS units presented in this Report represent the post-remediation condition (i.e., trench for underground utility removal).

The FSS did not identify residual radioactivity in excess of the applicable soil radioactivity release criteria. For the portions of the Site provided in this report, fifteen survey units were created in support of the FSS, including eight Class 1 survey units, three Class 2 survey units and four 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 following the Site FSS Plan (FSSP). Site-specific soil derived concentration guideline levels (DCGLs) have been derived as part of the decommissioning process. The DCGLs established for soil are 557 picocuries per gram (pCi/g) for total uranium and 5 pCi/g for cobalt 60 (Co-60), representing the byproduct source term. Site-specific DCGLs for two additional radionuclides (radium and thorium) have also been derived, but do not apply to the portions of the Site addressed by this FSS Report.

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. For this FSS Report, the Sign Test will be the statistical test for compliance evaluation since background concentrations of the DCGLs are insignificant. As described in the FSSP (AMEC, 2011), the Sign Test is a one-sample, non-parametric test that can be used to evaluate compliance with the DCGL.

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.

For the areas addressed by this FSS Report, the final radiological status survey of the soils at the Site concludes that in each survey unit all of the conditions and requirements for unrestricted radiological release have been met. This FSS Report submittal supports the regulatory decision to terminate the license following completion of all FSS report submittals for the Site.

1.0 INTRODUCTION

This radiological FSS Report documents the radiological status of a portion of the CE Windsor Site in Windsor, Connecticut. Presently, 2000 Day Hill Rd., Windsor, Connecticut is subject to U.S. NRC Radioactive Materials License No. 06-00217-06 (NRC, 2004) 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 the License Termination Rule at 10 CFR Part 20, Subpart E and to terminate NRC license No. 06-00217-06. The Site has been undergoing phased decommissioning, and this FSS Report is the first of seven reports that will document the final condition of the Site in preparation for license termination. This report documents the final radiological status of the Industrial Waste Lines, Waste Water Treatment Plant and East Main Sanitary Line. 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 revised DP (MACTEC, 2010a), and site-specific DCGLs (MACTEC, 2003 and MACTEC, 2010b). 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 the FSSP (AMEC, 2011) which incorporates methods outlined in 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, 2006); and
- NRC Radioactive Material License No. 06-00217-06 (NRC, 2011).

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 and sampling results and data evaluations are presented in Section 3. Section 4 evaluates FSS 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. Appendices are included for discrete survey units 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 612 acres and is located at 2000 Day Hill Road, in Windsor, Connecticut. An overview of the site layout is shown on Figure 1.2. The NRC issued a license amendment to Byproduct License 06-00217-06 which authorizes a partial site release of 365 contiguous acres of the 612 acre facility for unrestricted use (NRC, 2011). The remaining 248 acres remains under NRC jurisdiction for completion of decommissioning and eventual license termination for unrestricted use.

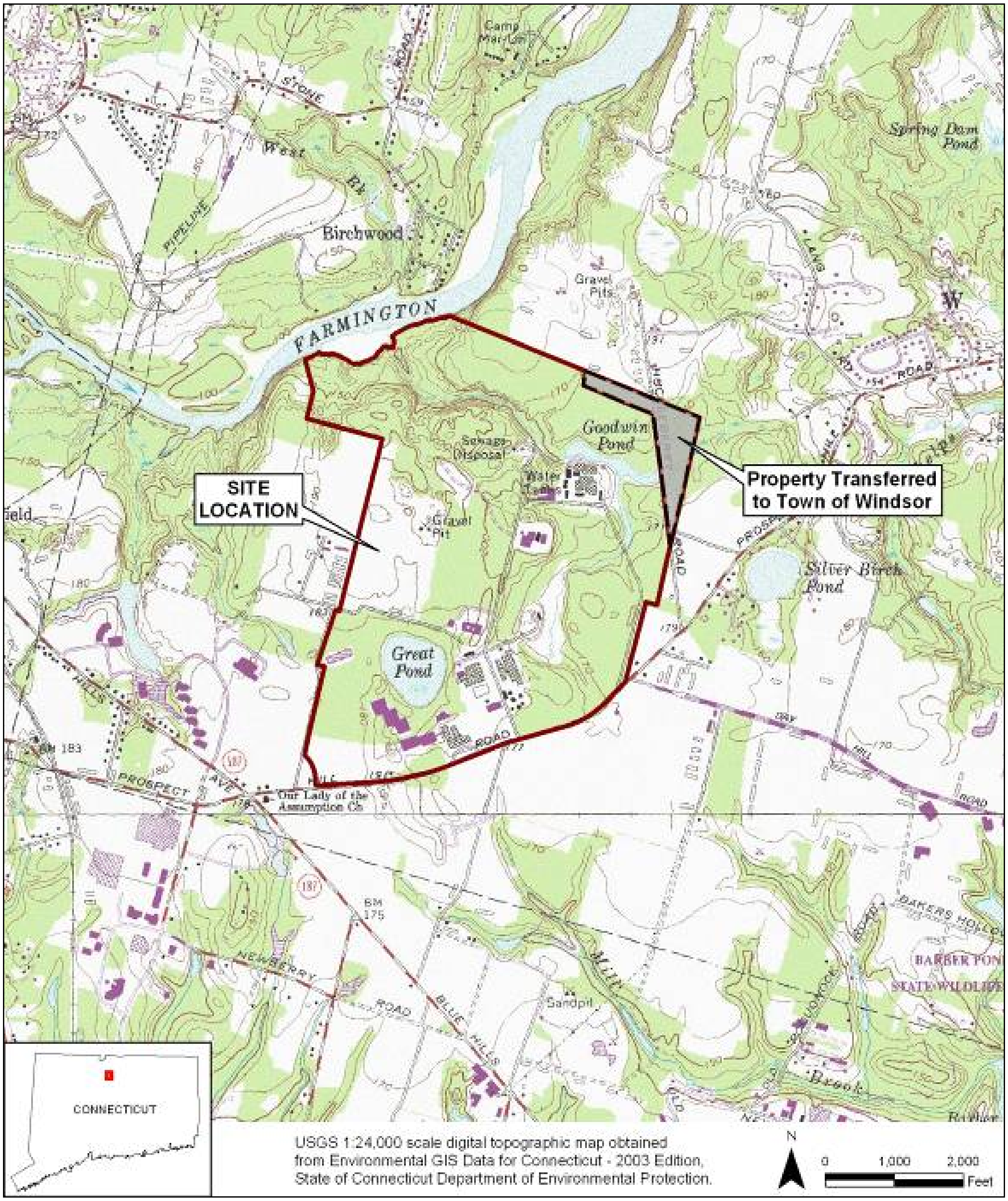

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Figure 1.1: Site Location Map

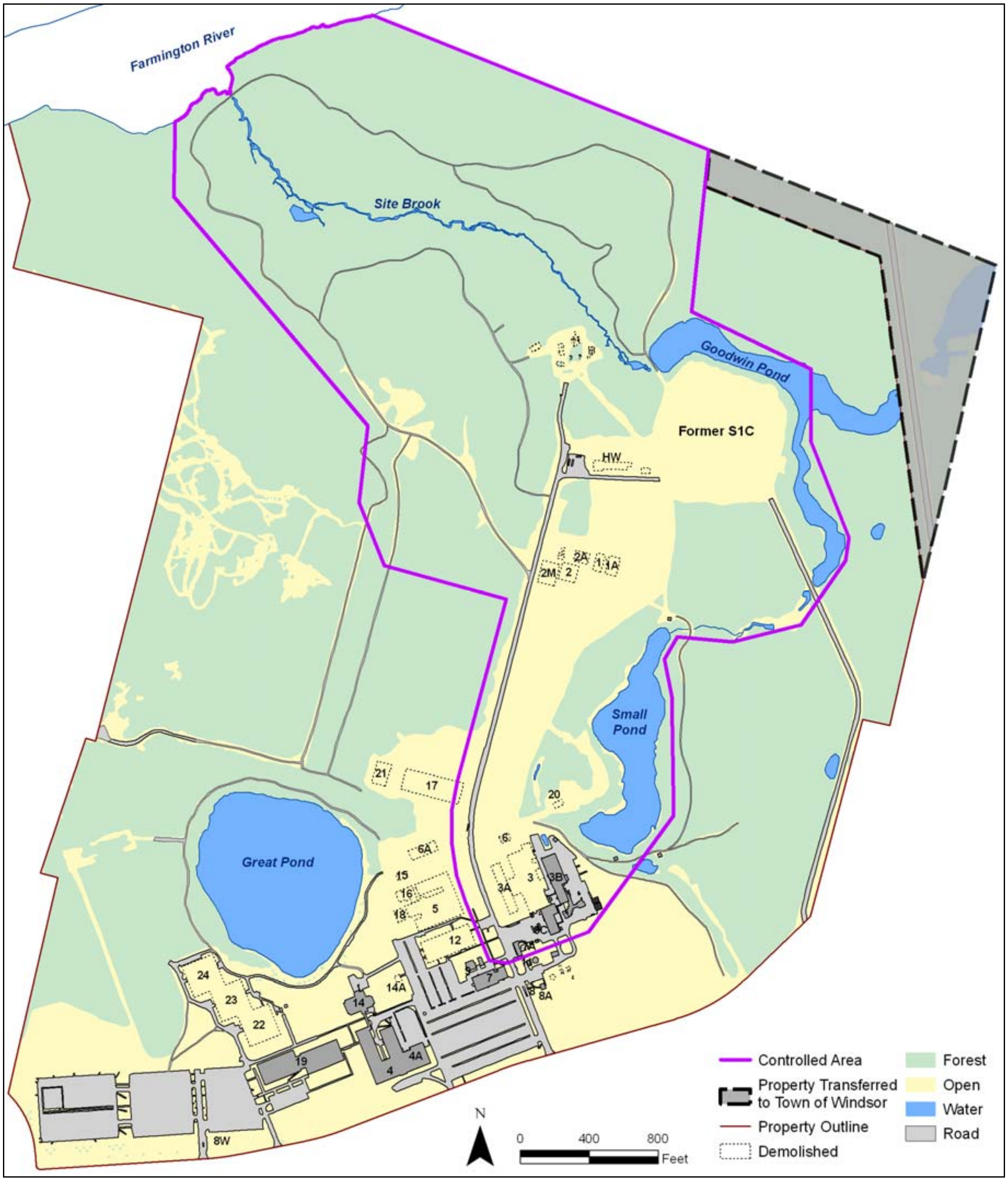

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Figure 1.2: Site Overview

Currently, the Site is commercial use 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; commercial use and a sand and gravel quarry to the west; the Windsor/Bloomfield Sanitary Landfill and Recycling Center and the Rainbow Reservoir portion of the Farmington River to the north; and forested land with some residential and commercial development to the east.

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, the Site 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. Buildings 3 and 6 initially were designed and built for Naval nuclear fuel manufacturing at the Site. 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 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

As part of the current Site activities, Building Complexes 3 and 6 have been decontaminated and dismantled and the below-ground utilities have been removed. The south end of Building 3 (High Bay) remains and currently is used for fossil fuel research and development, conducted by ABB's tenant.

The remaining radiologically impacted areas of the Site will be remediated as necessary. This will include removal of soil, piping, debris and other materials that are identified during decommissioning activities. Potentially impacted portions of the Site consist of land and surface water bodies adjacent to commercial licensed areas or other impacted areas on the Site. Figure 1.3 shows the areas at the Site, and identifies the current status for each.

This FSS Report specifically addresses the Industrial Waste Lines, Wastewater Treatment Plant and East Main Street Sanitary Line areas. These areas are depicted on Figure 1.3, with an aerial photo presented on Figure 1.4. The remaining areas within the licensed portion of the Site will be addressed in future FSS Report submittals.

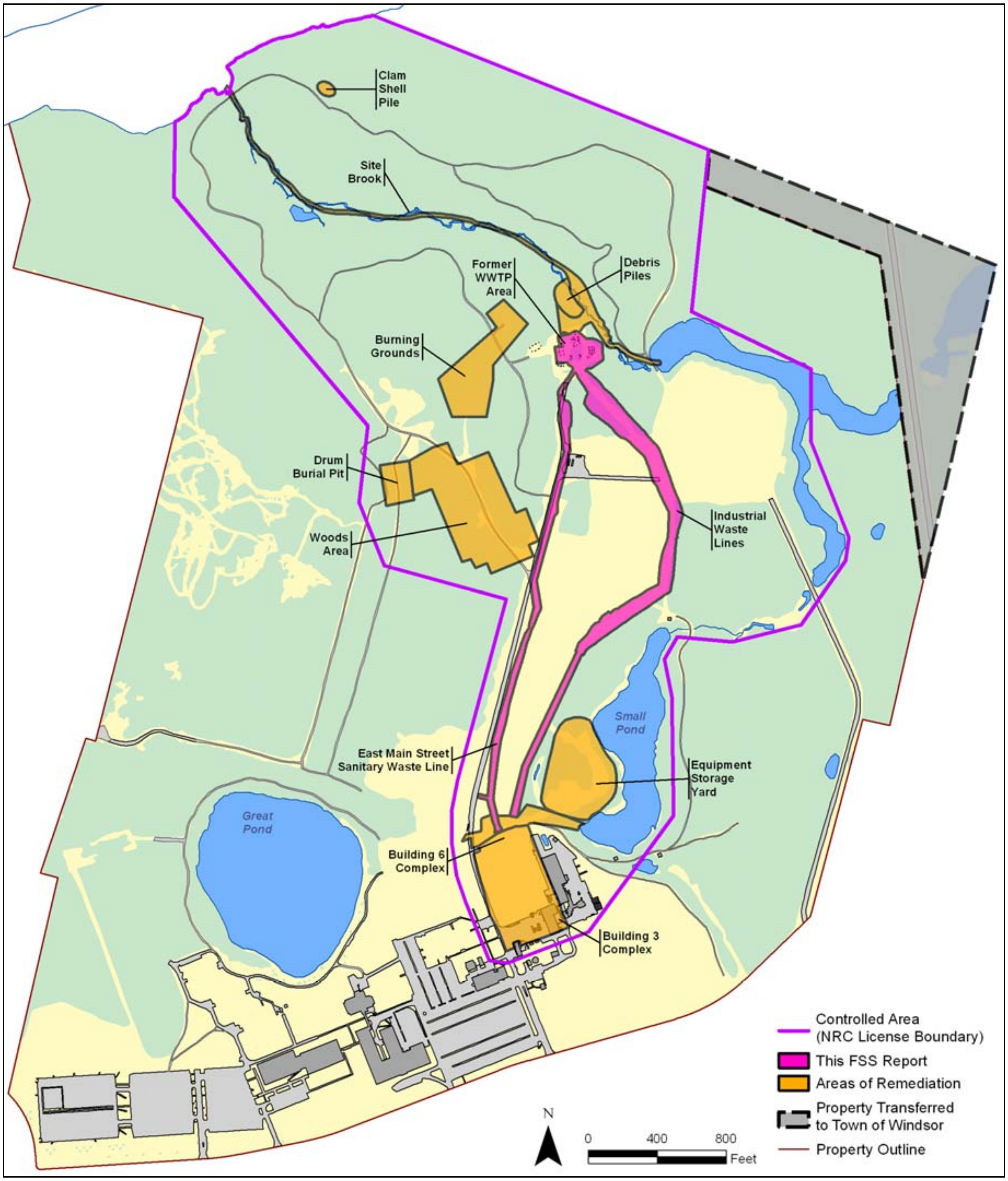

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Figure 1.3: Site Areas

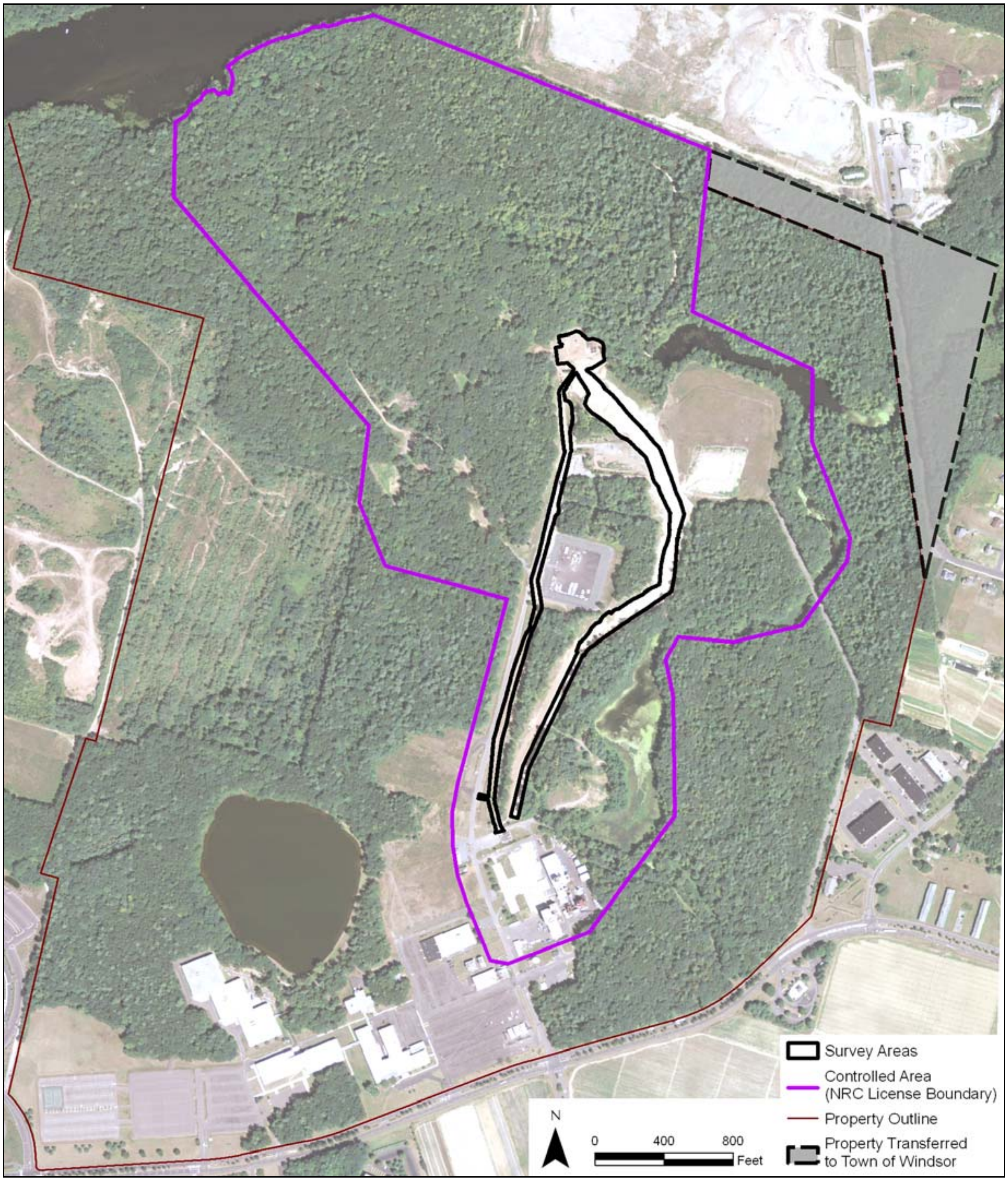

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Figure 1.4: Aerial Photo

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 primary 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 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.

In addition, thorium and radium have been identified in a few isolated areas of the Site. These areas are not included in this FSS Report and will be addressed in a future FSS Report submittal.

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 (now AMEC) 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, 2003). Results from dose modeling were used to select an enrichment of 3.5% to represent the uranium series. Co-60 is used to represent the reactor byproduct series.

1.6 DECISION FRAMEWORK

Since remediation is complete for the east main street sanitary waste line, industrial waste line, and waste water treatment facility areas, the results of the FSS performed outside of trench excavations or spoils piles demonstrate that the potential dose from any residual radioactivity is below the release criterion for each survey unit. Results of the trench release surveys (inside trench excavations and the spoils piles generated during excavation activities) demonstrate that the potential dose from any residual radioactivity is below the release criterion for the trench areas.

1.6.1 Compliance Testing

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

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

In trench areas when survey or sampling results were greater than investigation levels or greater than the established $DCGL_W$ values, immediate remediation of the identified area was performed and post-remedial sampling and analysis conducted. During the Industrial Waste Line removal, soil remediation was occasionally necessary. The characterization survey and sample protocol for Radioactive Waste Line and Industrial Waste Line removal is described in detail in the FSSP (AMEC, 2011).

The combination of volumetric sampling and gamma walkover (scan) survey data was used to demonstrate compliance with the release criterion. As described in the FSSP (AMEC, 2011), soils material excavated from trenches were divided into two categories based on sodium iodide (NaI) screening results. Soils that exceed the $DCGL_W$ values were placed into piles or containers for disposal as low-level radioactive waste while the remaining soils were placed into storage piles close to the trench for possible use as back fill once the underground utility has been removed. 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, and the DCGLs were derived independently, the unity rule was used to evaluate compliance with the dose criterion. 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.

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

Where:

C_U = uranium concentration

C_B = byproduct (cobalt 60) concentration

$DCGL_U$ = derived concentration guideline level for uranium

$DCGL_B$ = derived concentration guideline level for byproduct

1.6.3 Elevated Measurement Comparison Decision

Another factor in the decision rule is the EMC. Each measurement in the survey unit (systematic and walkover) is compared to the investigation levels. Any measurement that is greater than the investigation level was investigated. 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.

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

Where:

- δ_U = estimate of average uranium residual radioactivity in the survey unit
- δ_B = estimate of average byproduct residual radioactivity in the survey unit
- $\overline{\chi}_U$ = average uranium concentration in elevated area
- $\overline{\chi}_B$ = average byproduct concentration in elevated area
- A_m = area factor for the actual area of elevated concentration
- $DCGL_U$ = derived concentration guideline level for total uranium
- $DCGL_B$ = derived concentration guideline level for byproduct

If there were more than one area of elevated residual radioactivity in a survey unit then additional terms were 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 Co-60 is 5 pCi/g. Additional information can be found in the report *Derivation of the Site-Specific Soil DCGL*s (MACTEC, 2003). Calculations were performed using the Residual Radioactivity computer program 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.

Table 1.2: Calculated $DCGL_{EMC}$ Values

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

Prepared/Date: GSM 07/28/11
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2.0 FIELD IMPLEMENTATION

This section of the report documents the FSS in the Industrial Waste Lines, Wastewater Treatment Plant and East Main Street Sanitary Line areas. The remaining areas within the licensed portion of the Site will be addressed in future FSS Report submittals.

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 O).

Gamma walkover and direct static surveys were performed on soils using a 2" x 2" thallium-activated NaI detector coupled to an appropriate scaler/rate meter instrument to form a complete survey instrument package. In addition, for the industrial waste line and the wastewater treatment plant excavation survey units, the gamma walkover surveys were performed with the survey instrument coupled to the GPS unit for real-time data logging capability with GPS coordinates. Soil volumetric samples were collected and then analyzed on the on-site gamma spectroscopy system using a high purity germanium (HPGe) detector. 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 FSS Units was performed following the FSSP (AMEC, 2011). Individual survey units were identified and created based upon the potential likelihood of soils containing residual radioactivity. The presence of impacted underground (below grade) pipelines and utilities posed some unique conditions for FSS. Underground pipelines and utilities, that are not necessary to support the continued operation of Building 3 High Bay have been removed from the Site. The FSS report for these areas will be included in a future submittal. These underground pipelines and utilities include:

- Industrial and radiological waste lines;
- sanitary waste lines;
- storm water lines; and
- other underground utilities.

