FINAL STATUS SURVEY REPORT

SUBMITTAL NUMBER 2
BUILDING 3 HIGH BAY

CE WINDSOR SITE
WINDSOR, CONNECTICUT

US NRC LICENSE NUMBER 06-00217-06
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ABB	ABB Inc.
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
cm^2	centimeter(s) squared
Co-60	cobalt 60
cpm	counts per minute
CT	Connecticut
DCGL	derived concentration guideline level
$DCGL_{emc}$	derived concentration guideline level, elevated measurement comparison
$DCGL_w$	derived concentration guideline level, survey unit average (median) concentration
DP	Decommissioning Plan
dpm	disintegrations per minute
DQA	Data Quality Assessment
DQI	Data Quality Indicator
EMC	elevated measurement comparison
EPA	Environmental Protection Agency
FSS	Final Status Survey (radiological)
FSSP	Final Status Survey Plan
HSA	Historical Site Assessment
LBGR	lower bound of the gray region
m^2	meter(s) squared
MACTEC	MACTEC, Inc.
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
MDC_{scan}	minimum detectable concentration for scan surveys
N	sample size
NIST	National Institute of Standards and Technology
NORM	naturally occurring radioactive material
NRC	Nuclear Regulatory Commission

LIST OF ACRONYMS AND ABBREVIATIONS

QA	quality assurance
QC	quality control
RSA	removable surface activity
Site	2000 Day Hill Rd., Windsor, Connecticut
Tc-99	Technetium 99
Th-230	Thorium 230
TSA	total surface activity
UCL_{95}	95% upper confidence level
VSP	Visual Sample Plan computer program

LIST OF REFERENCES

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PNNL, 2010; Visual Sample Plan, Version 6.1b, (developed by John Wilson and James Davidson, Jr., Pacific Northwest National Laboratory) http://dqo.pnl.gov/vsp/index.htm.

10 CFR 20, Title 10, Code of Federal Regulations, Part 20, Subpart E, "Radiological Criteria for License Termination."

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 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 second 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 Building 3 High Bay.

No remediation was required in the areas addressed in this FSS Report. Building 3 was originally designed and constructed as a nuclear fuel manufacturing (NFM) facility. When nuclear operations ceased in the early 1960s, the building was decontaminated and renovated for fossil fuel research and development. The south end of Building 3 referred to as the High Bay currently houses unique fossil fuel research facilities and requires release for unconditional use since it will remain operational at the time of license termination.

The FSS did not identify residual radioactivity in excess of the applicable radioactivity release criteria. For the portions of the Site provided in this report, three survey units were created in support of the FSS, all three survey units were 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 building surface derived concentration guideline levels (DCGLs) have been derived as part of the decommissioning process. The DCGLs established for uranium is total surface activity concentration of 20,148 dpm per 100 cm^2 and reactor byproduct is total surface activity concentration of 6,980 dpm per 100 cm^2.

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 Building 3 High Bay is 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 derived concentration guideline level (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 Building 3 High Bay 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, 2011) 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 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 second 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 Building 3 High Bay. This FSS Report 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 Decommissioning Plan (DP) (MACTEC, 2010), and site-specific building DCGLs (MACTEC, 2008). 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 (Figure1.1). 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.

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 Winsor/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.

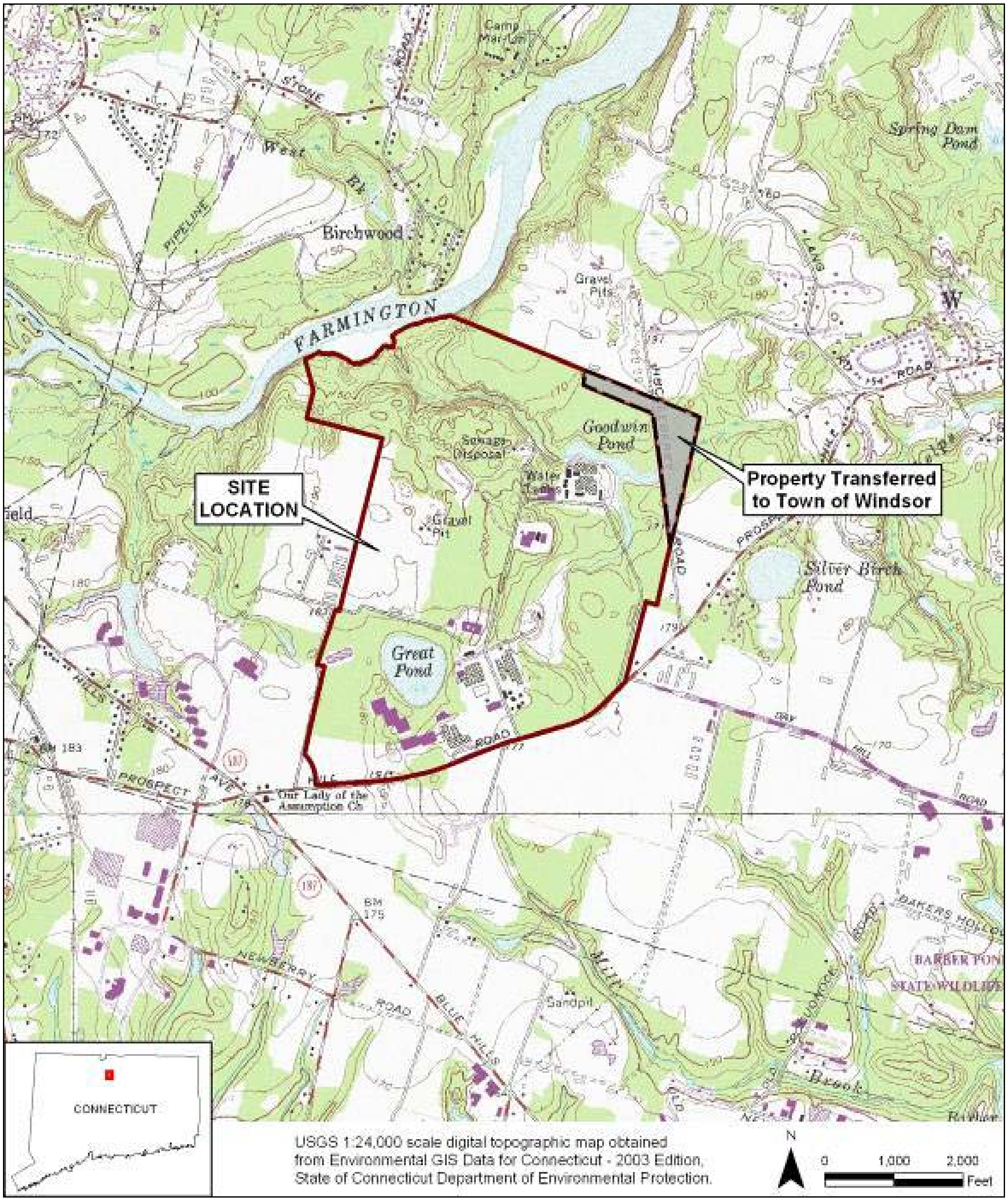

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Figure 1.1: Site Location Map

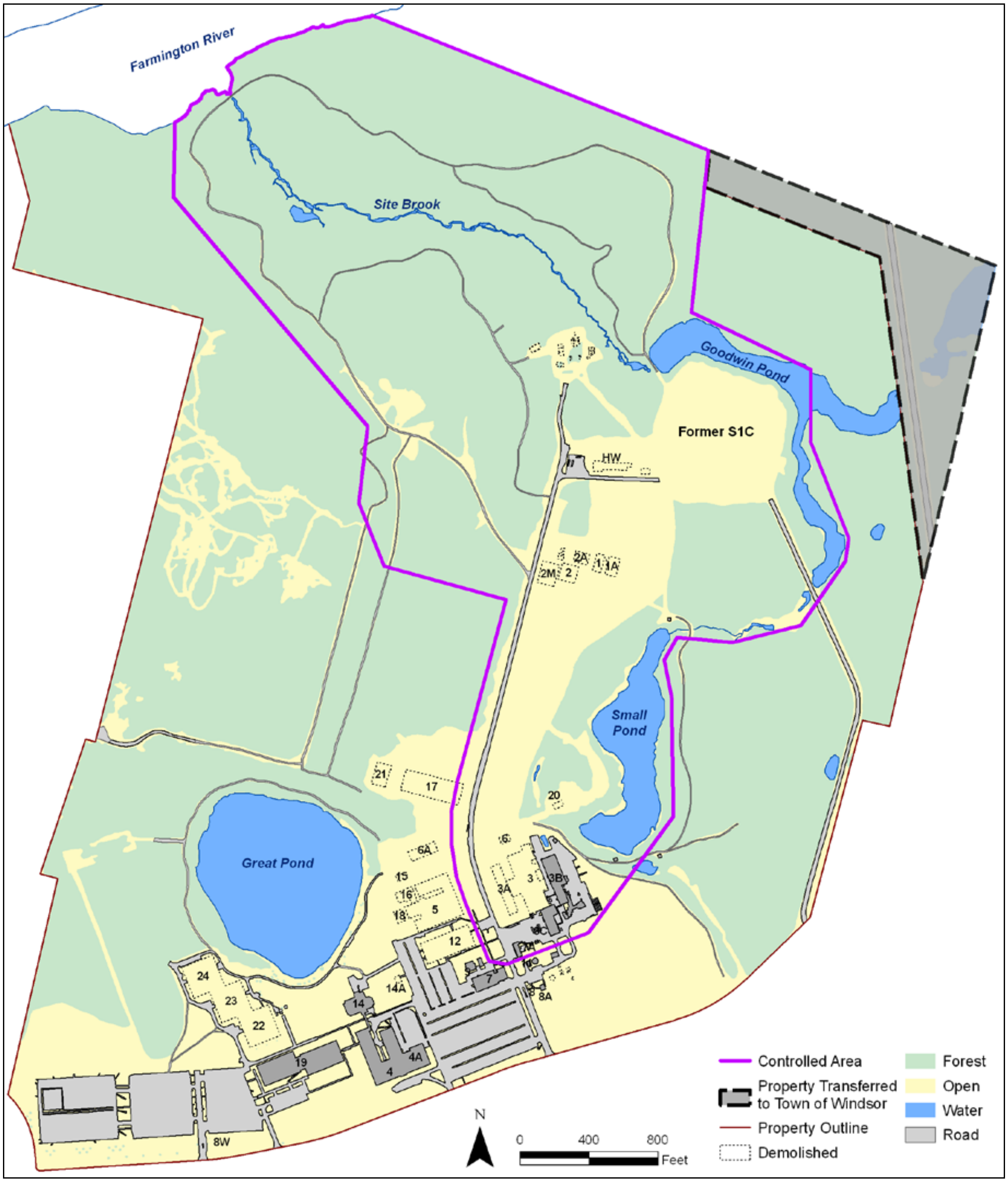

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Figure 1.2: Site Overview

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 ESE, 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 Building 3 High Bay. This area is depicted on Figure 1.3 with an overview of the building on Figure 1.4. The remaining areas within the licensed portion of the Site will be addressed in other FSS Report submittals.

1.5 RESIDUAL RADIOACTIVITY PROFILE

Based on the review of historical record, process knowledge, and the results of radiological surveys at the Site, the residual radioactivity potential for the Site has 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 cobalt 60 (Co-60). Based upon the results of soil sampling and analysis, it is evident that radionuclides associated with enriched uranium are the predominant radioisotopes found 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 other FSS Report submittals.

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 AMEC Environment & Infrastructure (formerly MACTEC) (AMEC) in support of the characterization, decontamination, and dismantling of the buildings as part of the decommissioning and license termination for the CE Windsor Site. This data is important because it was used to:

- Identify the radionuclides that are 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 appropriately detect and quantify the residual radioactivity that may be present on building surfaces; 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.

