FINAL STATUS SURVEY PLAN

REVISION 1

CE WINDSOR SITE
WINDSOR, CONNECTICUT

US NRC LICENSE NUMBER 06-00217-06
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ABB	ABB Inc.
AEC	Atomic Energy Commission
ANSI	American National Standards Institute
BNFL	British Nuclear Fuel
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
cm	centimeter
Co-60	cobalt 60
COC	chain of custody
cpm	counts per minute
CT	Connecticut
CTDEP	Connecticut Department of Environmental Protection
DCGL	derived concentration guideline level
$DCGL_{EMC}$	derived concentration guideline level, elevated measurement comparison
$DCGL_W$	derived concentration guideline level, survey unit average (median) concentration corresponding to the permissible limit
DP	Decommissioning Plan
DQA	Data Quality Analysis
DQI	Data Quality Indicator
DQO	Data Quality Objective
EMC	elevated measurement comparison
EPA	Environmental Protection Agency
FSS	Final Status Survey (radiological)
FSSP	Final Status Survey Plan
FUSRAP	Formerly Utilized Site Remedial Action Program
g/cm^3	grams per cubic centimeter
GIS	geographic information system
GPS	global positioning satellite
HEU	highly-enriched uranium
HPGe	high purity germanium detector
HSA	Historical Site Assessment
keV	kilo-electron volts
LEU	low-enriched uranium
LLRW	low-level radioactive waste

m	meter(s)
m^2	meters squared
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCA	multichannel analyzer
MDC	minimum detectable concentration
MDC_{SCAN}	minimum detectable concentration for scan surveys
MeV	million electron volts
μR/h	micro-Roentgen per hour
mrem	milli-Roentgen equivalent man
NAD	North American Datum
NaI	sodium iodide detector
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Engineering
pCi/g	picocuries per gram
QA/QC	quality assurance/quality control
RCRA	Resource Conservation and Recovery Act
RESRAD	residual radioactivity computer code
SAIC	Science Applications International Corporation
SOP	standard operating procedure
U-234	uranium 234
U-235	uranium 235
U-238	uranium 238
UCL	upper confidence level
USACE	United States Army Corps of Engineers
VCA	Voluntary Corrective Action
WWTP	wastewater treatment plant
y	year

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1.0 INTRODUCTION

From the mid-1950's, the Combustion Engineering (CE) Site in Windsor, Connecticut has been involved in research, development, engineering, production, and servicing of nuclear fuels, systems, and services. The objective of the licensee, ABB Inc. (ABB) is to decommission the Site such that it will meet the criteria for unrestricted use in accordance with the requirements of the License Termination Rule at 10 Code of Federal Regulations[CFR] Part 20, Subpart E and to terminate U.S. Nuclear Regulatory Commission (NRC) License No. 06-00217-06.

Certain buildings and areas on the Site were being addressed by the U.S. Army Corps of Engineers (USACE) under the Formally Utilized Sites Remedial Action Program (FUSRAP). Recently, an agreement was reached between the USACE and the U.S. NRC that will allow ABB to complete decommissioning at the Site. Since there is extensive commingling of FUSRAP and NRC-licensed material, the USACE proposed to suspend FUSRAP activities at the Site in order to allow cleanup under NRC decommissioning (USACE, 2007). The NRC accepted this approach which allows ABB to supplement the existing decommissioning plan (DP), decommission the remainder of the Site pursuant to NRC regulations and complete license termination (NRC, 2007a). When the remediation effort is complete, ABB will demonstrate that NRC dose criteria are met for unrestricted license termination of the entire CE Site.

The decommissioning process consists of a series of integrated activities that includes development of a DP, remediation of the site, performance of the final status surveys (FSS), and license termination. The final status survey plan (FSSP) provides a systematic approach to FSS sampling based upon the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NRC, 2000) guidance. This includes:

- FSS design
- Statistical tests that will be used to evaluate the FSS results
- Scanning instruments, methods, calibration, operational checks, and minimum detectable concentrations (MDCs)
- Laboratory analytical instruments methods, calibration, operational checks, and MDCs
- FSS investigation levels
- Sample collection and control

This FSSP is designed for FSS of the remaining portions of the site that have not yet been released by the NRC, including the previously identified FUSRAP areas. This FSSP is intended to provide the basis for FSS in support of license termination for the remainder of the Site.

1.1 METHODOLOGY AND GUIDANCE USED TO DEVELOP THE FSSP

Development of the FSSP has relied principally on the methods advocated in MARSSIM and incorporates guidance from the U.S. Environmental Protection Agency (EPA) and the NRC. Principal guidance documents used include:

- NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual” (NRC, 2000)
- EPA QA/G-4, “Guidance for the Data Quality Objectives Process” (EPA, 2000)
- NUREG-1757 Vol. 2, “Consolidated NMSS Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria” (NRC, 2003)
- Draft NUREG-1505, “A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys” (NRC, 1998)
- NUREG-1507, “Minimum Detectable Concentrations with Typical Survey Instruments for Various Contaminants and Field Conditions” (NRC, 1997)

Following this brief introduction will be a discussion of the CE Windsor Site history in Section 2 and a summary of the existing data in Section 3. The seven-step data quality objective (DQO) process is described in Section 4. Section 5 discusses the design of the FSS and Section 6 details measurement techniques and the equipment and tools to be used as part of the FSS. Section 7 discusses data quality and data assessment issues.

2.0 SITE HISTORY

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

The CE Windsor property is located in the Town of Windsor, eight miles north of Hartford, Connecticut (Figure 2–1). The entire property consists of approximately 612 acres and is located at 2000 Day Hill Rd. in Windsor, Connecticut . An overview of the current site layout is shown on Figure 2–2. The NRC issued a license amendment to Byproduct License 06-00217-06, which authorized a partial site release of 365 contiguous acres of the 612 acre facility for unrestricted use (NRC, 2009). The remaining 248 acres remain 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 Windsor/Bloomfield Sanitary Landfill and Recycling Center (Landfill) and the Rainbow Reservoir portion of the Farmington River to the north; and forested land with some residential and commercial development to the east.

ABB’s activities at the Site started in 1955 with an Atomic Energy Commission (AEC) contract to begin research, development, and manufacturing of nuclear fuel 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 manufacturing and research and development activities for naval nuclear fuel were terminated by AEC by 1962.

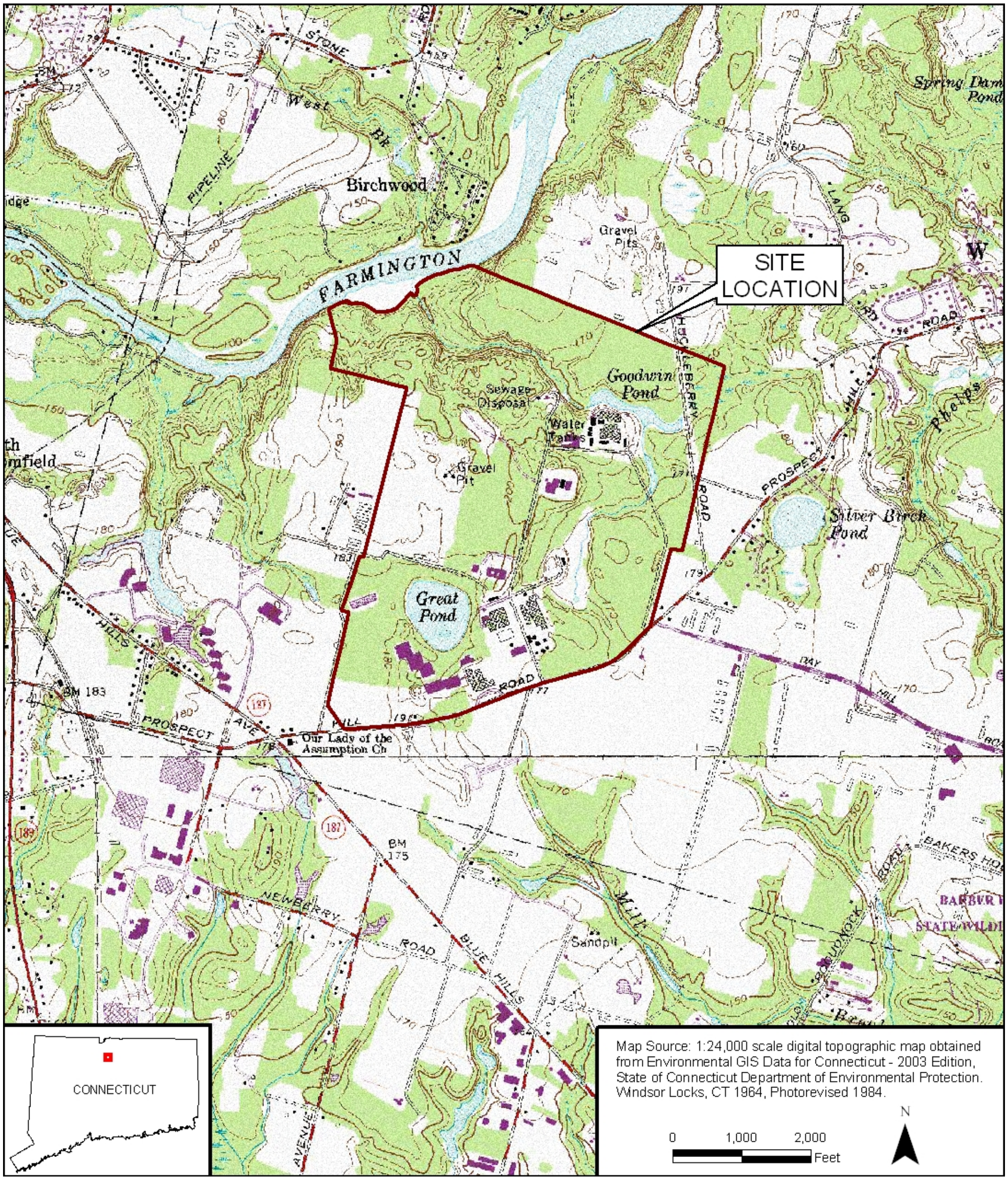

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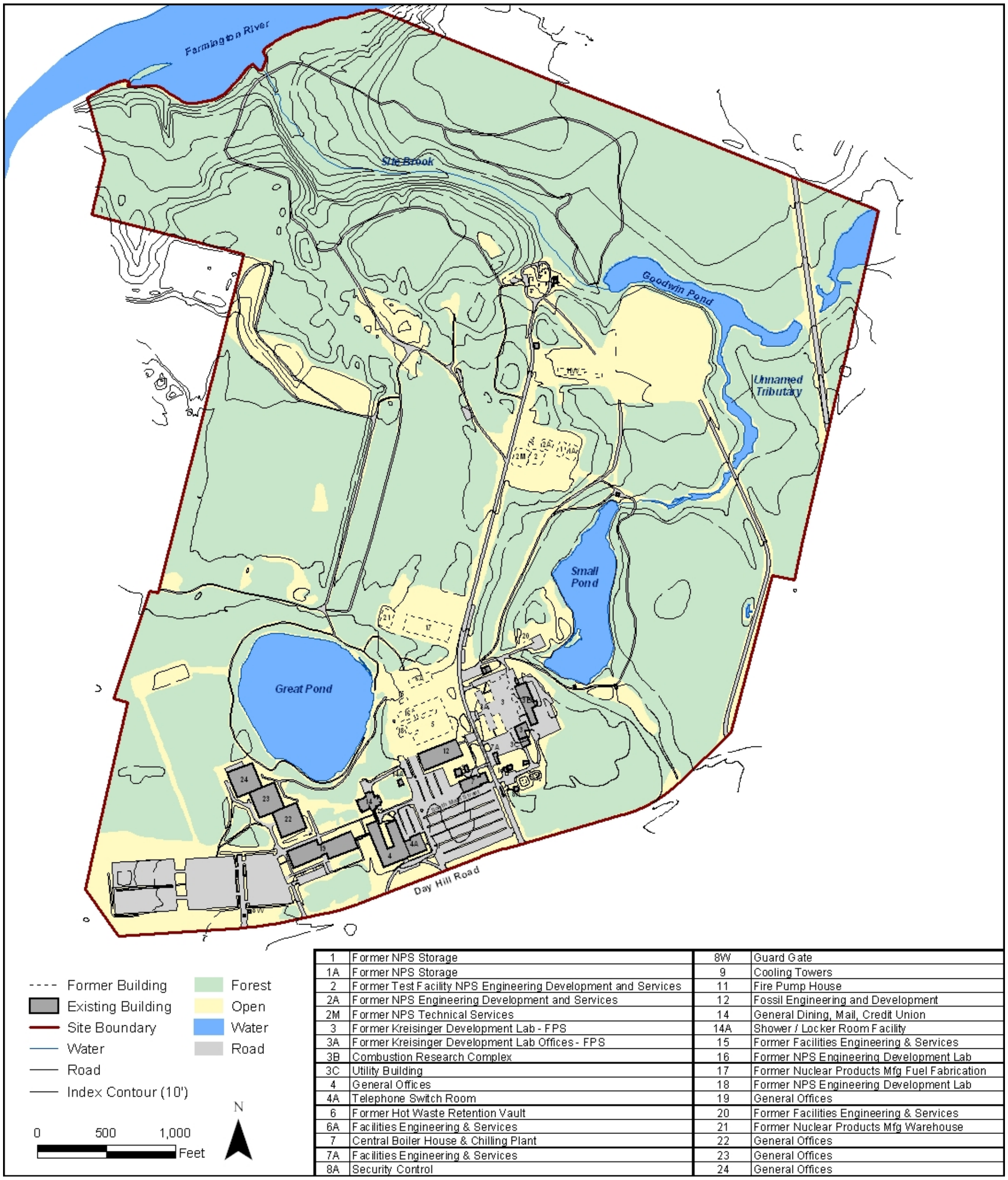
Figure 2-1 Site Location Map



1	Former NPS Storage	8W	Guard Gate
1A	Former NPS Storage	9	Cooling Towers
2	Former Test Facility NPS Engineering Development and Services	11	Fire Pump House
2A	Former NPS Engineering Development and Services	12	Fossil Engineering and Development
2M	Former NPS Technical Services	14	General Dining, Mail, Credit Union
3	Former Kreisinger Development Lab - FPS	14A	Shower / Locker Room Facility
3A	Former Kreisinger Development Lab Offices - FPS	15	Former Facilities Engineering & Services
3B	Combustion Research Complex	16	Former NPS Engineering Development Lab
3C	Utility Building	17	Former Nuclear Products Mfg Fuel Fabrication
4	General Offices	18	Former NPS Engineering Development Lab
4A	Telephone Switch Room	19	General Offices
6	Former Hot Waste Retention Vault	20	Former Facilities Engineering & Services
6A	Facilities Engineering & Services	21	Former Nuclear Products Mfg Warehouse
7	Central Boiler House & Chilling Plant	22	General Offices
7A	Facilities Engineering & Services	23	General Offices
8A	Security Control	24	General Offices

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Figure 2-2 Site Overview

From the early 1960s to 2000, ABB was involved in the research, development, engineering, production, and servicing of nuclear and fossil fuel systems. These activities were performed under both commercial and federal contracts. Projects included nuclear and combustion research for commercial use, as well as large-scale boiler test facilities and coal gasification. Nuclear fuel research and development and reactor outage servicing was conducted in Buildings 2 and 5, and components were manufactured in Building 17. Large-scale fossil fuel boiler tests were conducted in Building 3 after 1962. Wastewater pumping and dilution was conducted in Building 6 for nuclear operations.