The most likely underground utilities to have residual radioactivity are the radioactive waste piping and the industrial waste piping. Development of survey units for the Industrial Waste Lines area concluded in the establishment of six individual Class 1 survey units, with distinguishable and independent characteristics.

The only other underground utilities that have residual radioactivity are the sanitary sewer and storm drains. These underground utilities have a small potential for residual radioactivity in the surrounding soils, but it would not have originated from the utility. One portion of the sanitary sewer line originated just north of Building 6 and paralleled East Main Street, terminating at the wastewater treatment plant. This resulted in 3 large survey units. The survey units were classified as Class 3 areas since no significant concentrations of residual radioactivity were detected during previous characterization survey activities.

One area that processed sanitary waste and contained industrial wastewater areas is the former waste water treatment plant area. The above ground structures of the waste water treatment plant were removed in 2001. However, the pH adjustment tanks, which received and buffered industrial waste from 1974 through 1992, had not previously been removed. Because these tanks received industrial wastewater with elevated concentrations of uranium and Co-60 relative to the DCGLs, two survey units were designated and classified as Class 1 areas.

As described in the FSSP, the soils from the ground surface to the top of the waste piping ('top layer') or overburden were not expected to have any elevated soil concentrations. Therefore, cutback and toe areas surrounding trench excavation areas were considered having a lower potential for radioactive contamination, and were not expected to exceed the DCGLs, so three Class 2 survey areas were designated around the perimeter of the Class 1 survey areas. Areas surrounding the Class 2 areas where heavy equipment operations relating to excavation activities were performed were designated as a large Class 3 area, since this area was not expected to contain residual activity greater than a small fraction of the DCGLs.

A summary of the survey units for the Site FSS Areas is presented in Table 2.1 and depicted in Figures 2.1 through 2.3.

Table 2.1: Summary of FSS Units

Survey Unit ID	Class	Area (m^2)	Description
CE-FSS-41-01	3	3,382	East Main Street Sanitary Waste line
CE-FSS-41-02	3	2,537	East Main Street Sanitary Waste line
CE-FSS-41-03	3	4,312	East Main Street Sanitary Waste line
CE-FSS-42-01	1	1,721	IWL Excavation (MH-1 to past MH-2)
CE-FSS-42-02	1	1,977	IWL Excavation (MH-2 to past MH-4)
CE-FSS-42-03	1	1,671	IWL Excavation (MH-4 to past MH-6)
CE-FSS-42-04	1	1,317	IWL Excavation (MH-6 to past MH-7)
CE-FSS-42-05	1	1,294	IWL Excavation (MH-7 to past MH-8)
CE-FSS-42-06	1	1,623	IWL Excavation (MH-8A to MH-10/13)
CE-FSS-42-07	2	2,359	IWL Excavation Buffer Area
CE-FSS-42-08	3	8,738	Remainder of cutback area out from Class 2 Buffer area
CE-FSS-43-01	1	690	WWTP IWL Excavation Area-Includes Catch Basin line
CE-FSS-43-02	2	2,526	WWTP buffer areas around unit 43-01
CE-FSS-43-03	1	542	pH Adjustment Tanks Footprint including IWL from MH-10A to MH-15
CE-FSS-43-04	2	914	pH Adjustment Tank excavation buffer area

Notes:
MH-Manhole
WWTP-Wastewater Treatment Plant
IWL-Industrial Waste Line

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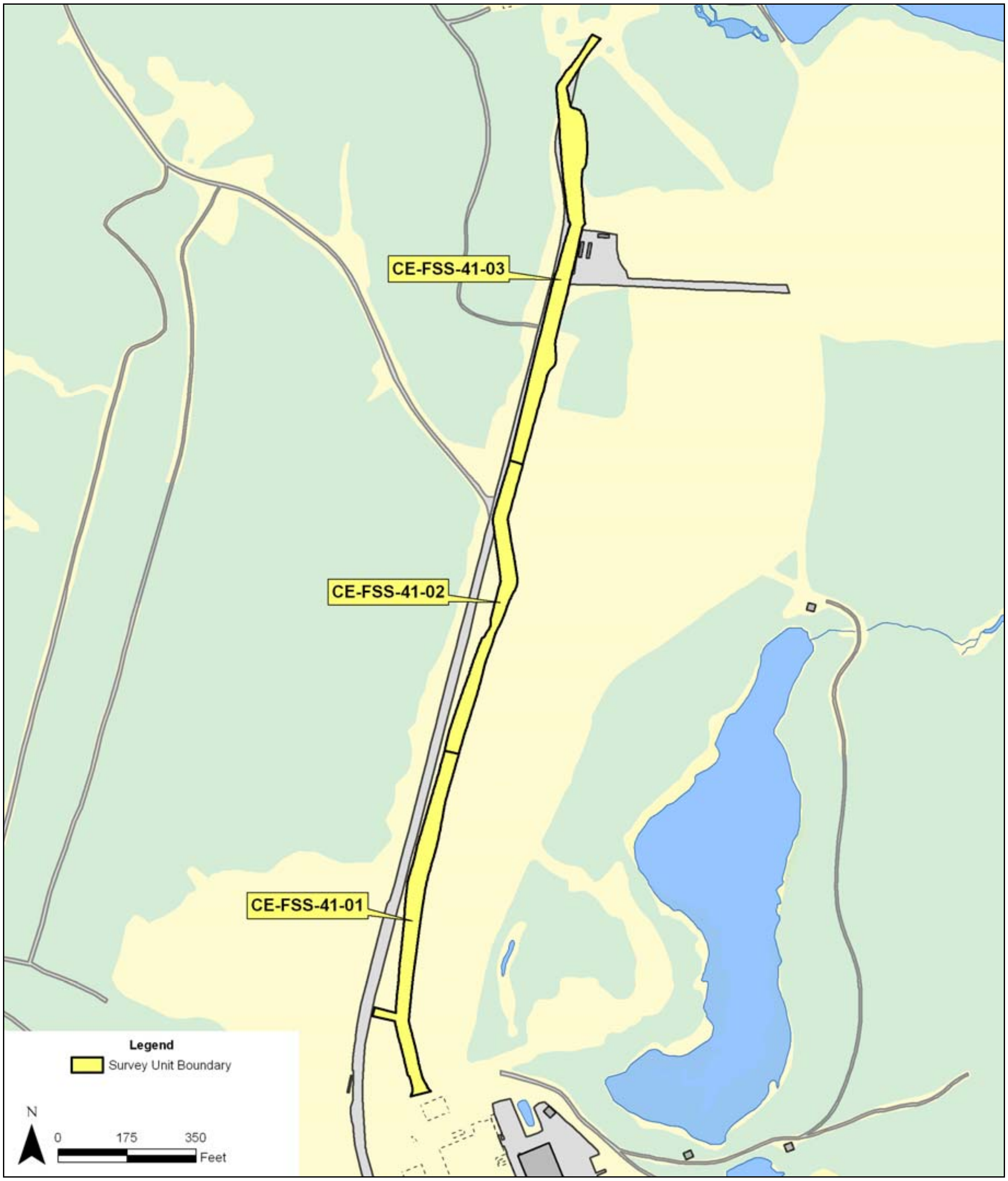

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Figure 2.1: Overview of East Main Street Sanitary Waste Line Excavation FSS Units

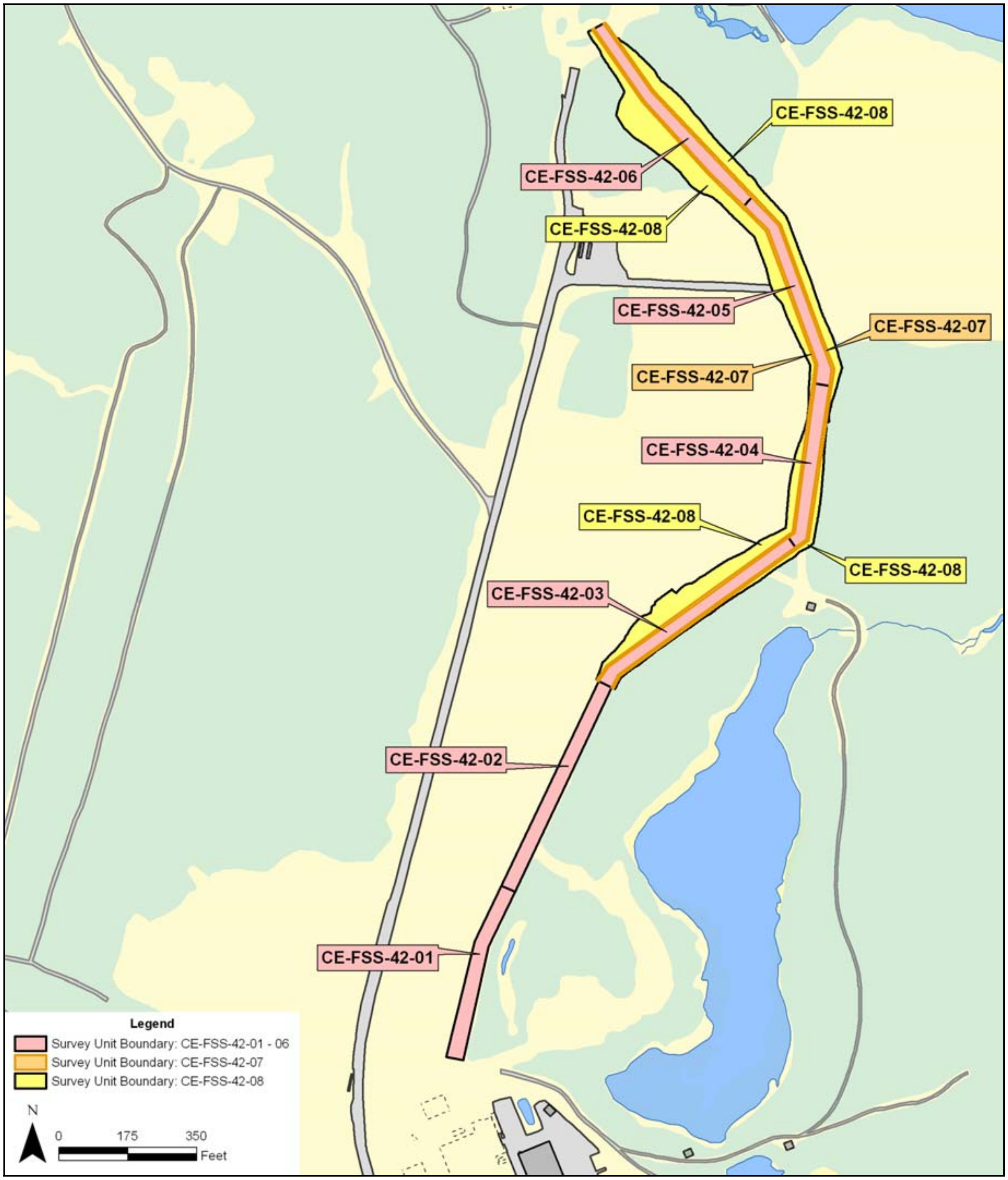

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Figure 2.2: Overview of Industrial Waste Line Excavation FSS Units

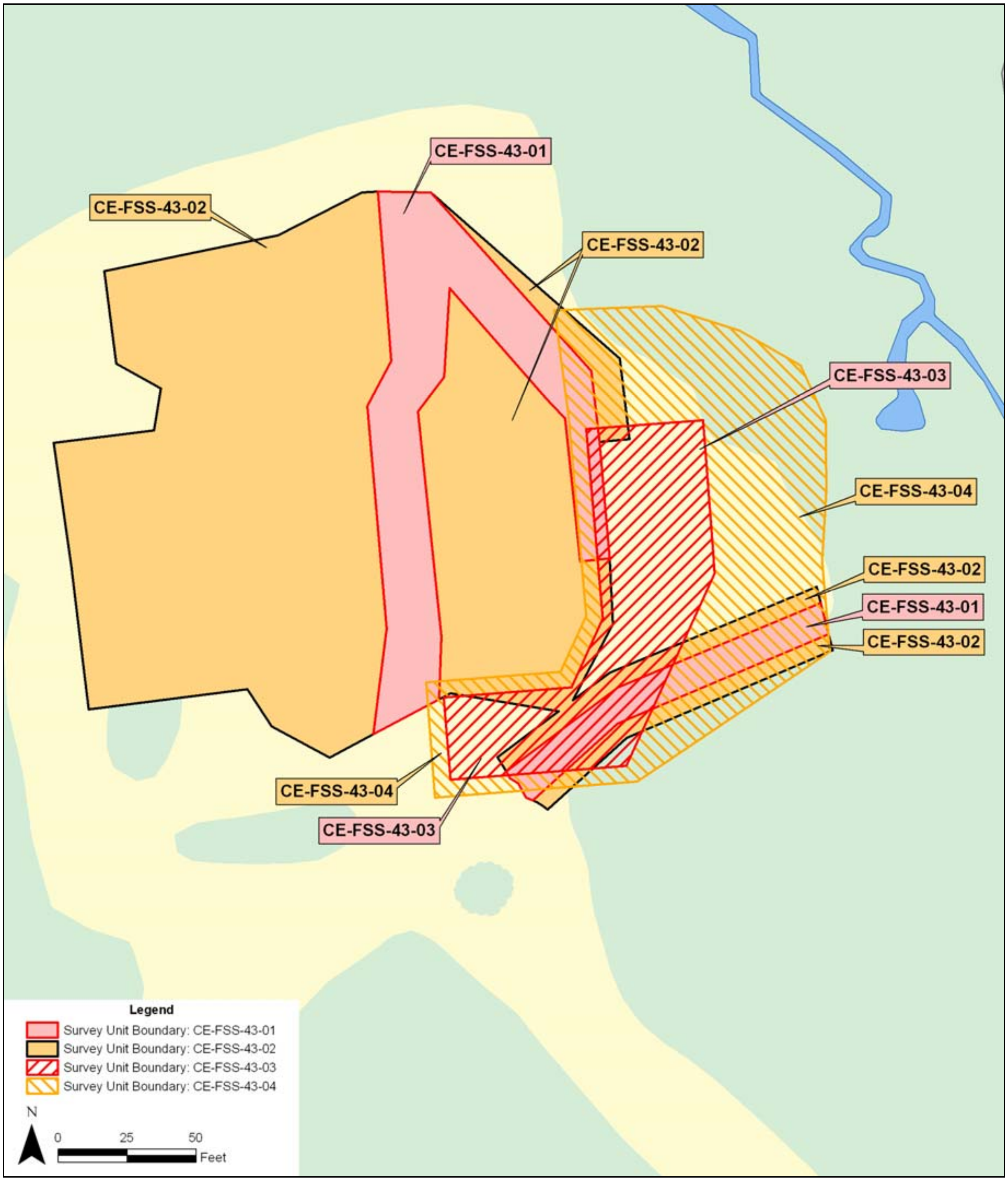

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Figure 2.3: Overview of Waste Water Treatment Plant Footprint and Industrial Waste Line Excavation FSS Units

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, 2010). 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 produced a sample distribution on scale drawings of the area(s) 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. Additionally, for comparison, since both source terms could be present in unrelated ratios, the weighted sum standard deviation was estimated for the unity sample size calculation using the guidance provided in Appendix I of MARSSIM (NRC, 2000). A discussion of sampling design methodology as well as α and β decision error is found in the FSSP (AMEC, 2011).

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 70% of the DCGL. The standard deviation was also conservatively approximated high (30%) as a safety margin to reduce the chance of failing the decision criteria. The 30% assumption for the coefficient of variation value used is cited in MARSSIM (NRC, 2000) as reasonable when preliminary data are not obtained, and are reasonable compared with the large variations of values from both the uranium and Co-60 data. The survey design parameters used to calculate the minimum required sample size for Class 1 Survey Units are shown in Table 2.2.

Table 2.2: 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)	390	3.5
Standard Deviation (σ) (pCi/g)	167	1.5
Relative Shift (Δ/σ)	1.0	1.0
Sample Size (N)	23	23
Additional 20%	5	5
FSS Sample Size	28	

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For this scenario, VSP calculated one additional sample, which correlates to the Sign Test Table 5.5 of MARSSIM (NRC, 2000) which lists an FSS total sample size of 29 using the same parameters in Table 2.2 because of rounding. Since having an additional sample is conservative, the VSP calculated sample size was used. This FSS report contains a total of eight Class 1 Survey Units.

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 50% of the DCGL. The standard deviation was conservatively assumed to be 30% (as described previously) 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 2.3. This FSS report contains a total of three Class 2 Survey Units.

Table 2.3: Class 2 Survey Unit Sample Size

Parameter	Total Uranium	Co-60
α decision error	0.05	0.05
β decision error	0.05	0.05
$DCGL_W$ (pCi/g)	557	5
LBGR (maximum estimated mean/median) (pCi/g)	278	2.5
Standard Deviation (σ) (pCi/g)	167	1.5
Relative Shift (Δ/σ)	1.6	1.6
Sample Size (N)	14	14
Additional 20%	3	3
FSS Sample Size	17	

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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 10% of the DCGL. The same standard deviation was used for Class 3 areas as was used for the Class 1 and Class 2 areas which 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 2.4. This FSS report contains a total of four Class 3 Survey Units.

Table 2.4: 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)	56	0.5
Standard Deviation (σ) (pCi/g)	167	1.5
Relative Shift (Δ/σ)	3.0	3.0
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 2.5. In every survey unit, the number of samples obtained met or exceeded the number of samples planned.

Table 2.5: Number of FSS Volumetric Samples Obtained per Survey Unit

Survey Unit ID	Class	Number of Samples Planned	Number of Samples Obtained
CE-FSS-41-01	3	14	16
CE-FSS-41-02	3	14	14
CE-FSS-41-03	3	14	14
CE-FSS-42-01	1	29	29
CE-FSS-42-02	1	29	29
CE-FSS-42-03	1	29	29
CE-FSS-42-04	1	29	29
CE-FSS-42-05	1	29	29
CE-FSS-42-06	1	29	29
CE-FSS-42-07	2	17	17
CE-FSS-42-08	3	14	14
CE-FSS-43-01	1	29	29

Table 2.5: Number of FSS Volumetric Samples Obtained per Survey Unit

Survey Unit ID	Class	Number of Samples Planned	Number of Samples Obtained
CE-FSS-43-02	2	17	17
CE-FSS-43-03	1	29	29
CE-FSS-43-04	2	17	17
Total Number of Samples		**339**	**341**

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2.4 SURVEY AND SAMPLE LOCATIONS

During FSS activities for the East Main Street Sanitary Waste Line and the Industrial Waste Line excavation areas, randomly chosen sampling and survey locations were used to place Class 3 survey locations within or around those survey units. Systematic grid patterns were used to place Class 1 and Class 2 survey locations within the Industrial Waste Line and Waste Water Treatment Plant excavation survey units. For the Class 1 and Class 2 survey units, a random start location was selected and used to provide an unbiased set of measurement locations for the FSS. Gamma walkover survey coverage was 100% for Class 1 survey units, minimum of 10% for Class 2 survey units, and judgmental for Class 3 survey units.

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) 83 (units of feet) as its reference datum. Sample locations were identified and marked within the survey units using a Trimble GeoXH 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 83 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 for the areas included in this submittal report during 2010. Figures of sample locations for each survey unit are provided in the survey unit data appendices (A through O). 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 defined by VSP using the assumptions stated earlier in this report.

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 6 inches of the soil at the sample location. Various sampling methods were used to collect the soil samples in the survey units. 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. No samples were relocated due to inaccessibility issues.

One minute static soil measurements were obtained with the 2” by 2” NaI detector system prior to sample collection of all volumetric samples. Once static measurements and QC duplicate measurements were completed as applicable, a 0.35 square foot area was demarcated and the top 6 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 homogenized and placed in a plastic pint jar and labeled in accordance with the FSSP.

Volumetric soil sampling in excavated trench areas was performed in a similar manner, except that sampling in the trenches was performed at both biased sample locations and non-biased locations (see Section 3 for greater detail).

2.5 INVESTIGATION LEVELS

Investigation levels (Table 2.6) for the volumetric sample results were developed in accordance with the guidance found in MARSSIM. Any sample result greater than the investigation level was identified, marked, and further investigation performed to determine the extent of contamination at greater than the $DCGL_W$.

Table 2.6: 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 gross counts per minute (cpm) value was generated at the stated $DCGL_W$ value for the scanning measurement investigation level (Table 2.7). For the purpose of this report, all reported cpm values, unless otherwise specified, should be considered gross values uncorrected for instrument background.

Table 2.7: Final Status Survey Scanning Gross Investigation Levels

Survey Unit Classification	Scanning Measurement Investigation Level (most conservative)
Class 1	> 4,104 cpm
Class 2	> 4,104 cpm
Class 3	> 4,104 cpm

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2.6 ON-SITE GAMMA SPECTROSCOPY INSTRUMENTATION

Soil volumetric samples analyzed on-site were analyzed by an HPGe gamma spectroscopy system throughout the entire FSS sampling campaign and in accordance with the Genie-2000 Spectroscopy System Operations Instructions (Canberra, 2009a).

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 (serial #9706) 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 2.4. The system is energy calibrated so the MCA data channels are given an energy equivalence and displays counts versus energy. An efficiency calibration is performed for each sample 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, 2009b).

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Figure 2.4: On-Site HPGe 25% 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 Q.

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 is 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 (target levels of 10-50% of the DCGL), 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.

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

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 $DCGL_W$, ABB chose to include soil background radioactivity as part of the residual activity attributable to licensed activities. No attempt was made to adjust the FSS soil gamma results data by subtracting 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) and to correct or normalize the detector's response to this "instrument background."

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 Q.

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.

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 are 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, overestimate of the total uranium in the sample.

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 uranium totals to the actual uranium total are shown in Table 2.8. The table demonstrates that the multiplier of 31 used to evaluate FSS data overestimates actual total uranium and is therefore conservative.

Table 2.8: Evaluation of Total Uranium Calculation

Parameter	NRC Equation 3.5%	Variation 1 3.5%	Variation 2 3.5%	NRC Equation 90%
Specific Activity (Ci/g)	1.8E-6	2.4E-6	2.6E-6	6.2E-05
U-234	77.49	83.38	84.66	96.82
U-235	4.27	3.15	2.91	3.13
U-238	18.24	13.47	12.43	0.05
Actual U Total	100	100	100	100
Calculated U Total (U-235 X 23)	98	72	67	72
Calculated U Total (U-235 X 31)	132	98	90	97

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 look or screen for the presence of localized areas of elevated radioactivity in soils.

Gamma walkover surveys 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. When such an area was found, a static one-minute count was performed. If the one-minute static measurement confirmed that the gross scan investigation level specified in Table 2.7 was exceeded (suggesting the presence of an elevated area), a biased soil sample was collected at that location. No locations of elevated residual radioactivity that exceeded the investigation level were identified during the walkover scan surveys by the surveyors. However, during review of the GPS data logger scan survey files for two survey units, CE-FSS-43-02 and CE-FSS-43-03, indicated that two small areas exceeded the gross scan investigation level. Discussion of the two investigations performed for survey units CE-FSS-43-02 and CE-FSS-43-03 are presented in Sections 3.2.13 and 3.2.14 respectively.