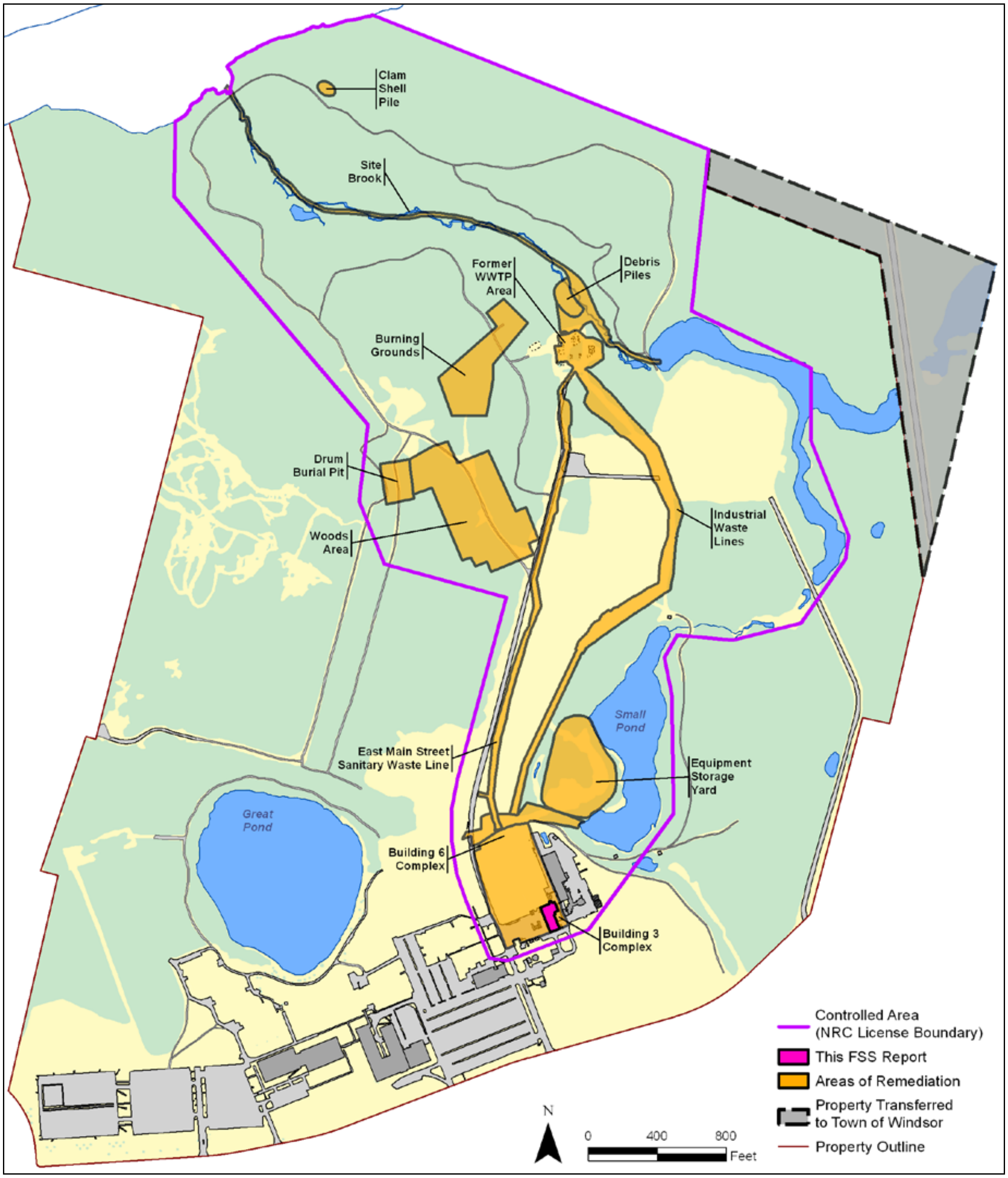

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Figure 1.3: Site Areas

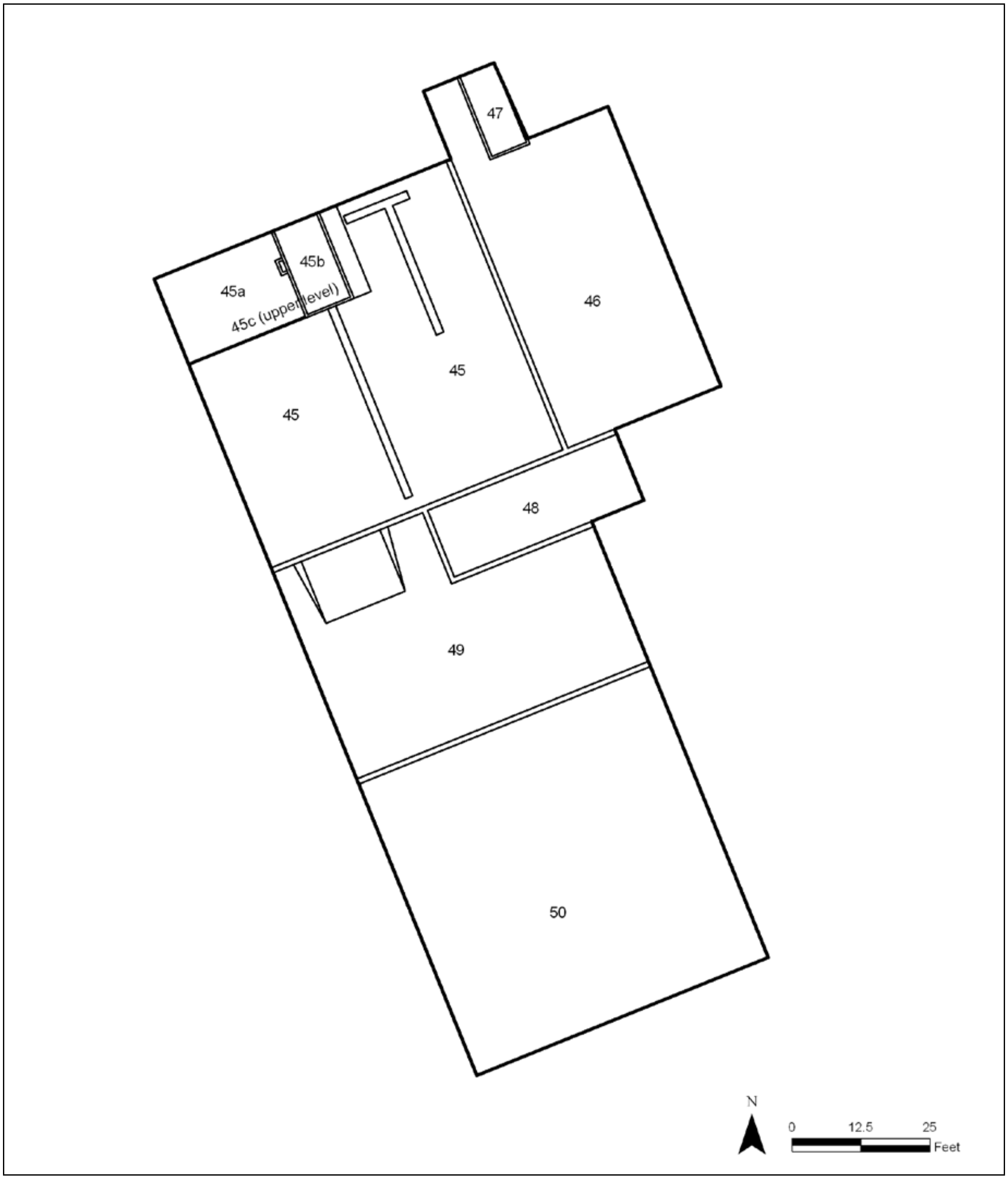

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Figure 1.4: Building 3 High Bay Overview

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, 2008). Results from dose modeling were used to select an enrichment of 90% to represent the uranium series for the Building 3 High Bay. Co-60 is used to represent the reactor byproduct series.

1.6 DECISION FRAMEWORK

The results of the FSS performed of the Building 3 High Bay demonstrate that the potential dose from any residual radioactivity is below the release criterion for each survey unit.

1.6.1 Compliance Testing

The Sign Test was used to evaluate compliance with derived concentration guideline level, survey unit average (median) concentration corresponding to the permissible limit ($DCGL_w$) for FSS. 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 surface activity measurements taken for the Building 3 High Bay.

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” surface activity concentration. This decision was made because background activity concentrations in building materials are appreciably lower than the DCGL values used during Site FSS.

The combination of total and removable radiation survey data was used to demonstrate compliance with the release criterion. In addition to single-point comparisons of the measurement against the limit, the Sign Test was conducted. The decision to release a survey unit was based upon the outcome of the comparisons made in Table 1.1.

Table 1.1: Summary of Decision Rules

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

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1.6.2 Unity Rule Testing

Given that there are two different source terms that are unrelated, 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.

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

Where:

C_U = uranium concentration
C_B = byproduct (cobalt 60) concentration
$DCGL_U$ = derived concentration guideline level for uranium
$DCGL_B$ = derived concentration guideline level for byproduct

1.6.3 Elevated Measurement Comparison Decision

Another factor in the decision rule is the EMC. Each measurement in the survey unit 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.

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

Where:

$DCGL_{EMC}$ = derived concentration guideline level for small areas of elevated activity
A_m = area factor for the area of the systematic grid (*a priori*) or actual area of elevated concentration (*a posteriori*)
$DCGL_W$ = derived concentration guideline level for average concentrations

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

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

Where:

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

If there 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 this building and accepted by the NRC as part of the DP. The approved Site-specific building surface $DCGL_w$ for uranium is 20,148 dpm/100cm2 and the $DCGL_w$ for Co-60 is 6,980 dpm/100cm2. Additional information can be found in the report *Development of Building DCGLs* (MACTEC, 2008). Calculations were performed using the Residual Radioactivity BUILD 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 $DCGL_{EMC}$ (dpm/100cm^2)	Co-60 Area Factor (A_m)	Co-60 $DCGL_{EMC}$ (dpm/100cm^2)
1	6418.9	129,320,938	41.3	288,304
2	3380.8	68,112,095	23.2	161,929
5	1496.1	30,140,943	12.0	83,937
10	826.1	16,643,042	8.0	55,723
100	107.3	2,162,655	3.3	22,905
500	23.0	463,426	2.3	15,788

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2.0 FIELD IMPLEMENTATION

This section of the report documents the FSS in the Building 3 High Bay area. 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 briefed on the FSS package requirements associated with each individual survey unit which referenced the appropriate field sampling equipment and procedures to be used. A set of simple architectural drawings of each survey unit for the building surfaces within the Building 3 High Bay was created. These drawings were then used in laying out the sampling and survey locations. Sample maps have been made as part of survey unit data in the appendices.

Three types of radiation detection instruments were selected for this survey application. The first type of instrument employed an instrument and detector connected by cable. The radiation detector was a solid state, dual-phosphor, scintillation detector designed to measure both the beta and alpha radiation emitted from a surface (direct measurement). The detector was coupled to a scaler/ratemeter to form a complete instrument/detector probe package. Direct readings were performed utilizing a Model 2224 scaler/ratemeter with a Ludlum Model 43-89 detector or a Ludlum Model 43-93 detector.

The second type of instrument used was a large area floor monitor manufactured by Ludlum as model 239-1F. The 239-1F floor monitor is comprised of a large area gas proportional detector (Model 43-37) coupled to a Model 2224 scaler/ratemeter.

In addition to these field measurements, removable radioactivity samples (smears) were made in survey units. The third instrument used was a low-background gas proportional system for counting smear samples in the on-site counting laboratory. A Canberra Series 5 XLB was utilized for counting smear samples from Building 3 High Bay.

The instruments used in the surveys were calibrated and frequently response checked and verified to be in working order and within established tolerance limits prior to use (ANSI 1997).

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 surfaces containing residual radioactivity.

The Building 3 High Bay was originally a part of Building 3 which has been decontaminated and dismantled at the time of this report. The building was used to make nuclear fuel under AEC contracts. Prior to 1961, Building 3 was used for US Navy nuclear fuel fabrication. The fuel was constructed with uranium and enriched in the isotope uranium 235 (U-235) to greater than 20 percent. After 1961, Building 3 was used for fossil fuel research and development.

The High Bay was also called the Core Assembly Building during the time in which Building 3 was used to fabricate nuclear fuel. The Core Assembly Building (High Bay) was located on the south end of the building and it was intended to be maintained radiologically clean. However, there were times when final assemblies contained residual uranium, and had to be cleaned before it was released from the building. The footprint of the High Bay was doubled in the early 1970s to support the commercial (fossil) power plant safety valve testing program. The soil under this addition was surveyed prior to construction and was clean of radiological contamination from the plant operations. Additionally, a detailed characterization survey was performed in 2008. No residual radioactivity readings above typical background levels of naturally occurring radioactive material (NORM) were detected in surveyed areas. Therefore, MARSSIM Class III survey unit classification is appropriate for all area surfaces of the facility. The facility is planned to remain in use until time of license termination.

The Building 3 High Bay was divided into three survey units. The first survey unit covered the interior surfaces (walls, ceilings, and floors) of Rooms 45, 45A, 45B, 45C, 48, and 49. This area currently contains unique fossil fuel research equipment. The second survey unit covered additional interior surfaces (walls, ceilings, and floors) of rooms 46, 47, and 50 (Building 3C) which was added in 1998 as a utility support building for the fossil fuel operations. The third survey unit covered Building 3 High Bay and Building 3C exterior surfaces (roof and the sides of the facility). A summary of the survey units for the Building 3 High Bay is presented in Table 2.1 and depicted in Figure 2.1.

Table 2.1 Summary of FSS Units

Survey Unit ID	Class	Area (m^2)	Description
CE-FSS-30-01	3	4,296	Building 3 High Bay Rooms 45, 45A, 45B, 45C, 48, 49
CE-FSS-30-02	3	2,269	Building 3 High Bay Rooms 46, 47, 50
CE-FSS-30-03	3	3,501	Building 3 High Bay Exterior Walls and Roof

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2.3 SURVEY UNIT SAMPLE SIZE DETERMINATION

The minimum sample size (N) and location of those samples for each survey unit was determined using the statistical sampling software, Visual Sample Plan (VSP) (PNNL, 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).

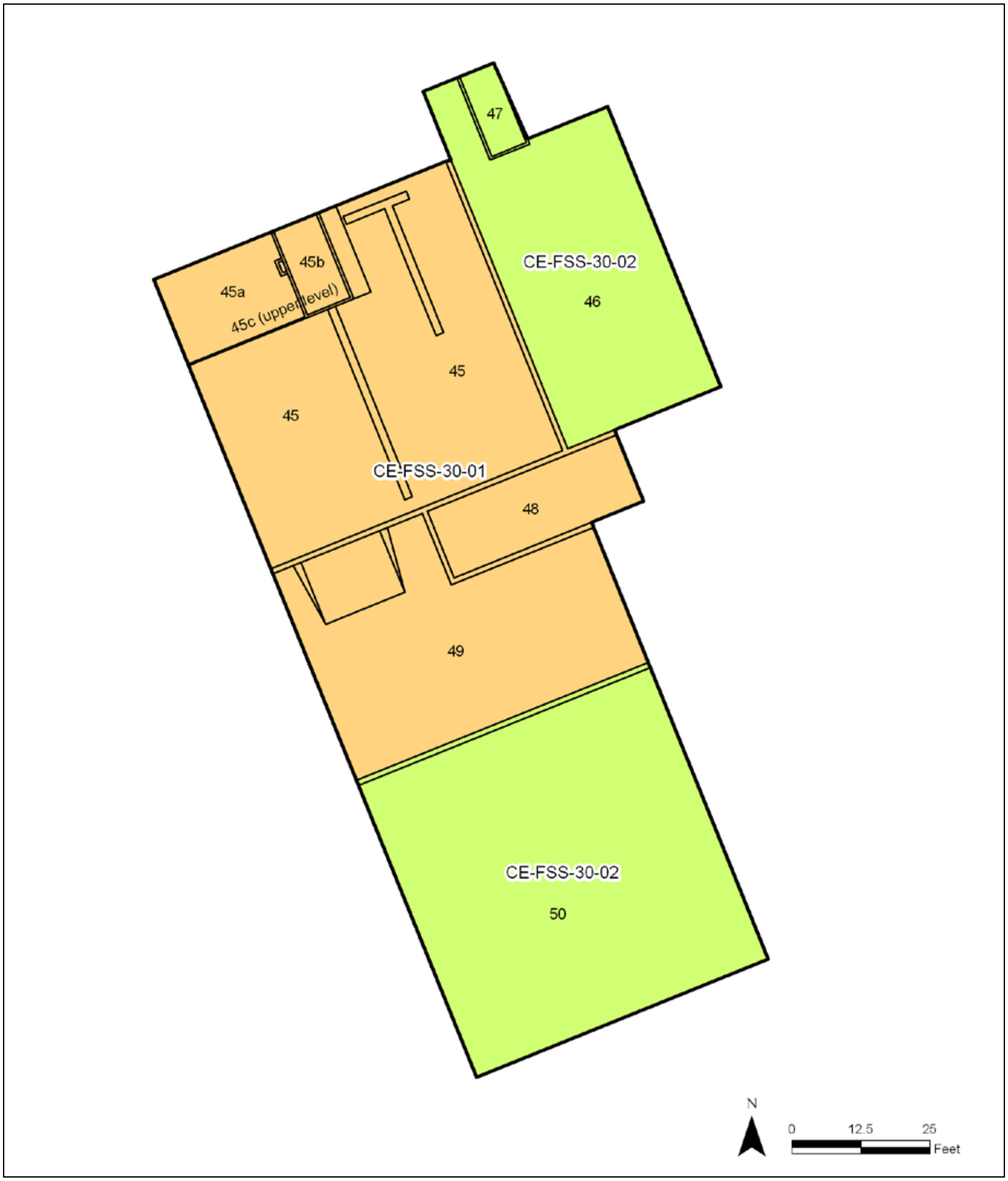

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Figure 2.1: Overview of Building 3 High Bay FSS Units

2.3.1 Class 3 Survey Unit Sample Size

For survey unit sample size calculations, the $DCGL_w$ was not used. Instead, the survey unit design was based on the more restrictive non-dose based total surface activity limits described in USNRC Regulatory Guide 1.86 (NRC, 1974) Since Class 3 survey units are not expected to have measurable residual radioactivity at only a small fraction of the DCGLs, the lower bound of the gray region (LBGR) was selected to be 60% of the DCGL. The standard deviation was also conservatively approximated high (40%) as a safety margin to reduce the chance of failing the decision criteria. The survey design parameters used to calculate the minimum required sample size for Class 3 Survey Units are shown in Table 2.2. This FSS report contains a total of three Class 3 Survey Units.