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

The historical processes at the Site generated both low-level radioactive wastes (LLRW) as well as Resource Conservation and Recovery Act (RCRA) hazardous chemical wastes. 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).

However, additional radiological constituents of concern were identified during investigational sampling of the Burning Grounds, which were associated with licensed incineration activities involving thorium (for disposal). More detailed information regarding the additional radiological constituents of concern is contained in the Site DP (MACTEC, 2010a).

2.1 CURRENT USAGE

Some of the Site's buildings are currently occupied by a tenant: ALSTOM Power, which conducts fossil fuel research and engineering design.

The Site was a permitted hazardous waste storage facility in the 1980s. Because it was permitted, it is subject to RCRA corrective action. In 1997, ABB entered into an agreement with the EPA to perform a Voluntary Corrective Action (VCA) at the Site. The VCA program is restricted to the investigation and remediation of chemical releases at the Site. This program is also ongoing.

Due to the sale of the Site businesses to Westinghouse and ALSTOM in 2000, the Site is also subject to the Connecticut Transfer Act with respect to Site investigation and cleanup. This program is being conducted concurrently with the VCA.

As part of Commercial D&D, Building Complexes 2, 5, 6A and 17 have been decontaminated and dismantled and the below grade utilities have been removed. In addition, soil remediation to meet derived concentration guideline levels (DCGLs) has been completed, Final Status Surveys for these Complexes have been completed and accepted by the NRC (NRC, 2007b).

Building Complexes 3 and 6 were abandoned years ago, and remaining equipment, machinery, etc., and interior systems will be decontaminated and dismantled as necessary. Above grade structures are planned to be deconstructed including the removal of the building slabs and foundations. The south end of Building 3 (High Bay) will be released for unrestricted use using existing license criteria and will remain in use for fossil fuel research and development.

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.

3.0 PRELIMINARY DATA

A great deal of radiological data has been collected by CE Site Remediation Services Group in support of the ongoing Radiation Protection Program, and by MACTEC (now AMEC) in support of the characterization, decontamination, dismantlement and FSS as part of decommissioning and license termination for the CE Windsor Site. This preliminary data is important because it is 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 detect and quantify residual radioactivity that may be present, and
- Estimate the minimum sample size needed to achieve sufficient statistical power to either accept or reject the null hypothesis within the bounds of the accepted decision errors.

3.1 RESIDUAL RADIOACTIVITY PROFILE

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

However, the Burning Grounds were associated with licensed incineration activities involving thorium (for disposal). Radionuclides associated with these activities include thorium 232 and radium 226. Therefore, these additional radionuclides will be evaluated in addition to total uranium and cobalt 60 and shall be included to demonstrate compliance with the release criterion for survey unit areas deemed to be impacted from thorium incineration activities. Evaluation of these additional radionuclides is appropriate based on information acquired/obtained during decommissioning activities. However, FSS including these additional radionuclides is only appropriate for areas clearly impacted or likely to be impacted from thorium incineration activities. These areas potentially include the Burning Grounds, the Woods area, the Drum Burial Pit and the Equipment Storage Yard.

3.1.1 Uranium Series Radionuclide Profile

The relative concentrations of uranium series isotopes vary from one sample location to another across the Site due to the variability in the isotopic ratios in the materials handled. Overall, Site operations included the use of uranium enrichments ranging from small quantities of natural and depleted uranium to greater than 90% enriched uranium. The ratio of the three primary isotopes in nuclear fuel changes with enrichment and is displayed in Figure 3-1. The relationship between these isotopes of uranium for enriched nuclear fuels created by gaseous diffusion can be described by equation 3-1 (NRC, 1974).

$$SA = (0.4 + 0.38E + 0.0034E^2)10^{-6} \qquad (3\text{-}1)$$

Where: SA = specific activity of enriched uranium in Ci/g
E = % U-235 by weight

It is important to note that U-238 dominates the mixture in depleted uranium, U-238 and U-234 are essentially equal in natural uranium, and U-234 dominates the mixture for enriched uranium.

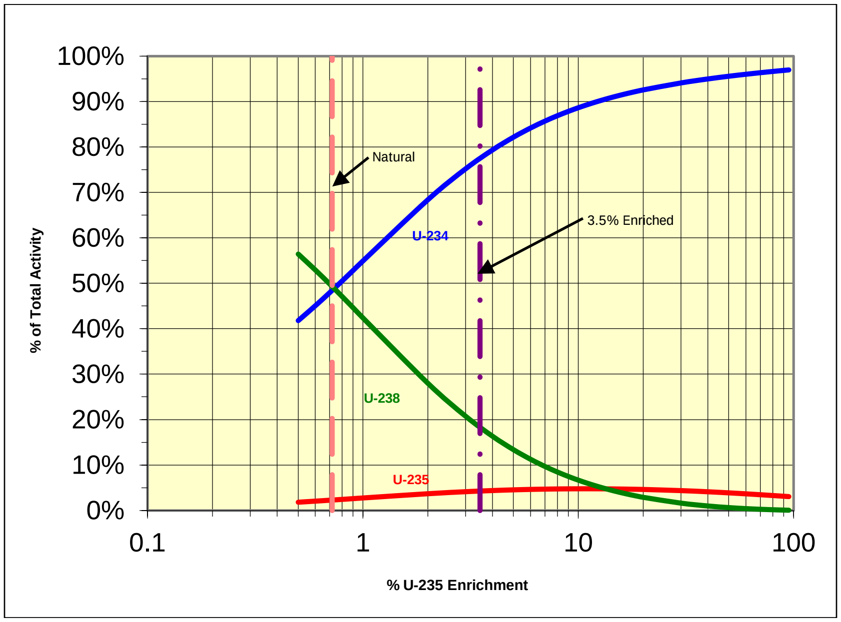

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Figure 3-1 Isotopic Contributions to Total Uranium Activity vs. Percent Enrichment

An enrichment of 3.5% was selected to represent the uranium series at the Site. Figure 3-2 portrays the relative isotopic abundance of the uranium isotopes for 3.5% enriched uranium as calculated from equation 3-1.

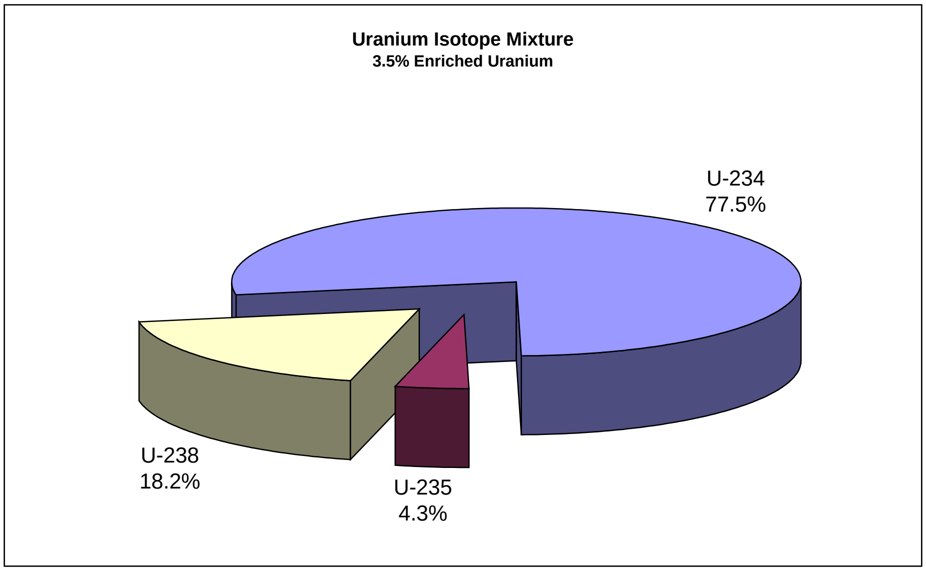

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Figure 3-2 Uranium Isotope Activity Fractions for 3.5% Enriched Uranium

Each of the three uranium isotopes decays by alpha emission. The uranium decay chain progeny that are in secular equilibrium with the 3 primary uranium isotopes includes thorium 234 (beta decay) and protactinium 234m (beta decay) from uranium 238 and thorium 231 (beta decay) from uranium 235. The radionuclide data for these uranium isotopes is presented in Table 3-1. The uranium series radionuclides are the primary constituent of any residual radioactivity within the Site.

Table 3-1 Properties of Uranium Series Radionuclides

Nuclide			Energies (MeV) and Abundances of Major Radiations		
Primary Uranium Isotopes	**Decay Products**	**Half-Life**	**Alpha**	**Beta***	**Gamma**
U-234		245,700 y	4.72 (28%) 4.77 (72%)		0.053 (0.12%) 0.121 (0.03%)
U-235		703,800,000 y	4.21 (5.7%) 4.32 (4.4%) 4.37 (17%) 4.40 (55%) 4.56 (4.2%) 4.60 (5.0%)		0.144 (11%) 0.163 (5.1%) 0.186 (57%) 0.205 (5.0%)
	Th-231	25.5 h		0.077 (100%)	0.026 (14%) 0.084 (6.6%) 0.090 (1.0%)
U-238		4,468,000,000 y	4.15 (21%) 4.20 (79%)		0.050 (0.06%) 0.114 (0.01%)
	Th-234	24.1 d		0.045 (100%)	0.063 (4.8%) 0.092 (2.8%) 0.093 (2.8%)
	Pa-234m	1.17 m		0.813 (100%)	0.766 (0.3%) 1.001 (0.8%)
* (average energy of total beta emitted spectrum)					

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The DCGL for total uranium (regardless of enrichment) is 557 picocuries per gram (pCi/g) for soils and sediment and 20,148 dpm/100 centimeters $(cm)^2$ gross alpha plus beta activity for building surfaces. Additional information can be found in the Derivation of the Site-Specific Soil DCGLs (MACTEC, 2003) and the Development of Building DCGLs (MACTEC, 2008).

3.1.2 Reactor Byproduct Series Radionuclide Profile

Radionuclides produced in the operation of a nuclear reactor are classified as byproduct materials, as they are the "byproduct" of a nuclear reaction. There are two subcategories of isotopes collectively classed as byproduct materials. They are described by their production mechanisms: 1) fission products, and 2) activation products. The nuclear fuel services work performed by CE and later by Westinghouse at the Site involved the repair, maintenance, and testing of reactor plant components. Since nuclear fuel itself is clad, or jacketed, to prevent a significant release of fission products, the principal radionuclides associated with plant components handled at the Site are activation products. For purposes of radiological survey of the buildings in the Complex, it is unimportant to further distinguish between these production mechanisms; however, it does serve to understand the byproduct material isotopic mixture encountered at the Site.

Isotopes found in byproduct materials are generally characterized by short half-lives and beta decay mechanisms. The shortest-lived isotopes rapidly decay away and are essentially gone before components can be removed from a reactor plant for service. After one year (no new byproduct radioactive materials have been introduced at the Site for more than one year), only a small number of the longest-lived radionuclides remain in potentially significant quantities.

The isotopes in the byproduct radionuclide mixture were characterized with data from three different sources at the Site (waste characterization data, 'hard-to-detect' investigation, and Connecticut Annual Low-Level Radioactive Waste Reports) (MACTEC, 2003). The following radionuclides were evaluated as part of the byproduct profile:

Ag-110m	cobalt 60 (Co-60)	Pu-239
Am-241	Cs-134	Pu-240
C-14	Cs-137	Pu-241
Cm-243	Fe-55	Sb-125
Cm-244	H-3	Sr-90
Cm-245	Mn-54	Zn-65
Cm-246	Ni-63	
Co-57	Pu-238	

Cobalt 60 clearly dominates the relative contribution to total activity in each of the three data sources evaluated (MACTEC, 2003). This dominance is amplified by the fact that cobalt 60 is by far the most potent dose producer among the byproduct nuclides present. To gauge the sensitivity of the potential future dose to a receptor with respect to variability in the isotopic ratios that might be reasonably expected in byproduct material, a sensitivity analysis was performed. The sensitivity analysis utilized RESRAD (residual radioactivity computer code) to assess the relative dose of each byproduct radionuclide detected at the site. The results of this analysis were that only cobalt 60 produces greater than 10% of the total dose by isotope. Thus, the byproduct source term can be effectively simplified to consider only cobalt 60.

Since the DCGL was derived for cobalt 60, it is the only byproduct radionuclide that will be considered in the design of the FSS. Cobalt 60 decays by beta emission and has no progeny. The radionuclide data for cobalt 60 is presented in Table 3-2.

Table 3-2 Properties of Byproduct Series Radionuclides

Nuclide	Half-Life	Energies (MeV) and Abundances of Major Radiations		
		Alpha	Beta*	Gamma
Co-60	5.2714 y		0.096 (100%)	1.173 (100%) 1.333 (100%)
* (average energy of total beta emitted spectrum)				

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The DCGLs for reactor byproduct series radionuclides is 5.0 pCi/g of Co-60 for soils and sediment and 6,980 dpm/100 cm^2 gross beta activity for building surfaces. Additional information can be found in the Derivation of the Site-Specific Soil DCGLs (MACTEC, 2003) and the Development of Building DCGLs (MACTEC, 2008).