2.7.1 Gamma Walkover Instruments

Gamma walkover survey instrumentation consisted of a NaI detector and an appropriate survey meter. The Ludlum Model 12 coupled with the Ludlum 44-10 NaI detector attached to a Trimble GeoXH GPS was used during FSS activities of the Industrial Waste Line and Waste Water Treatment Plant areas. GPS data logging capability was not instituted until after the East Main Street Sanitary Line FSS had been completed.

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 P.

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. The process to establish the MDC for the scanning instrument 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 detector's counting rate to exposure rate (cpm per μR/h[microrem per hour]) 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. Finally, all three parameters were utilized to calculate the MDC_{SCAN}.

The parameters used to develop the relationship between the NaI detector's counting rate to exposure rate and the assumptions used to determine (model) the relationship between radionuclide concentration in soil and exposure the relationship between radionuclide concentration in soil and exposure are described in the FSSP (AMEC, 2011).

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 2.9 and indicate that 786 cpm above background is the minimum value for 95% true positive detection.

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

Where:

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

Table 2.9: 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 centimeter [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 2.10. Since the manufacturers reported different efficiencies for the same size NaI detector, both were used to calculate MDC_{SCAN} values in order to show what range of MDC_{SCAN} might be expected.

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

Where:

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

Table 2.10: MDC_{SCAN} Values For 2" x 2" NaI Detector

Parameter		Byproduct	Uranium
$MDCR_{surveyor}$	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 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) cannot 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 2.11 presents the walkover (scan) and static survey background readings.

Table 2.11: Walkover and Static Survey Background Measurements

Walkover and Static Background Measurements	
Survey Unit	**Recorded Background Reading (gross cpm)**
CE-FSS-41-01	2,800 and 3,000
CE-FSS-41-02	3,000
CE-FSS-41-03	3,000
CE-FSS-42-01	2,400
CE-FSS-42-02	2,400
CE-FSS-42-03	2,400
CE-FSS-42-04	2,200
CE-FSS-42-05	2,200
CE-FSS-42-06	2,200
CE-FSS-42-07	2,800
CE-FSS-42-08	2,800
CE-FSS-43-01	3,000
CE-FSS-43-02	2,800
CE-FSS-43-03	2,381
CE-FSS-43-04	2,200

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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.

Instrument and detector combinations used for scanning of trench bottoms and trench excavation spoil piles were identical to scanning instruments used for the gamma walkover survey and carry the same detection limitations identified in Section 2.7.3. Instrumentation used for the trench walkover and static surveys is identified Table 2.12. Calibration certificates for the scanning instrumentation are presented in Appendix P.

Table 2.12: Walkover and Static Instrumentation

Instrumentation				
Instrument	**Inst Model**	**Serial #**	**Detector Model**	**Serial #**
Gamma 1	Model 12	145982	44-10	PR-150916
Gamma 5	Model 16	74100	44-10	PR-150296
Gamma 10	Model 12	172705	44-10	277925
Gamma 11	Model 12	172718	44-10	277926
Gamma 12	Model 12	172698	44-10	262444

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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 5 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. Data associated with each survey unit and its associated evaluations are provided in the appendices (A through O) of this report.

3.1 FIELD SURVEY AND VOLUMETRIC SOIL SAMPLING RESULTS OVERVIEW

341 volumetric soil samples from 15 survey units were collected and analyzed as part of FSS areas for this report. Sample locations where a single sample was collected and split into a duplicate sample are indicated as ‘split’ samples. Twenty-three 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 split sample measurement result were used as the reported value for the sample location. Additionally, as an internal laboratory control QC metric, a laboratory instrument replicate or “Laboratory Recount” was performed on randomly selected samples to measure instrument precision. Further information about split samples and laboratory recounts and the assurance of precision and variability is presented in Section 5.

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 patterns, relationships, or potential anomalies in the distribution of the data. A key test of the data set is for goodness-of-fit. Goodness of fit 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 locally elevated residual radioactivity. Posting plots are provided in the survey unit data appendices (A through O).

Once the survey unit data was assessed and verified that it is acceptable for comparison to the release criteria, it was evaluated against the $DCGL_{WS}$.

This section of the report provides a summary of the FSS data and statistical data assessment. The data associated with each survey unit and its associated evaluations are provided in the survey unit data appendices (A through O) of this report.

3.2.1 Survey Unit CE-FSS-41-01

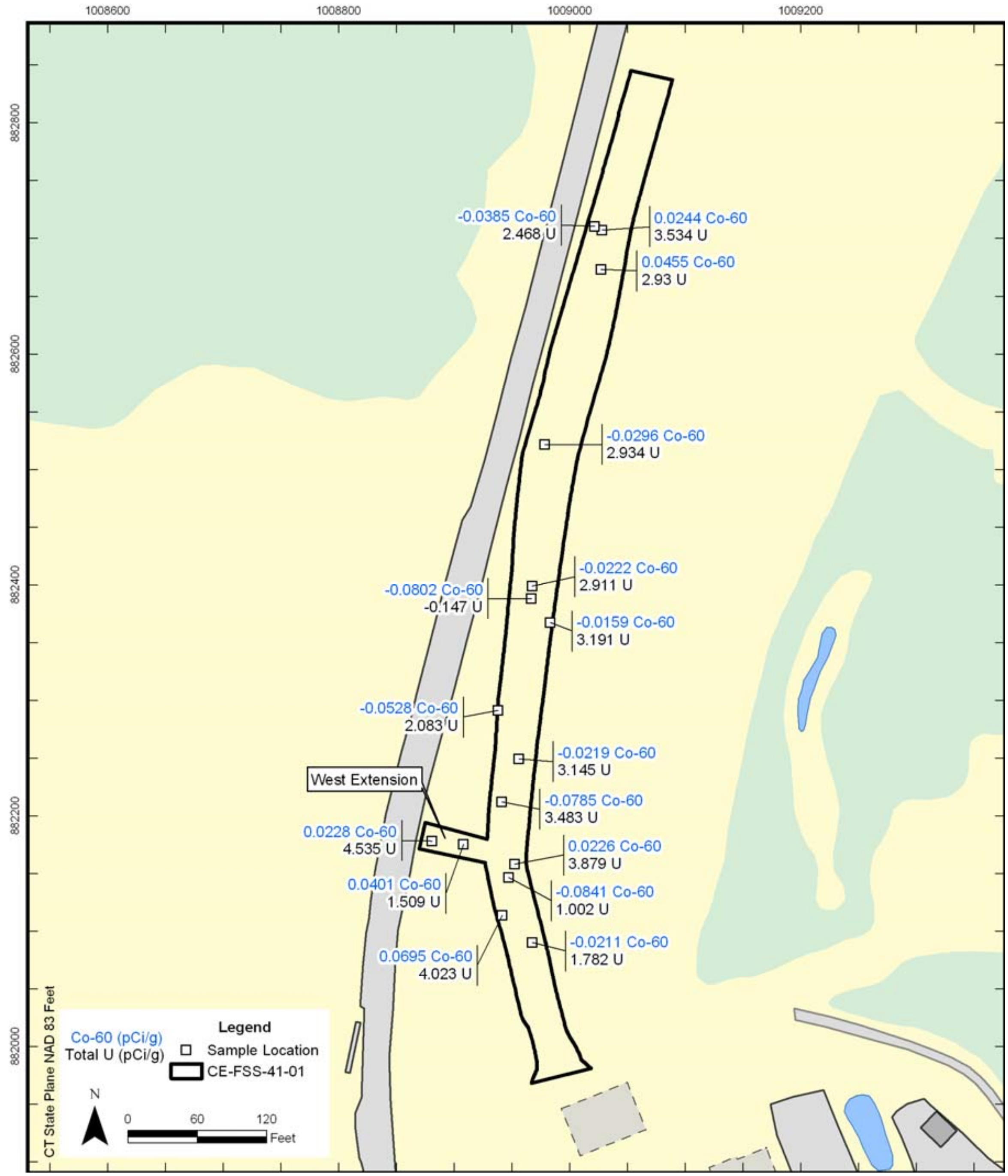
Survey Unit CE-FSS-41-01 covers the first portion of the East Main Street sanitary waste line excavation and starts about twenty feet north of the Building 6 pad, and continues north for approximately 880 feet along East Main Street and consists of approximately 3,382 m^2 of land area. Figure 3.1 presents an overview of the survey unit. Fourteen original survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit. Two additional non-biased samples were obtained in the area identified as the West Extension provided in a survey package addendum, as well as judgmental scans of the trench floor. Data associated with this survey unit are provided in Appendix A.

Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-41-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 2,800 cpm to 3,600 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the gross investigation level of 4,104 cpm (as listed in Table 2.7) were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected. As mentioned previously in Section 2.7, the scan survey performed for this survey unit predated implementation of the GPS data logger.

Volumetric Soil Sample Results

Sixteen randomly-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-41-01 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-41-01.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.1: Survey Unit CE-FSS-41-01 Total U and Co-60 Activities (pCi/g)

3.2.2 Survey Unit CE-FSS-41-02

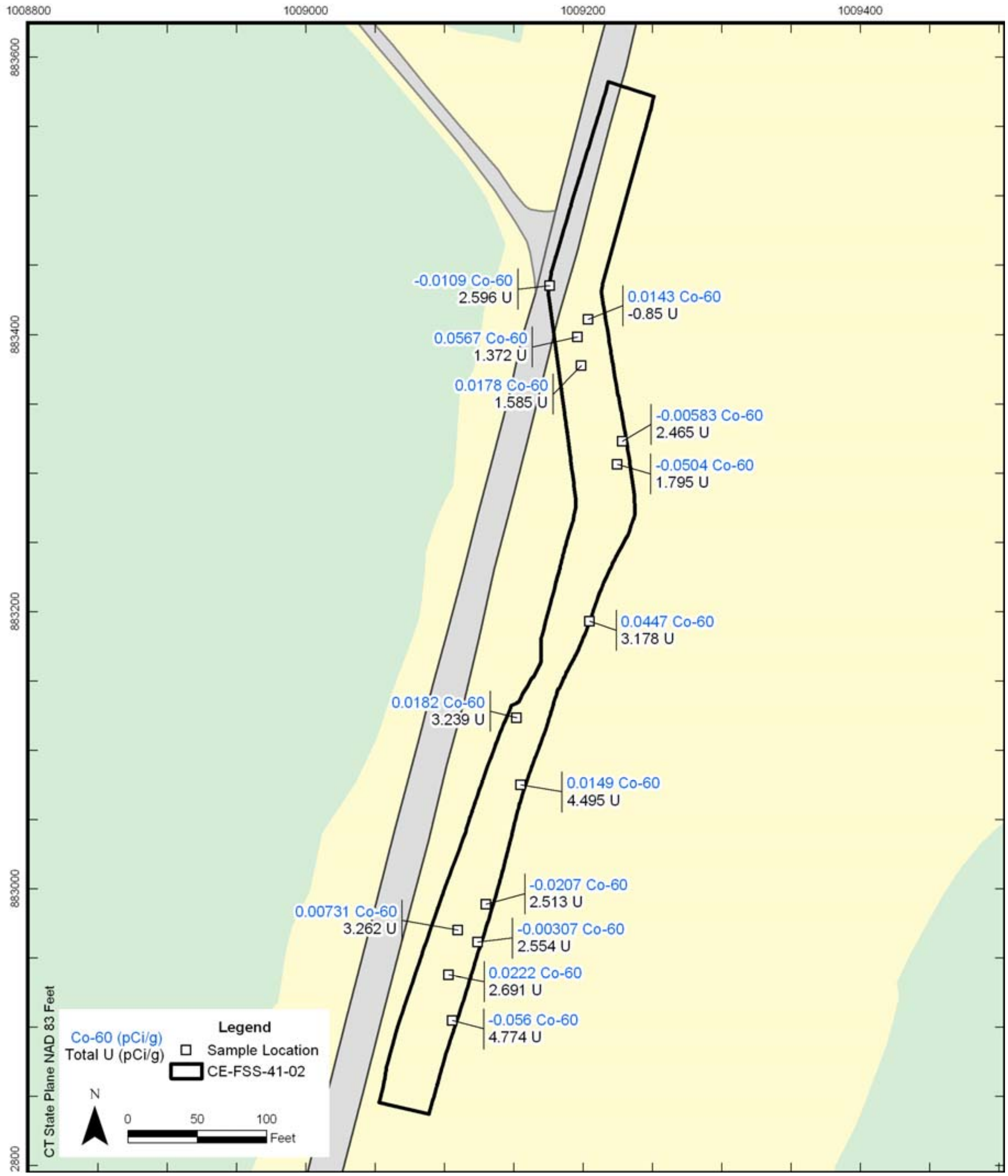
Survey Unit CE-FSS-41-02 is a continuation of the East Main Street sanitary waste line excavation which is bounded on the south by Survey Unit 41-01, and continues north for approximately 750 feet along East Main Street and consists of approximately 2,537 m^2 of land area. Figure 3.2 presents an overview 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.

Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-41-02 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 2,800 cpm to 3,600 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the gross investigation level of 4,104 cpm (as listed in Table 2.7) were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected. As mentioned previously in Section 2.7, the scan survey performed for this survey unit predated implementation of the GPS data logger.

Volumetric Soil Sample Results

Fourteen randomly-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-41-02 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-41-02.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.2: Survey Unit CE-FSS-41-02 Total U and Co-60 Activities (pCi/g)

3.2.3 Survey Unit CE-FSS-41-03

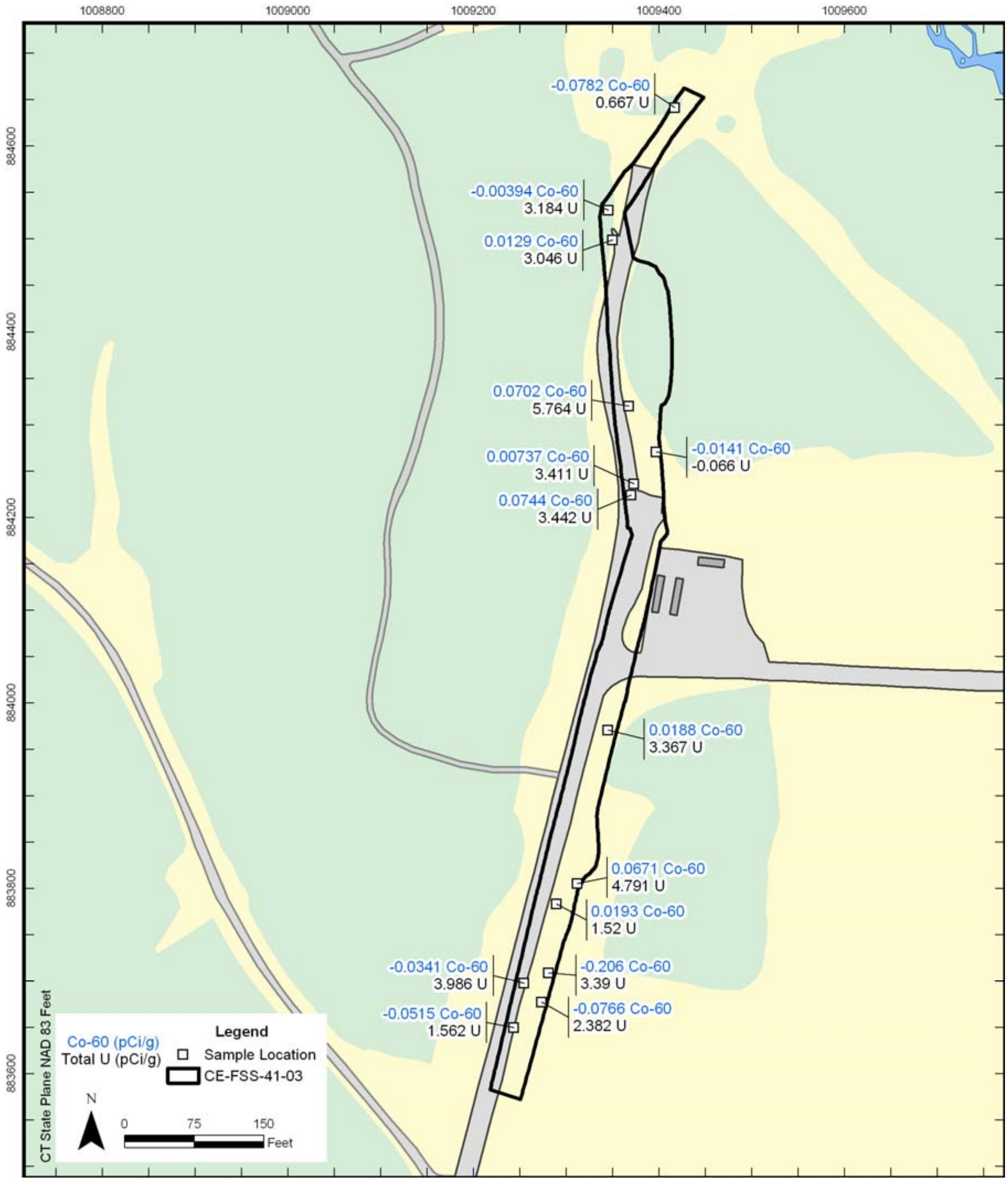
Survey Unit CE-FSS-41-03 is a continuation of the East Main Street sanitary waste line excavation which is bounded on the south by Survey Unit 41-02, and continues north for approximately 1,100 feet along East Main Street terminating at the former wastewater treatment plant area and consists of approximately 4,312 m^2 of land area. Figure 3.3 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 C.

<u>Gamma Walkover Survey Results</u>

Approximately 10 percent of the surface area for Survey Unit CE-FSS-41-03 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 2,800 cpm to 3,600 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the gross investigation level of 4,104 cpm (as listed in Table 2.7) were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected. As mentioned previously in Section 2.7, the scan survey performed for this survey unit predated implementation of the GPS data logger.

<u>Volumetric Soil Sample Results</u>

Fourteen randomly-placed volumetric soil samples were obtained for FSS in Survey Unit 41-03 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-41-03.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.3: Survey Unit CE-FSS-41-03 Total U and Co-60 Activities (pCi/g)

3.2.4 Survey Unit CE-FSS-42-01

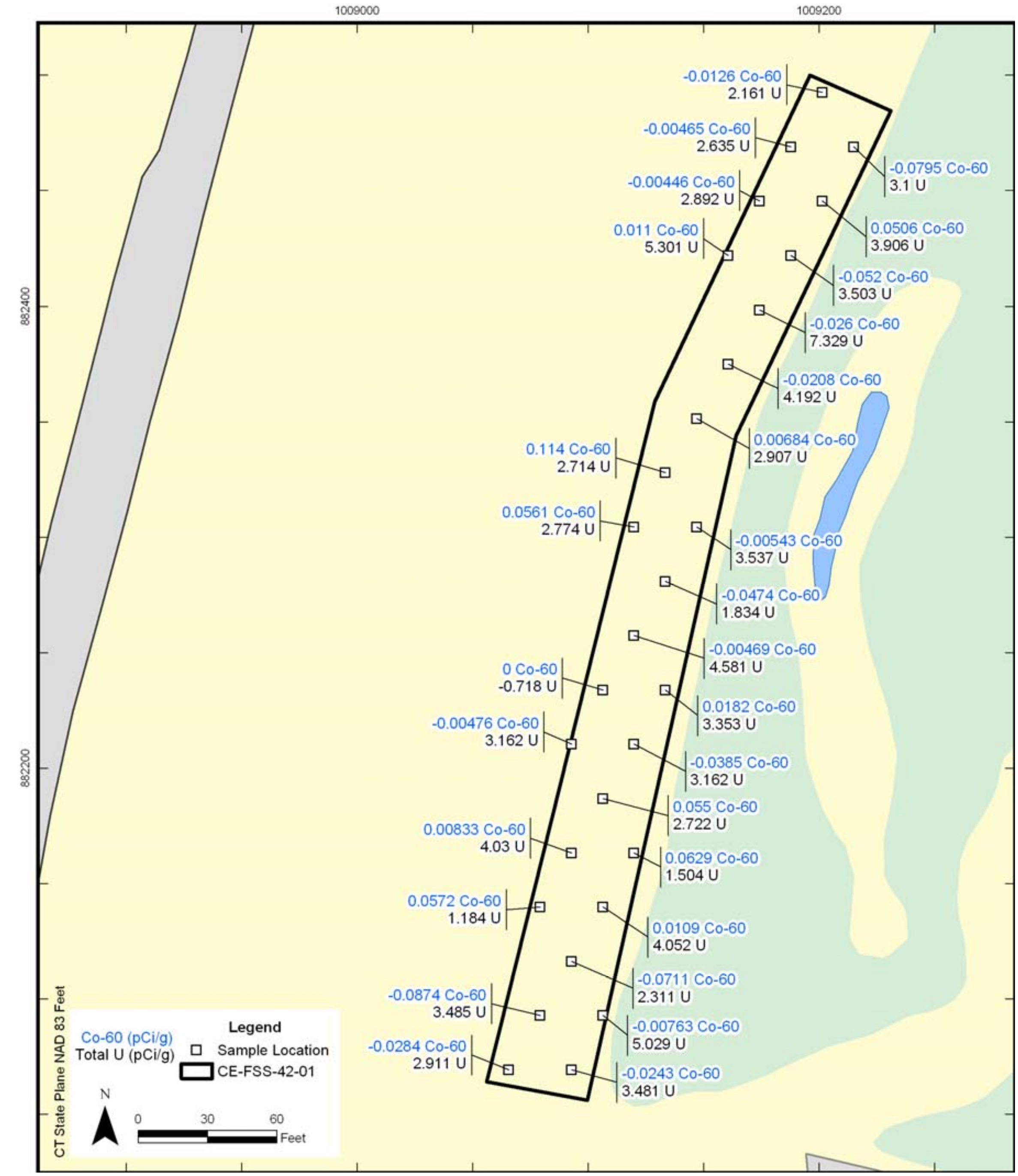
Survey Unit CE-FSS-42-01 covers the Industrial Waste Line excavation area starting north of Building 6 at manhole 1. The survey unit is approximately 460 feet long and consists of approximately 1,721 m^2 of land area. Figure 3.4 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix D.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-42-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 780 cpm to 4,440 cpm (gross) were recorded via the GPS datalogger during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. One scan data logger measurement value exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but was discounted as false positive because the four one-second scan results prior to and after the result (which was identified during data review) including the elevated value, averaged 3,440 gross cpm. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-01 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-01.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.4: Survey Unit CE-FSS-42-01 Total U and Co-60 Activities (pCi/g)

3.2.5 Survey Unit CE-FSS-42-02

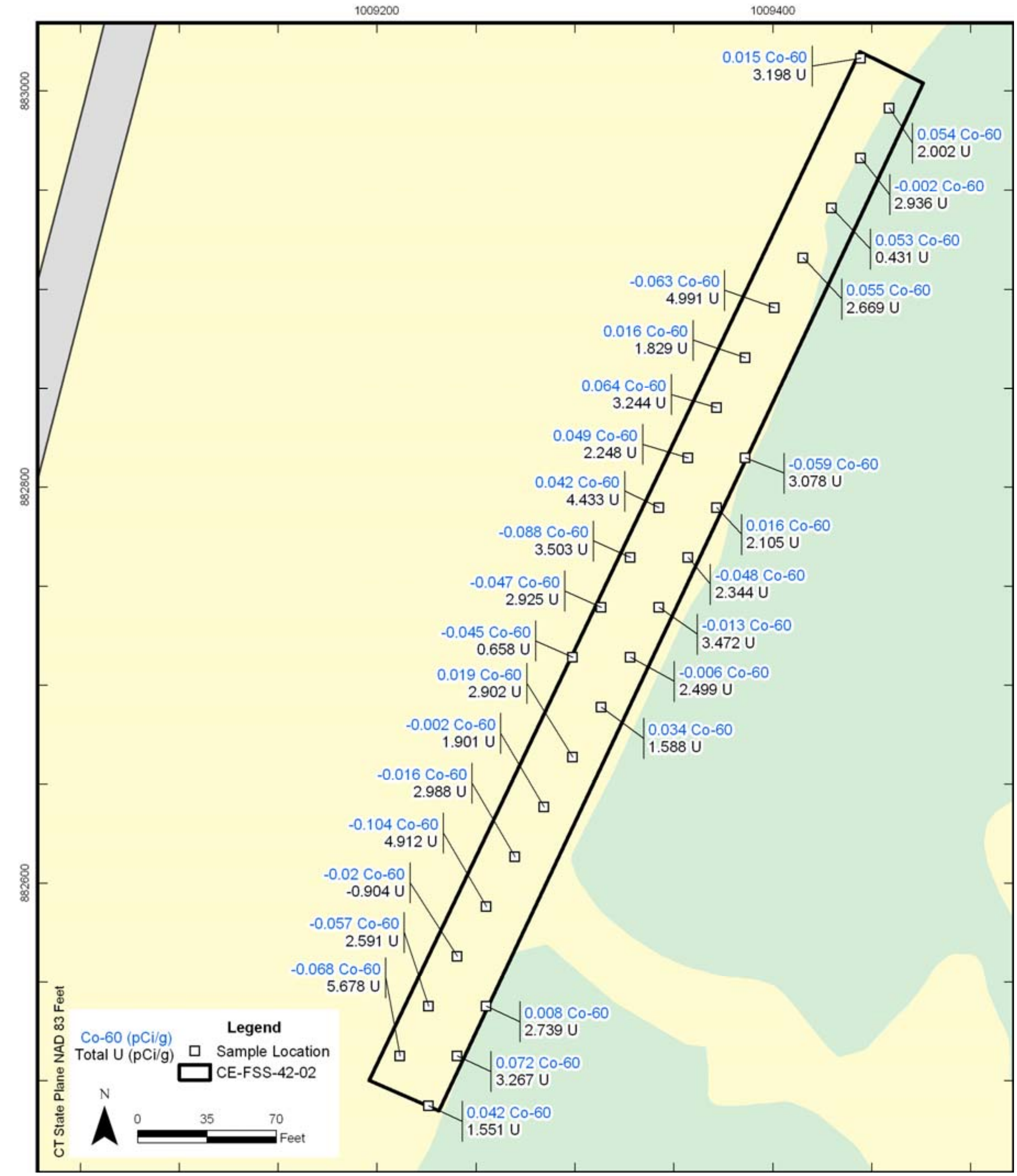
Survey Unit CE-FSS-42-02 is a continuation of the Industrial Waste Line excavation which is bounded on the south by Survey Unit 42-01. The survey unit is approximately 575 feet long and consists of approximately 1,977 m^2 of land area. Figure 3.5 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix E.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-42-02 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 1,200 cpm to 4,380 cpm (gross) were recorded via the GPS datalogger during the walkover survey. One scan measurement value exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but was discounted as false positive because the four one-second scan results prior to and after the result (which was identified during data review) including the elevated value averaged 3,100 gross cpm. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-02 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-02.