Table 2.2: Class 3 Survey Unit Sample Size

Parameter	Uranium	Co-60
α decision error	0.05	0.05
β decision error	0.05	0.05
$DCGL_w$ (dpm/100cm^2)	5000	5000
LBGR (maximum estimated mean/median) (dpm/100cm^2)	3000	3000
Standard Deviation (σ)(dpm/100cm^2)	2000	2000
Relative Shift (Δ/σ)	1.0	1.0
Sample Size (N)	24	24
Additional 20%	5	5
FSS Sample Size	29	

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The total number of samples planned and the number of samples obtained per survey unit is presented in Table 2.3. In every survey unit, the number of samples obtained met or exceeded the number of samples planned.

Table 2.3: Number of FSS Surface Measurements Obtained per Survey Unit

Survey Unit ID	Class	Number of Surface Measurements Planned	Number of Surface Measurements Obtained
CE-FSS-30-01	3	29	33
CE-FSS-30-02	3	29	32
CE-FSS-30-03	3	29	29

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2.4 SURVEY AND SAMPLE LOCATIONS

The proposed location of each measurement was laid out using a simple random sample allocation protocol. Electronic drawings of the survey units were created with the walls and ceilings “unfolded” and set flat to render a two-dimensional layout and assist the process of spatial distribution and sample location recording. The sampling design software *Visual Sample Plan, Version 6.1b* (PNNL 2010) was used to lay out the required number of measurements locations at random within the survey unit.

Drawings of each surface within the survey unit and actual sample locations, as determined in the field, are provided in Appendix A. After the measurement locations were allocated, an inspection of each survey unit was conducted to ensure that each sample location selected could be accessed and measured safely. Selected measurement locations that were inaccessible or presented safety hazards were relocated. The measurements relocated are annotated on the drawings.

Sample locations were next laid out on the building surfaces within the survey units. Each sample location was located in the field and marked on the surface with a sample location identifier unique to each particular survey unit outside the measurement location outline with an indelible marker or a sample label.

2.4.1 Building FSS Sample Locations

Surface measurements were collected for FSS evaluation for the areas included in this submittal report during 2011. Figures of measurement locations for each survey unit are provided in the survey unit data appendices (A through C). Measurement locations were placed such that a sample would be representative of the sample media. Measurement density was defined by VSP using the assumptions stated earlier in this report.

2.5 INVESTIGATION LEVELS

Investigation levels (Table 2.4) for the direct measurement results were developed in accordance with the guidance found in MARSSIM. Any surface location measurement 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 2.4: Final Status Survey Direct Measurement Investigation Levels

Survey Unit Classification	**Direct Measurement 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 scan survey were derived using the most conservative assumption basis: the least sensitive instrument of the inventory being used for the survey. Using conservative assumptions of data, gross counts per minute (cpm) values were generated at the stated $DCGL_w$ values for the scanning investigation level (Table 2.5). For the purpose of this report, all reported cpm values, unless otherwise specified, should be considered gross values uncorrected for instrument background. Only the Floor Monitor scanning investigation levels were presented because over 95% of the surfaces scanned were accessible floor surfaces.

Table 2.5: Final Status Survey Scanning Gross Investigation Levels

Survey Unit Classification	**Ludlum 239-1F Floor Monitor Scanning Alpha Investigation Level** (most conservative)	**Ludlum 239-1F Floor Monitor Scanning Beta Investigation Level** (most conservative)
Class 3	> 1,705 cpm	> 3,854 cpm

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2.6 DIRECT MEASUREMENTS AND SCAN SURVEYS

Direct measurements of the radiation emission from surfaces were made using static, 60-second counting intervals, over which the total counts were integrated. The measurements recorded were gross values. In the context of this sampling evolution, a “gross measurement” means a measurement made with a radiation detection instrument to which no background correction has been applied. Raw or gross data is important when measurements will be used to make statistical inferences, since not all data will necessarily have the same correction factors applied to properly reduce them to numbers that are meaningful in the context of the analysis. Reporting gross or raw data also permits one to analyze the functionality of the instrument with which the measurement was made, and to verify the appropriateness of the data reduction process. The data reduction process for the field measurement data collected in this surface measurement sampling program involves corrections for the efficiency of the radiation detector to the subject radiation and for the instrument response to background sources of radiation (excluding surface media contribution to background).

Biased scan surveys of building floor surfaces and high traffic walkways were performed since characterization and HSA surveys indicate that there were no areas with surface activity concentrations greater than a small fraction of the $DCGL_w$. Additionally, biased scans were performed on wall surfaces and areas inside drains, pipe trenches and penetrations.

2.6.1 Portable Instruments

The field measurement instrument used for direct static measurements of surface-deposited residual radioactivity was the Ludlum Model 2224 Scaler/Ratemeter Portable Multi-purpose Radiation Survey Instrument with a Ludlum Model 43-89 or 43-93 dual-phosphor scintillation detector probe. The direct measurement data was collected in accordance with the Final Status Survey Plan (AMEC, 2011). Scan surveys were performed with the Ludlum Model 2224 with 43-89 or 43-93 probe, and Ludlum 439-1F floor monitor.

2.6.2 Portable Instrument Calibration

MARSSIM guidance was considered in establishing efficiency factors (calibration constants) used to reduce the instrument count rate data to units comparable to those used in the surface standards along with guidance from other sources including NUREG-1507 (NRC,1997), NCRP Report 112 (NCRP, 1991), and ANSI N323A (ANSI, 1997).

As defined in MARSSIM and NUREG-1507, instrument efficiency is that derived by measuring the surface emission rate of a clean, calibrated and certified National Institute of Standards and Technology (NIST) traceable, reference source. The observed emission rate (counts per unit time) is compared to the certified emission rate (betas or alphas per unit time) to arrive at the instrument efficiency. The source efficiency relates the amount of activity truly present on the surface being measured to the observable particle emission rate. As such, the source efficiency captures the effects of backscatter, and self absorption inherent in the surface being measured.

In addressing the issues associated with the derivation of the appropriate total efficiency for the measurement, MARSSIM states (page 6-24) that the use of a total efficiency derived from measurements made on certified 4π activity traceable sources "…is not a problem, provided that the calibration source exhibits characteristics similar to the surface contamination (i.e., radiation energy, backscatter effects, source geometry, self-absorption)."

For the Building 3 High Bay, each of these four parameters was addressed as follows:

Radiation Energy. Radiation energy was addressed by selecting a calibration source that was manufactured using a pure beta emitter with an energy that approximates the beta energy of Co-60, the beta emitter of predominant concern in byproduct materials. For the Building 3 High Bay, a NIST traceable, technetium 99 (Tc-99) source was selected and used to establish the total efficiency. Tc-99 emits betas with a maximum energy (0.294 MeV) slightly lower than, but comparable to, those emitted by Co-60 (0.318 MeV), introducing a small but conservative bias.

Source Geometry. The source geometry of interest is the infinite plane source geometry. This is true for two reasons: (1) the average surface activity guideline values (DCGLs of 20,147 dpm/100 cm^2 Alpha and 6,980 dpm/100 cm^2 Beta) are established over the entire survey unit, and (2) a lack of perfect uniformity in response to radiation over the surface of the active face of the detector leading to what is generally termed "edge effects." Thus a detector is typically slightly less efficient when measuring distributed activity than it is when measuring a discrete point source of radioactivity on a surface. A wide area calibration source, with an active area measuring 10 cm by 15 cm was used to establish a calibration geometry that is essentially infinite with respect to the detector. The large area geometry of the source also accounts for any variability in efficiency across the face of the detector especially near the edges, introducing another small, but conservative bias. A second aspect of source geometry is that associated with the source to detector distance. Variability in the detector distance in the field introduces an uncertainty in the measurement made. To control this source of uncertainty, the detectors themselves were fitted with feet that are approximately 3/16" thick (1/8" to 1/4" thick depending on the detector housing). The source to detector geometry is thus controlled through the calibration and field measurement process.

Backscatter and Self-Absorption. Backscatter and self-absorption are more difficult to control in the measurement process since they are impacted by field variables, beyond the control of the surveyor. The desire is to represent the total surface activity on a given surface as accurately as possible. To do this, consideration for the surface(s) to be measured and their effect on backscatter and self-absorption must be taken into account. For the building surveys at the Site, this was recognized very early in the process and controlled. It was known that concrete surfaces would likely pose the most challenging of surfaces that would require measurements associated with release decisions. It was also recognized that the errors that can arise because of the characteristics of surface (often non-conservative errors) are due to distinct and measurable differences between the backscatter and self absorption characteristics of the calibration source as compared to the surface of interest. As suggested in NRC, 1997 and NCRP, 1991, AMEC chose to specify efficiency calibration sources specifically designed to closely approximate the backscatter and self-absorption characteristics of concrete.

By adopting this efficiency calibration source, it can be said with reasonable confidence that the total efficiency (as measured by exposing the detector to the check source and comparing its response to the stated total 4π activity) is appropriate for making measurements on concrete surfaces, having taken into account both source efficiency and instrument efficiency.

The alpha channel of the probe was calibrated to a thorium 230 (Th-230) National Institute of Standards and Technology (NIST) traceable calibration source. The beta channel of the probe was calibrated to a Tc-99 National Institute of Standards and Technology (NIST) traceable calibration source. The calibration certificates for the sources and the calibration data sheets for the instruments are provided in Appendix D. The probe face was fitted with small feet placed around the inactive perimeter so that consistent measurement geometry was maintained during calibration, response checks, and field measurements.

Background and response checks were normally performed at least twice a day when in use, as a preoperational check, and post use to ensure the instrument was operating properly during the survey period. Background results are discussed in this Section and response check results are discussed in Section 5. The calibration certificates for these sources and the calibration data sheets for the instruments are provided in Appendix D.

2.6.3 Measurement Detection Limitations

In order to calculate the statistically significant surface radioactivity, which could be distinguished from background (*a posteriori* minimum detectable concentration [MDC]), it was necessary to convert the background measurement units from dpm/100 cm^2 to units of cpm. In this case, the more conservative metric, the geometric mean, was chosen to calculate the detection sensitivity achieved to prevent overstating the actual sensitivity achieved. The converted mean background count rates for the sampling period (cpm) along with other actual field measurement parameters are presented in Table 2.6. Using the actual instrument field measurement parameters, a calculation of the actual field measurement MDC achieved can be determined by solving Equation 2–1.

$$MDC = \frac{3 + 4.65\sqrt{C_b}}{T_s \times \frac{A_p}{100cm^2} \times \in_T} \qquad \textit{(Equation 2–1)}$$

Where: MDC = the minimum surface radioactivity concentration above background radioactivity (in dpm/100 cm^2) that can be detected with 95% confidence.

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

T_s = sample count time (in minutes).

A_P = probe size (in cm^2).

$\in_T$ = counting system efficiency in count/disintegration.

Table 2.6 Static Surface Contamination Measurement MDC Parameters

Parameter		Ludlum 2224 with Ludlum 43-89 or 43-93 (Alpha)					
		AB-2 α		AB-7 α		AB-8 α	
		Alpha	**Beta**	**Alpha**	**Beta**	**Alpha**	**Beta**
C_b	Background Counts	1	149	2	138	1	114
T_s	Sample count time (minutes)	1	1	1	1	1	1
A_P	Probe Size	125	125	100	100	100	100
$\in_T$	Instrument system efficiency in counts per disintegration	0.0807	0.1090	0.1040	0.1230	0.0794	0.0749
MDC	dpm/100 cm^2	60	438	81	468	77	700

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Therefore, the "gross" field instrument reading that can be distinguished as different from background (the maximum adjusted gross MDC from the three instruments used), is 2,053 dpm/100 cm^2 (700 + 1,353). Having identified the *a posteriori* MDC for the field sampling measurements and the adjusted gross MDC, a simple sort of the gross field measurement data points can be performed to identify those measurements from a survey unit that are greater than 2,053 dpm/100 cm^2. Those locations with gross residual surface radioactivity greater than the adjusted gross MDC are credited as having statistically distinguishable amounts of added radioactivity, while those less than the adjusted gross MDC are statistically indistinguishable from background values.

It is further important to note that the net MDC distinguishable above background (700 dpm/100 cm^2) is lower than the most limiting DCGL (6,980 dpm/100 cm^2).

2.6.3.1 MDC Scan Calculations

For any of the instrument systems, the detection sensitivity is affected not only by the factors influencing static measurements (as described above) 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 following formulation (NRC, 2000) is used to calculate the minimum detectable concentration, in dpm/100 cm^2, for each of the two scanning instrument systems used:

$$s_i = d' * \sqrt{b_i} \qquad \textit{(Equation 2–2)}$$

Where: s_i = the minimum detectable number of net source counts in the counting interval, i (probe residence time over a given source area).

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.

$$MDCR = s_i * (60 / i) \qquad \textit{(Equation 2–3)}$$

Where: $MDCR$ = the minimum detectable count rate (above background) in cpm.

s_i = the minimum detectable number of net source counts in the counting interval, i (probe residence time over a given source area).

i = the length of the counting interval in seconds.

$$MDC_{SCAN} = \frac{MDCR}{\sqrt{\mathrm{p}} \times \frac{\mathrm{A_P}}{100\,\mathrm{cm}^2} \times \varepsilon_{\mathrm{T}}} \qquad \textit{(Equation 2–4)}$$

Where: MDC_{scan} = the minimum surface radioactivity concentration above background radioactivity (in dpm/100 cm^2) that can be reliably detected.

P = Surveyor efficiency.

A_P = Probe size (in cm^2).