3.1.3 Thorium Series Radionuclide Profile

As mentioned previously, thorium was identified during characterization surveys of the Burning Grounds as a result of licensed incineration activities occurring from approximately 1956 to 1961. This area is the former zirconium and magnesium thorium burning grounds. Zirconium tailings and turnings generated during fuel element assembly processes were transported in drums and burned at this location. By burning the tailings, the zirconium was stabilized and could then either remain in place or be transported off-site. After 1964, zirconium scrap was reportedly no longer burned on-site but was sent directly off-site to be reprocessed. The magnesium and thorium burning area was co-located with the zirconium burning ground. CE was licensed under the AEC to burn the magnesium and thorium wastes during the late 1950s. During this time, CE also accepted thorium wastes from off-site sources for burning. It is evident from waste and soil area characterization results that a majority of the source term released during these processes involved natural thorium (almost 100% Th-232) with small amounts of Ra-226 (present as an impurity in the thorium separation, zirconium tailings and turnings processes). Table 3-3 shows a list of Th-232 and associated daughters from the thorium decay series.

Table 3-3 Properties of Thorium Series Radionuclides

Nuclide			Energies (MeV) and Abundances of Major Radiations		
Primary Thorium Isotope	**Decay Products**	**Half-Life**	**Alpha**	**Beta***	**Gamma**
Th-232		14,056,000,000 y	3.95 (23%) 4.01 (77%)		0.059 (0.19%)
	Ra-228	5.753 y		0.010 (100%)	
	Ac-228	6.13 h		0.375 (97%)	0.209 (3.6%) 0.328 (3.2%) 0.338 (11%) 0.463 (4.4%) 0.794 (4.6%) 0.911 (28%) 0.965 (5.2%) 0.969 (17%) 1.588 (3.5%)
	Th-228	1.9132 y	5.34 (27%) 5.42 (73%)		0.084 (1.2%)
	Ra-224	3.62 d	5.45 (5%) 5.69 (95%)		0.241 (4.0%)
	Rn-220	56 s	6.29 (100)		
	Po-216	0.145 s	6.78 (100%)		
	Pb-212	10.643 h		0.099 (100%)	0.238 (45%) 0.300 (3.4%)
	Bi-212	60.55 m	6.05 (25%) 6.09 (10%)	0.717 (64%)	0.727 (12%) 0.785 (2%) 1.620 (3%)
	Tl-208	3.053 m		0.559 (99%)	0.510 (22%) 0.583 (84%) 0.860 (12%) 2.615 (100%)
	Po-212	0.3 µs	8.78 (100%)		
* (average energy of total beta emitted spectrum)					

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The DCGL for Th-232 is 4.0 pCi/g for soils and sediment. Additional information can be found in the *Derivation of the Site-Specific Soil DCGL Addendum* (MACTEC, 2010b).

3.1.4 Radium Series Radionuclide Profile

Ra-226, which is a radionuclide present in the uranium decay series and is considered to be an impurity in the thorium oxide extraction process, it is likely present in areas impacted by AEC licensed thorium incineration activities. The radionuclide data from Ra-226 is presented in Table 3-4.

Table 3-4 Properties of Radium Series Radionuclides

Nuclide			Energies (MeV) and Abundances of Major Radiations		
Parent Isotope	**Decay Products**	**Half-Life**	**Alpha**	**Beta***	**Gamma**
Ra-226		1,600 y	4.60 (5.6%) 4.78(94.4%)		0.186 (3.3%)
	Rn-222	3.8 d	5.49 (99.9%)		
	Po-218	3.11 m	6.00 (100%)		
	Pb-214	26.8 m		0.219 (100%)	241.9 (7.5%) 295.1 (19.2%) 351.9 (37.1%)
	Bi-214	19.9 m		0.632 (100%)	609.3 (46.3%) 768.4 (5.0%) 934.0 (3.2%) 1120.3 (15.1%) 1238.1 (5.9%) 1377.7 (4.1%) 1764.5 (15.8) 2204.2 (5.0%)
	Po-214	163.7 μs	7.69 (100%)		
	Pb-210	22.3 y		0.007 (100%)	46.5 (4.1%)
	Bi-210	5.0 d		0.389 (100%)	
	Po-210	138.4 d	5.30 (100%)		
* (average energy of total beta emitted spectrum)					

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The DCGL for Ra-226 is 4.5 pCi/g for soils and sediment. Additional information can be found in the *Derivation of the Site-Specific Soil DCGL Addendum* (MACTEC, 2010b).

3.2 BACKGROUND REFERENCE AREA

Since the residual radioactivity that may be present at the Site includes uranium, which is also a naturally-occurring radionuclide, a background reference area was initially established for the land areas. The initial background reference area was located in the northeast corner of the Site (Figure 3-3) and is non-impacted. This area was selected since it has very similar properties to the rest of the Site soils and historically no operations have been performed in this area. This area is undeveloped and has not been used for agricultural or other applications. In addition, the background reference area is separated from the rest of the Site by Huckleberry Road, which acts as a physical barrier as well.

Surface soil samples were collected from 37 locations within the initial reference area. The locations were randomly selected from a grid established across the reference area using 75-foot spacing. These samples were sent to a radiochemistry laboratory for uranium analysis by alpha spectroscopy and byproduct radionuclide analysis by gamma spectroscopy.

The results were that no cobalt 60 was detected in any of the samples and the total uranium had an average concentration of 1 pCi/g with a range of 0.6 pCi/g to 3 pCi/g. Cobalt 60 was not expected to be present in the background since it is not naturally occurring. The uranium concentration matches reported mean levels of naturally occurring uranium in soils of 1 pCi/g to 2 pCi/g. Since the background levels of uranium are less than 1% of the DCGL, background subtraction will not have any significant impact. Therefore background concentrations of radionuclides in soils will not be considered as part of the FSS process except as noted below. This will simplify data manipulation, statistical evaluations, and statistical tests for most of the FSS units.

However, it should be noted that certain impacted areas such as the Burning Grounds contain thorium (Th-232) and radium (Ra-226) produced from AEC licensed incineration activities. As mentioned previously, Th-232 and Ra-226 are naturally-occurring radionuclides. If the site has physical, chemical, geological, radiological, or biological variability that is not represented by the single reference background area described above, selecting more than one reference area may be necessary. Since any difference between the survey unit and the reference area is attributed to residual radioactivity, the choice of reference area may materially affect the decision on whether or not to release a survey unit to which it is compared. To minimize systematic biases in the comparison, the same sampling procedure, measurement techniques, and type of instrumentation should be used at both the survey unit and the reference area. It may be difficult to find a reference area from non-impacted areas for comparison to a particular survey unit since background may vary greatly due to different construction activities that have occurred at the site. Variations in background of a factor of five or more can occur in the space of a few hectares. To address these concerns, one or more of the following actions may be necessary:

- Reviewing and reassessing the selection of reference areas.
- Selecting different reference areas to represent individual survey units.
- Selecting survey units and their boundaries with respect to different areas of potential or actual background variability.
- More detailed scoping or characterization surveys may be needed to better understand background variability.
- If available, use published studies of radionuclide distributions.

The above concerns and actions were considered in the selection of four reference background areas shown in Figure 3-4. It should be noted that the initial reference background area described in the previous version (Figure 3-3) was redesigned and for this version designated as Reference Area 4.

To design each reference area survey plan, it was necessary to consider the likely maximum number of data points expected for the WRS test. Using the methodology described later in this document, a sample size of 26 was calculated for each reference area survey unit. Each reference area survey unit sample plan was designed using the criteria specified elsewhere in this document. The locations of the reference areas as shown on the overview map (Figure 3-4) roughly correlate to the north, south, east and west corners of the site boundaries, were deemed to be unaffected or non-impacted by past site process operations, and were expected to provide physical, chemical, geological, radiological, or biological variability not completely captured by selection of a single reference area.

From the sample data evaluated, it was difficult to conclude that significant background variability existed between any single reference area relative to another, or from the entire reference area sample population. Therefore, since no significant background variation exists between individual reference areas, it is assumed that the selection of any particular reference area to use in the WRS test will not materially affect the decision on whether or not to release a survey unit to which it is compared.

A summary of all reference sample data is shown in the table below. It should be noted that all reference area soil samples were analyzed on the 25% high purity germanium detector (HPGe) using an approved protocol. Also, since two split samples and two laboratory instrumentation replicate counts were performed on each reference area sample batch, these results were also included in the data summary results below. A grand total of 120 samples results from the background reference area population were used to generate the summary table.

Table 3-5 Statistical Summary of Background Reference Area Soil Data

Nuclide	Mean	Median	Standard Deviation	Min	Max
Ra-226	5.86E-01	5.69E-01	1.07E-01	3.22E-01	8.55E-01
U-235	8.25E-02	7.91E-02	3.54E-02	-5.78E-02	2.32E-01
Th-232	6.90E-01	6.79E-01	1.20E-01	4.70E-01	1.04E+00
Note: Above values listed are in pCi/g					

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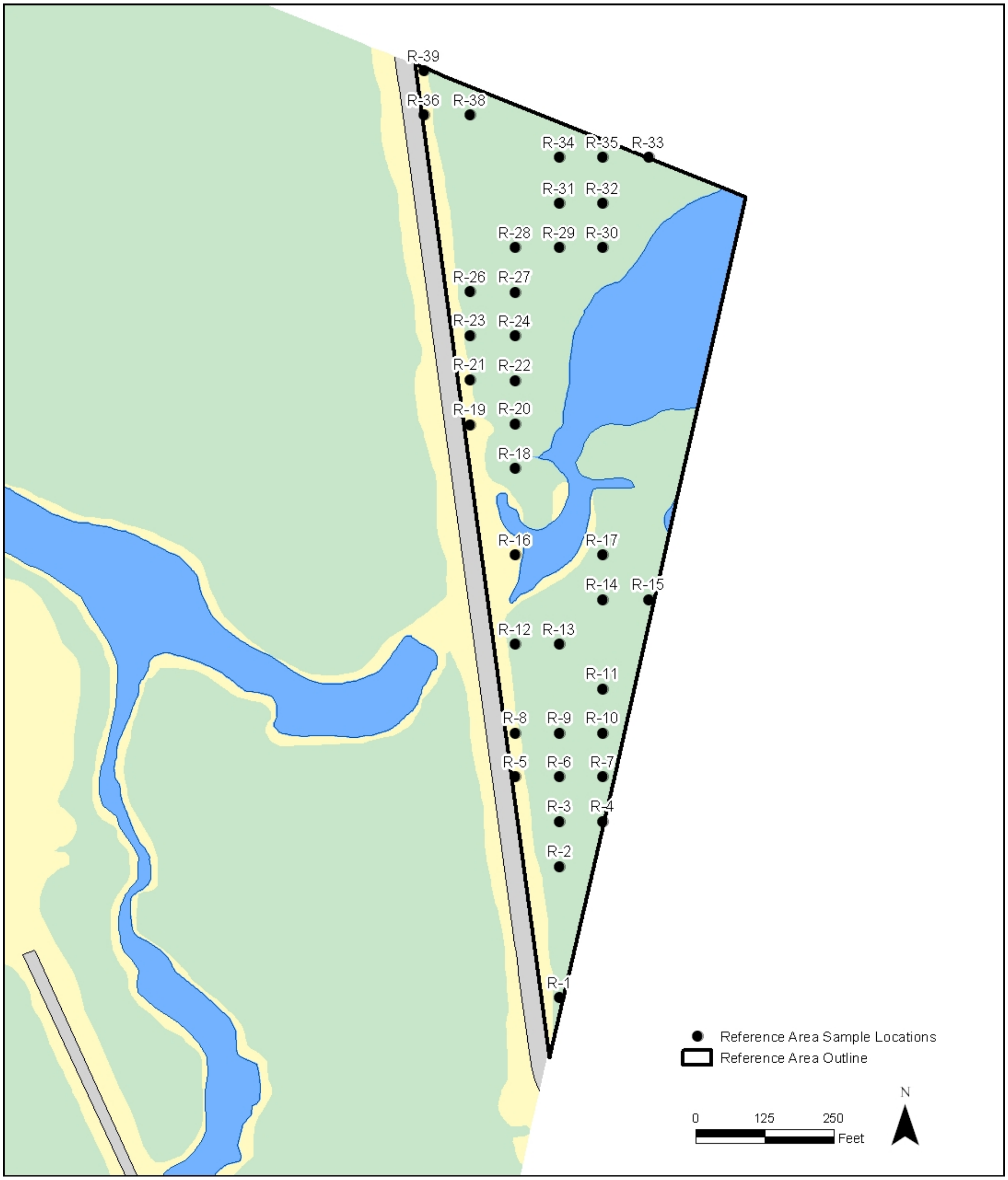

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Figure 3-3 Initial Background Reference Area