Prepared/Date: BRP 07/28/11
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Figure 3.5: Survey Unit CE-FSS-42-02 Total U and Co-60 Activities (pCi/g)

3.2.6 Survey Unit CE-FSS-42-03

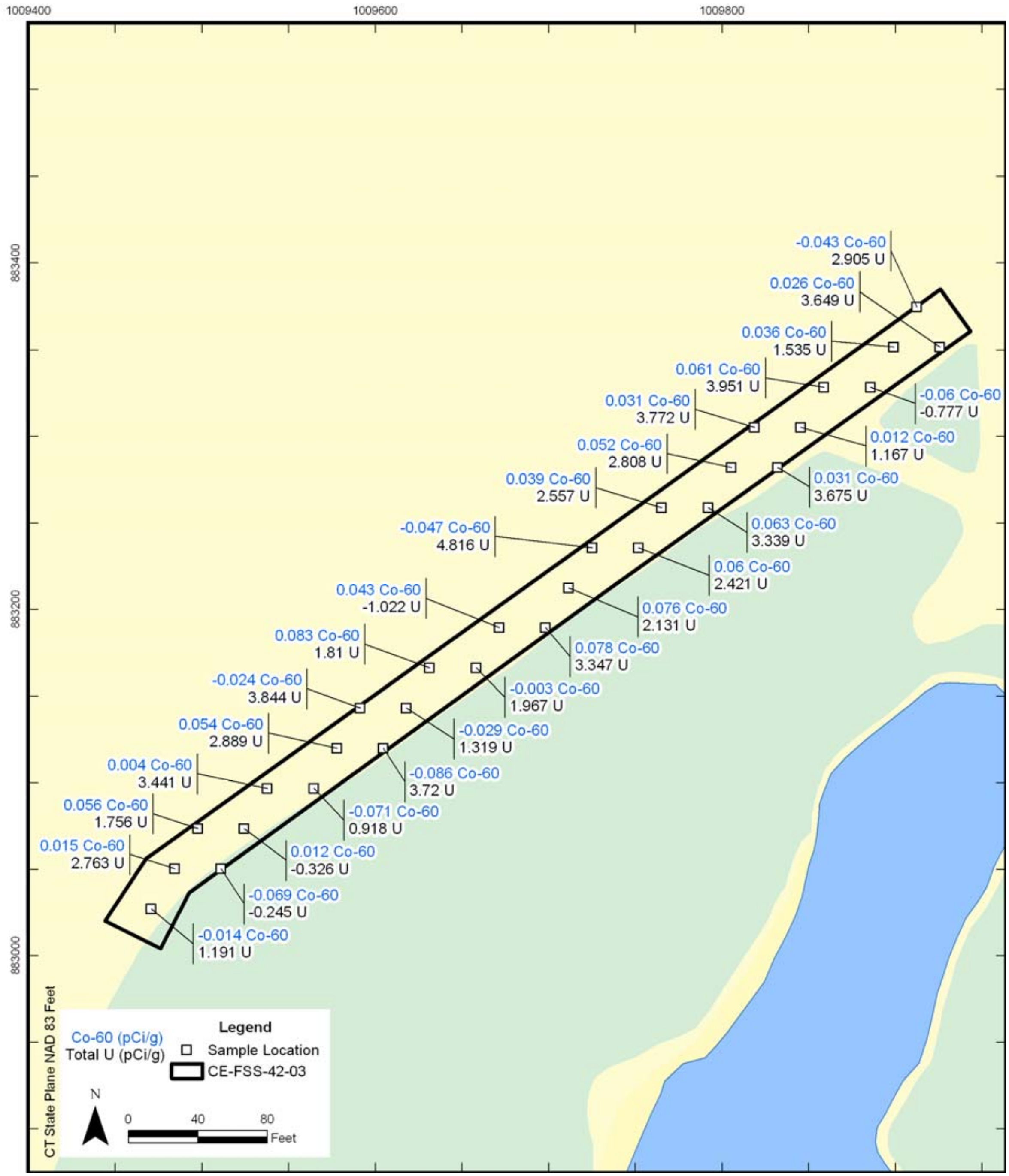
Survey Unit CE-FSS-42-03 is a continuation of the Industrial Waste Line excavation which is bounded on the south by Survey Unit 42-02. The survey unit is approximately 600 feet long and consists of approximately 1,671 m^2 of land area. Figure 3.6 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix F.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-42-03 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 1,140 cpm to 4,620 cpm (gross) were recorded via the GPS datalogger during the walkover survey. Several scan measurement values exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but were all discounted as false positive because the four one-second scan results prior to and after each result (which were identified during data review) including the elevated measurement results averaged 4,073 gross cpm or less. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-03 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-03.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.6: Survey Unit CE-FSS-42-03 Total U and Co-60 Activities (pCi/g)

3.2.7 Survey Unit CE-FSS-42-04

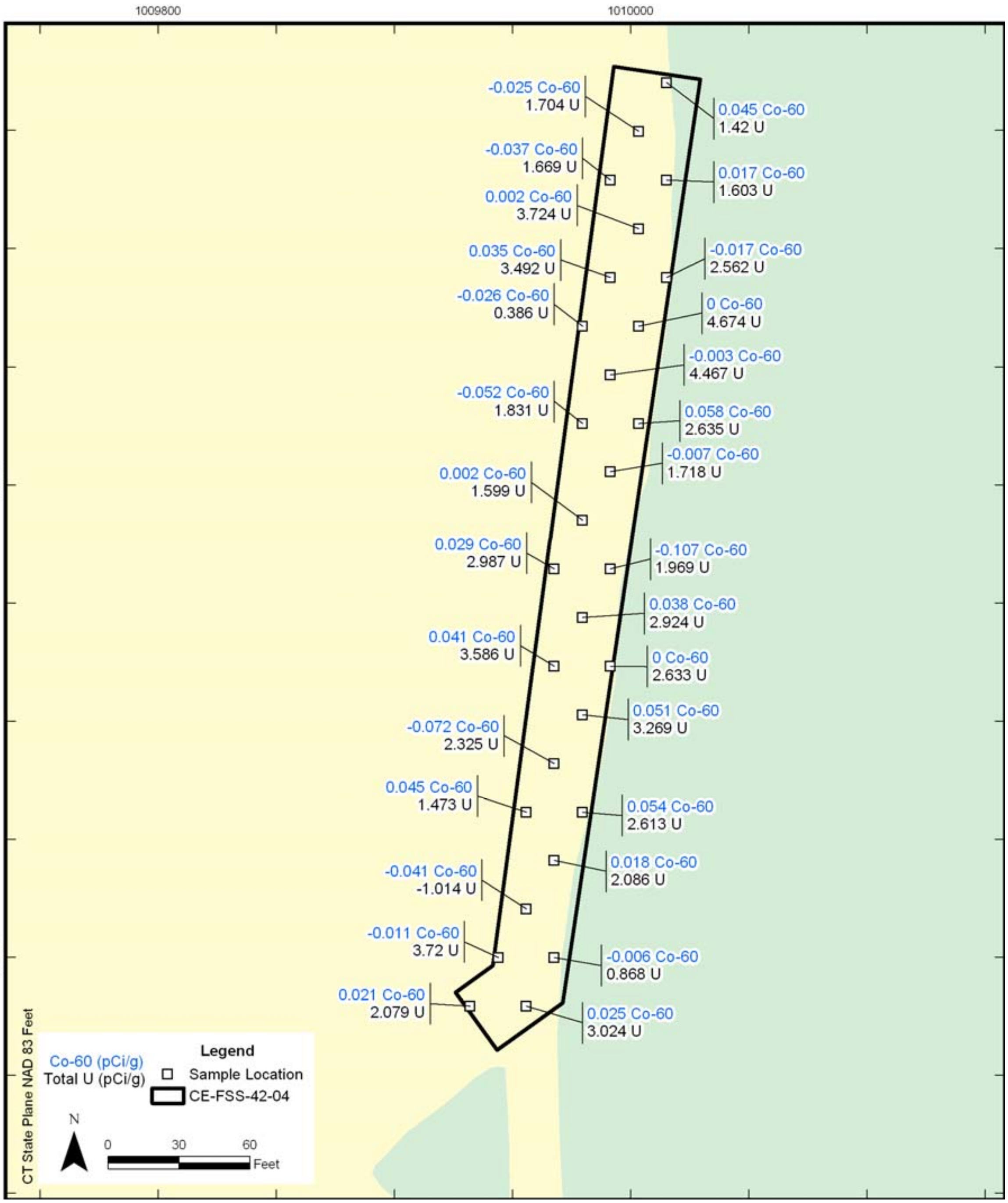
Survey Unit CE-FSS-42-04 is a continuation of the Industrial Waste Line excavation which is bounded on the south by Survey Unit 42-03. The survey unit is approximately 420 feet long and consists of approximately 1,317 m^2 of land area. Figure 3.7 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix G.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-42-04 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 780 cpm to 4,200 cpm (gross) were recorded during the walkover survey. Several scan measurement values exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but were all discounted as false positive because the four one-second scan results prior to and after each result (which were identified during data review) including the elevated measurement results averaged 3,613 gross cpm or less. No elevated readings exceeding the gross investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-04 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-04.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.7: Survey Unit CE-FSS-42-04 Total U and Co-60 Activities (pCi/g)

3.2.8 Survey Unit CE-FSS-42-05

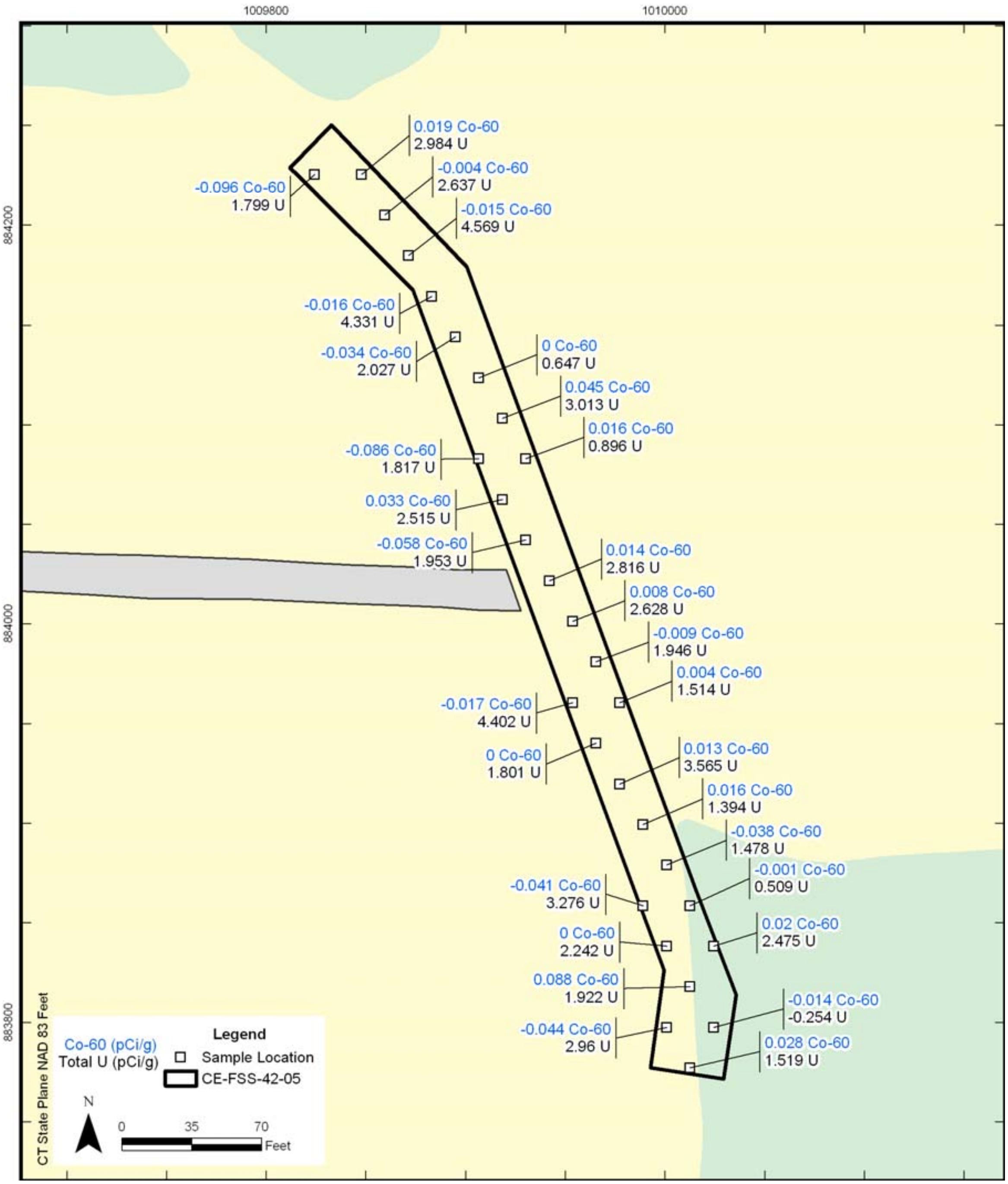
Survey Unit CE-FSS-42-05 is a continuation of the Industrial Waste Line excavation which is bounded on the south by Survey Unit 42-04. The survey unit is approximately 510 feet long and consists of approximately 1,294 m^2 of land area. Figure 3.8 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix H.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-42-05 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 1,140 cpm to 3,540 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the gross investigation level of 4,104 cpm (as listed in Table 2.7) were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-05 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-05.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.8: Survey Unit CE-FSS-42-05 Total U and Co-60 Activities (pCi/g)

3.2.9 Survey Unit CE-FSS-42-06

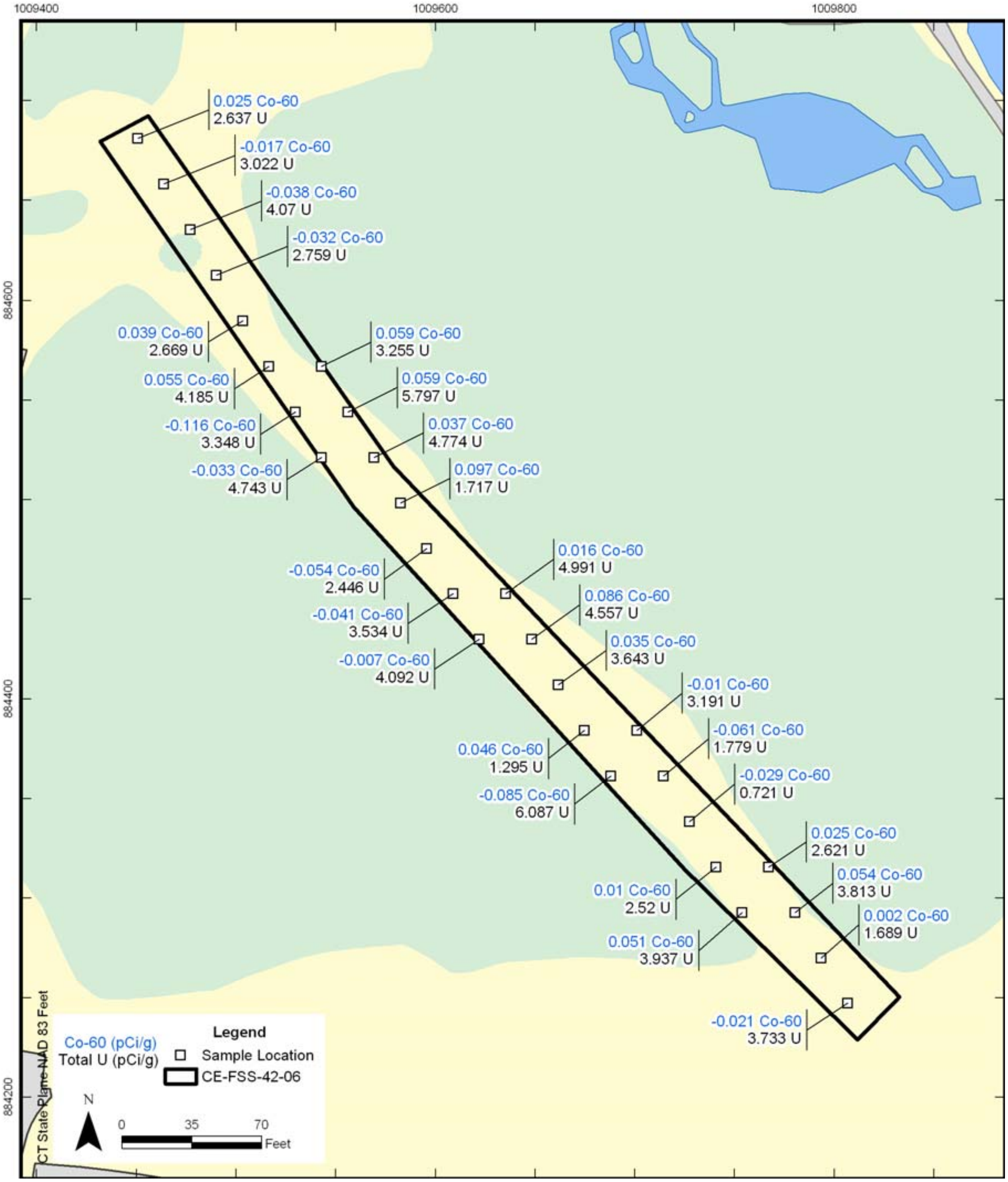
Survey Unit CE-FSS-42-06 is a continuation of the Industrial Waste Line excavation which is bounded on the south by survey unit 42-05. The survey unit is approximately 590 feet long and consists of approximately 1,623 m^2 of land area. Figure 3.9 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix I.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-42-06 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 1,380 cpm to 4,920 cpm (gross) were recorded during the walkover survey. Several scan measurement values exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but were all discounted as false positive because the four one-second scan results prior to and after each result (which were identified during data review) including the elevated measurement results averaged 3,920 gross cpm or less. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-06 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-06.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.9: Survey Unit CE-FSS-42-06 Total U and Co-60 Activities (pCi/g)

3.2.10 Survey Unit CE-FSS-42-07

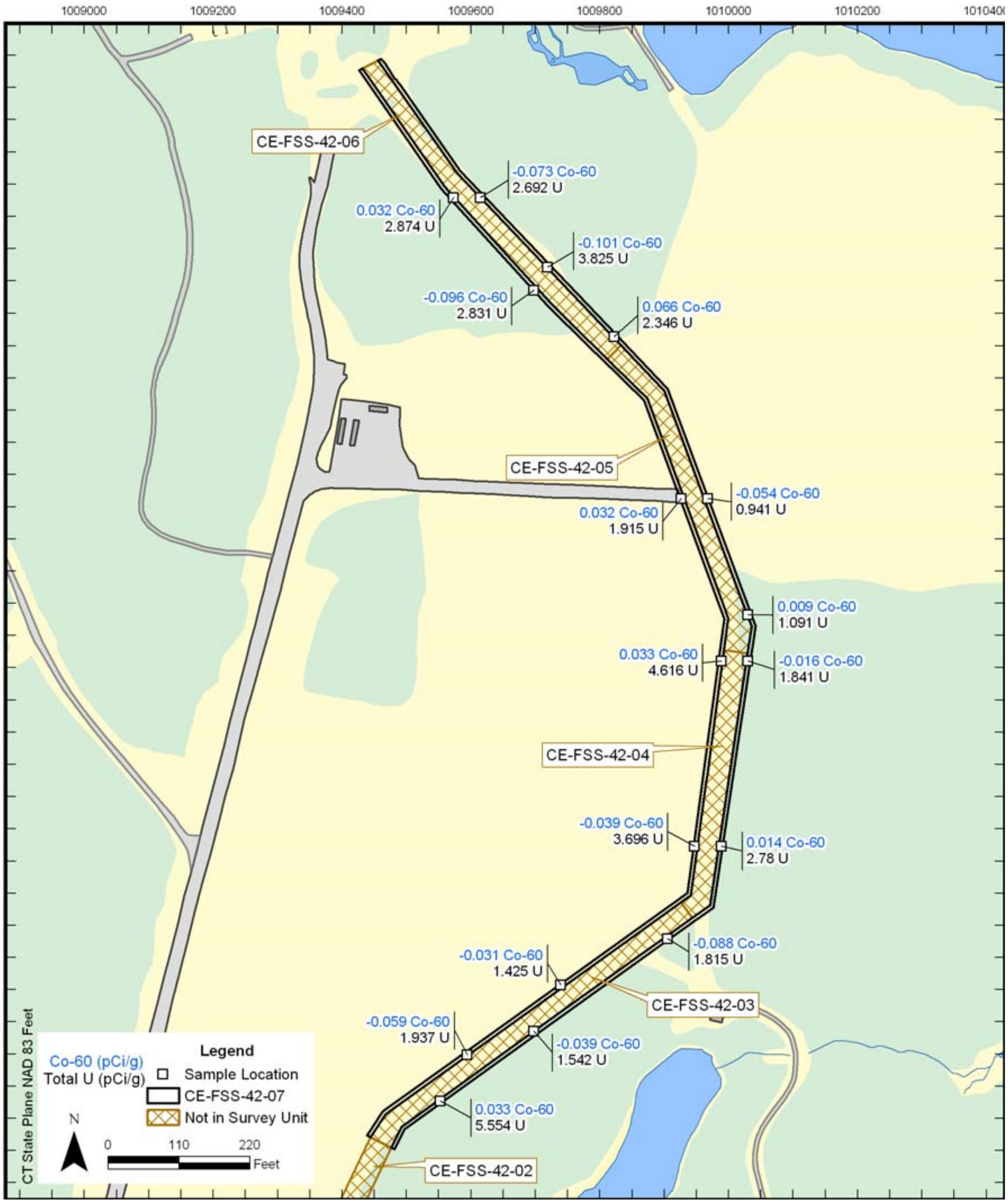
Survey Unit CE-FSS-42-07 is a 6' wide buffer area that covers the Industrial Waste Line excavation area starting at survey unit 42-03. The survey unit is approximately 2,150 feet long and consists of approximately 2,359 m^2 of land area. Figure 3.10 presents an overview of the survey unit. This survey unit is divided into two non-contiguous sections since it is bounded by the east and west sides of the Class 1 survey units. Seventeen 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. Data associated with this survey unit are provided in Appendix J.