ε_T = Counting system efficiency in counts/disintegration.

$$MDC_{SCAN} = \frac{MDCR}{\sqrt{\mathrm{p}} \times \frac{\mathrm{A_P}}{100\,\mathrm{cm}^2} \times \varepsilon_{\mathrm{T}}} \qquad \textit{(Equation 2–5)}$$

Where: MDC_{scan} = the minimum surface radioactivity concentration above background radioactivity (in dpm/100 cm^2) that can be reliably detected.

P = Surveyor efficiency.

A_P = Probe size (in cm^2).

ε_T = Counting system efficiency in counts/disintegration.

Some of these parameters were derived from guidance in MARSSIM. The index of sensitivity (*d'*) was selected to allow for a 95% probability of accepting true positive responses and a 60% probability of returning false positive results. Surveyor efficiency (*P*) was determined to be 0.5 for portable (hand-held) instruments and 0.75 for floor monitors since efficiency should improve with the use of mechanized survey equipment. The surveyor efficiency accounts for the uncertainty of the operator performing the non-static scanning procedure and the judgment involved in determining the presence of elevated counts during scanning.

Table 2.7 presents the observed site conditions and instrument specific parameters affecting the minimum detectable scanning concentration (MDC_{scan}) for the scanning surveys performed. Using these values, an *a posteriori* assessment of the building or material surface MDC_{scan} can be determined. Because scan speeds cannot be strictly regulated with the instruments used, and because there is no convenient method of actually measuring the scan speed, the *a posteriori* MDC_{scan} was evaluated for a range of scan speeds. The scan speeds considered include a wide range substantially broader than the nominally achieved scan speed of 2 inches per second. Figure 2.2, Figure 2.3, Figure 2.4, Figure 2.5, Figure 2.6 and Figure 2.7 present the MDC_{scan} sensitivities (for scan speeds between 0.1 and 10 inches per second) for the Ludlum 2224 with 43-89 probe (αβ-2), the Ludlum 2224 with 43-93 probe (αβ-8) and the Ludlum 239-1F floor monitor, respectively.

The *a posteriori* MDC_{scan} for alpha and beta radioactivity achieved under the conditions described and observed during surveys in the Building 3 High Bay (assuming nominal scan speeds of 2 inches per second) are listed at the bottom of Table 2.7. These *a posteriori* MDC_{SCANS} (even over the wide range of scan speeds considered) are well below the average surface residual radioactivity DCGL benchmark concentrations of 20,147 dpm/100 cm^2 alpha or 6,980 dpm/100cm^2 beta providing a solid basis for confidence that the scanning surveys employed are capable of detecting localized concentrations of significance.

Table 2.7 Surface Scanning Measurement MDC_{scan} Parameters and Values

Parameter		αβ-2 Ludlum Model 2224 w/43-89 AB		αβ-8 Ludlum Model 2224 w/43-93 AB		FM-1 Ludlum Model 2224 w/43-37 AB	
		Alpha	**Beta**	**Alpha**	**Beta**	**Alpha**	**Beta**
C_b	Background Count Rate (cpm)	1	117	1	110	3	614
i	The residence time of the detector probe over a given surface area (the counting interval) in seconds.	2.0		1.8		3.2	
d'	Index of sensitivity	1.38		1.38		1.38	
p	Surveyor efficiency	0.5		0.5		0.75	
A_P	Probe size (cm^2)	125		100		584	
ε_T	Instrument system efficiency in counts/disintegration	0.0807	0.109	0.0794	0.0749	0.0871	0.1803
MDCR	Minimum count rate above background	8	96	8	85	10	149
MDC_{scan}	dpm/100 cm^2	106	996	144	1600	24	164

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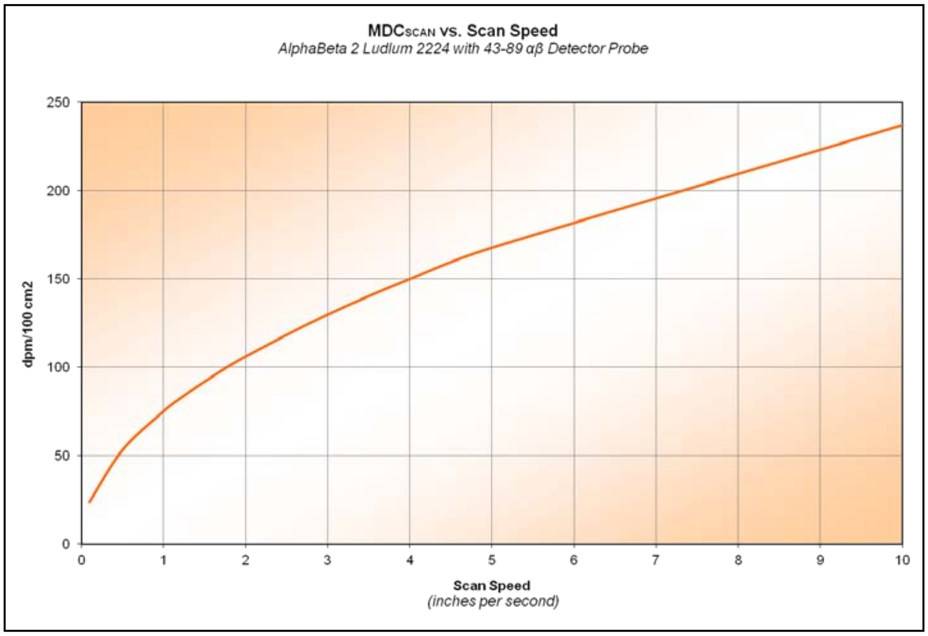

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Figure 2.2 Scan MDC vs. Scan Speed - Alpha for the αβ-2 Detector.

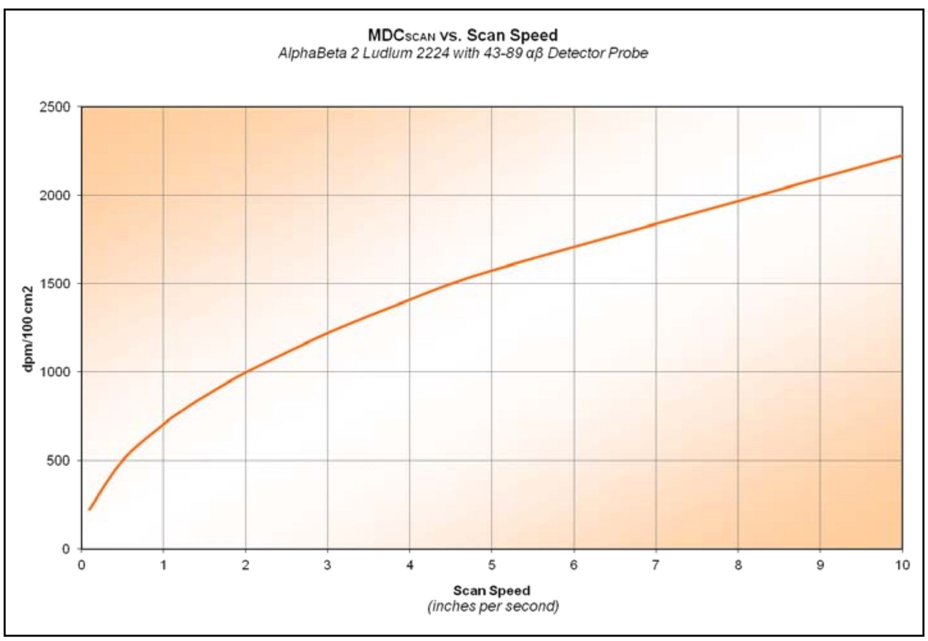

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Figure 2.3 Scan MDC vs. Scan Speed - Beta for the αβ-2 Detector

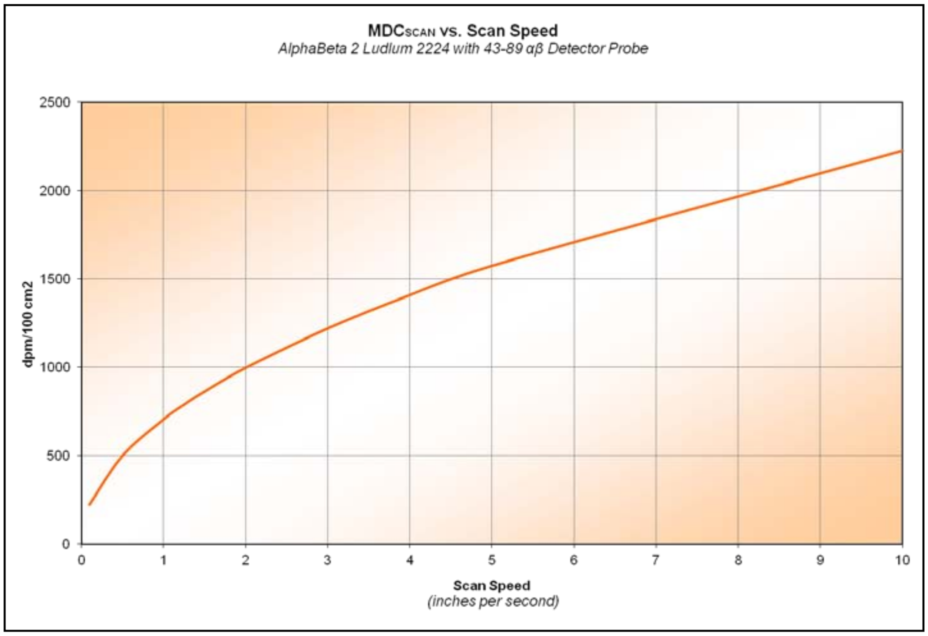

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Figure 2.4 Scan MDC vs. Scan Speed - Alpha for the αβ-8 Detector.

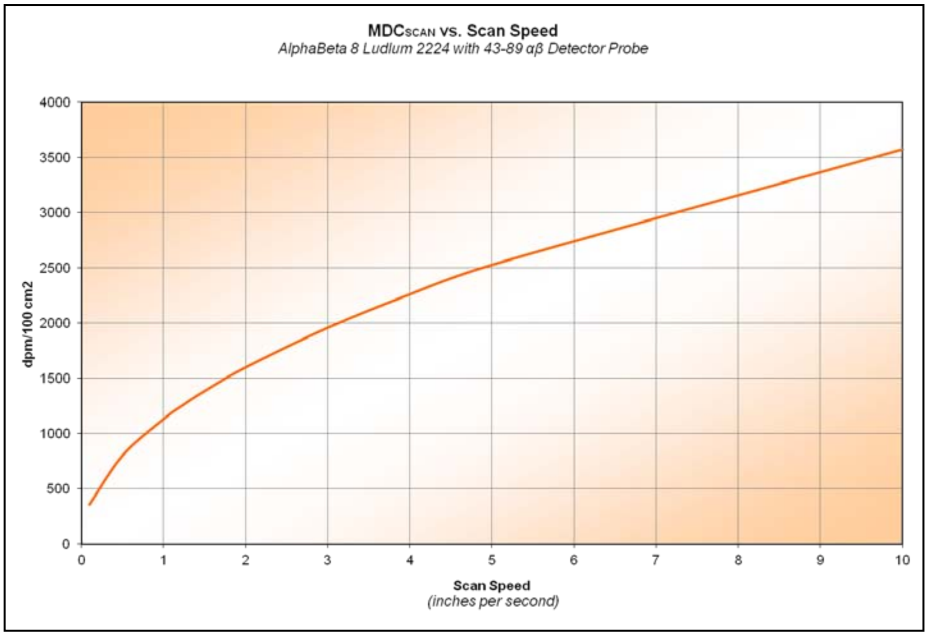

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Figure 2.5 Scan MDC vs. Scan Speed - Beta for the αβ-8 Detector.

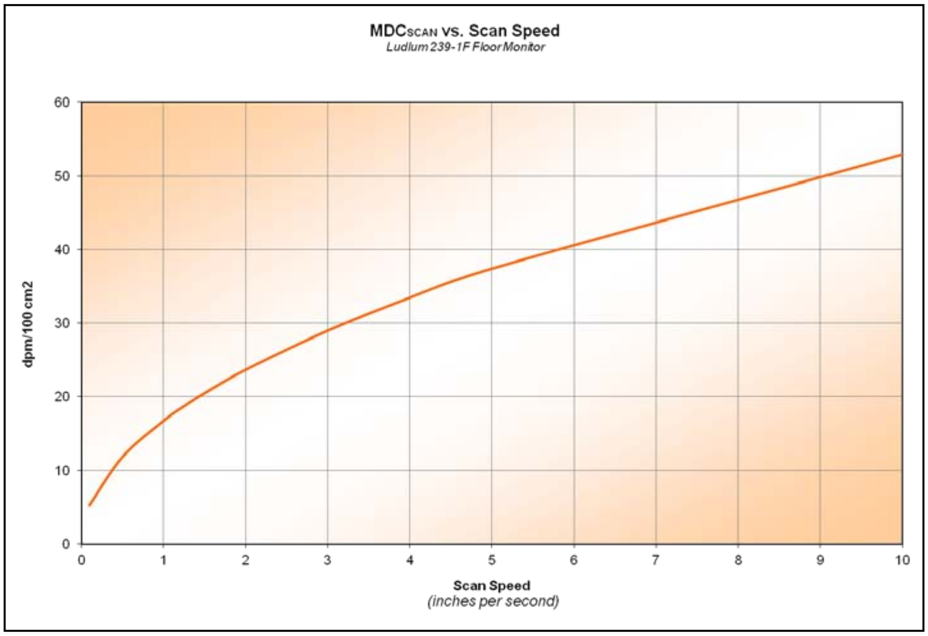

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Figure 2.6 Scan MDC vs. Scan Speed - Alpha for the Floor Monitor.

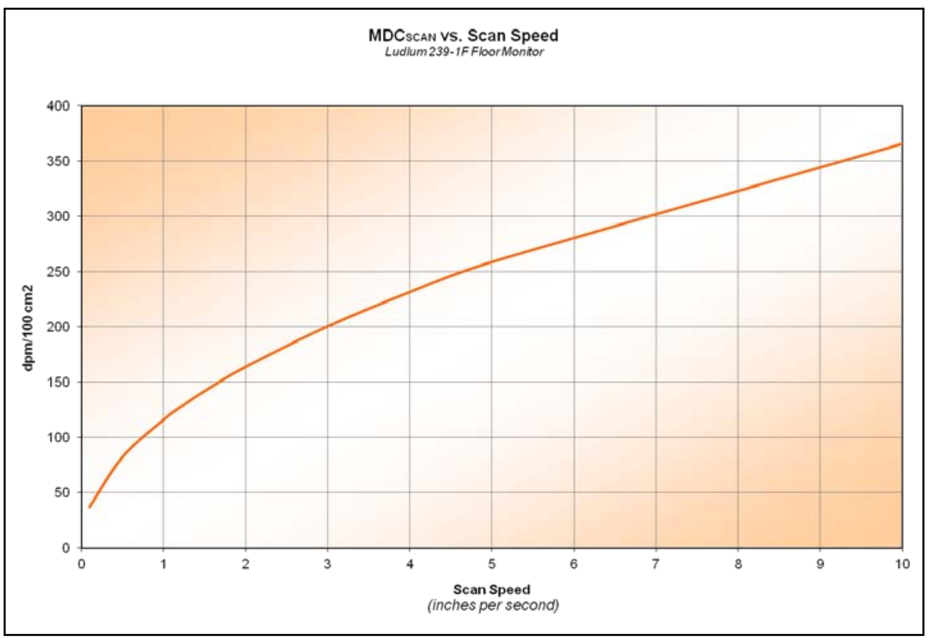

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Figure 2.7 Scan MDC vs. Scan Speed - Beta for the Floor Monitor.