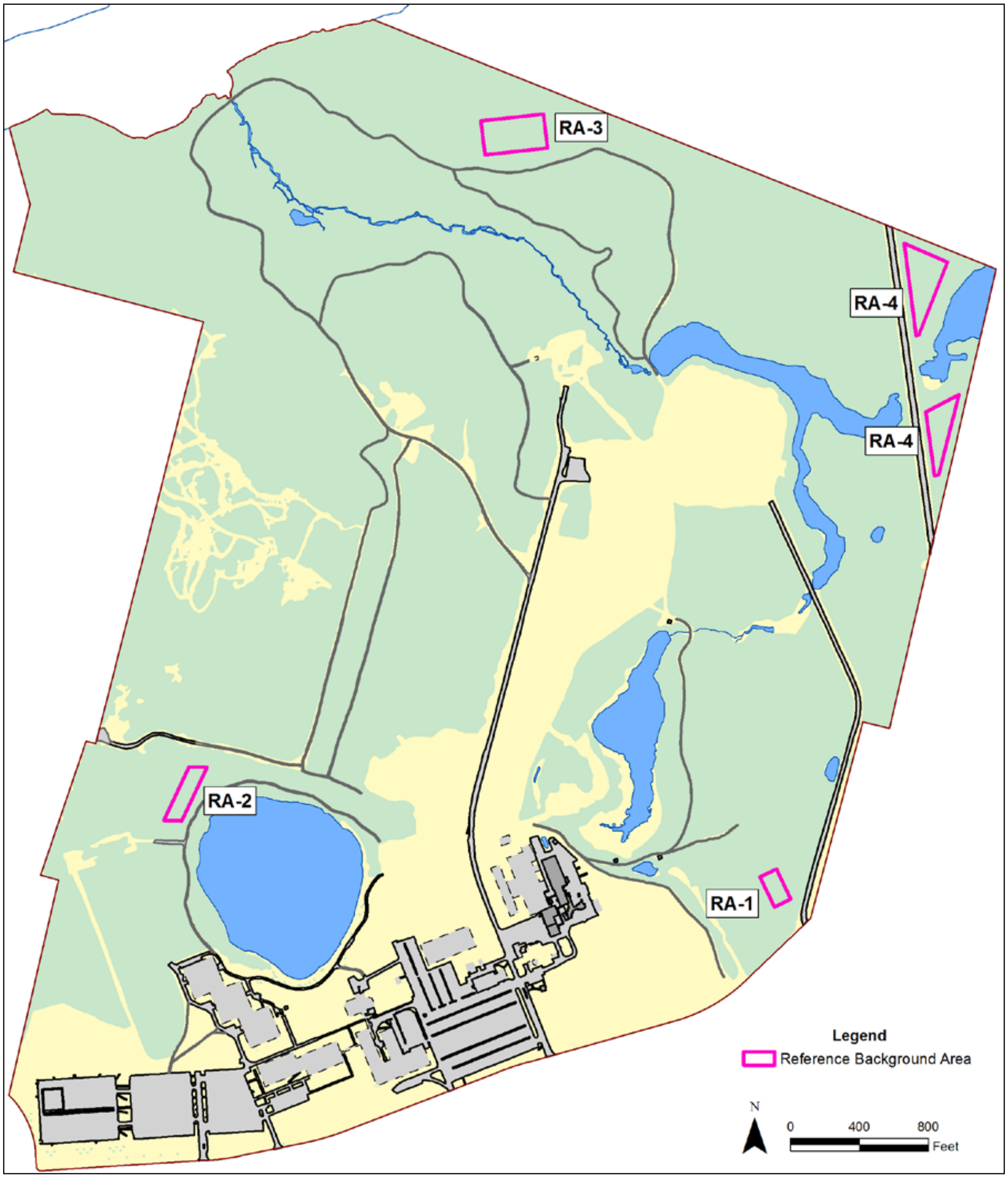

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Figure 3-4 Overview of Background Reference Areas

3.3 CHARACTERIZATION DATA

Various investigations and characterizations of the Site soils have occurred over time. Some of these were part of Site operations (CE, ABB), some part of FUSRAP (ORISE, SAIC, ENSR Corporation), some part of the RCRA VCA program (MACTEC), and others part of decommissioning activities (British Nuclear Fuel [BNFL], MACTEC). Specific details and references for these investigations have been presented within the HSA (Harding ESE, 2002) and DP (MACTEC, 2010a). A large amount of this characterization data has been input to the geographic information system (GIS) database for the Site. A query of the database produced more than 3,000 results of soil concentration data (not including any sediment or pipe sludge data) for Co-60 and total uranium across the entire site (including FUSRAP areas). The summary statistics of these two source terms is presented in Table 3-6.

Table 3-6 Statistical Summary of Site Soil Data

Source Term	$DCGL_W$	Mean	Standard Deviation	Median	Maximum	95^{th} Percentile
Co-60 (pCi/g)	5.0	0.13	0.64	0.02	12.7	0.34
Uranium (pCi/g)	557	162	2280	3.5	110,236	313

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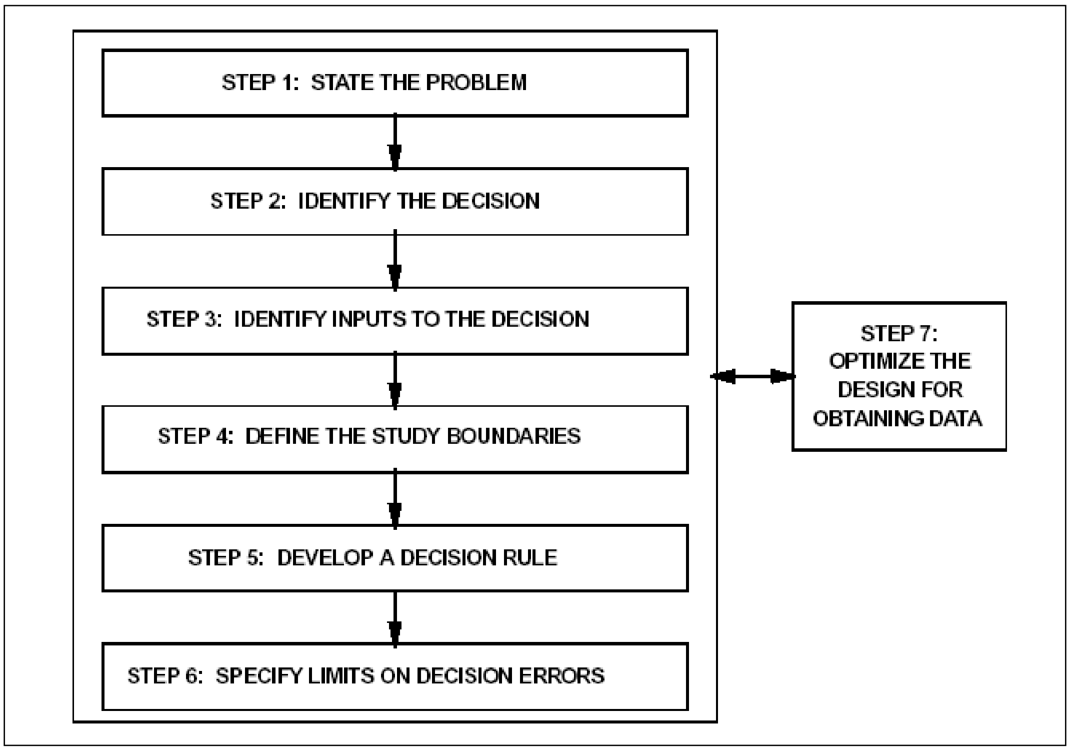
The large difference between the mean and the median along with the elevated maximum values indicate that there are a small percentage of high concentration samples that are skewing the mean. Although the mean values are below the DCGLs, it is not appropriate to make release decisions based upon this pooled data since the high concentration areas are being diluted by the low concentration areas. The vast majority of the elevated samples are located within FUSRAP areas. As survey units are designed, the characterization data specific to that location will be used as the basis for classification and survey design for statistical tests.

4.0 DATA QUALITY OBJECTIVES

This section identifies appropriate DQOs in planning and designing the FSSP. The DQO process is a series of planning steps based on the scientific method that are designed to ensure that the type, quantity, and quality of data used in the decision making are appropriate for the intended application. As indicated in Figure 4-1, the DQO process provides systematic procedures for defining the criteria that the survey design should satisfy, including what type of measurements to perform, when and where to perform measurements, the level of decision errors for the survey and how many measurements to perform. The DQO process consists of seven steps and the output of each step influences the decisions that follow with the intent that the final result is a more effective, efficient and defensible survey design.



(NRC, 2000)

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Figure 4-1 DQO Process

The process of identifying the applicable DQOs ensures that the FSS requirements, results, and data evaluation are of sufficient quality, quantity, and robustness to support the decision on whether cleanup criteria have been met using statistical tests. The DQO process is a flexible approach in planning and conducting FSS and for assessing whether survey results support the conclusion that the release criteria have been met. The DQO process is an iterative process that continually reviews and integrates new information in the design of survey units and decision-making. As such, this section of the plan will summarize the inputs to the DQO process. More specific details will be provided in other sections of this plan.

4.1 STEP 1: STATE THE PROBLEM

ABB seeks to terminate the NRC radioactive materials license for the CE Windsor Site in accordance with the DP. There is potential for residual radioactivity from past nuclear fuel manufacturing, research and development, and nuclear power reactor support operations to be present in soils and sediment and within structures (Building 3 High Bay)of the Site that may prevent the Site from being released from radioactive controls (unrestricted release). Release criteria are based upon NRC regulations (10 CFR 20.1401) and guidance, and State of Connecticut Department of Environmental Protection (CTDEP) requirements.

4.2 STEP 2: IDENTIFY THE DECISION

The identification of related decision statements and alternative actions is needed in order to demonstrate compliance. The decision that the FSS will attempt to resolve can be stated: "Determine whether residual radioactivity remaining in soils or on building structures after remediation in each survey unit is less than the approved release criteria."

4.3 STEP 3: IDENTIFY INPUTS TO THE DECISION

Inputs to the decision include the type, quality, and quantity of data that will be sufficient to make decisions. These inputs include identifying survey units, classifying survey units, and identifying appropriate measurement techniques. These inputs provide the framework for measuring the residual radioactivity in each survey unit. Inputs required to make decisions involve developing estimates of the average (median) residual radioactivity concentration or surface activity, average residual radioactivity concentration or surface activity in locally elevated areas, and maximum residual radioactivity concentrations and surface activities.

4.4 STEP 4: DEFINE THE STUDY BOUNDARIES

The definition of the site physical, temporal, and spatial boundaries for all media, including reference areas, will be covered by the decision process. Survey units are the smallest subsets of the site for which decisions will be made. The size of the survey unit and the measurement frequency within a survey unit are based upon the other steps of the DQO process.

4.5 STEP 5: DEVELOP A DECISION RULE

The purpose of this step is to define the parameter of interest (measurement), specify the action level (DCGL), and integrate previous DQO steps into a logical basis for choosing among alternative actions. The decision rule to consider is if the data collected in the FSS provides sufficient evidence that each survey unit has residual radioactivity below the applicable DCGLs, then conclude that the Site meets the criteria for release from radiological controls without restriction.

It should be noted that under normal circumstances, Scenario A is employed wherein the null hypothesis is that the concentration of residual contamination exceeds the DCGL. However, when the DCGL is low and the background variability is large, it is recognized that it may be impossible to employ Scenario A in an effective manner. For instance, the relatively low soil DCGLs of Ra-226 and Th-232 (4.5 pCi/g and 4.0 pCi/g respectively) suggest that unnecessary remediation may be likely for some survey units because these contaminants are present in the background. Therefore, it may be necessary to employ

Scenario B, wherein the null hypothesis states that the mean concentrations of the contaminants in the survey unit are indistinguishable from those in the background. This hypothesis will be restated as: The difference in the median concentration in the survey unit and in the reference area is less than the LBGR. If it is necessary to employ Scenario B, the Kruskal-Wallis Test might be used to demonstrate that there are significant differences in the background concentrations of the contaminant(s) in the potential reference areas (NRC, 1998). This test may be bypassed under the assumption that there are significant differences between the reference areas. The WRS test is then performed. If the null hypothesis is not rejected, a second statistical test, the Quantile test, is then used to detect non-uniform concentrations of residual radioactivity that may be excess of the release criterion, but that might be missed by the WRS test.

4.6 STEP 6: SPECIFY LIMITS ON DECISION ERRORS

This step is to specify the limits for Type I and Type II decision errors in support of the null hypothesis. This will include using prospective power curves and evaluating the impacts on sample size for different amounts of decision error. Site measurement data used to estimate the actual site conditions and decisions based on the measurement data could be in error (known as decision error). Statistical sampling designs in accordance with MARSSIM attempt to control design error by defining the types of errors and incorporating them in the statistical sampling design process.

4.7 STEP 7: OPTIMIZE THE DESIGN FOR OBTAINING DATA

This step is optimization of the data collection process and updating the design of the survey plan while meeting all DQOs. It includes documenting the operational details and theoretical assumption of the selected design. The decisions that will be made based upon the data collected during the survey should be specified along with the alternative actions that may be adopted.

4.8 INTEGRATION OF THE DQO PROCESS

These DQOs will be detailed in this FSSP in the appropriate sections. DQOs that pertain to the design of survey units will be provided in Section 5. Survey methods and instrumentation and their associated DQOs will be presented in Section 6. Finally, quality control, data quality indicators (DQIs), and data assessment will be offered in Section 7.

5.0 FINAL STATUS SURVEY DESIGN

This FSS is designed for the CE Windsor site, for remaining areas that are impacted or potentially impacted and for the Building 3 High Bay, and will be performed to demonstrate that residual radioactivity in each survey unit satisfies the predetermined criteria for release for unrestricted use. The survey results provide data used to demonstrate that radiological parameters do not exceed the established DCGLs. The survey unit represents the fundamental element for compliance demonstration using statistical tests. There are numerous factors that influence the delineation of a survey unit and the design of the survey within the unit. This section of the plan will focus on the parameters and decisions that affect the delineation of survey units, the classification of survey units, and the number and location of measurement and sampling points within survey units.

The first part of the section will address the boundaries of FSS at the Site and provide the process to be followed for creation of the survey units. The second part of this section will provide the sampling design process to generate the appropriate number and locations of samples within the survey units to provide the required confidence in the statistical tests for decision making.

5.1 SURVEY UNIT DESIGN

Different areas of the site will not have the same potential for residual radioactivity and, accordingly, will not need the same level of survey coverage to achieve the required confidence that the release criteria have been satisfied. The FSS process will be more efficient if the survey is designed so areas with higher potential for contamination will receive a higher degree of survey effort. The first part of the survey unit design process is to establish the potential for residual radioactivity across the Site and then to classify the areas based upon the guidance in MARSSIM. In addition, there are several unique situations at this Site (e.g., underground pipelines) that will require additional decision processes. This portion of the FSS design section will provide the process to delineate impacted areas and to further subdivide them into various FSS units.

5.1.1 Impacted Areas

The first step in the process of designing survey units is to determine what areas are considered impacted by radiological operations/activities at the Site. Areas determined to be non-impacted will not require FSS since, by definition, there is no reasonable potential for the presence of residual radioactivity.

Based upon the HSA, characterization surveys performed at the Site, and areas that have been recently remediated, the remaining impacted or potentially-impacted areas have been identified and are listed in Table 5-1 and shown in Figure 5-1. The rest of the Site is considered non-impacted or previously remediated/released and therefore not subject to FSS. More details on each of these areas as well as the basis for determination of impacted or non-impacted are provided in the HSA (Harding ESE, 2002), the various site DPs, FUSRAP information, and recent FSS reports and acceptance letters for the previously remediated areas.

With the incorporation of the FUSRAP areas into the Site NRC license and DP, the entire Site is now under decommissioning jurisdiction of the NRC. In addition to any remaining buffer or generally impacted areas, Buildings 3 High Bay, underground utilities, Equipment Storage Yard, Small Pond, Woods Area, Drum Burial Pit, Burning Grounds, Clamshell Pile, Debris Piles and the Site Brook are now identified areas that will undergo FSS.