Gamma Walkover Survey Results

Approximately 50 percent of the surface area for Survey Unit CE-FSS-42-07 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 900 cpm to 5,340 cpm (gross) were recorded during the walkover survey. Several scan measurement areas contained measurement values that exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but were all discounted as false positive because the four one-second scan results prior to and after each result (which were identified during data review) including the elevated measurement results averaged 3,933 gross cpm or less. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Seventeen systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-07 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-07.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.10: Survey Unit CE-FSS-42-07 Total U and Co-60 Activities (pCi/g)

3.2.11 Survey Unit CE-FSS-42-08

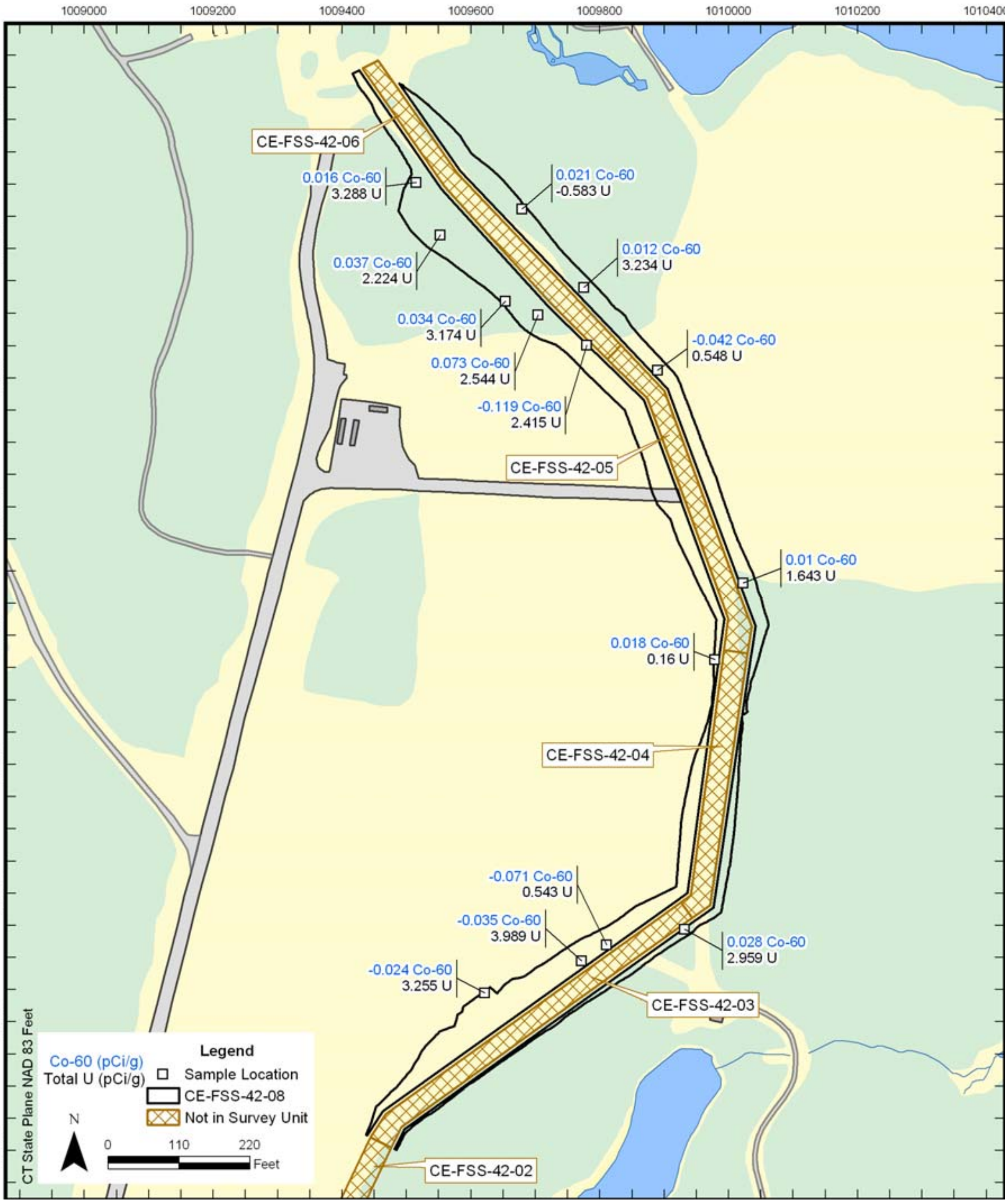
Survey Unit CE-FSS-42-08 includes the remainder of the cutback area excavated before the utility pipe was uncovered and consists of approximately 9,378 m^2 of land area. Figure 3.11 presents an overview of the survey unit. This survey unit is divided into two non-contiguous sections since it represents the boundaries of the Class 2 trench buffer areas. 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.

Gamma Walkover Survey Results

Approximately 10 percent of the surface area for Survey Unit CE-FSS-42-08 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 900 cpm to 4,140 cpm (gross) were recorded during the walkover survey. One scan measurement value exceeded the most conservative gross scan measurement investigation level of 4,104 cpm as listed in Table 2.7 of this report, but was discounted as false positive because the four one-second scan results prior to and after the result (which were identified during data review) including the elevated measurement result averaged 3,400 gross cpm. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Fourteen randomly-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-42-08 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-42-08.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.11: Survey Unit CE-FSS-42-08 Total U and Co-60 Activities (pCi/g)

3.2.12 Survey Unit CE-FSS-43-01

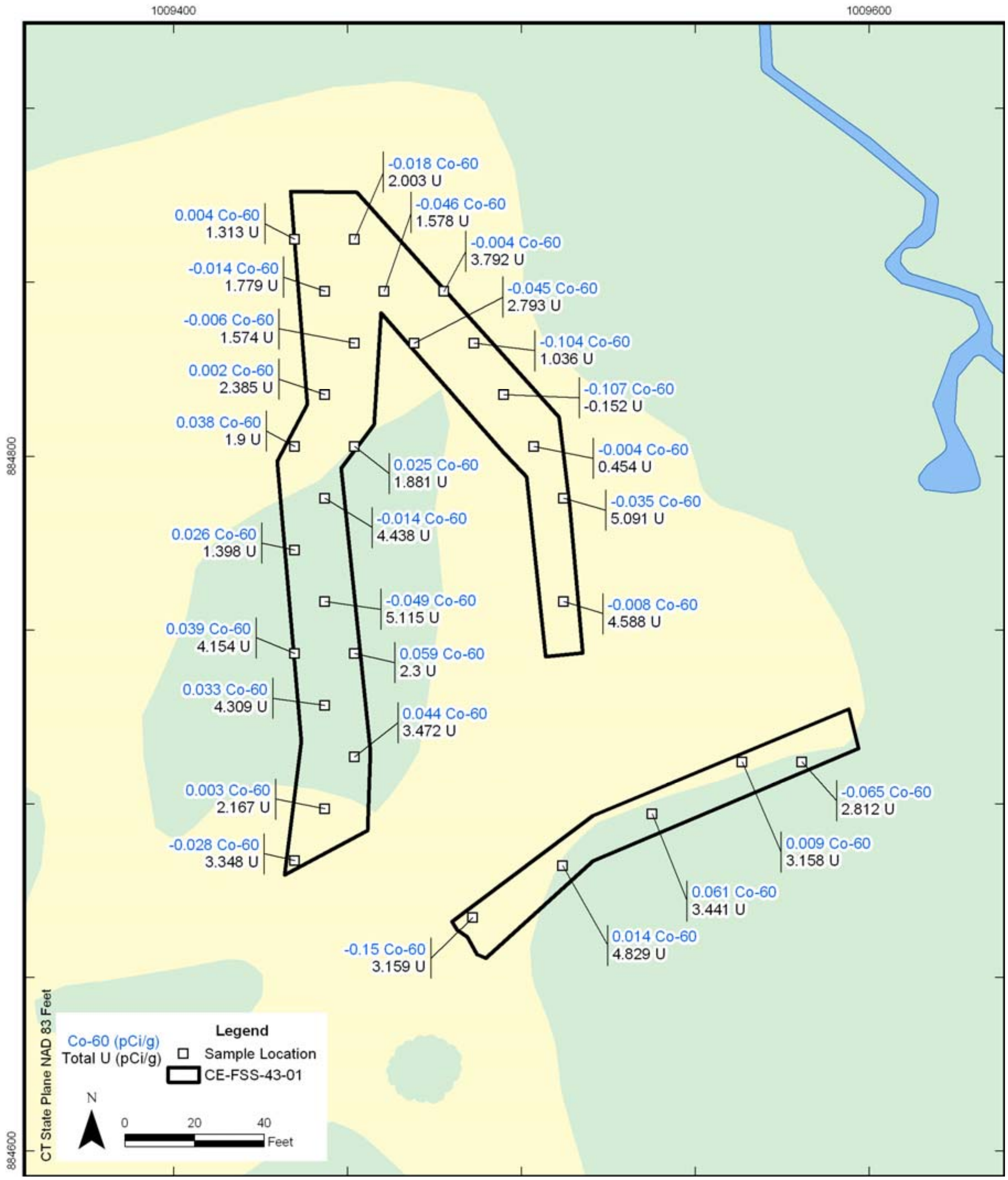
Survey Unit CE-FSS-43-01 is the Industrial Waste Line trench excavation area located within the former WWTP area, including the catch basin and valve vault footprint and consists of approximately 690 m^2 of land area. Figure 3.12 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix L.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-43-01 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 660 cpm to 4,020 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the gross investigation level of 4,104 cpm (as listed in Table 2.7) were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-43-01 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality 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-43-01.



Prepared/Date: BRP 07/28/11
Checked/Date: HTD 07/28/11

Figure 3.12: Survey Unit CE-FSS-43-01 Total U and Co-60 Activities (pCi/g)

3.2.13 Survey Unit CE-FSS-43-02

Survey Unit CE-FSS-43-02 is the WWTP footprint and the 6' buffer area located around the catch basin line excavation, excluding the pH Adjustment Tanks and Survey Unit 43-01, and consists of approximately 2,526 m^2 of land area. Figure 3.13 presents an overview of the survey unit. Seventeen 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. Data associated with this survey unit are provided in Appendix M.

Gamma Walkover Survey Results

Approximately 25 percent of the surface area for Survey Unit CE-FSS-43-02 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 840 cpm to 4,980 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the gross investigation level were identified by the FSS technician during the walkover survey. During review of the scan data file, three measurement areas exceeded the gross scan investigation level of 4,104 cpm as listed in Table 2.7 of this report, two of which were discounted as false positive because the four one-second scan results prior to and after each result (which were identified during data review) including the elevated measurement results averaged 4,013 gross cpm or less.

The other scan measurement area contained measurement values that averaged 4,233 gross cpm. Since this area exceeded the investigation criteria, a rescan of the elevated area was performed to a diameter of about 10 feet. The highest elevated location within the area was located, and a one minute static measurement of the location indicated a gross activity level of 4,134 cpm. Since the one-minute static measurement exceeded the investigation criteria specified in the survey package, a biased soil investigation sample was obtained at the static measurement location. The investigation soil sample activity results did not indicate elevated activity from potential source term radionuclides. However, the activity levels of naturally occurring radioactive material (NORM) found in the investigation sample explain the net count rate for the elevated area. Since the single volumetric sample of soil to investigate the scan anomalies was collected and the gamma analysis results were less than the Site $DCGL_W$s for total uranium and cobalt 60 (and the cause of the elevated scan measurement was due to NORM), no additional investigation volumetric samples of soil were warranted. The investigation sample certificate of analysis (COA) and the investigation summary table can be found in Appendix M of this report.

Volumetric Soil Sample Results

Seventeen systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-43-02 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.13 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-43-02.

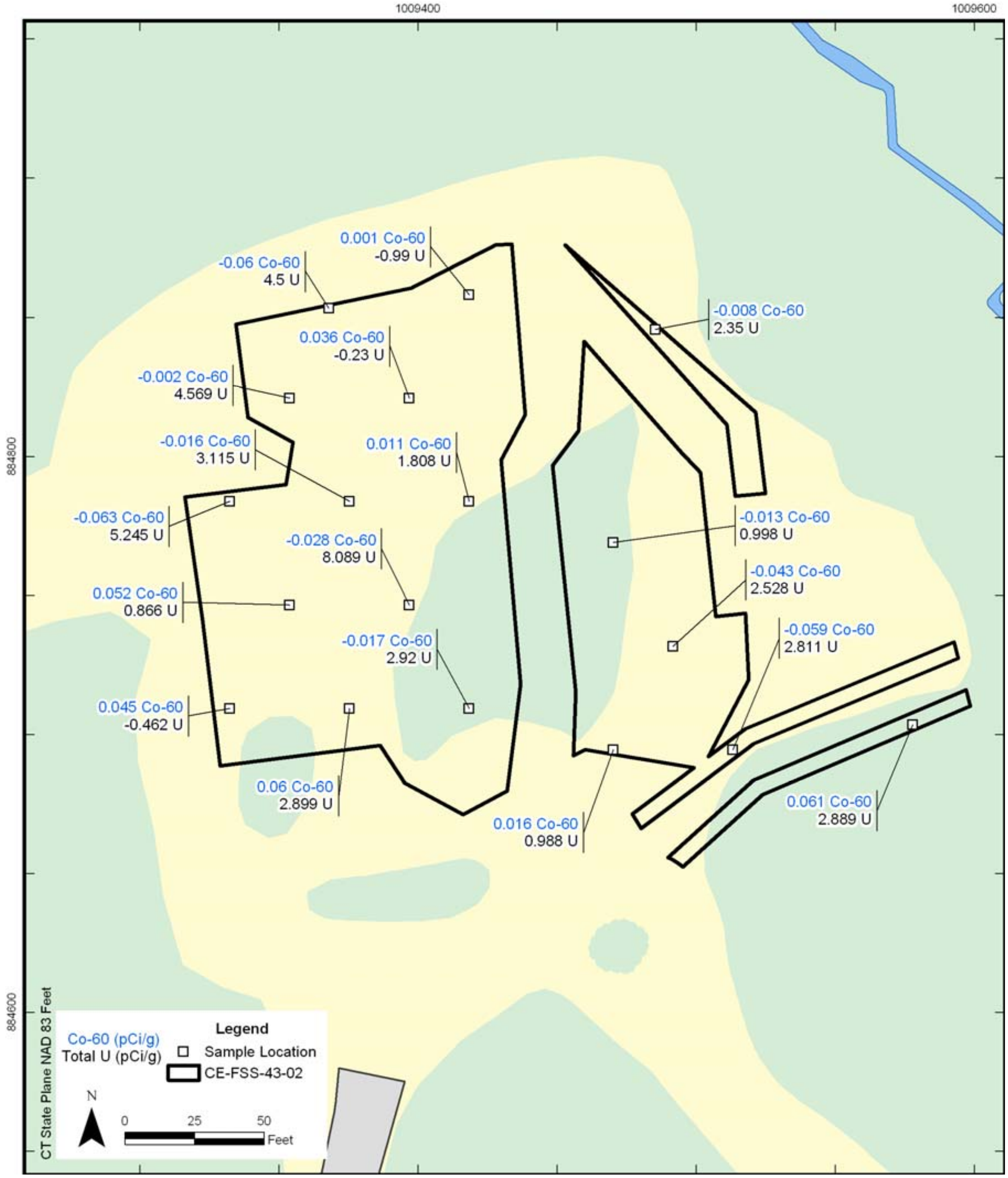


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Figure 3.13: Survey Unit CE-FSS-43-02 Total U and Co-60 Activities (pCi/g)

3.2.14 Survey Unit CE-FSS-43-03

Survey Unit CE-FSS-43-03 is the pH Adjustment Tanks footprint area and associated Industrial Waste Line excavation area and consists of approximately 542 m^2 of land area. Figure 3.14 presents an overview of the survey unit. Twenty-nine survey locations were placed on a systematic grid pattern within the Class 1 survey unit to represent the distribution of residual radioactivity for the survey unit. Data associated with this survey unit are provided in Appendix N.

Gamma Walkover Survey Results

Approximately 100 percent of the surface area for Survey Unit CE-FSS-43-03 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings from 1,140 cpm to 4,500 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified by the FSS technician during the walkover survey. However, during review of the scan data file, it was discovered that one scan measurement area contained measurement values that exceeded the most conservative scan measurement gross investigation level of 4,104 cpm as listed in Table 2.7 of this report. The elevated area averaged just above the scan measurement investigation level at about 4,200 gross cpm, but contained measurement values up to a maximum level of 4,500 gross cpm. A rescan of the elevated area was performed to a diameter of about 10'. The highest elevated location within the area was located, and a one minute static measurement of the location indicated a gross activity level of 3,405 cpm. Even though the one-minute static measurement did not exceed the investigation criteria specified in the survey package, a biased soil investigation sample was obtained at the static measurement location. The investigation soil sample activity results did not indicate elevated activity from potential source term radionuclides. Additionally, the activity levels of NORM found in the investigation sample explain the net count rate for the elevated area. Since the gamma analysis results of the volumetric sample of soil to investigate the scan anomalies were less than the Site $DCGL_{W}$s for total uranium and cobalt 60, no additional investigation volumetric samples of soil were warranted. The investigation sample COA and an investigation summary table can be found in Appendix N of this report.

Volumetric Soil Sample Results

Twenty-nine systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-43-03 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_{W}$. Data quality assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.14 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-43-03.

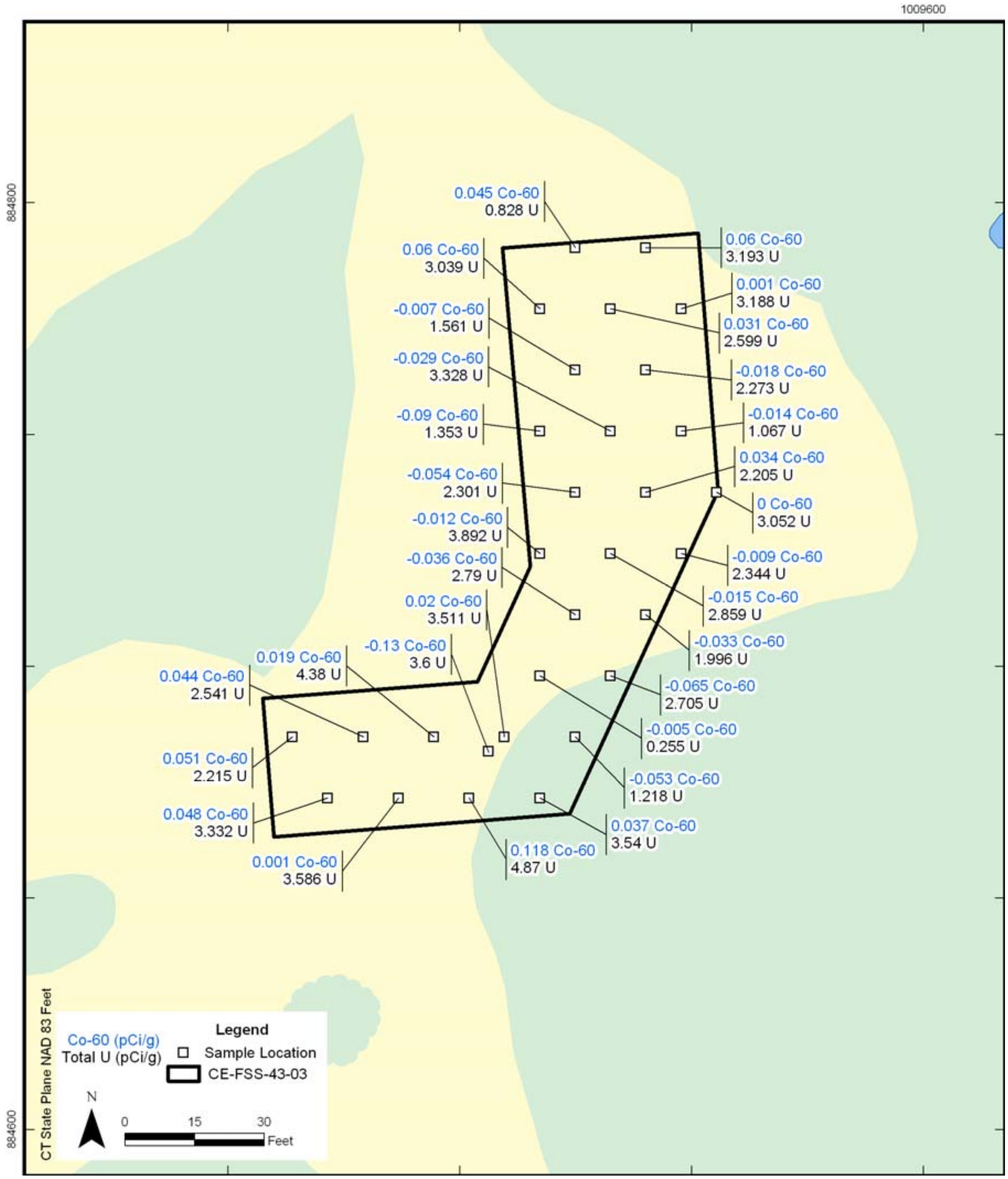

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Figure 3.14: Survey Unit CE-FSS-43-03 Total U and Co-60 Activities (pCi/g)

3.2.15 Survey Unit CE-FSS-43-04

Survey Unit CE-FSS-43-04 is the pH Adjustment Tanks excavation buffer area located around Survey Unit 43-03 and consists of approximately 914 m^2 of land area. Figure 3.15 presents an overview of the survey unit. Seventeen 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. Data associated with this survey unit are provided in Appendix O.