2.6.4 Portable Instrument Background Measurements

Background measurements were made to assess the instrument background and not to assess the background concentration of a contaminant of concern in some media. Instrument Background is defined as: “the response of the radiation-detecting instrument to sources of radiation in the environment such as cosmic radiation and to electronic noise in the instrumentation that may produce a measurable signal not due to radiation. It does not include the instrument’s response to concentrations of radioactive materials that might be present in the media being measured but which are considered to be part of the background environment.” Thus the instrument background as collected is not a measure of the inherent background in some media (such as naturally occurring uranium activity in concrete) for which a media specific reference area background would be required to permit compensation. Media specific backgrounds with their associated reference area measurements have not been used in the assessment of surface activity in the Building 3 High Bay survey units.

Instrument background measurements were made frequently throughout the sampling/survey period (not just once per day) and in the immediate vicinity of the survey area. In this way, the instruments response to variability in cosmic and terrestrial sources of radiation (sources unassociated with the beta radioactivity on the surfaces of a survey unit being measured) was captured. All field measurements were recorded as raw uncorrected or “gross” measurements.

The local area instrument background measurements were made by placing the detector a 180 degree orientation (or using a suitable shield or detector cover for the floor monitor) from a background location surface designated at a designated suitable area selected by either the Radiological Controls Supervisor or the FSS Lead Technician which were repeated during all surveys associated with the B3 High Bay FSS. The local area background measurements were performed three times daily, coinciding with the pre-operational, mid-day, and post use performance checks. Background measurements were made using the timed static count of 60 seconds. Measurements were recorded manually.

The assessment of an instrument’s response to background radiation is important from two perspectives. First, it permits the assessment of the minimum sensitivity (detection limit) for the instrument and measurement process in the presence of background radiation. The *a posteriori* MDC is calculated from this actual background data. Second, by assessing the instrument’s response to background radiation in terms of the units that field data will be collected, a correction can be applied to the field measurement data to permit determination of radioactivity present in excess of background. Because the naturally occurring concentrations of background radioactivity in building materials used in the construction of the buildings were expected to be below and well within the DCGL benchmarks for residual radioactivity on building surfaces, it was decided to conservatively assign all building material background radioactivity as part of the residual activity attributable to licensed activities for comparison against the DCGL. As a result, no attempt was made to measure the concentrations of naturally occurring radioactivity measurable on surfaces in an unaffected area or “reference survey unit”.

Still, there was the need to measure and account for the instrument’s response to other ubiquitous sources of background radiation (e.g., cosmic radiation) that could otherwise not be distinguished from the contaminant of concern. To correct the data for instrument sensitivity to background radiation, excluding that present in the substrate of the surfaces being measured, instrument background measurements were made periodically over the sampling period. In all, 32 measurements of the alpha and beta background radiation levels were recorded during the sampling period. Additionally, since the alpha instrument backgrounds results were typically one or zero cpm, statistical analysis was not presented on the alpha instrument background data.

The variance in the recorded background data was small and within the range expected for beta-gamma background radiation (see Appendix D).

Time series plots of the background data sets, segregated according to the specific instrument/detector probe with which the measurement was made, illustrate the lack of trend in the data over time and the overall stability of the instrument background count rate over the sampling period are provided in Appendix D. Coupled with the instrument response check measurements also performed over the entire sampling period, the stability in the measured beta background provides evidence of instrument stability. The time series plots of the background data set also reveal that the variability in the data set is small and that the frequency of values falling outside of the $\pm$ 50% range is within the expected range. Again, since the alpha instrument backgrounds results were typically one or zero cpm, a time series plot was not presented using the alpha instrument background data.

2.6.5 Portable Instrument Background Adjustment

Since the contribution from naturally occurring radionuclides present in the Building 3 High Bay survey units was anticipated to be relatively small compared with the DCGLs (i.e., notably less than 20,147 dpm/100 cm^2 alpha or 6,980 dpm/100cm^2 beta), it was conservatively decided to accept their contribution to background as attributable to Site activities. Consequently, in this report, data sets are presented without background corrections.

2.7 REMOVABLE RADIOACTIVITY MEASUREMENTS

Technical smears were used to collect a sample of the removable radioactivity on building surfaces in all three survey units. The smear samples were collected by wiping the cloth filter over a 100 cm^2 area of the surface to be sampled using moderate pressure applied with two fingers. The smears were packaged to prevent sample contamination, labeled with a unique identification number linked to the location from which the smear sample was collected, and then measured for radioactivity.

Because the historical radiological surveys performed in the buildings consistently showed the absence of measurable removable surface radioactivity, and because the total (fixed + removable) residual surface radioactivity was expected to be low relative to the 20,147 dpm/100 cm^2 alpha and 6,980 dpm/100cm^2 beta DGCLs in all survey units identified above, removable surface activity (RSA) measurements were not necessary to demonstrate compliance with the building DCGLs. However, to demonstrate compliance with the unrestricted release criterion for removable residual radioactivity, and to validate the assumptions used in the derivation of the DCGLs for the dose model, RSA measurements were taken after the direct static total surface activity (TSA) measurements were collected. The RSA measurements demonstrate compliance with ABB's radioactive materials license, even though most of the measurement locations indicated that there was zero likelihood for the existence of removable radioactivity in excess of 1,000 dpm/100cm^2 above background given that practically all TSA direct alpha and beta instrument background corrected measurements were less than the removable radioactivity unrestricted release criterion. No attempt was made to adjust the RSA measurement data to account for smear collection efficiency.

2.7.1 Instrument

Smear samples were counted on a Canberra Series 5 XLB Automatic Low Background Alpha/Beta Gas Proportional counting system. Background and response checks were performed at least once a day when in use.

2.7.2 Instrument Calibration

The Canberra Series 5 XLB was calibrated for both alpha and beta. The alpha channel was calibrated with a Th-230 NIST traceable source. The beta channel was calibrated with a Tc-99 NIST traceable source. Efficiency was determined utilizing calibration sources created on a glass fiber filter inside an aluminum planchet, which is the same material and geometry for analysis of the smears. The calibration certificates for these sources and the calibration data sheets for the instrument are provided in Appendix D.

2.7.3 Measurement Detection Limitations

In order to calculate the statistically significant surface radioactivity, which could be distinguished from background (*a posteriori* MDC), Equation 2-1 was modified to remove the probe area term as shown in Equation 2-6. The parameters for calculating the MDC are presented in Table 2.8.

$$MDC = \frac{3 + 4.65\sqrt{C_b}}{T_s \times \in_T}$$

(Equation 2–6)

Where: MDC = the minimum surface radioactivity concentration above background radioactivity (in dpm/100 cm^2) that can be detected with 95% confidence.

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

T_s = sample count time (in minutes).

$\in_T$ = counting system efficiency in count/disintegration.

Table 2.8 Removable Contamination Measurement MDC Parameters

Parameter		Canberra Series 5 XLB 8516 Count Date					
		5/26/11		6/6/11		6/07/11	
		α	β	α	β	α	β
C_b	Background Counts	0.00	0.75	0.00	0.80	0.05	0.90
ε_T	Instrument system efficiency in counts/disintegration	0.2074	0.2900	0.2074	0.2900	0.2074	0.2900
MDC	dpm/100 cm^2	14	24	14	25	19	26

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It is further important to note that the net MDC distinguishable above background listed in the above table are significantly lower than the removable radioactivity unrestricted release criterion.

2.7.4 Instrument Background Measurements

For those survey units where smears were collected, they were analyzed by a Canberra Series 5 XLB Automatic Low Background Alpha/Beta Gas Proportional counting system. Background measurements were taken as part of the response checks for the instrument and periodically when in use. Alpha and Beta background measurements are provided in Appendix D.

2.7.5 Instrument Background Adjustment

Since the background measured by the Series 5 XLB was relatively small compared with the DCGLs (i.e., notably less than 20,147 dpm/100 cm^2 alpha or 6,980 dpm/100cm^2 beta), it was conservatively decided to accept their contribution to background as attributable to Site activities. Consequently, in this report, data sets are presented without background corrections.

3.0 FIELD SURVEY AND SURVEY 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 C) of this report.

3.1 FIELD SURVEY RESULTS OVERVIEW

94 direct static surface measurements (not counting replicate measurements) and an equal number of removable surface measurements from the wall, floor, ceiling, and roof surfaces from 3 survey units were collected and analyzed as part of FSS areas for this report. For data reduction purposes, the arithmetic mean of the initial sample measurement result and the corresponding QC replicate direct static measurement result were used as the reported value for a specific sample location at which a replicate measurement was made. Six replicate measurements were collected as part of the overall project quality assurance/quality control (QA/QC). Further information about the replicate measurements and the assurance of precision and variability is presented in Section 6.0.

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 C).

Once the survey unit data was assessed and verified that it is acceptable for comparison to the release criteria, it was evaluated against the $DCGL_W$s.

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 C) of this report.

3.2.1 Survey Unit CE-FSS-30-01

Survey Unit CE-FSS-30-01 covers the interior surfaces of Building 3 High Bay, (rooms 45, 45A, 45B, 45C, 48, and 49) and consists of approximately 4296 meters squared ($m^{2)}$ of surface area. Figure 3.1 presents an overview of the survey unit. Thirty-three survey locations were randomly selected within the Class 3 survey unit to represent the distribution of residual radioactivity for the survey unit Data associated with this survey unit are provided in Appendix A.

Total Surface Activity Scanning Results

Approximately 70 percent of the floor surface area for Survey Unit CE-FSS-30-01 was surveyed by walking the floor monitor (Ludlum 439-1F) in a systematic manner. Instrument readings ranging from 600 cpm to 800 cpm (gross) beta were recorded during the walkover survey. No elevated readings exceeding the gross alpha or beta investigation levels of 1,705 cpm or 3,854 cpm (as listed in Table 2.5) respectively were identified during the walkover scan survey by the FSS technician. Therefore, no additional direct static measurements to investigate anomalies were performed.

Total Surface Activity Results

Thirty-three randomly-placed direct static surface activity measurements were obtained for FSS in Survey Unit CE-FSS-30-01 with the Ludlum Model 2224 Portable Radiation Survey Instrument coupled with the 43-89 or 43-93 A/B detector probe. The analytical results show that the mean/median removable radioactivity is appreciably below the $DCGL_W$s. 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 alpha and beta total surface activity levels for Survey Unit CE-FSS-30-01.

Removable Results

Thirty-three randomly-placed removable surface activity measurements (at the direct static locations) were obtained for FSS in Survey Unit CE-FSS-30-01 and analyzed on Site with the automatic low background XLB Alpha/Beta smear counter. The analytical results show that the mean/median removable 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 alpha and beta removable surface activity levels for Survey Unit CE-FSS-30-01.

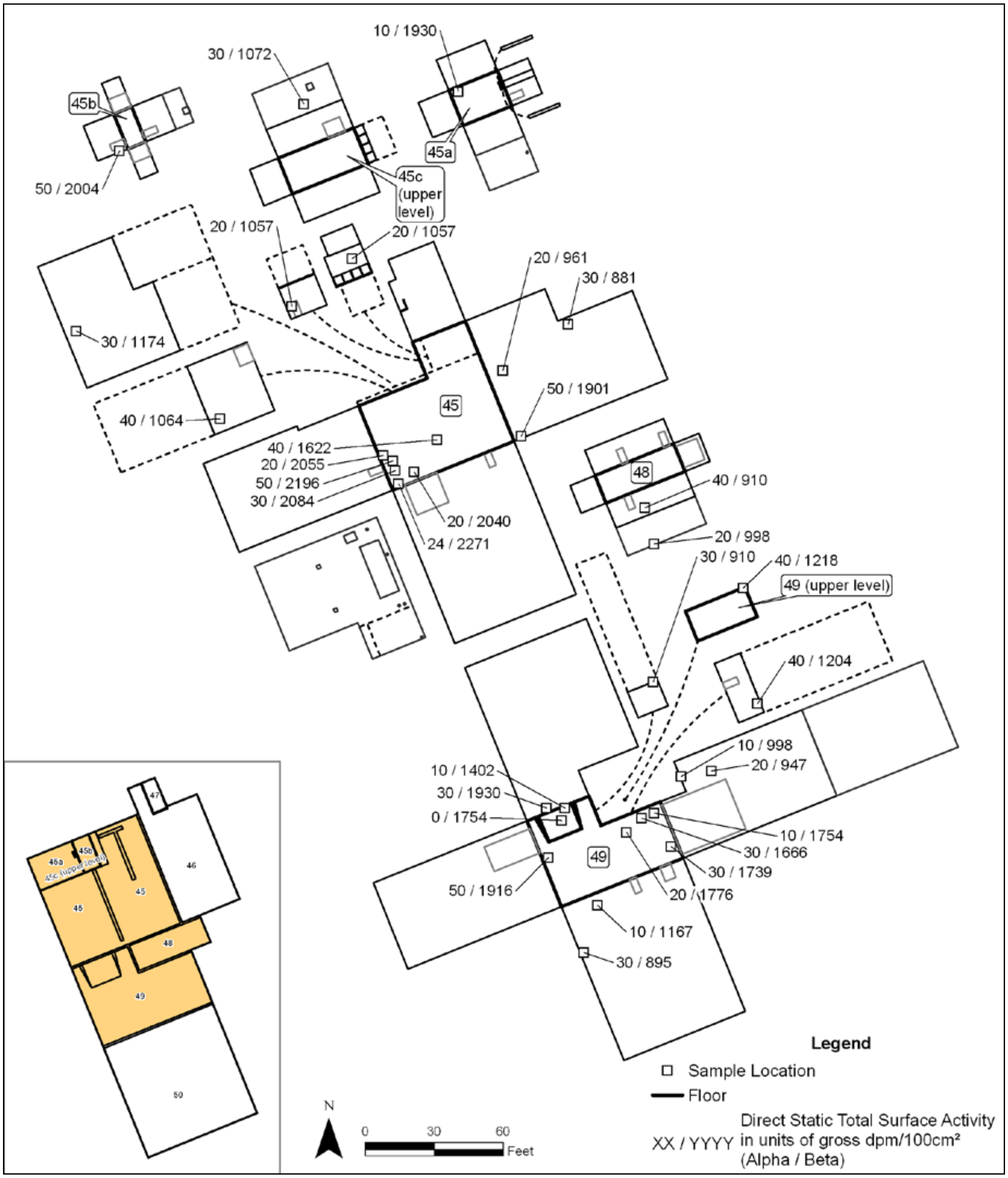

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Figure 3.1: Survey Unit CE-FSS-30-01 Total Surface Activity (dpm/100cm^2)