Table 5-1 Impacted Areas at the CE Windsor Site

Building 3 High Bay	Drum Burial Pit
Building 3 & 6 Complexes	Burning Grounds
Underground Utilities	Clamshell Pile
Buffer / General Areas	Debris Piles
Equipment Storage Yard	Site Brook
Small Pond	Former WWTP
Woods Area	Waste Lines

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The FSSP includes Building 3 High Bay structure surfaces and surface soils (up to 30 cm in depth) in the impacted or potentially impacted areas. The FSS will also be performed for sediments in surface water impacted areas such as Site Brook and Small Pond. Groundwater and surface water are not considered impacted as discussed in the DP (MACTEC, 2010a).

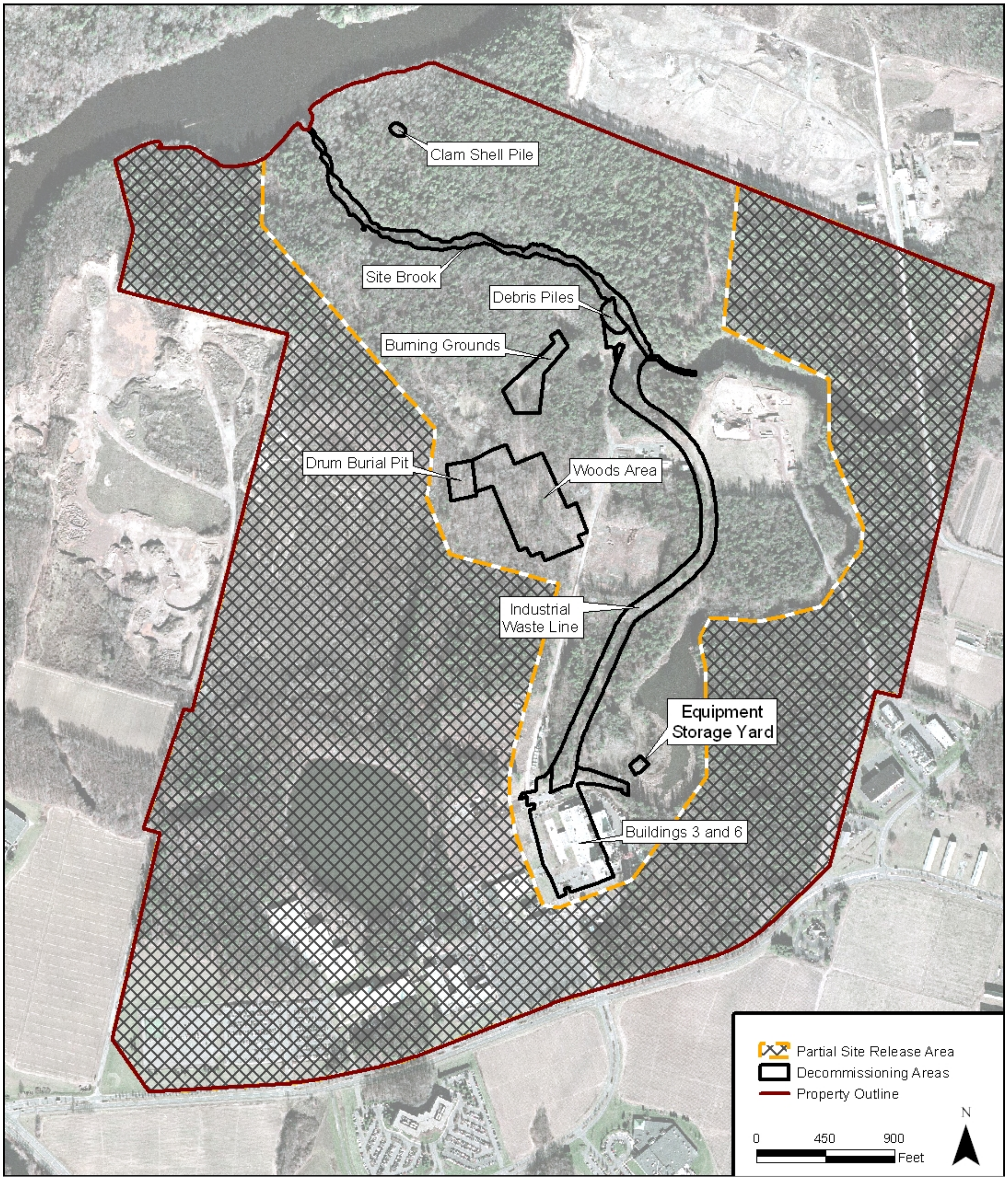

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Figure 5-1 Impacted Areas

5.1.2 Survey Unit Identification and Classification

The next step in the process of designing the FSS is to subdivide the impacted areas into survey units based upon the potential for the presence of residual radioactivity using the guidance in MARSSIM. This is a critical step since the survey unit is the basis for demonstrating compliance with the release criterion.

Survey unit demarcation is performed by evaluating each of the impacted areas with respect to historical uses, common uses, below ground impacts, or other characteristics such as naturally distinguishable portions. In some impacted areas, there will be former building locations or portions of buildings remaining underground that will also affect the manner in which survey units are created. The MARSSIM guidance for survey unit demarcation and classification is summarized in Table 5-2. Class 1 areas are the most likely to contain significant concentrations of residual radioactivity greater than the DCGLs, Class 2 areas are likely to have the potential for residual radioactivity less than the DCGLs, and Class 3 areas are the least likely to have residual radioactivity.

Table 5-2 Survey Unit Classification Guidelines

Classification	Basis	Suggested Area (Building)	Suggested Area (Land)
Class 1	Areas that are expected to exceed the $DCGL_w$.	Up to 100 m^2	Up to 2,000 m^2
Class 2	Areas that are not expected to exceed the $DCGL_w$.	100 to 1,000 m^2	2,000 to 10,000 m^2
Class 3	Areas that have a small fraction of the $DCGL_w$.	No limit	No limit

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Each survey unit will be identified by a unique alpha-numeric code. Survey unit identification throughout the course of the FSS shall be labeled by:

- a site designation (CE);
- a FSS designation (FSS);
- a two digit FSS area number (XX); and
- a two digit survey unit number (YY).

The format of the survey unit identification will be as follows:

CE-FSS-XX-YY

Survey units (YY) shall be numbered sequentially within a FSS area. The FSS areas and survey unit designatiors (XX) are described in Table 5-3 below.

Table 5-3 FSS Survey Unit Designation (XX)

03	Building 3 Complex
06	Building 6 Complex
23	Equipment Storage Yard
25	Small Pond
26	Buffer/General Areas
30	Building 3 High Bay
32	Underground Utilities
33	Site Brook
34	Debris Piles
35	Clamshell Pile
36	Drum Burial Pit
38	Woods Area
39	Burning Grounds
40	B2 Sanitary Waste Line
41	Sanitary Waste Line
42	Industrial Waste Lines
43	WWTP

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For example, the third survey unit in Building 6 Complex would be labeled CE-FSS-06-03.

Initial classification of the survey units is based upon historical information and characterization data. Survey data from routine operations or remediation may be used to change the initial classification of a survey unit. Once FSS of a survey unit begins, the basis for any reclassification will be documented. If reclassification of a survey unit will change the number or location of samples, then the survey may be terminated without completion and a new survey designed.

5.1.3 Survey Unit Design for Land Areas

The majority of the impacted area is open land. To facilitate survey design in open land areas, survey units will be comprised of land areas having a common history or other characteristics, or which are naturally distinguishable from other portions of the site. These survey units will be for surface soils (up to 30 cm) and any subsurface investigations will be designed as described in section 5.1.4 and 5.1.5. The MARSSIM guidance summarized in Table 5-2 will be used to create the survey units.

5.1.4 Survey Unit Design for Building Complex Land Areas

The building complex areas will be a mixture of land surface and subsurface impacted areas. In these areas, the buildings will be dismantled, including all below ground structural components to a depth of four feet. Underground utilities and pipelines will be removed along with paved surfaces in impacted areas. Again, the MARSSIM guidance summarized in Table 5-2 will be used to create the survey units.

The land area within the footprint of previous locations of the buildings will be considered different than the surrounding open land. In these areas, the footprint of the previously existing buildings will be extended out one meter and considered as a survey unit since this area has a higher likelihood of being impacted from past activities in and around the buildings.

In addition to the open land portions of these areas, the soil below any structures left in place (greater than four feet deep) will need to be evaluated as part of the FSS. The soils beneath subsurface building structures left in place below grade will have some samples collected to evaluate the potential for any radiological impact to exist beneath it. These samples will be considered part of the building footprint survey unit. The former location of the underground utilities and pipelines will be considered as part of the surface soil survey unit that overlays their position. Some sections will be part of a building footprint survey unit while other sections will be part of open land survey units. The specific details for the underground utilities and pipelines will be discussed in greater detail in section 5.1.5.

The combination of these different types of impacts creates an intricate overlay of survey units. An example of how survey units would be designed for a building complex area is shown in Figure 5-2. In addition, a cross-sectional view is shown in Figure 5-3.

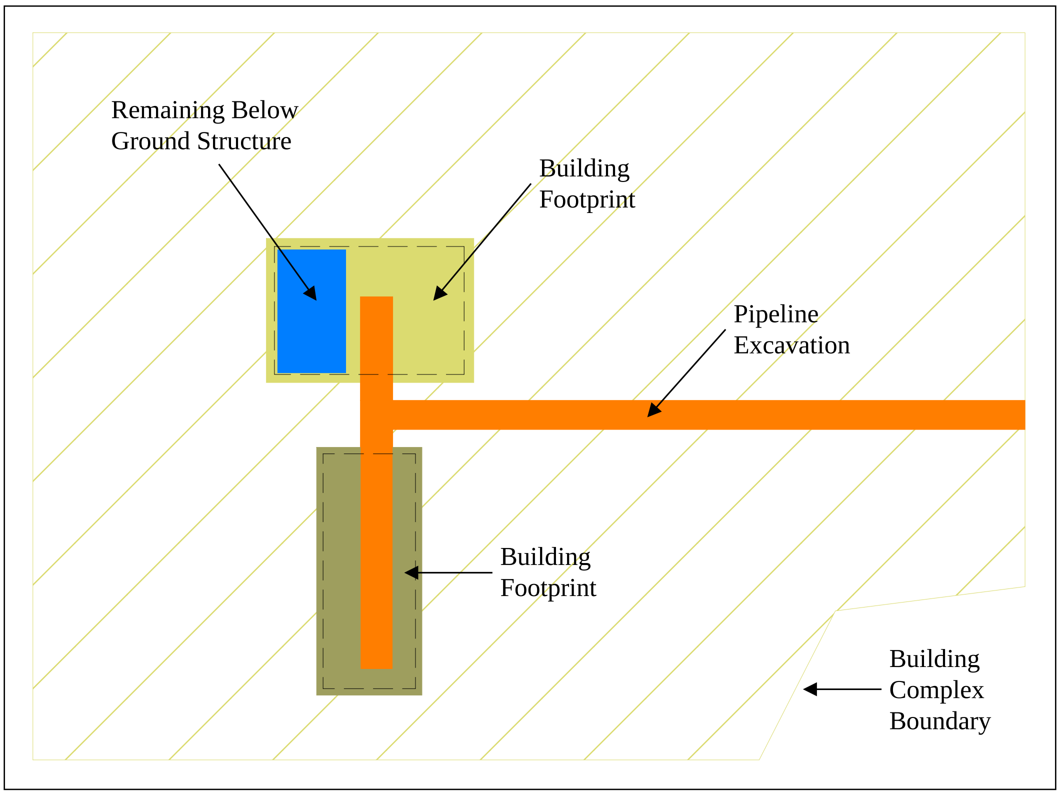

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Figure 5-2 Building Complex Survey Unit Overview

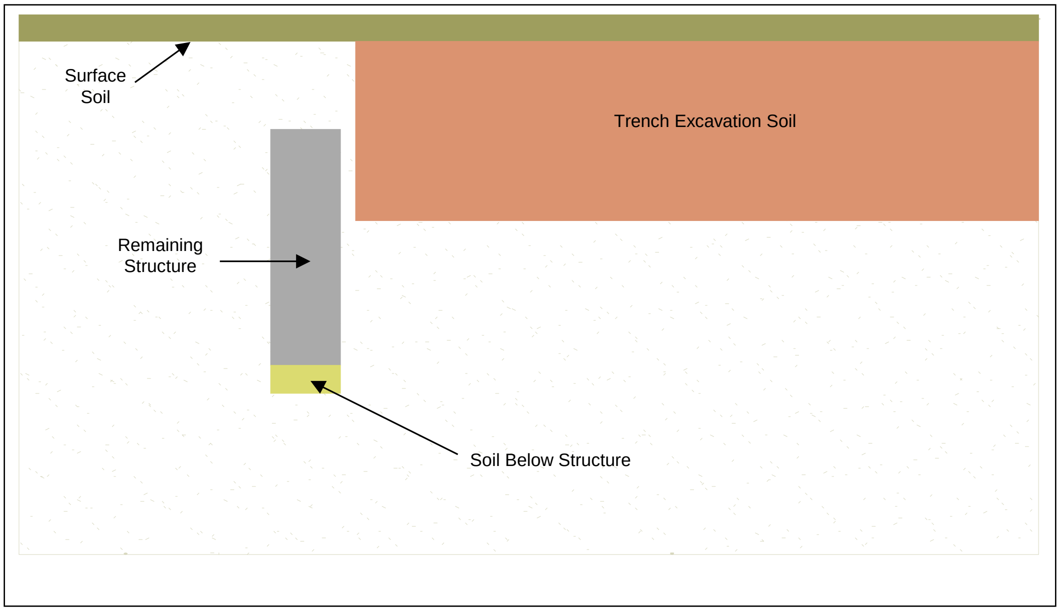

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Figure 5-3 Building Complex Survey Unit Cross-Section

5.1.5 Survey Unit Design for Pipeline Removal / Excavations

The presence of impacted underground (below grade) pipelines and utilities will pose some unique conditions for FSS. Underground utilities that are not necessary to support Building 3 High Bay continued use will be removed from the Building Complexes:

- Potable water lines and piping
- High temperature and chill water lines
- Fire protection system water piping and hydrants
- Communications
- Electrical distribution lines
- Natural gas lines
- Storm drains and associated piping
- Sanitary sewer piping and associated manholes
- Industrial waste piping and associated manholes
- Radioactive waste piping and associated manholes

The most likely utilities to have residual radioactivity are the radioactive waste piping and the industrial waste piping. The only other underground utilities that might have residual radioactivity are the sanitary sewer and storm drains. The rest of the underground utilities could have a small potential for residual radioactivity in the surrounding soils, but it would not have originated from the utility. The following processes will be utilized to characterize and remediate during the excavation and trenching associated with the removal of underground systems. In these situations, the characterization and remedial action support surveys will occur in quick succession and necessitates a standardized approach for these operations. The removal of utilities will require excavation and trenching as shown in Figure 5-4.