Gamma Walkover Survey Results

Approximately 25 percent of the surface area for Survey Unit CE-FSS-43-04 was surveyed by walking transects across the area, moving the detector from side-to-side in a serpentine motion. Instrument readings ranging from 1,260 cpm to 4,020 cpm (gross) were recorded during the walkover survey. No elevated readings exceeding the investigation level were identified during the walkover survey by the FSS technician. Therefore, no additional volumetric samples of soils to investigate anomalies were collected.

Volumetric Soil Sample Results

Seventeen systematically-placed volumetric soil samples were obtained for FSS in Survey Unit CE-FSS-43-04 and analyzed on Site. The analytical results show that the mean/median soil residual radioactivity is appreciably below the $DCGL_W$. Data quality assessments indicated that the results meet the data quality requirements and are acceptable for use. Figure 3.15 presents the FSS results for both Co-60 and total uranium concentrations for Survey Unit CE-FSS-43-04.

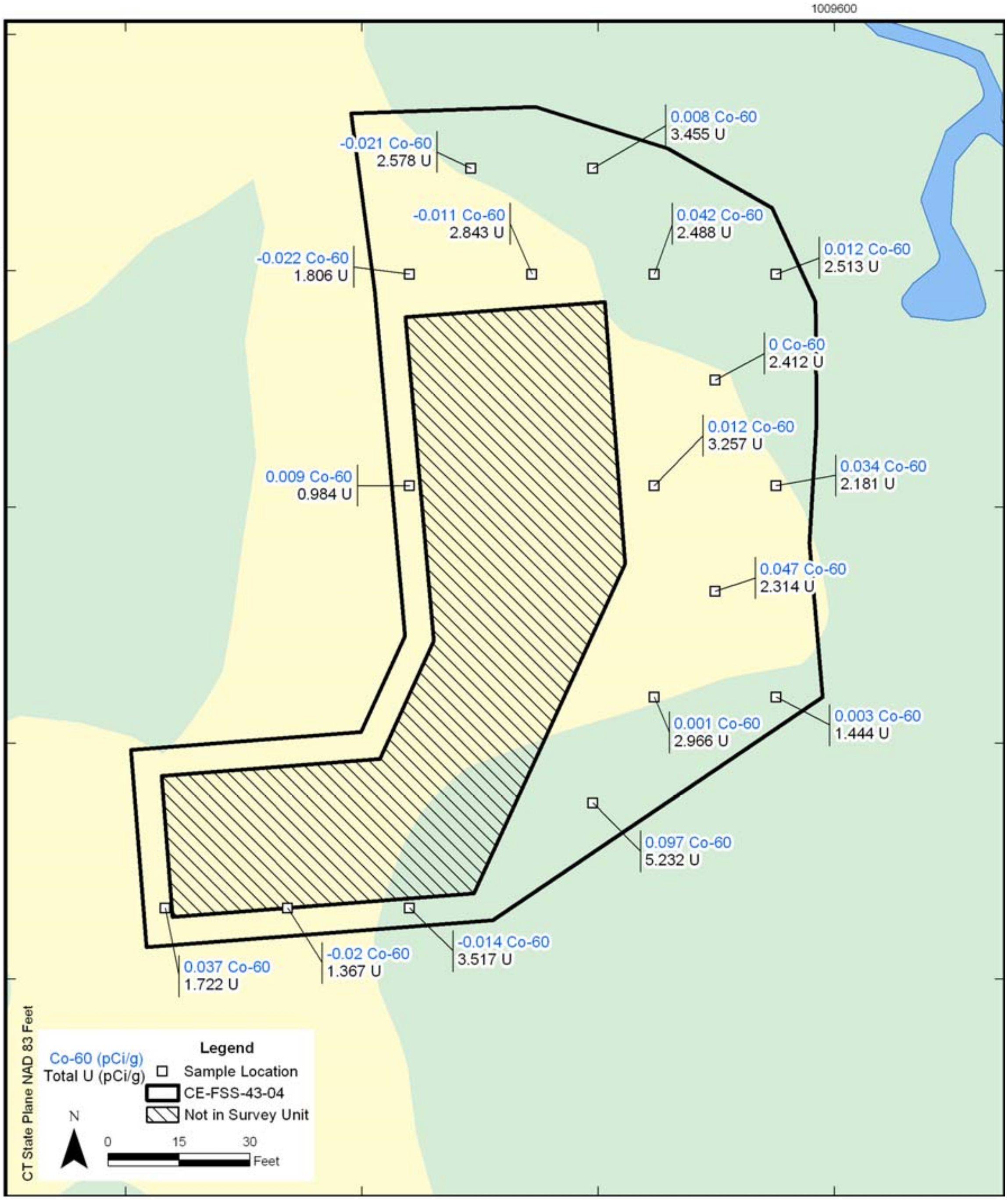

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Figure 3.15: Survey Unit CE-FSS-43-04 Total U and Co-60 Activities (pCi/g)

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. Figures 3.16, 3.17, and 3.18 indicate areas where gamma walkover surveys were performed for the East Main Street sanitary waste line, the industrial waste lines, and the WWTP excavation areas respectively.

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)	Highest Scan Reading (gross cpm)	Highest Scan Reading (net cpm)
41-01	3	10	0	2,800	3,624	824
41-02	3	10	0	3000	3,600	600
41-03	3	10	0	3000	3,600	600
42-01	1	100	0	2,400	4,400	2,000
42-02	1	100	0	2,400	4,380	1,980
42-03	1	100	0	2,400	4,620	2,220
42-04	1	100	0	2,200	4,200	2,000
42-05	1	100	0	2,200	3,540	1,340
42-06	1	100	0	2,200	4,920	2,720
42-07	2	50	0	2,800	5,340	2,540
42-08	3	10	0	2,800	4,140	1,340
43-01	1	100	0	3,000	4,020	1,020
43-02	2	25	1	3,000	4,980	2,180
43-03	1	100	1	2,381	4,500	2,119
43-04	2	25	0	2,200	4,020	1,820

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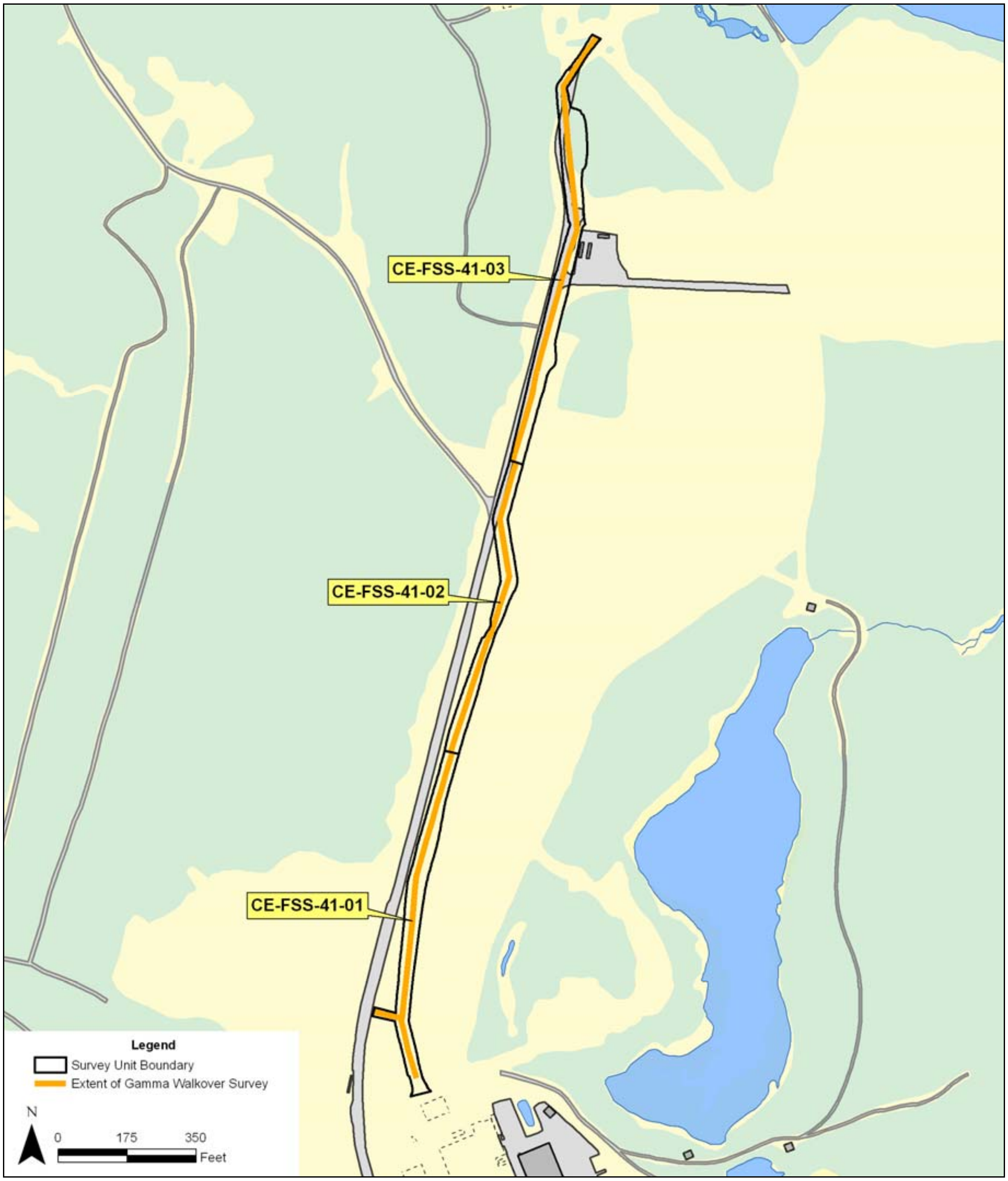

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Figure 3.16: East Main Street Sanitary Waste Line Gamma Walkover Surveys

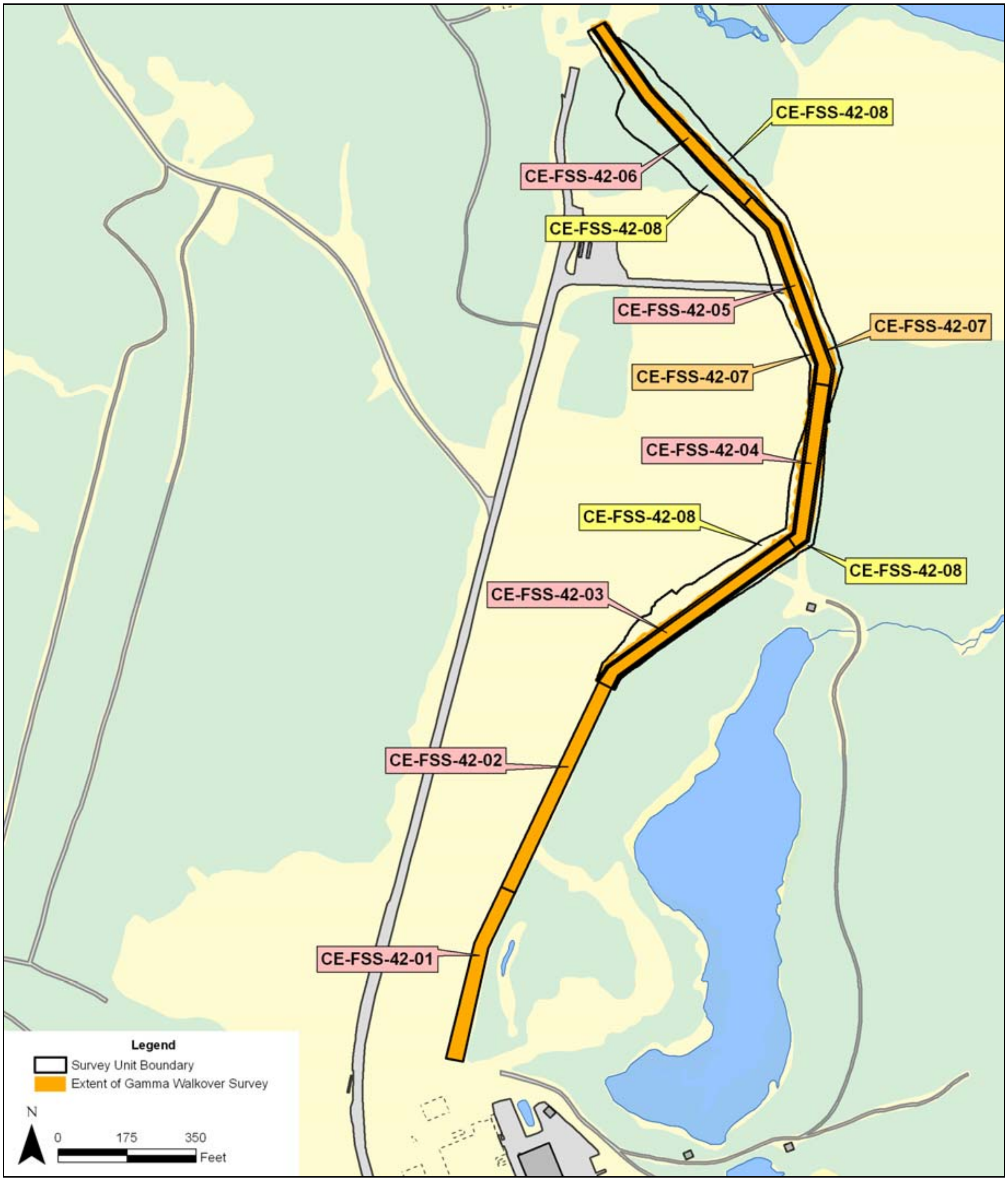

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Figure 3.17: Industrial Waste Line Gamma Walkover Surveys

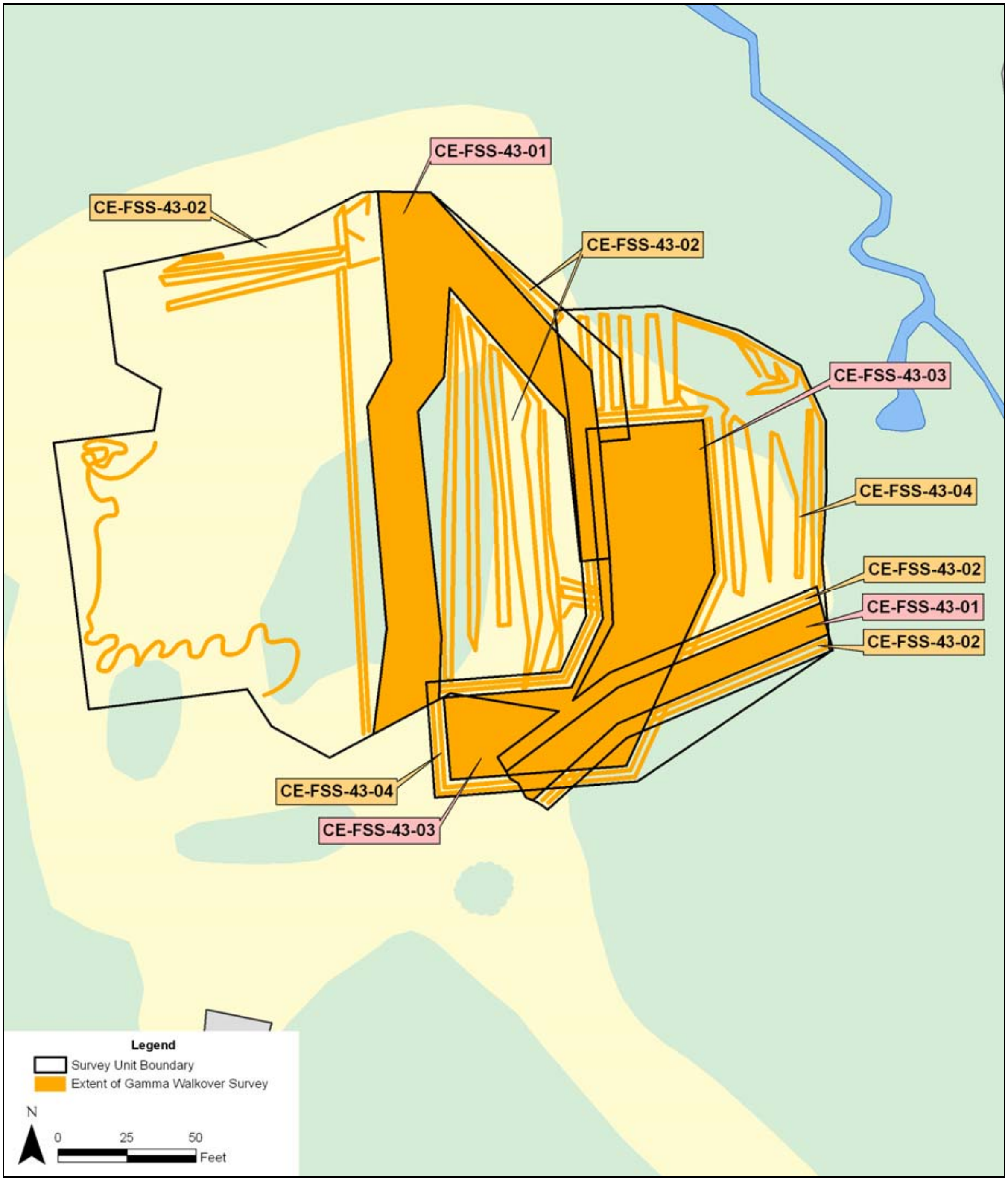

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Figure 3.18: WWTP Gamma Walkover Surveys

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 FSSP, 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 (gross cpm)**	**Avg. Static Measurement Result (net cpm)**
41-01	16	3,214	389
41-02	14	3,184	184
41-03	14	2,993	-7
42-01	29	2,934	134
42-02	29	2,836	36
42-03	29	2,656	256
42-04	29	2,353	323
42-05	29	2,182	-18
42-06	29	3,066	266
42-07	17	2,765	-35
42-08	14	2,604	204
43-01	29	2,957	-43
43-02	17	3,062	262
43-03	29	2,722	341
43-04	17	2,492	292

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

3.3.3 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 FSS units included with this FSS Report.

Table 3.3: Summary Statistics, Total Uranium

Statistic	Survey Unit (CE-FSS)														
	41-01	41-02	41-03	42-01	42-02	42-03	42-04	42-05	42-06	42-07	42-08	43-01	43-02	43-03	43-04
Number of Samples	16	14	14	29	29	29	29	29	29	17	14	29	19	29	17
Arithmetic Mean	2.70	2.56	2.89	3.21	2.68	2.25	2.35	2.25	3.35	2.55	2.05	2.77	2.58	2.84	2.54
Standard Deviation	1.20	1.39	1.57	1.43	1.36	1.51	1.21	1.15	1.30	1.27	1.40	1.41	2.19	1.60	1.01
Standard Error of the Mean	0.30	0.37	0..42	0.26	0.25	0.28	0.22	0.21	0.24	0.31	0.37	0.26	0.50	0.30	0.24
Coefficient of Variation	0.44	0.54	0.54	0.44	0.51	0.67	0.51	0.51	0.39	0.50	0.68	0.51	0.85	0.56	0.40
Geometric Mean	2.70	2.66	2.77	3.14	2.51	2.48	2.22	2.08	3.06	2.27	1.72	2.53	2.69	2.42	2.36
Maximum	4.50	4.8	5.80	7.30	5.70	4.80	4.70	4.60	6.10	5.60	4.00	5.10	8.10	8.04	5.23
Median	2.90	2.60	3.30	3.20	2.70	2.60	2.30	2.00	3.30	2.30	2.45	2.80	2.80	2.53	2.49
Minimum	-0.10	-0.90	-0.10	-0.70	-0.90	-1.00	-1.00	-0.30	0.70	0.90	-0.60	-0.20	-1.00	-0.99	0.98
Range	4.60	5.70	5.90	8.00	6.60	5.80	5.70	4.90	5.40	4.70	4.60	5.30	9.10	9.03	4.25
UCL95 (median)	3.50	3.20	3.40	3.50	3.10	3.30	2.90	2.60	3.90	2.90	3.20	3.40	3.10	3.29	2.97
LCL95 (median)	1.80	1.60	1.50	2.70	2.10	1.50	1.70	1.80	2.60	1.80	0.50	1.90	1.00	0.99	1.81

Note 1: All statistics reported above with the exception of the Number of Samples, the Standard Error of the Mean, and the Coefficient of Variation are in units of pCi/g.

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Table 3.4: Summary Statistics, Co-60

Statistic	Survey Unit (CE-FSS)														
	41-01	41-02	41-03	42-01	42-02	42-03	42-04	42-05	42-06	42-07	42-08	43-01	43-02	43-03	43-04
Number of Samples	16	14	14	29	29	29	29	29	29	17	14	29	19	29	17
Arithmetic Mean	-0.01	0.00	-0.01	0.00	0.00	0.01	0.00	-0.01	0.01	-0.02	0.00	-0.01	0.00	0.00	0.01
Standard Deviation	0.05	0.03	0.07	0.04	0.05	0.05	0.04	0.04	0.05	0.05	0.05	0.05	0.04	0.04	0.03
Standard Error of the Mean	0.01	0.01	0.02	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
Coefficient of Variation	-3.44	8.97	-5.35	-22.84	-14.69	3.69	14.86	-6.45	9.86	-2.35	-16.31	-4.28	-9.45	10.02	2.48
Geometric Mean	0.03	0.02	0.03	0.02	0.03	0.04	0.02	0.02	0.03	0.03	0.02	0.02	0.02	0.02	0.01
Maximum	0.07	0.06	0.07	0.11	0.07	0.08	0.06	0.09	0.10	0.07	0.07	0.06	0.06	0.12	0.10
Median	-0.02	0.01	0.00	0.00	0.00	0.03	0.00	0.00	0.01	-0.03	0.01	0.00	0.01	0.00	0.01
Minimum	-0.08	-0.06	-0.21	-0.09	-0.10	-0.09	-0.11	-0.10	-0.12	-0.10	-0.12	-0.15	-0.06	-0.09	-0.02
Range	0.15	0.11	0.28	0.20	0.18	0.17	0.17	0.18	0.21	0.17	0.19	0.21	0.12	0.21	0.12
UCL95 (median)	0.02	0.02	0.02	0.01	0.03	0.04	0.03	0.01	0.04	0.03	0.03	0.01	0.02	0.03	0.03
LCL95 (median)	-0.05	-0.02	-0.08	-0.02	-0.04	-0.01	-0.01	-0.02	-0.03	-0.06	-0.04	-0.03	-0.04	-0.02	-0.01

Note 1: The coefficient of variation statistics reported above are virtually meaningless since the measured activity for all survey units is at or near 0.00.

Note 2: All statistics reported above with the exception of the Number of Samples, the Standard Error of the Mean, and the Coefficient of Variation are in units of pCi/g.

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3.4 TRENCH SURVEY AND SAMPLING

Field survey and volumetric sampling results were assessed and compared to the release criteria. Where anomalies, or notable results, were identified, additional discussion and data are presented. QC data is presented separately in Section 5 of this report.