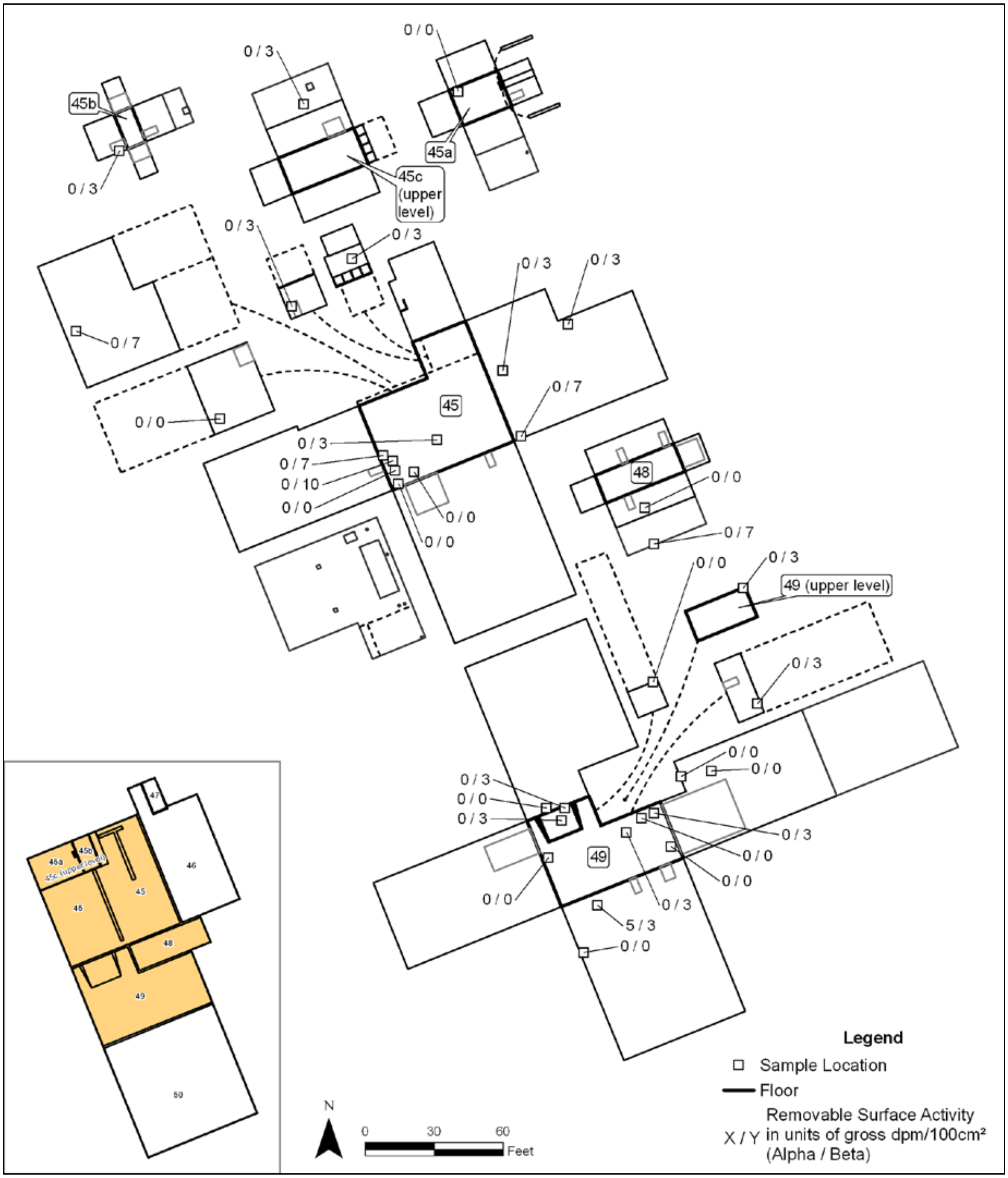

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Figure 3.2: Survey Unit CE-FSS-30-01 Removable Surface Activity ($dpm/100cm^2$)

3.2.2 Survey Unit CE-FSS-30-02

Survey Unit CE-FSS-30-02 covers the interior surfaces of Building 3C High Bay Annex, (rooms 46, 47, 50) and consists of approximately 2,269 m^2 of surface area. Figure 3.3 presents an overview of the survey unit. Thirty-two 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.

Total Surface Activity Scanning Results

Approximately 80 percent of the floor surface area for Survey Unit CE-FSS-30-01 was surveyed by walking the floor monitor (Ludlum 439-1F) in a systematic manner. Instrument readings ranging from 600 cpm to 800 cpm (gross) beta were recorded during the walkover survey. No elevated readings exceeding the gross alpha or beta investigation levels of 1,705 cpm or 3,854 cpm (as listed in Table 2.5) respectively were identified during the walkover scan survey by the FSS technician. Therefore, no additional direct static measurements to investigate anomalies were performed.

Total Surface Activity Results

Thirty-two randomly-placed direct static surface activity measurements were obtained for FSS in Survey Unit CE-FSS-30-02 with the Ludlum Model 2224 Portable Radiation Survey Instrument coupled with the 43-89 or 43-93 A/B detector probe. The analytical results show that the mean/median removable radioactivity is appreciably below the $DCGL_W$s. 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 alpha and beta total surface activity levels for Survey Unit CE-FSS-30-02.

Removable Results

Thirty-two randomly-placed removable surface activity measurements (at the direct static locations) were obtained for FSS in Survey Unit CE-FSS-30-02 and analyzed on Site with the automatic low background XLB Alpha/Beta smear counter. The analytical results show that the mean/median removable 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 alpha and beta removable surface activity levels for Survey Unit CE-FSS-30-02.

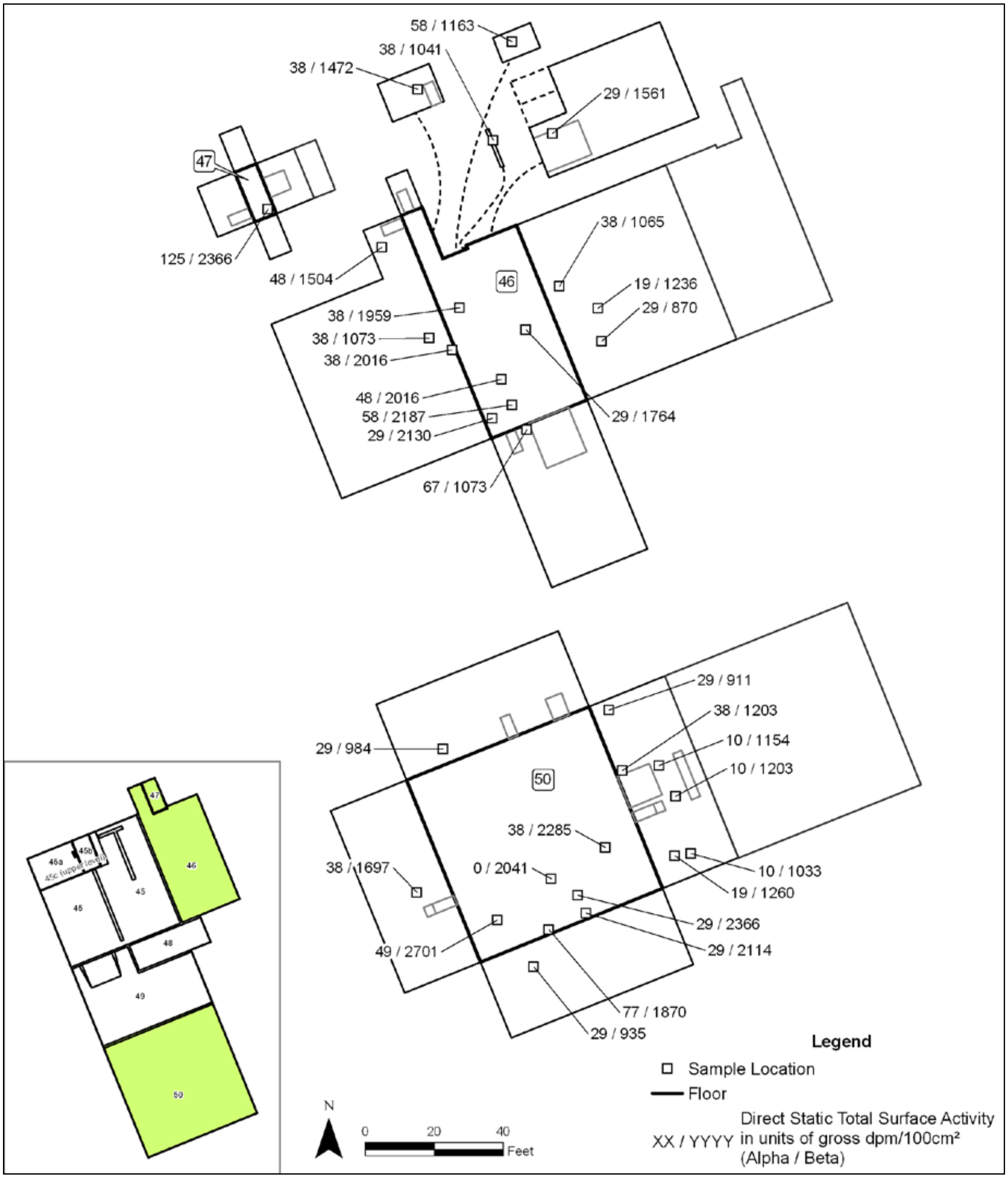

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Figure 3.3: Survey Unit CE-FSS-30-02 Total Surface Activity (dpm/100cm^2)

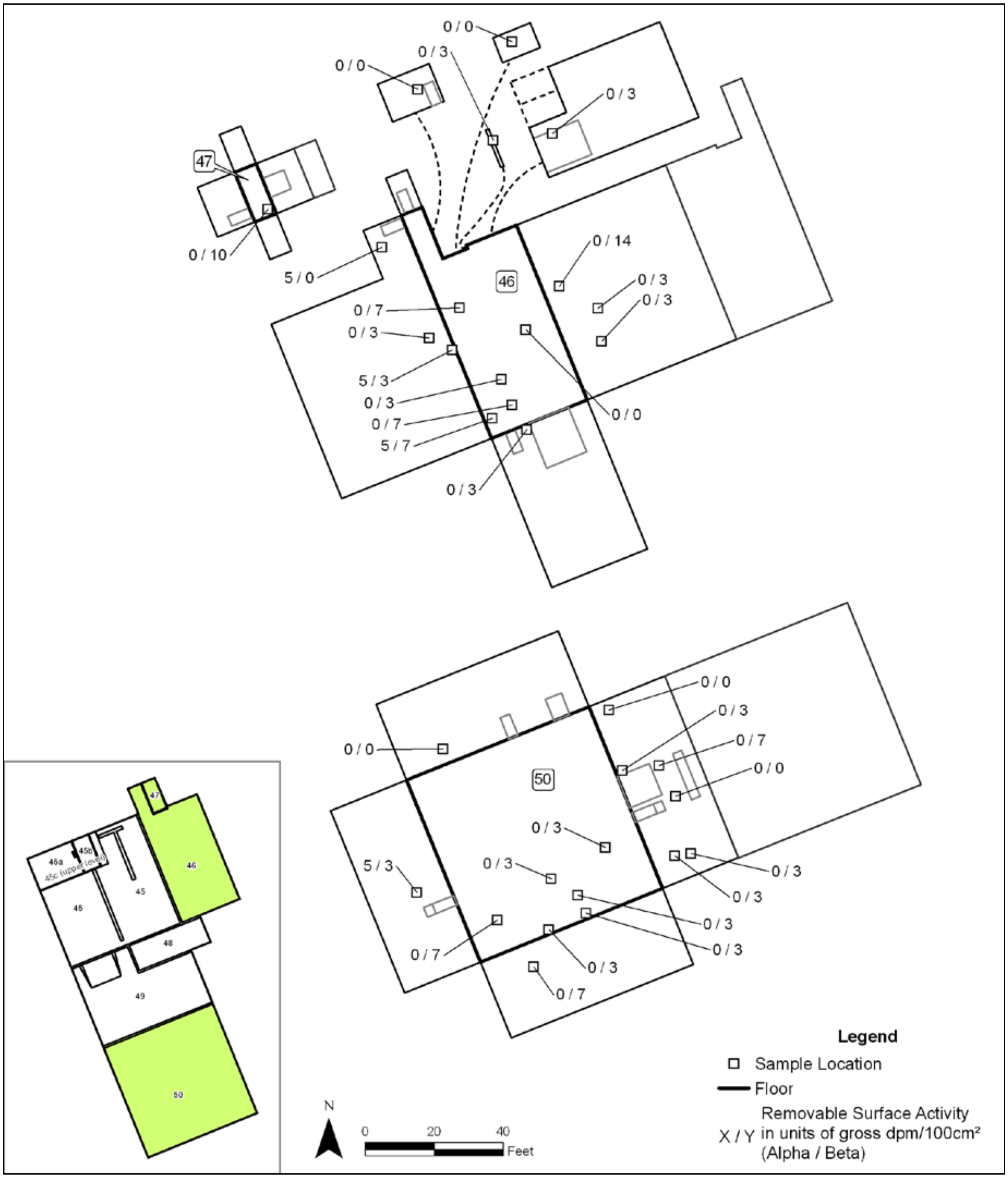

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Figure 3.4: Survey Unit CE-FSS-30-02 Removable Surface Activity ($dpm/100cm^2$)

3.2.3 Survey Unit CE-FSS-30-03

Survey Unit CE-FSS-30-03 covers the exterior surfaces of Building 3 High Bay, and consists of approximately 3,501 m^2 of surface area. Figure 3.5 presents an overview of the survey unit. Twenty-nine 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.

Total Surface Activity Scanning Results

Based on information presented in the HSA that the building roof had been replaced in the 1960's and again in the late 1970's and that recent characterization surveys have not identified residual radioactivity, no scans were performed in this survey unit.