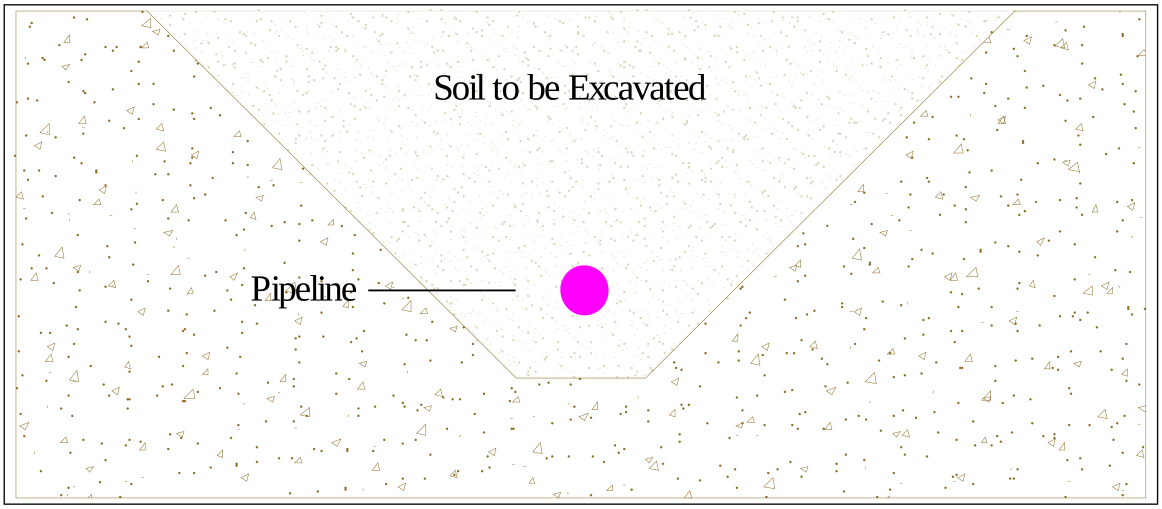

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Figure 5-4 Cross Section View of Below Grade Pipeline

During the excavation process of underground utilities, the soil that is removed will be screened (sodium iodide detector [NaI]) in order to provide worker protection and for characterization. The soils excavated from the trench will be divided into two categories based upon the screening results. Soils that exceed the DCGLs will be placed into piles or containers for disposal as low-level radioactive waste while the remaining soils will be placed into storage piles along the trench for possible use as back fill once the underground utility has been removed. Additional characterization or remediation surveys may be necessary depending on the potential for residual radioactivity within the underground utility

Given the probability and nature of any areas of elevated residual radioactivity originating from the underground utilities, this process of screening and surveying during excavation and remediation will provide effective coverage. Since there is no indication of widespread contamination in the subsurface soils in the vicinity of underground utilities, it is more likely that there are discrete areas with elevated concentrations of residual radioactivity arising from small leaks in the pipe. These localized areas will be excavated and removed for disposal as low-level radioactive waste during the excavation and removal of the underground utilities. The biased soil samples collected as part of this process will serve as documentation that the subsurface soils meet the release criteria. The surface soil in these areas (after remediation and back filling) will be considered as part of a FSS survey unit as described in Section 5.1.4. The process for each of these categories of underground utilities is detailed next.

5.1.5.1 Radioactive Waste Lines and the Industrial Waste Lines

The soils surrounding these utilities do not contain residual radioactivity that is significantly greater than background soil concentrations based on currently available sampling data. There is the potential for leakage of high concentration radioactive material from the piping into the surrounding soils at junctures near manholes and piping joints. Therefore the soils from the ground surface to the top of the piping (‘top layer’) is not expected to have any elevated soil concentrations as shown in Figure 5-5. If there are indications of elevated residual radioactivity of at least 50% of the DCGLs, then a soil sample will be collected and be analyzed with the on-site gamma spectroscopy system.

Once the pipeline has been exposed, it will be characterized by scanning with a NaI and collecting two samples of materials within the pipeline (if available) from approximately every twelve feet of pipe. Pipe joints and areas where leakage appears to have occurred due to soil discoloration, odor, etc. will be marked for further evaluation once the piping is removed. Analysis by on-site gamma spectroscopy of the samples collected from the pipeline will be used to determine the concentration, enrichment, and U-235 gram weight in sections of the pipeline. If the U-235 concentration is less than 1,080 pCi/g, then the material may be moved without concern for criticality. If the U-235 concentration is greater than 1,080 pCi/g, then the material can only be moved by limiting the U-235 weight to less than 350 grams as a criticality control. In addition, special nuclear materials accountability controls will be implemented as required by NRC license conditions to ensure that license limits are not exceeded.

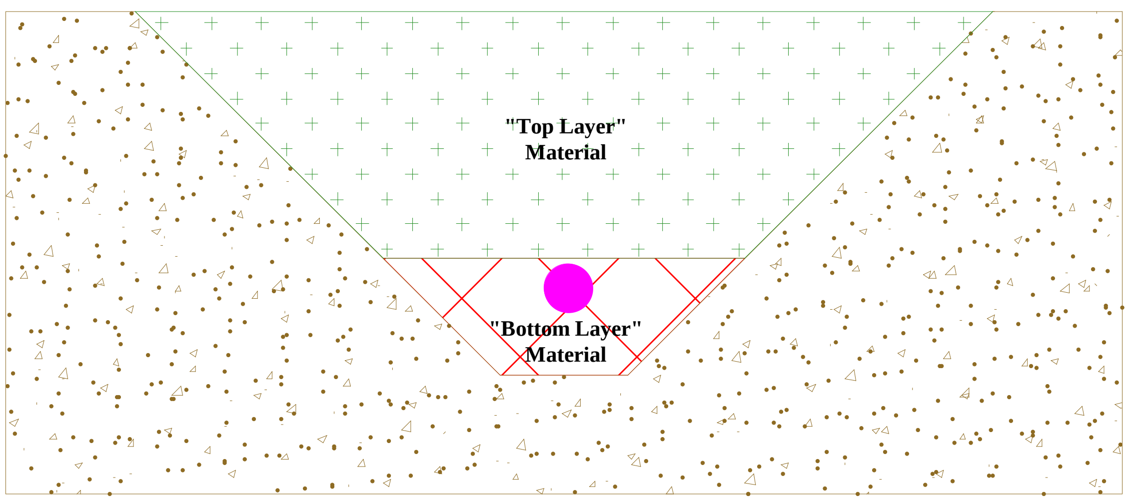

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Figure 5-5 Trench Excavation Soil Layers

Unless there is a potential cost-benefit to decontaminate radioactive waste lines, it is planned that they will be processed as contaminated such that the pipe will be disposed of as low-level radioactive waste along with its contents. For industrial waste lines, if the sampling indicates that there is no residual contamination, or they can be decontaminated, the pipes will be opened to the extent necessary to perform a surface scan using nominally 100 cm^2 scintillation or gas proportional detectors and a wipe survey of every square meter of surface area. The bottom interior section of the pipe will have surveys, and if contamination is found then the top interior section of the pipe will also have surveys. Exterior pipe surfaces will be surveyed if elevated residual radioactivity was detected during screening of the trench soil. The scanning levels will be compared with the equipment and materials surface release levels (Regulatory Guide 1.86[1], or other applicable standard) and if less than those values, the piping will be disposed of as normal construction debris waste. If the scanning shows levels greater than equipment and mateerials surface release levels, a determination will be made as to the cost-effectiveness of decontaminating pipe versus disposing of the pipe as low-level radioactive waste. None of the radioactive waste line or industrial waste line pipeline will be left in place.

After pipe removal, scanning of the bottom of the trench will be performed to identify areas of residual radioactivity where greater than 50% of the DCGLs exist. These areas will have further sampling and analysis performed to determine appropriate actions. Scanning will also evaluate locations where the pipe joint and leakage markers were placed during the previous phase of the excavation process. In addition, at least one soil sample for every 100 linear feet of the trench will be collected at a location where the highest elevated count rate can be found in the trench and in the associated excavated soil pile.

A layer of material (e.g., plastic fencing or white rock) will be used to mark the bottom of the trench once all surveys are complete. The trench may be back filled from the clean soil piles once soil exceeding the DCGLs has been removed.

5.1.5.2 Sanitary Sewers and Storm Drains

The soils surrounding these utilities do not contain residual radioactivity that is significantly greater than background soil concentrations. These lines would not contain sufficient concentrations of radioactive material to cause the surrounding soils to significantly exceed background concentrations, even with a leak. Therefore none of the soils from the ground surface to the surrounding soils of the piping are expected to have any elevated soil concentrations. Screening of the soils will be performed to detect any concentration anomalies and to provide worker protection. If elevated residual radioactivity of at least half the DCGL are identified, then additional evaluations will be made to ensure soil DCGLs are not exceeded.

Once the pipeline has been exposed, pipe joints and areas where leakage appears to have occurred due to soil discoloration, odor, etc. will be marked for further evaluation once the piping is removed. After the piping is removed from the trench, the pipes will be opened to the extent necessary to perform a surface scan using nominally 100 cm^2 scintillation or gas proportional detectors and a wipe survey of 10 percent of the interior surface area. In addition, biased surveys will be performed at locations where internal material accumulation is present. Exterior pipe surfaces will be surveyed if elevated residual radioactivity was detected during screening of the trench soil. The scanning levels will be compared with the equipment and materials surface release levels (Regulatory Guide 1.86, or other applicable standards) and if less than those values, the piping will be disposed of as normal construction debris waste. If the scanning shows levels greater than building surface release levels, a determination will be made as to the cost-effectiveness of decontaminating pipe versus disposing of the pipe as low-level radioactive waste.

[1] 5000 dpm/100 cm^2 for total surface activity and 1000 dpm/100 cm^2 for removable activity

After pipe removal, scanning of the bottom of the trench will be performed to identify areas of residual radioactivity where greater than half the DCGL exist. These areas will have additional evaluations performed to ensure soil DCGLs are not exceeded. Scanning will also be performed at locations where the pipe joint and leakage markers were placed during the previous phase of the excavation process. In addition, at least 1 soil sample for every 100 linear feet of the trench will be collected at a location where the highest elevated count rate can be found in the trench and in the associated excavated soil pile.

A layer of material (e.g., plastic fencing or white rock) will be used to mark the bottom of the trench once all surveys are complete. The trench may be back filled from the clean soil pile once soils exceeding the DCGLs have been removed.

5.1.5.3 All Other Underground Utilities

The rest of the underground utilities (e.g., potable water) are considered non-impacted and limited screening will be performed during excavation to detect any concentration anomalies and to provide worker protection. If there are any indications of elevated residual radioactivity of at least half the DCGL, then additional evaluations will be made to ensure soil DCGLs are not exceeded. No scans or surveys of the piping or the trenches will be performed unless elevated soil concentrations are detected during excavation of the surrounding soils.

5.1.5.4 Other Excavations

At other excavations, the same basic process will be followed. Soils above impacted areas will be screened during excavation. If there are any indications of elevated residual radioactivity, then additional evaluations will be performed. Once the impacted area has been uncovered, it will be characterized by scanning with a NaI and collecting representative samples. Areas where leakage appears to have occurred due to soil discoloration, odor, etc. will be marked for further evaluation. On-site analysis results of the samples collected from the impacted materials will be used to determine the concentration, enrichment, and U-235 gram weight.

After the impacted materials are removed, scanning of the bottom of the excavation will be performed to identify areas of residual radioactivity where greater than 50% of the DCGLs exist. These areas will have further sampling and analysis to determine appropriate actions.

A layer of material (e.g., plastic fencing or white rock) will be used to mark the bottom of the excavation once all surveys are complete. The excavation may be back filled from the clean soil pile once soils exceeding the DCGLs have been removed.

5.1.6 Survey Unit Design for Building Structures

The Building 3 High Bay may be left in place for continued use after decommissioning and license termination. The building's surfaces will be divided up into appropriate survey units following MARSSIM guidance, summarized in Table 5-2.

5.2 SAMPLING DESIGN

FSS sampling design entails determining the number of samples in order to achieve the desired level of statistical confidence and power, locating the samples such that they are representative of the survey unit, and evaluating the survey units for the presence of areas of elevated residual radioactivity. Since Co-60 is not present in background and the background concentrations of uranium in soil are such a small fraction of the $DCGL_W$ as to be considered insignificant, the background reference area will not normally be used, except as noted below. The FSS results will be directly compared to the DCGL and the one-sample Sign test will be used to determine compliance.

As previously noted, certain impacted areas such as the Burning Grounds also have additional contaminants of concern (i.e. Ra-226 and Th-232) which could be present in quantities in excess of natural background concentrations. Since Ra-226 and Th-232 could be present at a significant fraction of their respective $DCGL_w$, selection of a suitable background reference area may be necessary. The DQO process will be used to prepare an FSS plan to determine whether media specific backgrounds, ambient area background or no background will be applied to a survey area or unit. The approach used for a specific survey unit will be based on the survey unit classification and the DCGLs. If applied, media specific backgrounds will be determined via measurements made in one or more reference areas (for land areas) and on various materials (for building surfaces) selected to represent the baseline radiological conditions for the site. The determination of media specific background will be controlled with a documented survey plan, which will include the DQO process. The collected data may be used as the reference area data set when using the Wilcoxon Rank Sum test, or, for survey units with multiple materials, background data may be subtracted from survey unit measurements (using paired observations) if the Sign Test is applied. Reference and media specific backgrounds should not be confused with instrument background which is subtracted from a static measurement result to obtain an activity concentration which is used to demonstrate compliance with the release criterion.

Whether or not they are radionuclide-specific, background measurements should account for both spatial variability over the area being assessed and the precision of the instrument or method being used to make the measurements. Thus, the same materials or areas may require more than one background assessment to provide the requisite background information for the various survey instruments or methods expected to be used for FSS. The result of these background assessments will provide the basis for determining the mean and its associated standard deviation.