3.5 UTILITY TRENCH SURVEY AND SAMPLING OVERVIEW

The sanitary and industrial waste line and other underground utilities were removed in 2010. As-built drawings, as well as test excavations, were used to locate underground utilities. Trench volumetric soil sampling was performed mostly along the bottom of the trench floor and from the spoil piles generated as a result of the excavation to reach the utility. Volumetric sampling methods included use of hand trowels and stainless steel spoons to collect soil and sediment samples from the trench sampling areas. Trench radiological scan surveys were performed using hand-held instruments and appropriate detectors. Trench bottoms and spoil piles were randomly scanned to identify areas of elevated residual radioactivity. If suspect areas (stained or discolored soil) were identified, a biased scan was performed in that area. In areas where scanning measurements exceeded a predetermined action level, the area was marked and a volumetric soil sample taken.

3.5.1 Trench Radiological Surveys

During the excavation of the trench utilities, trench bottoms and the spoils removed from the trenches were radiologically surveyed and sampled. Trench bottoms were scanned using a 2” by 2” NaI detector with appropriate instrument. Spoils and trench bottoms were also scanned if there was indication of leakage from the sanitary and industrial waste line to the surrounding soils. A minimum of one volumetric soil sample was collected at approximately 100 linear foot intervals from the spoil piles and the trench bottoms, and collected in areas where scanning measurements exceeded the predetermined action level.

3.5.2 Trench Scanning Results

Trench bottoms were scanned using a 2” by 2” NaI detector and applicable instrument systems. In areas where scan measurement results exceeded the predetermined action level, the area was marked and a volumetric soil sample taken. None of the scans of the trench bottoms or spoil piles resulted in readings greater than the action level.

3.5.3 Trench Volumetric Sampling Results

A total of 343 volumetric soil samples were collected from the trench bottoms and spoil piles generated during utility excavation activities for the sanitary waste lines, industrial waste lines, and the waste water treatment facility. Each volumetric soil sample was analyzed on the on-site HPGe gamma spectroscopy system.

3.6 UTILITY TRENCH LOCATION

Trench excavation and utility removal was performed in sections to minimize the number of trench excavations left open at any one time. After removal of the utility from the trench, the trench was radiologically surveyed and sampled. Once the trench was radiologically checked by FSS and CTDEP, the trench was backfilled and graded to match surrounding grade. In the sanitary waste lines, industrial waste lines, and the waste water treatment facility areas, trench excavation, utility removal, and radiological survey and sampling were performed from May 2010 through November 2010. Utility trenches were segregated into five definitive trench groups: hot waste line, industrial waste line, sanitary sewer waste line, storm water drains, and other utilities. Other utilities include those that had minimal potential to contain residual radioactive materials or contribute to the release or spread of residual radioactive materials to the environment.

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 FSS data from a single survey unit indicates that:

- The mean/median volumetric soil sample measurement result 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 $DCGL_{EMC}$; **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 survey unit meets the criteria for release from radiological controls without restriction.

An ALARA analysis in agreement with NRC guidance provided in NUREG-1757 (NRC, 2006) was performed as part of the DP. The analysis shows that shipping affected 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 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-41-01	CE-FSS-41-02	CE-FSS-41-03	CE-FSS-42-01	CE-FSS-42-02	CE-FSS-42-03	CE-FSS-42-04	CE-FSS-42-05	CE-FSS-42-06	CE-FSS-42-07	CE-FSS-42-08	CE-FSS-43-01	CE-FSS-43-02	CE-FSS-43-03	CE-FSS-43-04
Unity	Power of Sign Test	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1	~1
Total U (pCi/g)	Median	3.05	2.60	3.25	3.20	2.70	2.80	2.30	2.00	3.40	2.30	2.48	2.79	2.53	2.71	2.49
	UCL_{95} of Median	3.60	3.20	3.40	3.50	3.10	3.30	2.90	2.60	3.90	2.90	3.23	3.44	3.39	3.29	3.19
	Arithmetic Mean	3.11	2.45	2.88	3.21	2.66	2.29	2.35	2.26	3.37	2.59	2.10	2.76	2.47	2.84	2.54
	Geometric Mean	2.95	2.53	2.76	3.14	2.47	2.52	2.22	2.08	3.08	2.28	1.74	2.52	2.59	2.42	2.35
	Maximum	8.90	4.80	5.70	7.30	5.70	4.80	4.70	4.60	6.10	5.60	3.99	5.09	8.04	9.08	5.22
Co-60 (pCi/g)	Median	-0.02	0.01	0.00	0.00	0.00	0.03	0.00	0.00	0.01	-0.03	0.01	0.00	0.00	0.00	0.01
	UCL_{95} of Median	0.02	0.02	0.02	0.01	0.03	0.04	0.03	0.01	0.04	0.03	0.03	0.01	0.03	0.03	0.03
	Arithmetic Mean	-0.01	0.00	-0.01	0.00	0.00	0.01	0.00	-0.01	0.01	-0.02	0.00	-0.01	0.00	0.00	0.01
	Geometric Mean	0.03	0.02	0.03	0.03	0.03	0.04	0.02	0.02	0.03	0.03	0.02	0.02	0.02	0.03	0.01
	Maximum	0.07	0.06	0.07	0.11	0.08	0.08	0.06	0.09	0.10	0.07	0.07	0.06	0.06	0.12	0.10

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, 3.9 pCi/g, is less than the $DCGL_W$ by a factor of more than 142, and the highest Co-60 UCL_{95} estimate of the median, 0.04 pCi/g, is less than the $DCGL_W$ by a factor of more than 125. 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 FSS 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 9.1 pCi/g and 0.12 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

A 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 O), 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 survey units.

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 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 AMEC supervisory personnel and the Site RSO. Electronic copies of original electronic data sets are preserved on a retrievable data storage device. No data reduction, filtering, or manipulation was performed on the original electronic versions of data sets.

Record copies of surveys, sampling, and analytical data (and supporting data) are provided in the survey unit data appendices (A through O).

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 a commercial laboratory for off-site analysis are provided in the survey unit data appendices (A through O).

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 23 duplicate samples were collected from an overall sample population of 341 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 (Split) Sample Measurement Results

Sample ID (CE-FSS)	Co-60 (pCi/g)			U-235 (pCi/g)		
	Activity	Uncert.	MDA	Activity	Uncert.	MDA
41-01-003	-9.16E-02	1.16E-01	1.25E-01	1.14E-01	9.31E-02	1.48E-01
41-01-003S	-6.54E-02	1.13E-01	1.27E-01	1.11E-01	8.46E-02	1.37E-01
41-02-013	-1.46E-02	8.95E-02	1.31E-01	4.93E-02	9.21E-02	1.35E-01
41-02-013S	-2.67E-02	9.82E-02	1.38E-01	1.13E-01	8.93E-02	1.45E-01

Table 5.1: Duplicate (Split) Sample Measurement Results

Sample ID (CE-FSS)	Co-60 (pCi/g)			U-235 (pCi/g)		
	Activity	Uncert.	MDA	Activity	Uncert.	MDA
41-03-003	-8.19E-02	1.44E-01	1.83E-01	9.54E-02	1.20E-01	1.83E-01
41-03-003S	7.40E-02	7.97E-02	1.27E-01	1.10E-01	8.05E-02	1.27E-01
42-01-018	3.65E-02	8.00E-02	1.39E-01	1.14E-01	9.68E-02	1.54E-01
42-01-018S	-4.74E-02	8.95E-02	1.09E-01	1.14E-01	8.96E-02	1.43E-01
42-01-020	-3.01E-02	9.31E-02	1.26E-01	9.10E-02	9.12E-02	1.46E-01
42-01-020S	4.38E-02	1.03E-01	1.71E-01	9.68E-02	8.21E-02	1.29E-01
42-02-009	6.32E-04	1.08E-01	1.62E-01	7.48E-02	8.85E-02	1.42E-01
42-02-009S	-3.73E-03	9.02E-02	1.40E-01	3.57E-03	1.04E-01	1.46E-01
42-02-029	3.80E-02	7.24E-02	1.41E-01	9.10E-02	9.42E-02	1.48E-01
42-02-029S	-8.06E-03	8.51E-02	1.28E-01	1.15E-01	9.14E-02	1.48E-01
42-03-011	-2.41E-03	8.64E-02	1.31E-01	6.47E-02	9.32E-02	1.40E-01
42-03-011S	-5.59E-02	9.13E-02	1.05E-01	2.03E-02	1.02E-01	1.45E-01
42-03-022	2.43E-02	8.20E-02	1.40E-01	8.23E-02	9.69E-02	1.49E-01
42-03-022S	3.77E-02	9.44E-02	1.59E-01	1.55E-01	9.43E-02	1.55E-01
42-04-008	6.11E-02	7.54E-02	1.53E-01	1.02E-01	8.41E-02	1.31E-01
42-04-008S	4.77E-02	4.94E-02	7.40E-02	6.39E-02	8.84E-02	1.36E-01
42-04-028	-1.77E-02	1.11E-01	1.24E-01	3.75E-02	8.91E-02	1.33E-01
42-04-028S	-3.23E-02	1.08E-01	1.29E-01	7.24E-02	9.07E-02	1.41E-01
42-05-006	2.00E-02	8.71E-02	1.36E-01	1.70E-01	8.96E-02	1.53E-01
42-05-006S	1.92E-02	6.23E-02	1.15E-01	-1.07E-02	8.42E-02	1.17E-01
42-05-019	3.70E-02	5.27E-02	8.98E-02	9.23E-02	8.52E-02	1.36E-01
42-05-019S	2.92E-02	8.65E-02	1.45E-01	6.99E-02	6.93E-02	9.72E-02
42-06-011	-5.17E-02	1.15E-01	1.49E-01	3.68E-02	1.05E-01	1.56E-01
42-06-011S	3.17E-02	8.89E-02	1.53E-01	1.69E-01	9.33E-02	1.59E-01
42-06-012	1.91E-02	1.05E-01	1.70E-01	1.17E-01	9.37E-02	1.52E-01
42-06-012S	5.01E-02	8.02E-02	1.59E-01	1.18E-01	9.77E-02	1.58E-01
42-07-007	2.94E-02	7.02E-02	1.33E-01	1.05E-01	7.98E-02	1.34E-01
42-07-007S	-2.39E-03	8.33E-02	1.31E-01	7.43E-02	8.52E-02	1.35E-01
42-08-002	-4.80E-02	8.04E-02	1.03E-01	3.83E-02	9.12E-02	1.36E-01
42-08-002S	-3.57E-02	1.00E-01	1.32E-01	-2.96E-03	8.61E-02	1.21E-01
43-01-004	-1.46E-02	1.02E-01	1.44E-01	1.31E-01	9.02E-02	1.44E-01

Table 5.1: Duplicate (Split) Sample Measurement Results

Sample ID (CE-FSS)	Co-60 (pCi/g)			U-235 (pCi/g)		
	Activity	Uncert.	MDA	Activity	Uncert.	MDA
43-01-004S	3.26E-02	6.60E-02	1.24E-01	7.32E-02	1.03E-01	1.56E-01
43-01-024	-7.36E-02	1.07E-01	1.08E-01	9.06E-04	9.02E-02	1.27E-01
43-01-024S	-1.34E-01	1.12E-01	1.07E-01	6.24E-02	7.12E-02	1.15E-01
43-02-015	-5.80E-02	1.10E-01	1.37E-01	1.32E-01	1.06E-01	1.67E-01
43-02-015S	-6.16E-02	1.03E-01	1.28E-01	1.47E-01	9.86E-02	1.60E-01
43-03-018	2.73E-02	1.01E-01	1.64E-01	2.41E-03	1.01E-01	1.42E-01
43-03-018S	4.02E-02	7.32E-02	1.34E-01	1.40E-01	8.65E-02	1.33E-01
43-03-024	-1.12E-02	9.76E-02	1.39E-01	5.84E-02	8.20E-02	1.24E-01
43-03-024S	-2.49E-02	6.99E-02	8.97E-02	8.83E-02	6.79E-02	1.10E-01
43-04-012	1.75E-02	8.87E-02	1.38E-01	2.66E-02	1.06E-01	1.53E-01
43-04-012S	-6.15E-02	9.70E-02	1.16E-01	8.97E-02	7.64E-02	1.20E-01

Note:
S=Split
MDA= minimum detectable activity

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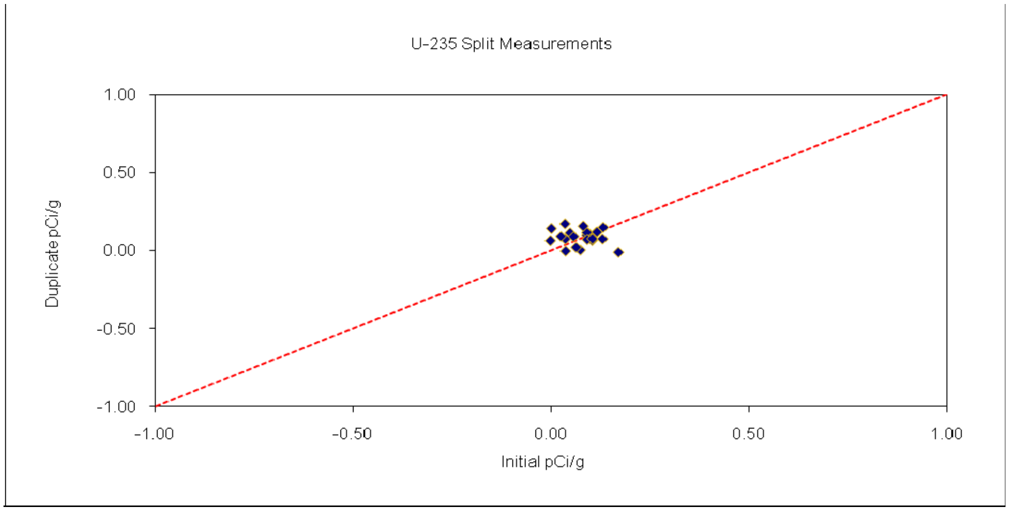
The overall quality of the volumetric soil sample data is evident in the graphic presentation in Figure 5.1 (U-235) and Figure 5.2 (Co-60).



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Checked/Date: HTD 07/28/11

Figure 5.1: U-235 Duplicate Measurement Result Comparisons

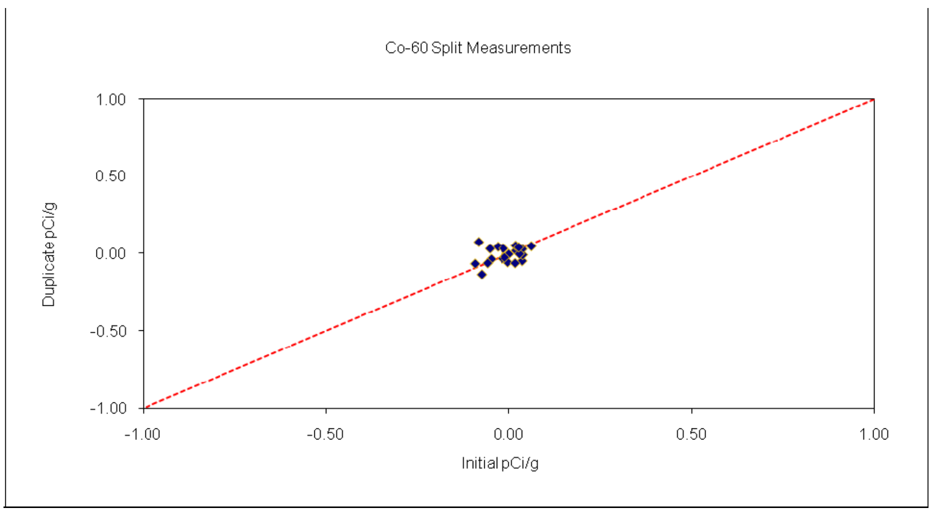

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Figure 5.2: Co-60 Duplicate Measurement Result Comparisons

5.1.3.2 Laboratory Instrumentation Replicate (Lab Recounts)

Another QC metric for monitoring instrument precision consists of a laboratory instrument replicate count, which is the repeated measurement of a sample that has been prepared for counting (i.e., laboratory sample preparation and radiochemical procedures have been completed). It is used to determine the precision of the analytical system (repeated measurements using the same instrument) and the instrument calibration (repeated measurements using two different instruments, such as two different germanium detectors with multichannel analyzers). Laboratory Instrumentation Replicate counts were performed in the HP count laboratory at a frequency of no less than 5 percent (1:20) for the sample population expected, and were performed on the on-site gamma spectroscopy system for analysis as laboratory recounts (LR). Every survey unit was represented with replicate samples being collected from that survey unit. A total of 23 laboratory instrument replicate analyses were performed from an overall sample population of 341 volumetric samples, equating to a sampling frequency significantly greater than the 1:20 minimum requirement. The results of the replicate sample analyses were evaluated in comparison to the results obtained from the original analysis. Each of the replicate sample results was within the expected tolerance for the analysis, providing additional evidence that the sample preparation, extraction, and measurement processes were precise (Table 5.2).

Table 5.2: Laboratory Instrumentation Replicate (Laboratory Recounts) Sample Measurement Results

Sample ID (CE-FSS)	Co-60 (pCi/g)			U-235 (pCi/g)		
	Activity	Uncert.	MDA	Activity	Uncert.	MDA
41-01-013	-3.16E-02	1.14E-01	1.59E-01	-1.60E-03	9.50E-02	1.33E-01
41-01-013LR	-1.07E-02	9.86E-02	1.38E-01	1.17E-01	9.48E-02	1.55E-01
41-02-010	9.44E-03	8.88E-02	1.36E-01	9.13E-02	8.22E-02	1.30E-01
41-02-010LR	5.18E-03	6.87E-02	1.13E-01	1.19E-01	7.80E-02	1.28E-01
41-03-010	5.34E-03	1.11E-01	1.75E-01	1.14E-01	9.93E-02	1.60E-01
41-03-010LR	9.39E-03	1.03E-01	1.64E-01	1.06E-01	1.07E-01	1.67E-01
42-01-005	-5.36E-02	1.16E-01	1.52E-01	4.93E-02	4.22E-02	7.36E-02
42-01-005LR	-8.86E-02	1.17E-01	1.44E-01	9.98E-02	8.65E-02	1.42E-01
42-01-007	3.68E-02	4.34E-02	6.85E-02	1.15E-01	9.48E-02	1.50E-01
42-01-007LR	-1.49E-02	7.93E-02	1.10E-01	1.47E-01	9.64E-02	1.56E-01
42-02-014	-4.74E-02	1.03E-01	1.11E-01	5.96E-02	9.50E-02	1.43E-01
42-02-014LR	-4.64E-02	1.06E-01	1.38E-01	1.29E-01	1.34E-01	2.23E-01
42-02-028	7.83E-02	8.20E-02	1.63E-01	9.29E-02	8.99E-02	1.42E-01
42-02-028LR	2.97E-02	8.67E-02	1.50E-01	3.63E-02	9.78E-02	1.44E-01
42-03-013	-4.09E-02	1.03E-01	1.42E-01	4.64E-02	9.03E-02	1.36E-01
42-03-013LR	3.54E-02	9.54E-02	1.52E-01	8.05E-02	9.40E-02	1.46E-01
42-03-020	8.32E-02	7.85E-02	1.64E-01	7.62E-02	9.94E-02	1.51E-01
42-03-020LR	4.36E-02	6.97E-02	1.38E-01	1.39E-01	7.87E-02	1.37E-01
42-04-014	1.04E-01	7.91E-02	1.76E-01	1.10E-01	1.12E-01	1.71E-01
42-04-014LR	-6.20E-02	1.02E-01	1.24E-01	2.39E-02	1.09E-01	1.56E-01
42-05-005	3.11E-03	8.62E-02	1.33E-01	1.35E-01	8.25E-02	1.41E-01
42-05-005LR	5.50E-02	8.29E-02	1.53E-01	5.79E-02	1.01E-01	1.51E-01

Table 5.2: Laboratory Instrumentation Replicate (Laboratory Recounts) Sample Measurement Results

Sample ID (CE-FSS)	Co-60 (pCi/g)			U-235 (pCi/g)		
	Activity	Uncert.	MDA	Activity	Uncert.	MDA
42-05-015	1.50E-02	8.85E-02	1.42E-01	6.93E-02	8.84E-02	1.37E-01
42-05-015LR	-1.56E-02	8.38E-02	1.42E-01	7.53E-02	8.19E-02	1.30E-01
42-06-006	-6.18E-02	1.11E-01	1.22E-01	7.02E-02	6.76E-02	1.11E-01
42-06-006LR	4.47E-02	7.14E-02	1.41E-01	5.53E-02	9.91E-02	1.49E-01
42-06-026	-2.04E-02	1.07E-01	1.54E-01	7.97E-02	8.12E-02	1.34E-01
42-06-026LR	6.95E-02	8.58E-02	1.74E-01	8.94E-02	1.05E-01	1.71E-01
42-07-006	-1.20E-02	1.06E-01	1.53E-01	7.70E-02	8.93E-02	1.42E-01
42-07-006LR	-5.14E-02	1.42E-01	1.87E-01	1.01E-01	1.11E-01	1.80E-01
42-08-010	-7.59E-02	1.26E-01	1.49E-01	1.48E-01	9.72E-02	1.59E-01
42-08-010LR	-2.35E-03	8.31E-02	1.24E-01	9.02E-02	9.89E-02	1.52E-01
43-01-014	4.28E-02	9.09E-02	1.60E-01	8.75E-02	7.32E-02	1.18E-01
43-01-014LR	-1.15E-02	7.64E-02	1.16E-01	1.25E-01	9.91E-02	1.61E-01
43-01-020	4.51E-02	5.24E-02	8.44E-02	4.46E-02	9.29E-02	1.37E-01
43-01-020LR	6.20E-03	7.56E-02	1.19E-01	4.56E-02	9.03E-02	1.34E-01
43-02-015	-3.67E-02	1.09E-01	1.05E-01	1.04E-01	7.28E-02	1.19E-01
43-02-015LR	4.10E-02	5.80E-02	1.14E-01	4.96E-02	7.46E-02	1.13E-01
43-03-022	-5.80E-02	1.10E-01	1.37E-01	1.32E-01	1.06E-01	1.67E-01
43-03-022LR	-5.84E-02	1.18E-01	1.49E-01	-7.33E-03	1.06E-01	1.47E-01
43-03-025	3.02E-04	6.07E-02	9.53E-02	3.94E-02	7.73E-02	1.13E-01
43-03-025LR	-2.76E-02	8.65E-02	1.22E-01	2.95E-02	8.16E-02	1.19E-01
43-04-013	4.50E-02	7.49E-02	1.35E-01	1.08E-01	7.91E-02	1.28E-01
43-04-013LR	7.59E-02	6.09E-02	1.35E-01	8.81E-02	7.79E-02	1.28E-01
41-01-013	-1.23E-02	1.03E-01	1.52E-01	1.34E-01	8.87E-02	1.48E-01
41-01-013LR	-1.01E-02	9.58E-02	1.42E-01	4.94E-02	9.80E-02	1.46E-01

Note:
LR=Laboratory Recount

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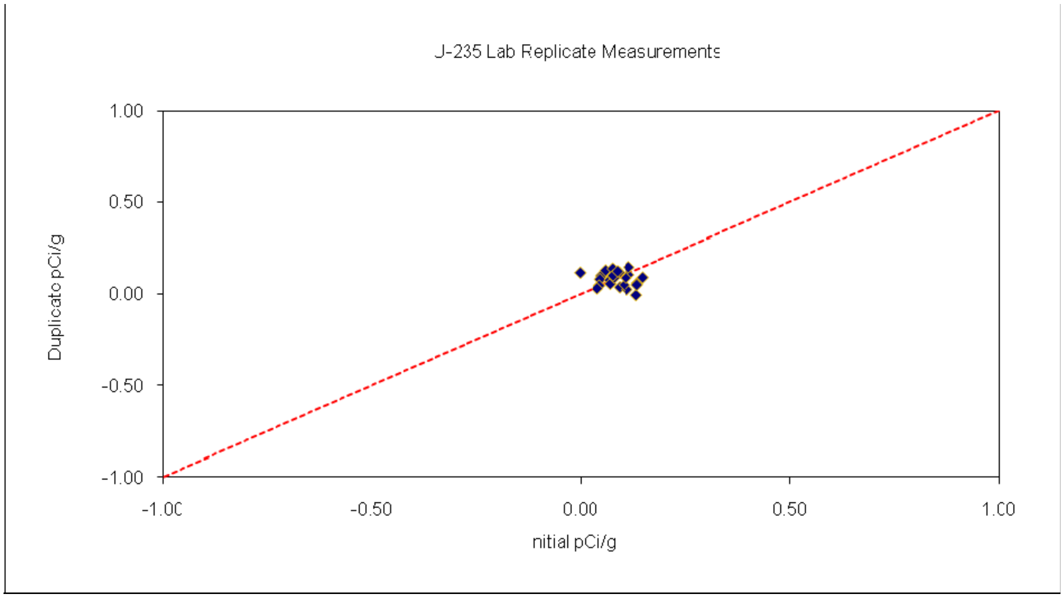
The overall quality of the volumetric soil sample data reproducibility is evident in the graphic presentation in Figure 5.3 (U-235) and Figure 5.4 (Co-60).