Total Surface Activity Results

Twenty-nine randomly-placed direct static surface activity measurements were obtained for FSS in Survey Unit CE-FSS-30-03 with the Ludlum Model 2224 Portable Radiation Survey Instrument coupled with the 43-89 or 43-93 A/B detector probe. The analytical results show that the mean/median removable radioactivity is appreciably below the $DCGL_W$s. 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 alpha and beta total surface activity levels for Survey Unit CE-FSS-30-03.

Removable Results

Twenty-nine randomly-placed removable surface activity measurements (at the direct static locations) were obtained for FSS in Survey Unit CE-FSS-30-03 and analyzed on Site with the automatic low background XLB Alpha/Beta smear counter. The analytical results show that the mean/median removable 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 alpha and beta removable surface activity levels for Survey Unit CE-FSS-30-03.

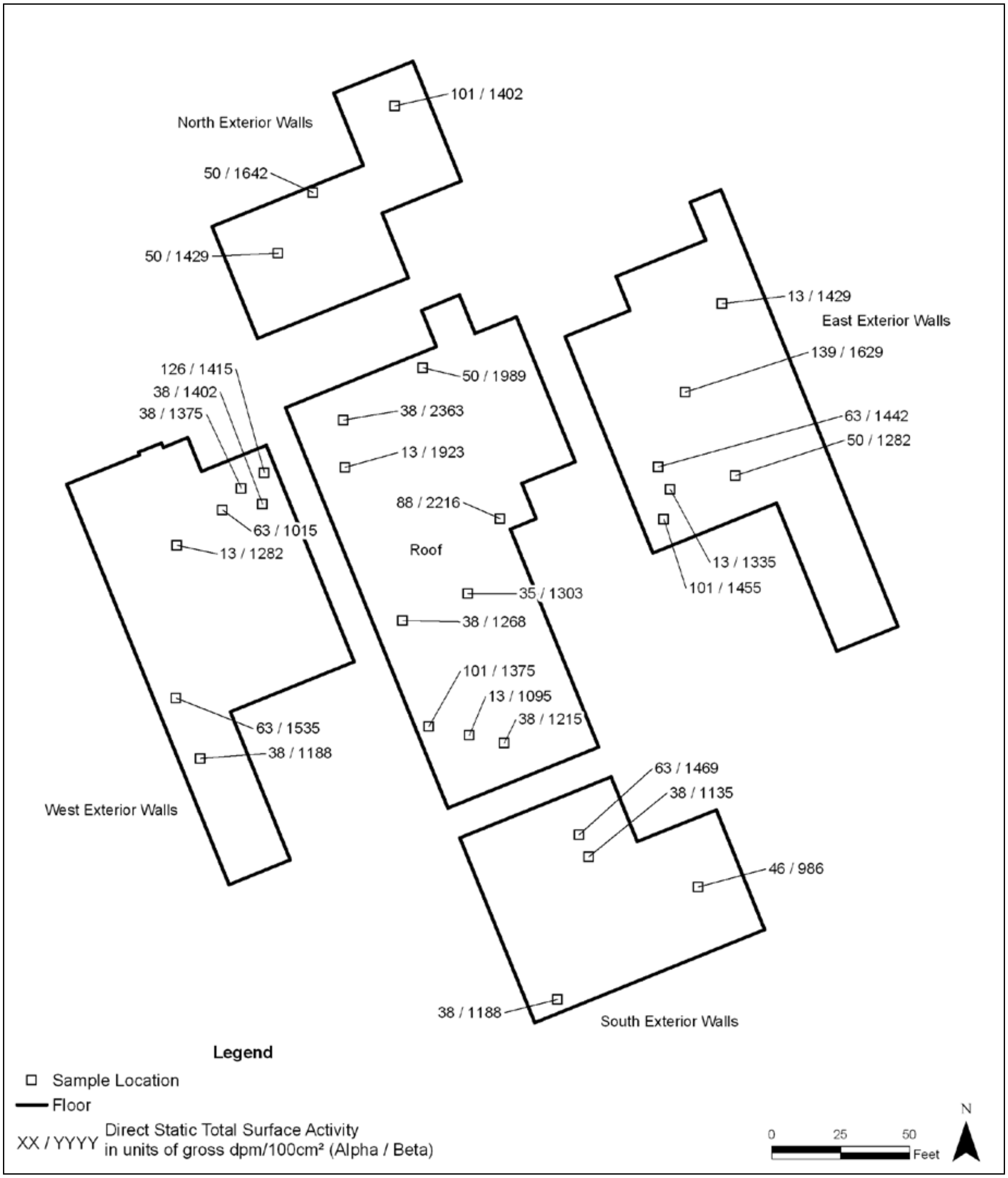

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Figure 3.5: Survey Unit CE-FSS-30-03 Total Surface Activity (dpm/100cm^2)

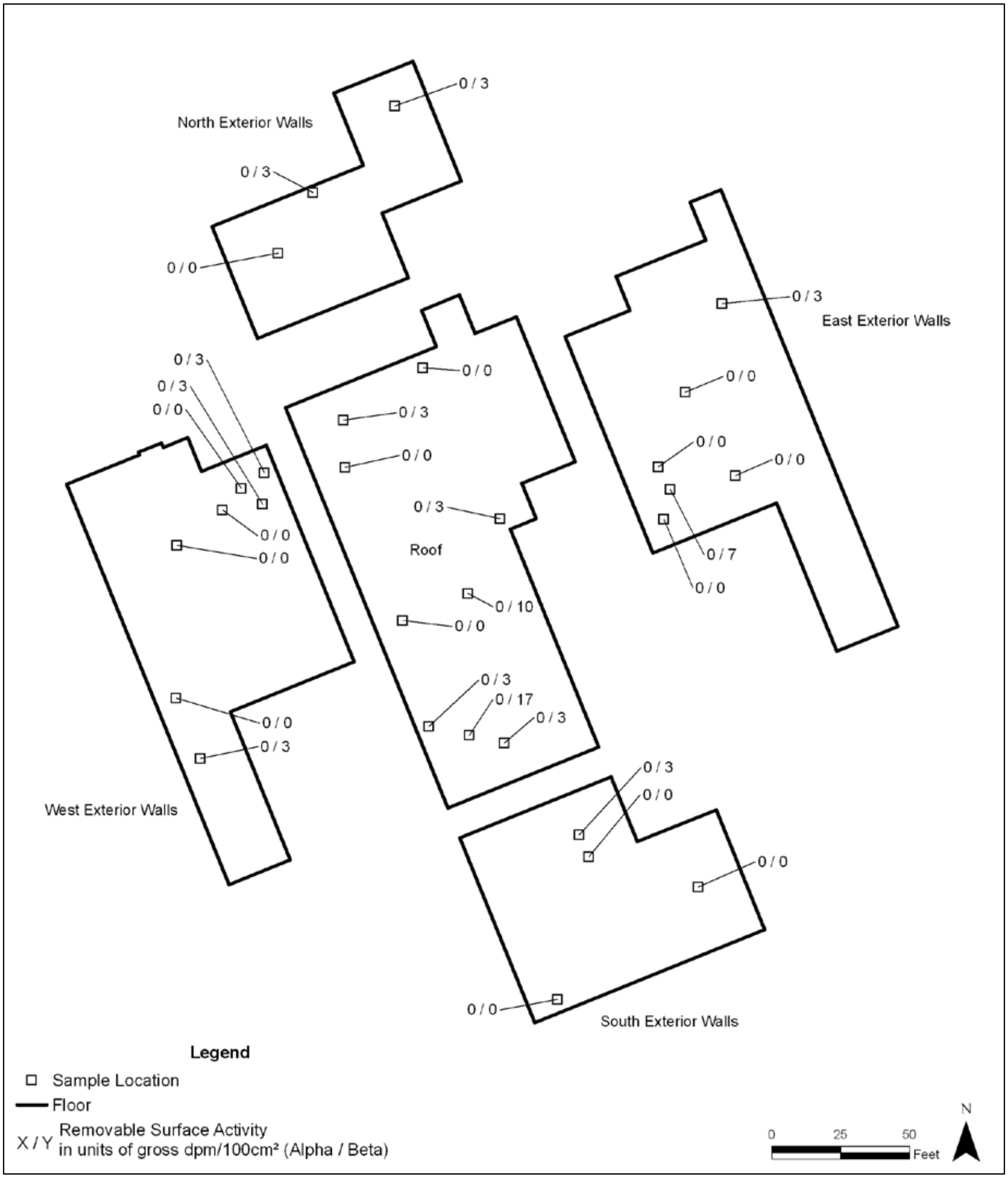

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Figure 3.6: Survey Unit CE-FSS-30-03 Removable Surface Activity (dpm/100cm^2)

3.3 SURVEY SUMMARY RESULTS

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

3.3.1 Scan Survey

Table 3.1 presents the summary results of the scan surveys, the number of volumetric samples obtained as a result of elevated scan survey readings, and the highest measurements obtained during static counts performed in locations where a discernable increase in the count rate was identified. Scan survey areas are identified on the applicable survey unit Radiological Survey Map, located in the survey unit specific appendix.

Judgmental scan surveys of the floor, pipe trench and pipe penetration surfaces were performed in Survey Units CE-FSS-30-01 and CE-FSS-30-02. While the scans ranged from background to levels expected from NORM levels expected in the concrete matrix, none of the reported gross scan results exceeded 10% of the beta investigation level which was calculated to be 80% of the $DCGL_w$ as provided in the FSSP (AMEC, 2011). Scans were not performed in survey unit CE-FSS-30-03 as this area did not have a potential for having residual surface radioactivity. None of these survey units were identified as having residual surface radioactivity in excess of the total surface DCGL or were areas that had a significant potential for having residual surface radioactivity. These survey units were never subject to remedial activities to reduce the surface radioactivity to within acceptable levels.

Table 3.1: Scan Survey Results Summary

Survey Unit (CE-FSS)	**Building Scan Results**									
	Survey Unit Class	**Percent of Survey Unit Surveyed (accessible floor)**	**Number of Elevated Locations Identified and Sampled**	**Recorded Background Reading** (cpm)		**Highest Scan Reading** (gross cpm)		**Highest Scan Reading** (net cpm)	
				α	β	α	β	α	β
30-01	3	45	0	2	559	2	800	0	241
30-02	3	75	0	3	614	3	800	0	186
30-03	3	0	0	n/a	n/a	n/a	n/a	n/a	n/a

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3.3.2 Total Surface Measurements

In addition to scan surveys, 1-minute direct static surface measurements were performed at FSS measurement locations using the alpha/beta scintillation detectors. These 1-minute static measurements were used 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 alpha direct static readings and Table 3.3 provides a summary of the beta direct static readings performed at each measurement location.

Table 3.2 Summary Statistics, Direct Alpha Static Measurement Data

Alpha - α Statistic	Survey Unit (CE-FSS)		
	30-01	**30-02**	**30-03**
Number of Measurements	33	32	29
Arithmetic Mean	27	38	54
Standard Deviation (sample)	13	23	34
Standard Error of the Mean	2.3	4.1	6.2
Coefficient of Variation	0.48	0.61	0.62
Geometric Mean	25	34	44
Maximum	50	125	139
Median	30	38	46
Minimum	0	0	13
Range	50	125	126
UCL95 (median)	30	38	63
LCL95 (median)	20	29	38

Note 1: Except for number of samples, standard error and the coefficient of variation (unitless) all statistics reported above are in units of dpm/100 cm^2.

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Table 3.3 Summary Statistics, Direct Beta Static Measurement Data

Beta - β Statistic	Survey Unit (CE-FSS)		
	30-01	**30-02**	**30-03**
Number of Measurements	33	32	29
Arithmetic Mean	1471	1570	1441
Standard Deviation (sample)	466	532	326
Standard Error of the Mean	81	94	61
Coefficient of Variation	0.32	0.34	0.23
Geometric Mean	1399	1485	1410
Maximum	2271	2701	2363
Median	1402	1488	1402
Minimum	881	870	986
Range	1390	1831	1377
UCL95 (median)	1776	1959	1442
LCL95 (median)	1064	1154	1282

Note 1: Except for number of samples, standard error and the coefficient of variation (unitless) all statistics reported above are in units of dpm/100 cm^2.

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3.3.3 Removable Surface Measurements

A summary of the removable results is presented by survey unit in Table 3.4.

Table 3.4 Summary Statistics, Removable Radioactivity

Statistics	Survey Unit (CE-FSS)					
	30-01		30-02		30-03	
	Alpha	Beta	Alpha	Beta	Alpha	Beta
Number of Measurements	33		32		29	
Arithmetic Mean	0.15	2.42	0.63	3.66	0	2.31
Standard Deviation (sample)	0.87	2.66	1.68	3.19	0	3.69
Standard Error of the Mean	0.15	0.46	0.30	0.56	0	0.69
Coefficient of Variation	5.74	1.10	2.69	0.87	0	1.60
Geometric Mean	5	3.82	5	4.10	0	3.93
Maximum	5	10	5	14	0	17
Median	0	3	0	3	0	0.00
Minimum	0	0	0	0	0	0
Range	5	10	5	14	0	17
UCL95 (median)	0	3	0	3	0	3
LCL95 (median)	0	0	0	3	0	0

Note 1: Except for number of samples, standard error and the coefficient of variation (unitless) all statistics reported above are in units of dpm/100 cm^2.

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4.0 ANALYSIS OF RESULTS FOR COMPLIANCE

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

4.1 DECISION RULES

IF the evaluation of the FSS data from a single survey unit indicates that:

- The mean/median surface activity concentration measurement result is less than the $DCGL_w$ (6,980 dpm/100cm^2 Co-60 and 20,148 dpm/100cm^2 uranium); **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 on the building surfaces greater than the $DCGL_{emc}$; **AND**
- The cost benefit analysis indicates that residual radioactivity on the building surfaces 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 building decontamination waste 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 and unacceptable radiological risk to humans under any reasonable and foreseeable future use or occupancy.

4.2 FIELD SURVEY RESULTS COMPARED TO THE $DCGL_S$

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.

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 Building Metrics

	Metric	CE-FSS-30-01	CE-FSS-30-02	CE-FSS-30-03
Unity	Power of Sign Test	~1	~1	~1
Total U ($dpm/100cm^2$)	Median	30	38	46
	UCL_{95} of Median	30	38	63
	Arithmetic Mean	27	38	54
	Geometric Mean	25	34	44
	Maximum	50	125	139
Co-60 ($dpm/100cm^2$)	Median	1402	1488	1402
	UCL_{95} of Median	1776	1959	1442
	Arithmetic Mean	1471	1570	1441
	Geometric Mean	1399	1485	1410
	Maximum	2271	2701	2363

1) No measure of the building 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, 63 $dpm/100cm^2$, is less than the $DCGL_w$ by a factor of more than 319, and the highest Co-60 UCL_{95} estimate of the median, 1,959 $dpm/100cm^2$, is less than the $DCGL_w$ by a factor of more than 3. 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 20,148 $dpm/100cm^2$ (Total U DCGL) or 6,980 $dpm/100cm^2$ (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 Building 3 High Bay meets all 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 (median) total surface residual radioactivity concentration on the building surfaces in each survey unit is below the $DCGL_w$ values of 20,148 dpm/100cm^2 (uranium) and 6,980 dpm/100cm^2 (byproduct).