This section will provide the methodology for designing a MARSSIM based FSS sample design. Visual Sample Plan (VSP) software (or equivalent) will be used as appropriate instead of hand calculating the various parameters. VSP is a software tool for selecting the right number and location of samples so that the results of statistical tests performed on the data collected via the sampling plan have the required confidence for decision making.

5.2.1 Sample Size

The MARSSIM methodology for evaluating whether a survey unit meets its applicable release criterion using fixed measurements plus scans is based on using non-parametric statistical tests for data assessment. Specifically, the methods of MARSSIM are based on two non-parametric tests: the Wilcoxon Rank Sum (WRS) test and the Sign test. The number of samples or measurements in a survey unit will be determined by the acceptable decision error rates, the estimated variability of the residual radioactivity concentration in the survey unit, the $DCGL_W$, and the lower bound of the gray region (LBGR). Selection of the required minimum number of data points depends on which statistical test is going to be used to evaluate the data, and thus depends on what type of measurements are to be made (gross measurement, net measurement or radionuclide specific) and if the radionuclide(s) of interest appear(s) in background. The following process will be used to calculate the sample size.

A decision error is the probability of making an error in the decision on a survey unit by failing a survey unit that should pass or by passing a survey unit that should fail. When using the statistical tests, larger decision errors may be unavoidable when encountering difficult or adverse measuring conditions. This is particularly true when trying to measure residual radioactivity concentrations close to the variability in the concentration of those materials in natural background.

The α decision error is the probability of passing a survey unit whose actual mean or median concentration exceeds the release criterion. A decision error α value of 0.05 or less is acceptable under most conditions. The β decision error is the probability of failing a survey unit whose actual mean or median concentration is equal to LBGR. Any value of β is acceptable to the regulator provided the null hypothesis is in the form that assumes the concentration of residual radioactivity exceeds the $DCGL_W$.

The estimated variability of the residual radioactivity (σ) is determined from survey results (characterization) or estimated based upon professional judgment. The standard deviation of the characterization data should provide a reasonable estimate of σ. The LBGR should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit based upon the characterization data.

The number of samples needed will depend on a ratio involving the concentration to be measured relative to the variability in the concentration. The ratio to be used is called the relative shift, Δ/σ. The relative shift is defined as shown in equation 5-1.

$$\Delta / \sigma = \frac{DCGL_W - LBGR}{\sigma} \qquad (5\text{-}1)$$

Where:
$DCGL_W$ = derived concentration guideline levels
LBGR = concentration at the lower bound of the gray region
σ = an estimate of the standard deviation of the concentration of residual radioactivity in the survey unit or the standard deviation established for the corresponding reference area if the survey data are to be evaluated against a reference area(s)

The relative shift should have a value between one and three. The following adjustments may be made to adjust Δ/σ. If Δ/σ is less than one, LBGR should be decreased until Δ/σ is greater than or equal to one, or until LBGR equals zero. If Δ/σ exceeds 3, the LBGR should be increased until Δ/σ is less than or equal to 3. It should be noted that the standard deviations in the contaminant level(s) will likely be available from previous survey data (*e.g.,*scoping or characterization survey data for unremediated survey units or remedial action support surveys for remediated survey units). If they are not available, it may be necessary to consult MARSSIM (NRC, 2000) for guidance regarding the estimation of distributions.

The number of samples for the sign test can then be determined by equation 5-2.

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(SignP - 0.5\right)^2} \qquad (5\text{-}2)$$

Where:

N = the recommended minimum sample size

$Z_{1-\alpha}$ = the value of the standard normal distribution for which the proportion of the distribution to the left of $Z_{1-\alpha}$ is 1-α

$Z_{1-\beta}$ = the value of the standard normal distribution for which the proportion of the distribution to the left of $Z_{1-\beta}$ is 1-β

Sign P = the probability that a random measurement from the survey unit will be less than the $DCGL_w$ when the survey unit median concentration is equal to the LBGR.

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, anomalous results which are judged to be erroneous, and other errors, minimum sample size computation will be increased by twenty percent and rounded up to obtain sufficient data points to yield the desired power.

As an example of this process, consider a Class 1 land survey unit that has characterization data of Co-60 mean concentration of 0.006 pCi/g with a standard deviation of 0.064 pCi/g and a uranium mean concentration of 5.1 pCi/g with a standard deviation of 3.6 pCi/g, a Th-232 mean of 1.0 pCi/g with a standard deviation of 0.5 pCi/g, and a Ra-226 mean of 2.8 pCi/g with a standard deviation of 1.8 pCi/g. An α decision error of 0.05 and a β decision error of 0.10 will be used. Using the unity rule, equation 7-8 was used as an independent source term for each sample in the data set to ensure that the total dose due to the sum of four discrete source terms does not exceed the release criteria (unity). The values for calculating the sample size based on unity is presented in Table 5-4. The soil/sediment $DCGL_W$s used are 5 pCi/g for Co-60, 557 pCi/g for uranium, 4.5 pCi/g for Ra-226, and 4 pCi/g for Th-232.

The example results below indicate that using the data from the unity calculation requires 23 samples (19+4), including the additional 20%. Therefore it will be used as the sample size for the land area survey design. It should also be noted that for some survey units, sample size based on unity may not be appropriate if characterization and/or remediation surveys indicate only one source term is present, such as uranium.

Table 5-4 Example Sample Size Calculation

Parameter	Unity
α decision error	0.05
β decision error	0.10
$DCGL_W$	1
LBGR (mean)	0.88
Standard Deviation (σ)	0.42
Relative Shift (Δ/σ)	0.29
Adjusted Relative Shift (Δ/σ)	1
FSS Sample Size for Sign Test	19+4

Additional samples may be necessary in order to demonstrate that there are no small areas of elevated activity. This process is detailed in Section 5.2.4.

The methods described above, for land area sampling, are also used to identify the number (N) of samples (direct measurements) for all classifications of the Building 3 High Bay area.

It should be noted from the above example that a unity sample calculation will be performed for each survey unit for survey areas with multiple contaminants of concern. However, it is possible that certain areas were not impacted from the commercial nuclear power support and/or thorium incineration activities processes. These areas potentially include, but are not limited to the Clamshell Pile, which only contains uranium contamination resulting from the nuclear fuels manufacturing process. Therefore, for areas with only one contaminant of concern, a separate unity calculation need not be performed.

Alternatively, the number of required samples (N) for the sign test can be obtained directly from Table 5-5 of MARSSIM (NRC, 2000).

For some survey units which have been impacted from thorium incineration activities, the background levels of contaminants (i.e., Th-232 and Ra-226) present will require additional statistical tests on the data. When comparison of measurements from a reference area and the survey unit is required, the WRS test will be used. For the WRS test, the following number of samples can be determined by Equation 5-3.

$$N = \frac{1}{2} \times \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3\left(P_r - 0.5\right)^2} \tag{5-3}$$

Where:

N = the minimum number of measurements required for each survey area or reference area;

P_r = the probability that a random measurement from the survey unit exceeds a random measurement from the background reference area by less than the $DCGL_W$ when the survey unit median is equal to the LBGR above background.

The value of N computed for the WRS test applies for both the survey unit and the reference area (i.e., at least N measurements should be performed in both areas). As with the sign test, to ensure against lost or unusable data, the value of N will be increased by at least a factor of 1.2 when assigning the number of measurements to be made. Additionally, it should be noted that the number of survey area samples may not always equal the number of reference area measurements, since a given set of reference area measurements may be used for more than one survey unit.

Alternatively, the number of required samples (N/2) for the WRS test can be obtained directly from Table 5-3 of MARSSIM (NRC, 2000).

Sample size will be determined for each survey unit prior to collecting FSS sample or measurement data. Sample size calculations, using VSP, will be performed using the most current and available radiological survey or characterization data, as available.

5.2.2 Sample Locations

Sample locations within a survey unit may be distributed randomly or on a systematic grid. Random measurement patterns are planned for Class 3 survey units to provide independent results. Systematic triangular grids are planned for Class 2 and Class 1 survey units since there is an increased probability of small areas of elevated activity. The systematic grid will allow for limitations on the size of an unsampled area with the potential for elevated activity based upon the area in between measurement locations. The systematic grids will have a random starting point to provide an unbiased set of measurement locations.

Random sample locations (Class 3) will be determined by generating sets of random numbers by calculator, computer, or mathematical tables. Each set of random numbers is multiplied by the appropriate survey unit dimension to provide coordinates. This process is repeated until enough locations have been created to meet the required sample size. Another option is to utilize specialized computer software designed for such applications such as VSP. An example of random sample placement is shown in Figure 5-6.

Systematic grid sample locations use a random number generation process to identify the starting point of the grid. The sample size and area of the survey unit determine the spacing of the triangular grid, L, as shown in equation 5-4. Survey locations are spaced at intervals of L from the starting point and rows are spaced at a distance of 0.866*L from the first row. Another option is to utilize specialized computer software designed for such applications such as VSP. An example of the systematic grid sampling approach is shown in Figure 5-7.

$$L = \sqrt{\frac{A}{0.866n}} \tag{5-4}$$

Where:
L = grid spacing
A = area of the survey unit
n = number of survey locations

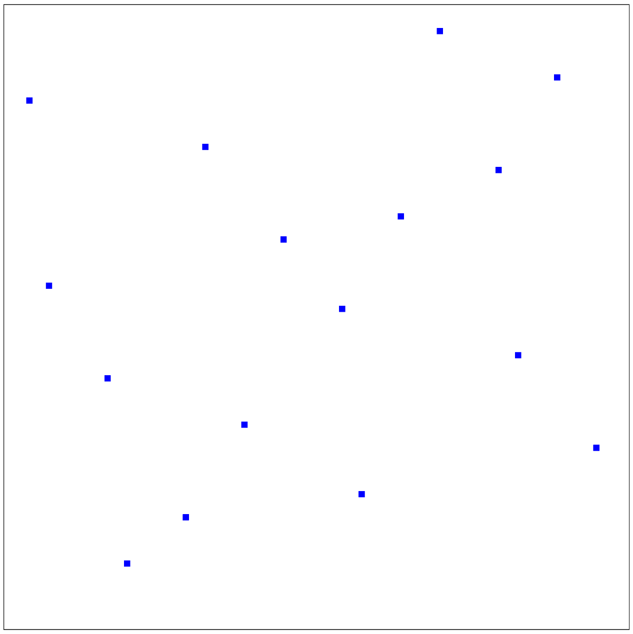

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Figure 5-6 Example Random Sampling Placement

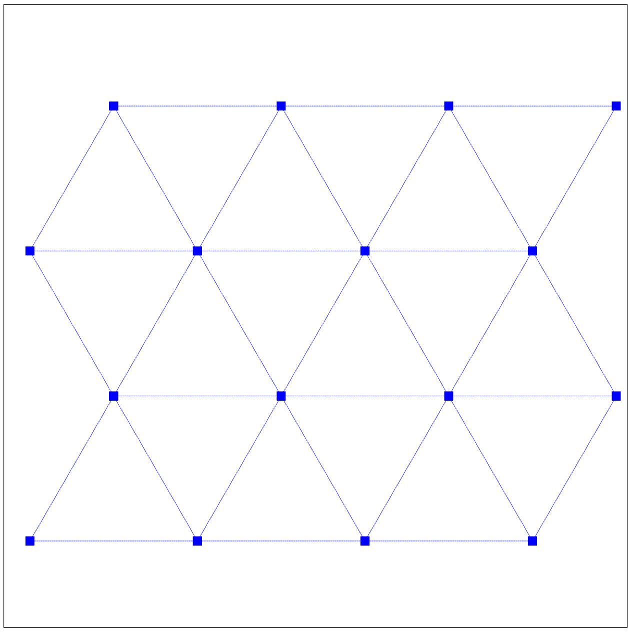

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Figure 5-7 Example Systematic Grid Sampling Placement

5.2.2.1 Reference coordinate system

A reference coordinate system will be established for each survey unit in order to provide a mechanism for referencing a measurement to a specific location so that the same survey point can be relocated. For land area surveys, sample locations will be marked in some visually discernable manner (pin flags, stakes, marking paint, etc.). For building structural surveys, sample locations will be marked using an appropriate method (e.g., paint, indelible marker, adhesive sticker).

A GIS has been created for the Site and the survey units and sample locations are planned to be integrated into the GIS for land area surveys. The Site GIS uses the Connecticut State Plane North American Datum (NAD) 83 (units of feet) established as its reference. Sample locations will be located using differential global positioning system (GPS) surveying. A Trimble GeoXH submeter GPS survey system (or equivalent) will be used for locating sample locations, locating pertinent site features, and navigating to previous sample locations

In general, land area survey unit reference coordinates will be based upon compass directions N (north), S (south), E (east), and W (west) and distance. For survey units that include subsurface samples, depth of the sample will be used to identify the vertical dimension. An example of the reference coordinate system is shown in Figure 5-8.