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Checked/Date: HTD 07/28/11

Figure 5.3: U-235 Laboratory Replicate Measurement Result Comparisons

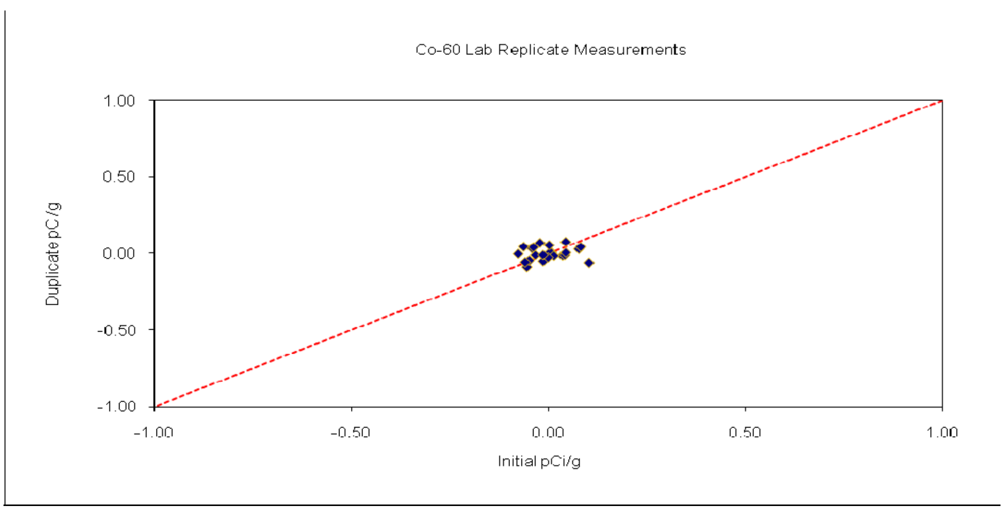

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Figure 5.4: Co-60 Laboratory Replicate Measurement Result Comparisons

5.1.4 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 P.

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 FSSs were 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 P.

Additionally, to provide a comparison between different technician survey technique and separate instrument systems, field duplicate static measurements were conducted by a different technician using a different instrument at randomly selected volumetric soil sample locations in situ prior to sample collection, at the rate of 5% per survey unit. This QC measurement is used to identify possible differences in the survey technique between operators, and instrument bias. The field duplicate count comparison results are presented in Appendix P. Since most of the net static results were near the detection level of the instrument, and both instruments used identical NaI detectors, raw (uncorrected for background) gross count results were evaluated in the comparison.

5.1.5 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 (1,332 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 Q. Examination of this data concludes that the gamma spectroscopy system was functioning correctly during FSS. Starting in mid-September 2010, the 59 keV peak FWHM exhibited a trend of about 5.3% over the establish mean of about 0.72 keV, as noted during review of the control chart. This condition occurred until about mid-November. Since the FWHM parameter provides a metric of the peak shape relative to the initial peak shape calibration, a trend could indicate problems with the instrument hardware and/or the detector crystal. The cause of the problem could include electronic noise, moisture in the detector preamplifier, or the formation of a dead layer in the detector crystal. Since the FWHM trend was not evident during the period from review of the mid (662 keV) and high (1,332 keV) energy ranges, and the decay corrected activity was within tolerance for all three energy ranges, it is apparent that there was no adverse impact to data quality. As a standard maintenance protocol to troubleshoot or correct resolution issues with the instrument, the manufacturer recommends performance of a detector warm-up thermo-cycle. Therefore, a detector warm-up thermo-cycle was performed during a weekend. As a result, the detector FWHM for the 59 keV peak returned to expected parameters after the thermo-cycle as indicated in the QA control chart data plotted in Appendix Q.

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 General Environmental Laboratories, Inc. (GEL). A total of 23 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 5.5 (U-235) and Figure 5.6 (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 Q.

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 5.7 (for U-235) and Figure 5.8 (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 (UCL) of 95% (UCL_{95}) was specified.

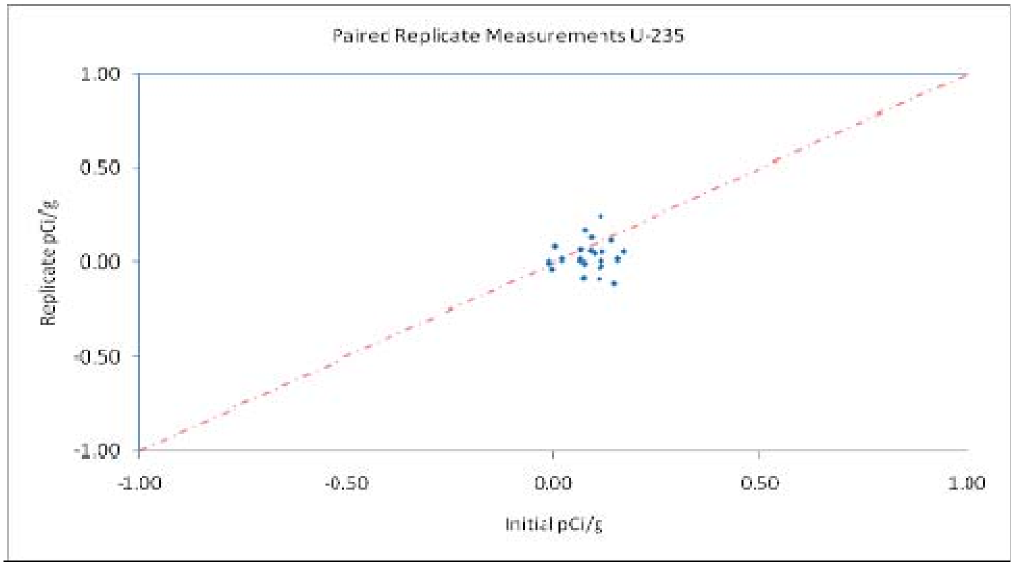

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Figure 5.5: U-235 Comparison Between Replicate Measurements

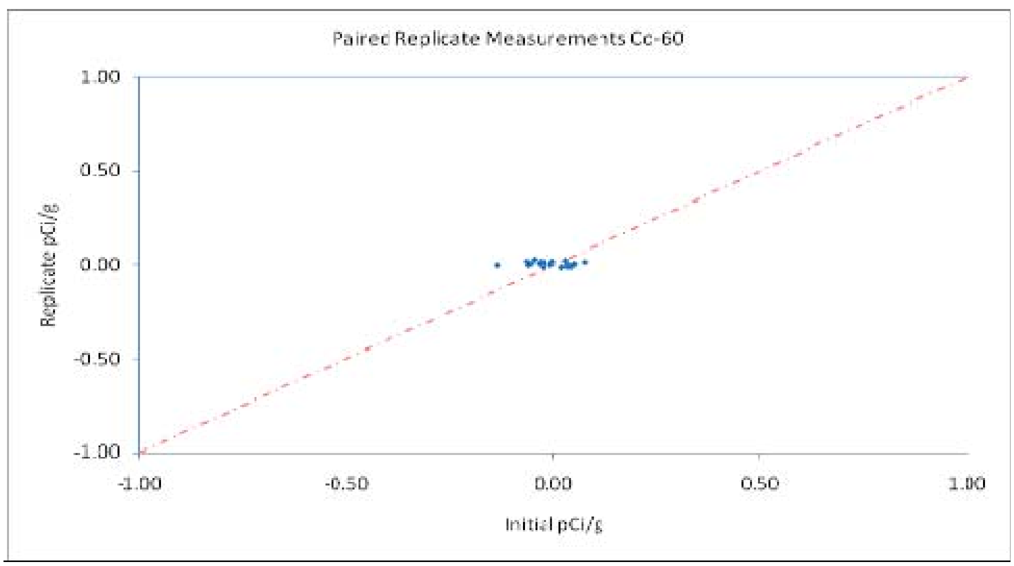

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Figure 5.6: Co-60 Comparison Between Replicate Measurements

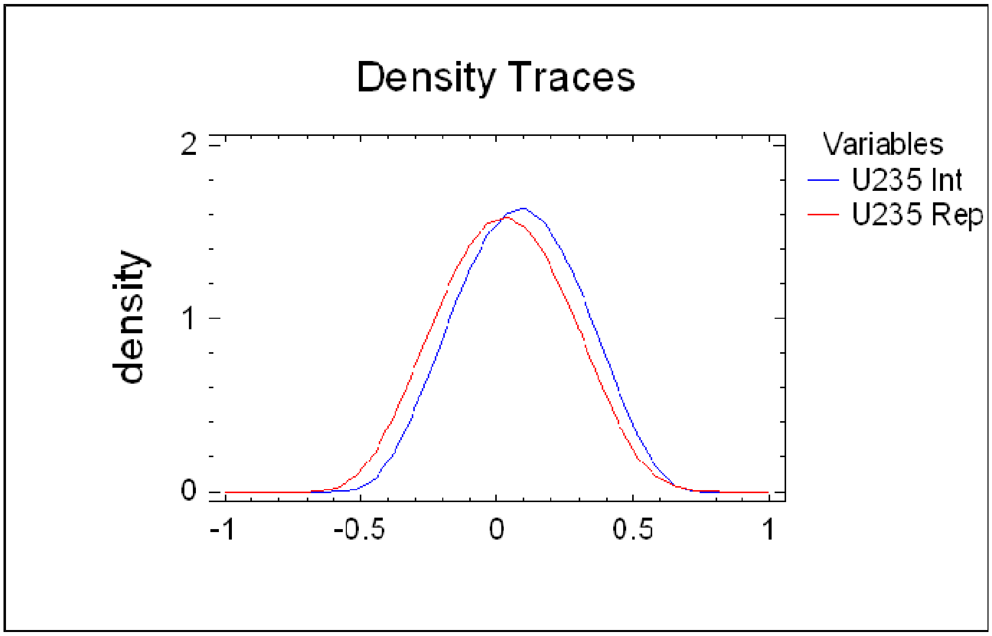

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Figure 5.7: U-235 Two-Sample Comparison of Density for Replicate Measurements

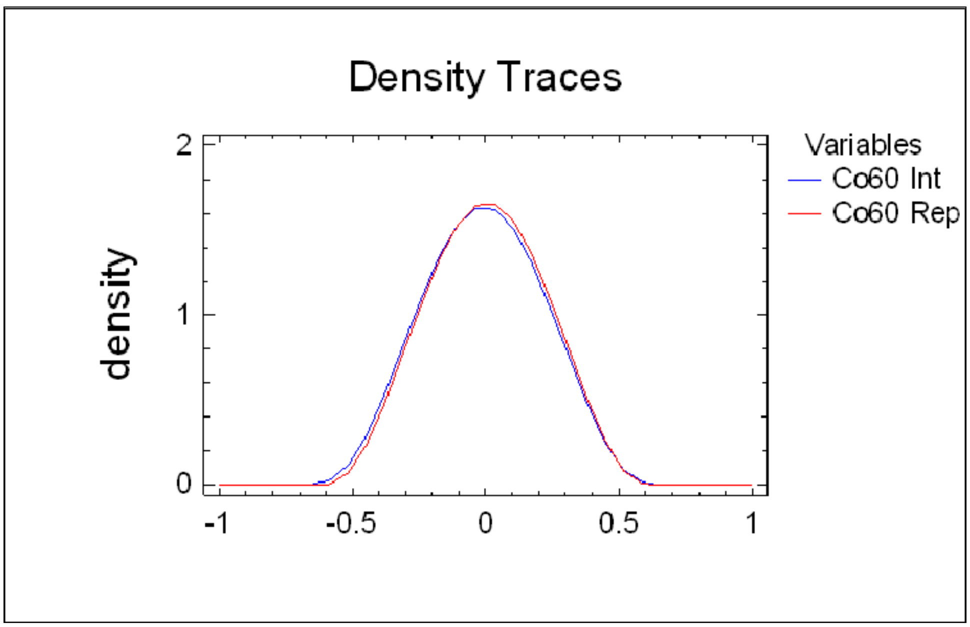

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Figure 5.8: Co-60 Two-Sample Comparison of Density for Replicate Measurements

5.1.5.1 Laboratory QC Comparisons

An internal QC method used to assess the accuracy and precision with laboratory measurements of volumetric soil media is to perform duplicate sample and laboratory replicate measurement comparisons analyzed with the Site's gamma spectroscopy system, using the NORM activity levels which are present in every soil sample. It is apparent that the above uranium and byproduct duplicate sample and laboratory replicate measurement results are very low compared to their corresponding $DCGL_{W}$s, at or below the detection capability of the instrument for many results. MARSSIM states that "Determining precision by replicating measurements with results at or near the detection limit of the measurement system is not recommended because the measurement uncertainty is usually greater than the desired level of precision."

Since several NORM nuclides are routinely identified during analysis of the FSS volumetric soil samples, a good test of accuracy and precision for a particular analytical program is to compare the detected radionuclide results for the samples homogenized and split from a single sample location, laboratory recounts of the same sample, and third party analysis of split samples. This comparison provides a more realistic view of the detection capability of the analytical method. Since there is much less uncertainty with a detected result that may be more than several times its detection threshold than a result near or less than its detection level, it is reasonable and appropriate to evaluate the accuracy and precision data quality indicators using quantifiable radionuclide concentrations.

To determine accuracy, the following protocol was used:

1. The resolution for each identified radionuclide was determined by dividing its reported activity by its corresponding 1 sigma (σ) uncertainty.
2. Determine the agreement ratio of each nuclide by dividing the reported value by the known value (or the reported value of the comparison result).
3. Compare the calculated agreement ratio value to the agreement ratio ranges listed in the following table.

RESOLUTION	AGREEMENT RATIO
<4	No Comparison
4 - 7	0.4 - 2.0
8 - 15	0.6 - 1.66
16 - 50	0.75-1.33
51- 200	0.80 - 1.25
> 200	0.85 - 1.18

To determine precision, the following protocol was used:

1. Determine the mean measured activity for each nuclide.
2. Determine the individual measurement percent deviation from the mean.
3. The results are in agreement if the percent deviation is within ±15% of the mean value.

When Co-60 and/or U-235 radionuclides were identified at levels of several times detection thresholds, they were automatically evaluated in the comparison.

The FSS QC comparison results for the split samples and the laboratory recounts are provided in each survey unit Appendix (A through O).

5.1.5.2 Laboratory Control Standard

As a separate internal QC benchmark, for every analysis batch of FSS unit samples, a laboratory control standard (LCS) count of the pint soil efficiency calibration source (No. 1316-45) was performed. Since the pint soil source has a similar matrix density to the FSS soil samples evaluated, and has the two analytes considered in this report to be credible source terms, U-235 and Co-60. In Figures 5-9 and 5-10 below, the reported interference corrected activities of the LCS (which were analyzed using the same protocol as the FSS soil samples) were ratioed to the source certificate activity and the results plotted on a control chart. Even though a daily QC check was performed using a source that trends peak resolution, peak activity, and peak centroid, to show instrument operability according to established laboratory analytical protocol, the QC check does not trend matrix efficiency for uranium and byproduct radionuclides. The LCS count is an internal QC method that adds additional confidence that there is no significant bias in the FSS soil sample results. Even though there were only 15 sample batches, a total of 16 LCS counts were performed because an additional count was performed as required by the 41-01 addendum package for the East Main Street sanitary waste line extension. Also, the last two LCS U-235 measurements are slightly at or outside of the lower 95% standard ratio control band. Since no anomalies were identified during review of the Daily QC control charts for the measurement periods, it is possible that the lower ratios could be explained by small random source positioning errors on the detector, since the corresponding Co-60 standard ratios during the period were 1.02 and 1.01 respectively. The higher energy Co-60 gamma lines are not as sensitive to source positioning differences as the lower energy U-235 gamma lines because of matrix self attenuation (the source may not have been centered directly on the detector).

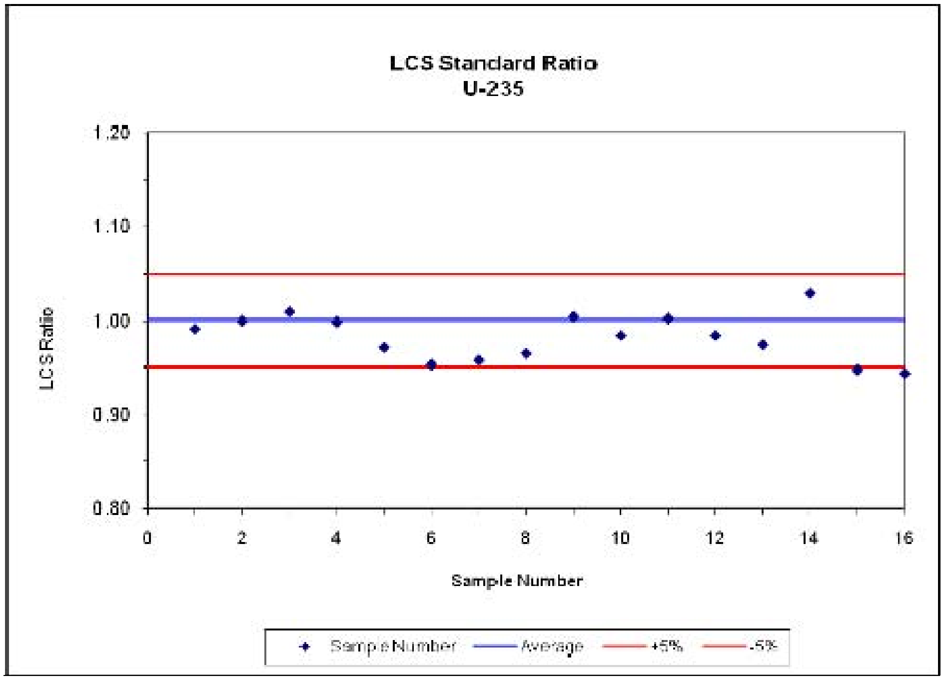

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Figure 5.9: LCS Standard Ratio U-235

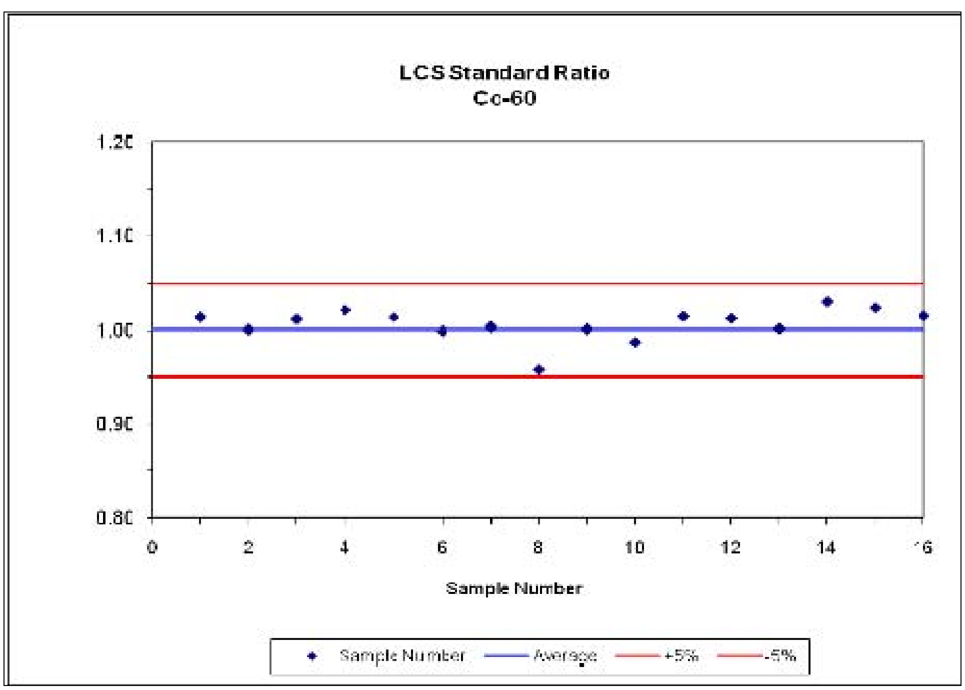

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Figure 5.10: LCS Standard Ratio Co-60

5.2 MEASUREMENT UNCERTAINTY AND DATA QUALITY INDICATORS

Measurement uncertainty in the techniques prescribed for the FSS 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 FSS is regarded as high quality data and acceptable for its intended use.

Table 5.3: 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 23, 14, or 11 volumetric soil samples from each of the Survey Units was planned, classified according to area contamination potential. 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 341 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

Table 5.3: Target Data Quality Indicators and Evaluation Results

DQI	Quality Objective	Significance	Action/Remark	Finding
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 1, 2 and 3 Survey Units were identified using the computer software program *Visual Sample Plan*. The survey was designed to produce a random sample allocation distribution within each of the Class 3 survey units and a random start for a triangular grid for Class 1 and Class 2 survey units. The sample locations selected meet the intent of the survey design and are considered representative of conditions of the Site soils. There are no analytical or measurement effects (e.g., holding times or compositing effects) affecting representativeness.	DQI accepted

Table 5.3: Target Data Quality Indicators and Evaluation Results

DQI	Quality Objective	Significance	Action/Remark	Finding
Precision	Measurement variability, due to techniques and/or technology, may increase uncertainty.	Field sampling and instrument operation were governed by procedures. Duplicate volumetric samples, laboratory replicate counts, laboratory control standard counts, 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 (23) was obtained. Duplicate volumetric sample analysis showed adequate precision even at the low activities encountered (many were below the detection limit for the method). Laboratory replicate analysis also showed adequate reproducibility. 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 AMEC 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

Table 5.3: Target Data Quality Indicators and Evaluation Results

DQI	Quality Objective	Significance	Action/Remark	Finding
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

Prepared/Date: GSM 07/28/11
Checked/Date: HTD 07/28/11

6.0 SUMMARY AND CONCLUSIONS

On the basis of the analyses presented in this report, FSS 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 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 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 residual radioactivity in the survey units 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 (survey units identified in this report) should be released from further radiological controls.