The median total surface residual radioactivity concentration in each survey unit has been demonstrated to be less than the $DCGL_w$ of values of 20,148 dpm/100cm^2 (uranium) and 6,980 dpm/100cm^2 (byproduct) 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 total surface activity measurement was identified as having uranium and Co-60 activity greater than 139 dpm/100cm^2 and 2701 dpm/100cm^2 respectively, significantly below the $DCGL_w$ value of 20,148 dpm/100cm^2 (uranium) and 6,980 dpm/100cm^2. Sum of fraction (unity) values were well below 0.1. No locally elevated concentrations of residual radioactivity were identified above the direct measurement or 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 C), and illustrate the power of the Sign Test to conclude that the null hypothesis (that the total surface residual radioactivity concentration on the building surfaces exceeds the allowable radioactivity concentration) should be rejected for all survey units.

5.0 QUALITY CONTROL AND DATA QUALITY ANALYSIS

An important aspect of any survey or sampling evolution is the effort made to assure the quality of data collected. It was critical to assure the quality of all 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 within the building 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 one primary approach: QC measurements.

5.1.1 Quality Control Measurements

A significant portion of the data comes from in situ field measurements using conventional health physics techniques and practices and from wipe samples measured by gas proportional system (laboratory). Both require additional steps in order to ensure accuracy of the sampling techniques and analysis methodologies.

5.1.1.1 Field Survey Replicate Measurements

The first of the two data sets evaluated contains the replicate measurements periodically made over the duration of the sampling period. In all, 6 replicate measurements were made in the 3 survey units under consideration in this survey report. Table 5.1summarizes the cumulative paired replicate measurement results collected over the course of the performance of the final status survey. A simple linear regression analysis was performed to assess the comparability between the initial and replicate measurements and is graphically presented in Figure 5.1. The correlation coefficient, R^2, was determined to be 0.87 for the paired data indicating remarkably good correlation (and, therefore, good measurement precision) between replicate total surface radioactivity measurements. Such a strong measure of correlation in this paired data set is remarkable because the vast majority of the replicate measurements were made on surfaces where little, if any, residual surface radioactivity was measured in excess of the minimum detectable activity (MDA). As a result, instrument background response (owing to the ubiquitous nature of background radiation) dominates the total activity reported, and natural variability in background radiation tends to produce smaller correlation coefficients as opposed to larger ones. In addition to the regression analysis of the replicate data set, a two-sample comparison of the data set is presented in Figure 5.2 and Figure 5.3. These figures graphically portray the virtually identical probability density functions of the initial and replicate data sets and offer solid evidence that the direct static measurements of building surfaces are highly reproducible. Thus, the figures serve as a good indicator of the measurement precision.

Table 5.1 Results of Replicate Direct Static Surface QC Measurements

Sample Location		Measured Activity (dpm/100cm^2)			
Initial Measurement	Replicate Measurement	Initial Measurement α	Replicate Measurement α	Initial Measurement β	Replicate Measurement β
D 30-01-022	D 30-01-022QC	10	38	2,099	2,443
D 30-01-023	D 30-01-023QC	50	50	2,055	2,336
D 30-02-003	D 30-02-003QC	50	50	1,923	3,271
D 30-02-009	D 30-02-009QC	38	38	1,244	2,150
D 30-03-002	D 30-03-002QC	50	19	1,282	1,325
D 30-03-018	D 30-03-018QC	25	67	1,028	943

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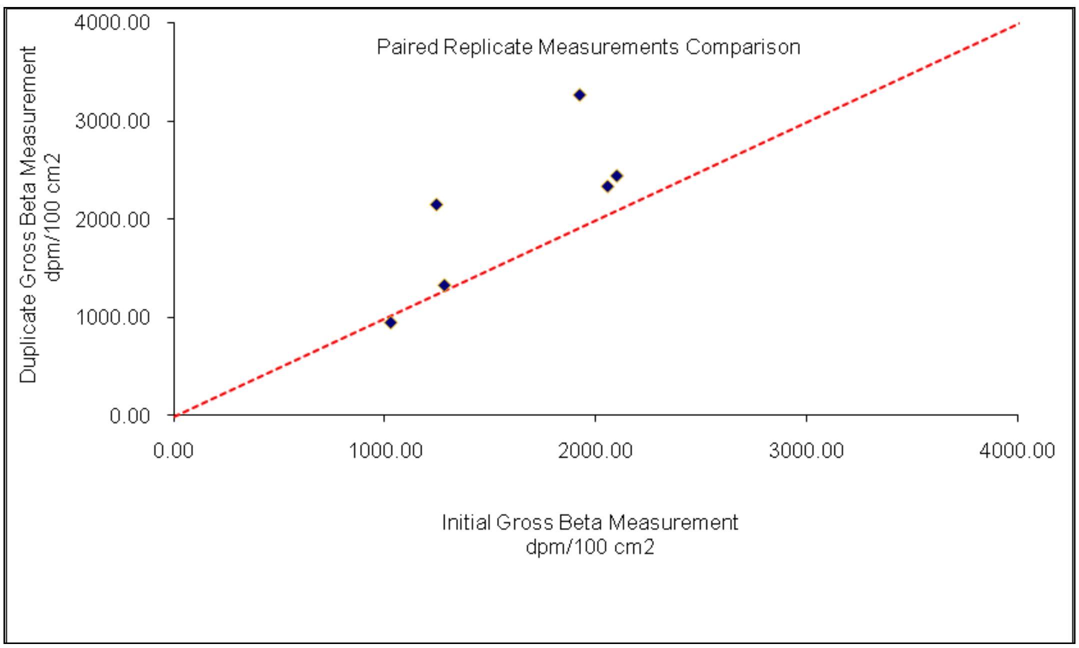

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Figure 5.1 Comparison Between Replicate Direct Static Surface Measurements

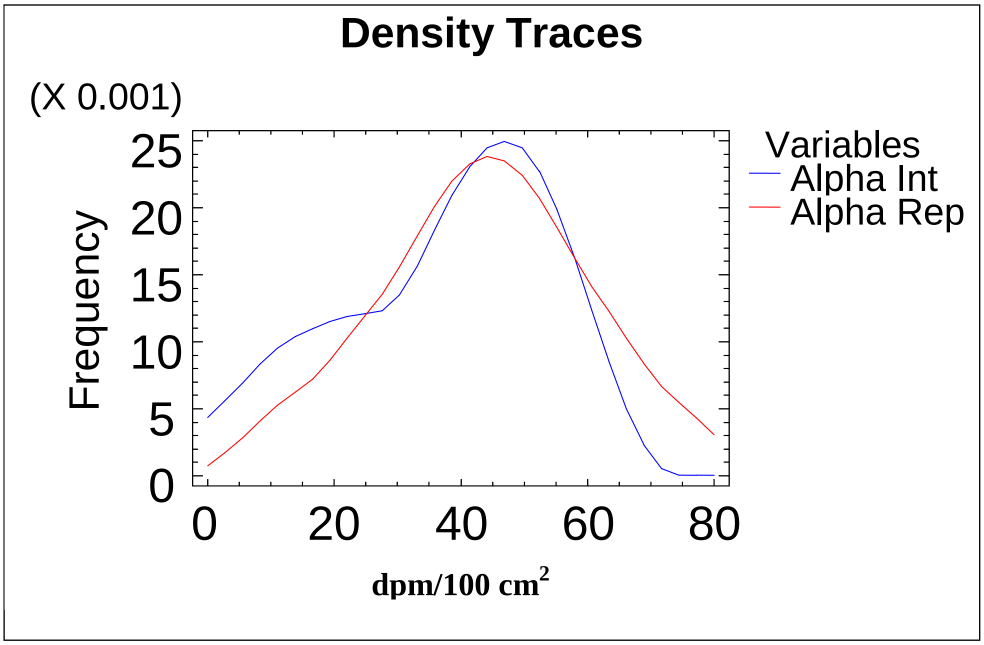

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Figure 5.2 Alpha Two-Sample Comparison of Density for Replicate Measurements

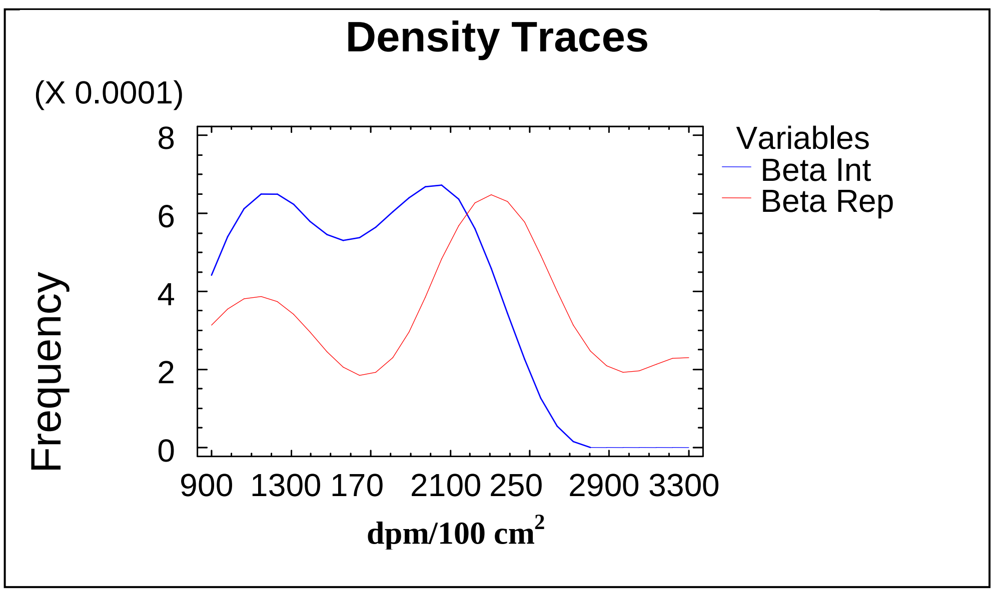

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Figure 5.3 Beta Two-Sample Comparison of Density for Replicate Measurements

5.1.2 Field Instrument Response Checks

The second of the two data sets used to present the quality of direct static surface measurements is the response of the instruments (Ludlum 2224 with the 43-89 or 43-93 αβ probe) to a surface deposited activity source with a known amount of radioactivity. It is an anodized surface source containing Tc-99 radioactivity. The source was manufactured and certified to be NIST traceable (see copy of manufacturer's certification in Appendix I).

Prior to initiating a survey each day, periodically, and at the end of a survey each day, the survey instrument in use was used to make a measurement on the known concentration source. Instrument response check data for each probe used during the final status survey is sorted and presented for individual probes. Response check data sheets are provided in Appendix D. A total of 57 response check measurements were made with the six combinations of instrument packages used during the survey period.

A control chart is provided for each of the individual probes to graphically portray the steadfastness of the instruments' responses to the source over the sampling period in Appendix D. Two different sources were used to perform response checks on the Ludlum 2224 with the 43-89 or 43-93 A/B probe, so there are two control charts for each probe. Notable is the relatively tight band within which the response checks fall. No degradation of the instruments' response was observed over the entire sampling period. It should be noted that even though the instruments presented were only used one or two days during the FSS, additional source response check data was presented to allow the reviewer more data for trend comparisons.

5.1.3 Laboratory Instruments

The quality of removable measurements can be measured by the response of the instruments (Canberra Series 5 XLB) to a source of known radioactivity. A Th-230 source for alpha response checks and a Tc-99 source for beta response checks were used.

Prior to counting smears, the sources of known concentration were counted on the instrument. Instrument response check data for the Series 5 XLB is presented for both sources. Response check data sheets are provided in Appendix D.

A control chart is provided to graphically portray the steadfastness of the instruments' responses to the source over the survey period in Appendix D. No degradation of the instruments' response was observed over the entire survey period.

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/ laboratory 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 referred 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 Building 3 High Bay 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 the project. Table 5.4 presents the target DQIs and summarizes the post-sampling data quality assessment.

Inspection of Table 5.4 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 total surface measurement results are less than the $DCGL_w$ (20,148 dpm/100cm^2 uranium and 6,980 dpm/100cm^2 byproduct); **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 on building surfaces 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 5.4 Target Data Quality Indicators and Findings

DQI	Quality Objective	Significance	Action/Remark	Finding
Completeness	90% completeness	Less than complete data set could decrease confidence in supporting information.	A minimum 29 direct static surface radioactivity measurements were planned in each of the 3 survey units of the Building 3 High Bay. 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 minimum of 29 direct static surface measurements were actually collected from each survey unit. Ninety-four direct surface emission measurements (87 was the specified minimum) were obtained (>100%).	DQI accepted.
Comparability	Affects ability to combine data sets produced using different sampling and/or analytical methods.	Data collected from randomly selected locations within a survey area are unbiased and comparable by design and can be combined. Combining of other data sets would be subject to appropriate two-sample statistical test methods designed to detect significant differences between samples or populations.	Sampling procedures and protocols were used throughout the FSS process for remaining impacted Site areas. No critical deviation from these procedures was encountered.	DQI accepted.
Representativeness	Non-representativeness increases or decreases Type I error depending on the bias.	Sample allocation included a minimum number of unbiased, randomly distributed sample locations based on survey design.	Sample allocation for Survey Units was 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. The sample locations selected meet the intent of the survey design and are considered representative of conditions of the Building 3 High Bay.	DQI accepted.

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. 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.
Accuracy	Sampling and data handling can introduce bias and affect Type I and Type II errors.	Sampling and measurements were governed by procedures. Instruments were calibrated with NIST traceable sources.	All sampling and field measurement processes were controlled by approved written procedures. Analytical measurements were controlled by approved procedures. Survey and sampling results were recorded in accordance with approved written procedures.	DQI accepted.

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6.0 SUMMARY AND CONCLUSIONS

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

More specifically, the FSS of the Building 3 High Bay demonstrates that:

- No unexpected results or trends are evident in the data.
- The sampling and survey results demonstrate that residual radioactivity in the potentially impacted area 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 Test, 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 Building 3 High Bay. The areas surveyed and sampled during FSS (survey units identified in this report) should be released from further radiological controls.