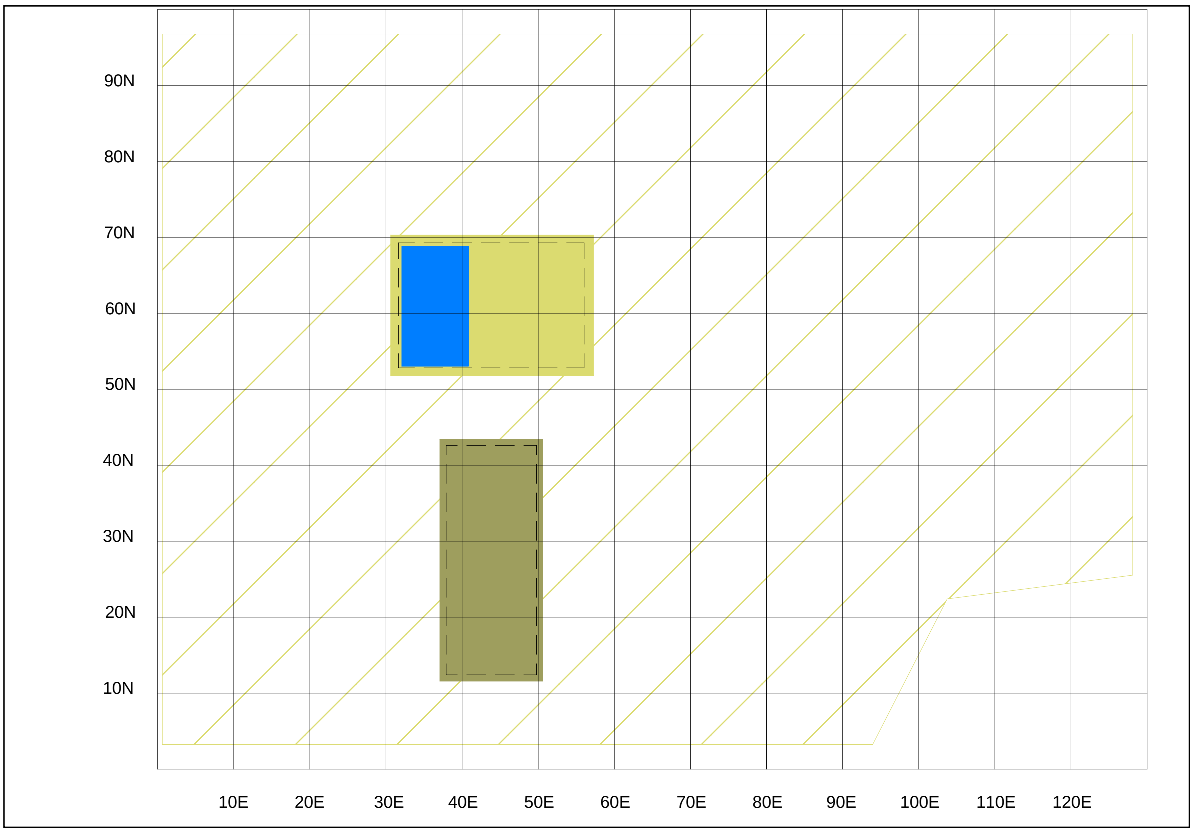

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Figure 5-8 Example Reference Coordinate System Land Area

A reference coordinate system will be established for each survey unit defined for building structures. Building structure survey unit reference coordinates will be placed on a two-dimensional exploded-view surface layout. An example of this reference coordinate system (excluding ceiling area) is shown in Figure 5-9.

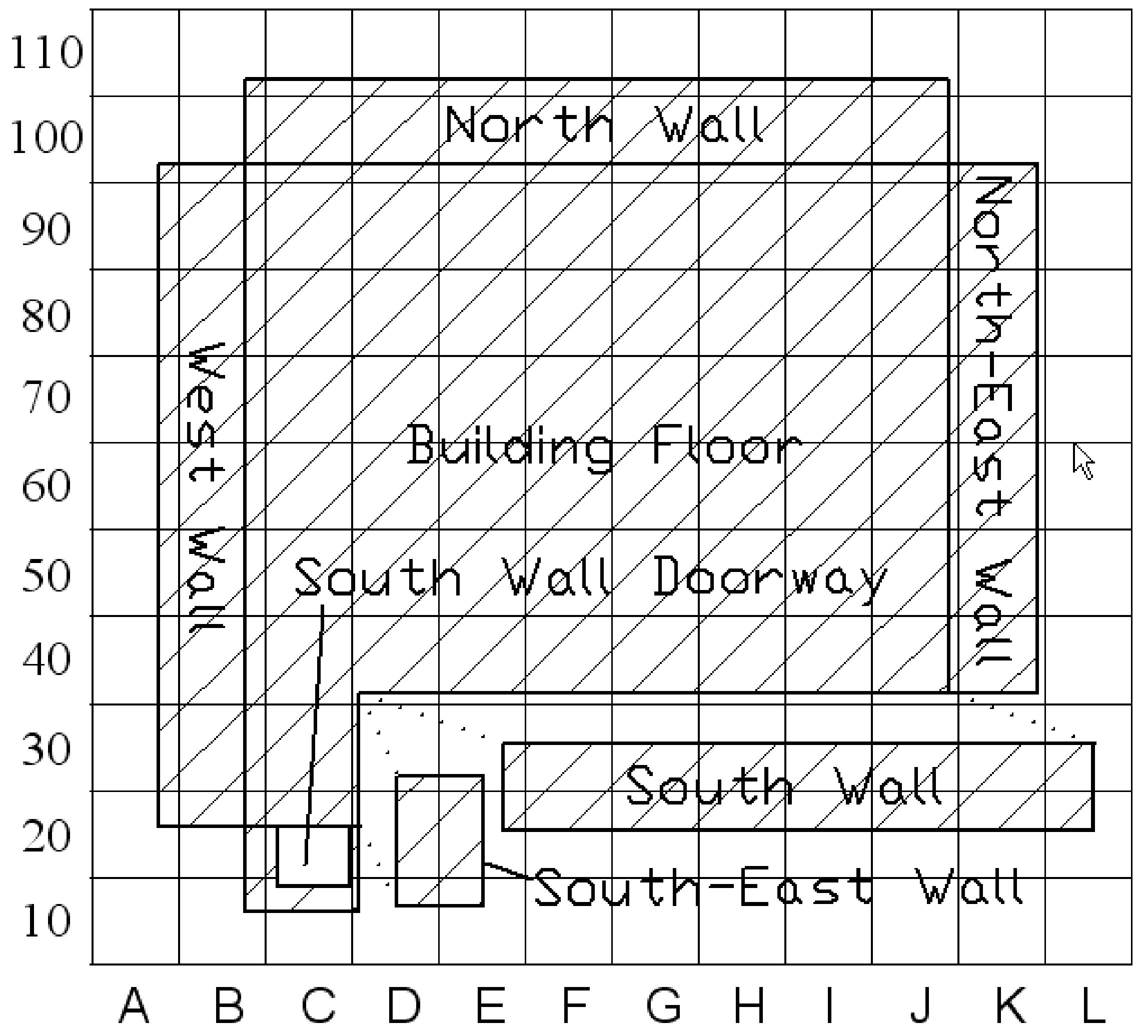

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Figure 5-9 Example Reference Coordinate System - Building

5.2.2.2 Relocating a Sample

If a sample location is placed in an inaccessible location or in a location which is deemed unsafe to access, the location will be moved to the nearest accessible location within the same survey unit while conforming to the overall spatial coverage theme. Generally, the nearest available location that compliments the goal of even spatial coverage would be the most defensible choice. Alternate measurement locations will be documented as a revised location selection indicating the reason for relocation.

5.2.3 Connecticut RSRs – Sampling

Soil samples collected as part of FSS may also be used for demonstrating compliance with the Connecticut RSRs. Additional soil samples may need to be collected in order to meet the RSR requirements. Soil samples collected that are not from the FSS design will be considered biased and may be used for statistical evaluations only if it can be shown that they are likely from the same population as samples placed using random distribution methods.

5.2.4 Investigation of Land Areas with Locally Elevated Concentrations

Additional measures may be necessary in order to be assured that small areas having significant concentrations of elevated residual radioactivity or elevated surface activity are not missed during FSS. One approach to investigate for small areas of elevated residual radioactivity is to perform scan surveys. Based upon the likelihood of finding elevated concentrations of residual radioactivity in surface soils, the scan survey coverage recommended is presented in Table 5-5 for land areas. In general, scan surveys will be performed along transects of the survey unit along with select areas based on professional judgment (e.g., specific areas known to have been involved with radioactive material activities). An example of a scan survey pattern is shown in Figure 5-10.

Table 5-5 Recommended Scan Survey Coverage – Land Areas

Survey Unit Classification	Scanning Coverage
Class 1	100 percent
Class 2	10 percent – 100 percent
Class 3	Judgmental (areas with greater potential for residual radioactivity)

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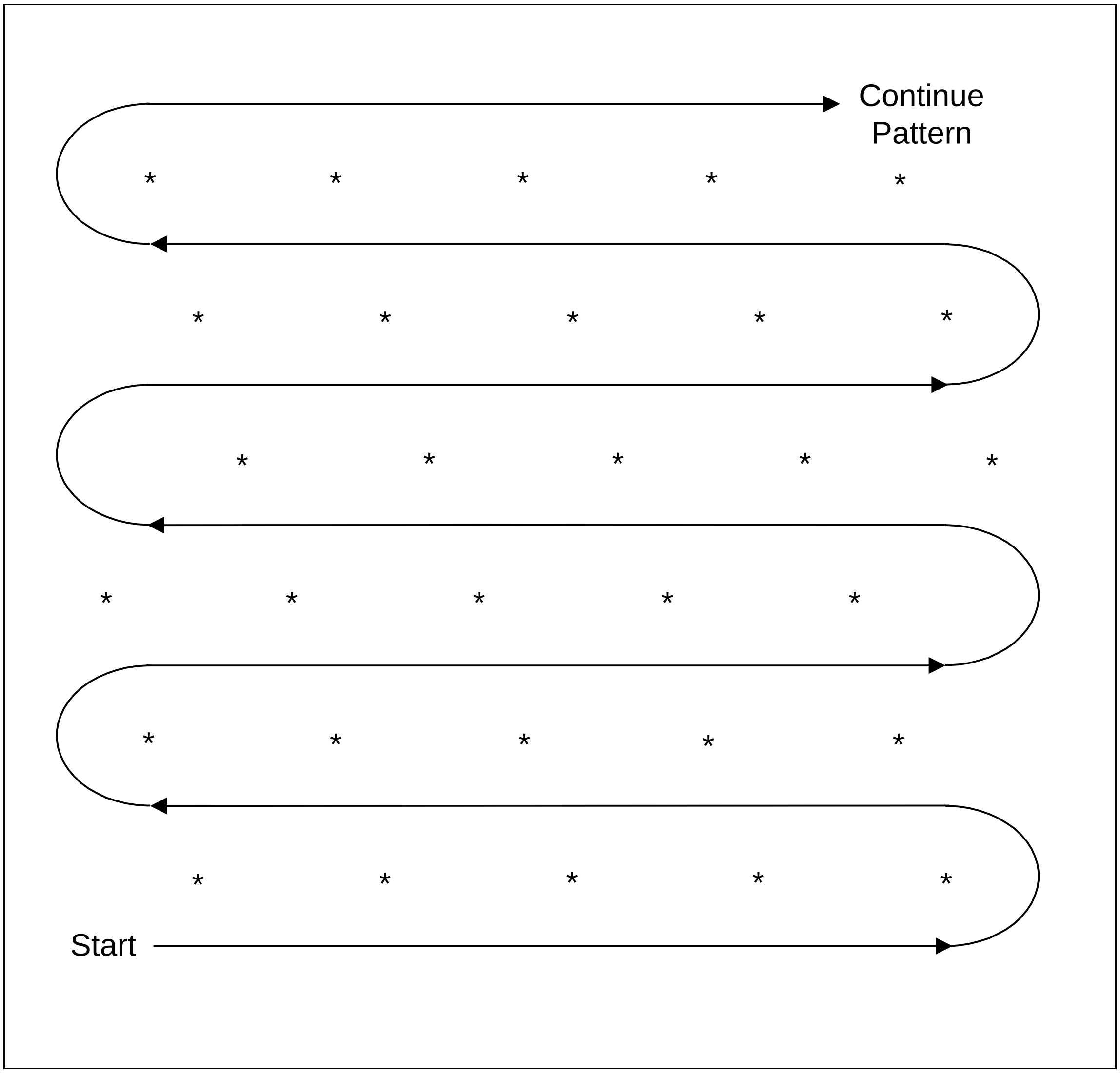

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Figure 5-10 Example Scan Survey Pattern

5.2.4.1 Land Area Scan Surveys

Since land area scan surveys are evaluating small areas, the $DCGL_W$ needs to be modified in order to have an equivalent measure. This can be achieved by using a correction factor that accounts for the difference in area and the resulting change in dose. The area factor is the magnitude by which the concentration within a small area of elevated activity can exceed the $DCGL_W$ and still maintain compliance with the release criterion. The area factors were derived by using the same RESRAD case used for derivation of the site-specific DCGLs. In RESRAD, the area of the contaminated area was sequentially reduced from 2,023,400 square meters (essentially infinite) to less than one square meter for uranium, Co-60, Ra-226 and Th-232. The area factors were calculated such that multiplying the $DCGL_W$ by the area factor will result in the concentration in the smaller area that will deliver the same calculated dose ($DCGL_{EMC}$). The calculation of the area factors is shown in Equation 5-5 and the calculation of the $DCGL_{EMC}$ is shown in Equation 5-6. The area factors calculated for the FSS are presented in Table 5-5.

$$A_m = \frac{D_a}{D_i} \qquad (5\text{-}5)$$

Where:
A_m = area factor
D_a = dose from $DCGL_W$
D_i = dose from small area i

$$DCGL_{EMC} = A_m * DCGL_W \qquad (5\text{-}6)$$

Where:
$DCGL_{EMC}$ = derived concentration guideline level for small areas of elevated activity
$DCGL_W$ = derived concentration guideline level for average concentrations
A_m = area factor

Table 5-6 Area Factors for $DCGL_{EMC}$ – Land Areas

Area (m^2)	0.25	1	2	5	10	100	500	1000
Uranium Area Dose	0.94	0.97	1.6	2.8	4.1	8.1	11.0	13.4
Uranium Area Factor	20.2	19.6	12	6.8	4.6	2.4	1.7	1.4
$DCGL_{EMC}$ Uranium pCi/g	**11,235**	**10,922**	**6,698**	**3,807**	**2,562**	**1,311**	**962**	**790**
Byproduct Area Dose	1.4	1.4	2.5	4.7	7.1	14.1	16.7	17.4
Byproduct Area Factor	13.4	13.4	7.6	4.1	2.7	1.4	1.1	1.1
$DCGL_{EMC}$ Co-60 pCi/g	**66.9**	**66.9**	**37.9**	**20.3**	**13.4**	**6.7**	**5.7**	**5.5**
Radium Area Dose	1.0	1.0	1.7	3.2	4.9	10.1	13.7	16.4
Radium Area Factor	19.4	19.4	11.0	5.9	3.9	1.9	1.4	1.2
$DCGL_{EMC}$ Ra-226 pCi/g	**87.6**	**87.1**	**49.5**	**26.6**	**17.5**	**8.5**	**6.3**	**5.2**
Thorium Area Dose	0.3	1.3	2.2	4.1	6.2	12.5	15.6	17.1
Thorium Area Factor	58.3	15.1	8.6	4.7	3.1	1.5	1.2	1.1
$DCGL_{EMC}$ Th-232 pCi/g	**233.1**	**60.3**	**34.6**	**18.6**	**12.3**	**6.1**	**4.9**	**4.4**

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5.2.4.2 Determining Data Points for Small Areas of Elevated Activity – Land Areas

The potential for localized areas of elevated residual radioactivity will vary across the Site. Given the uncertainty associated with gamma walkover scans of soils, some additional evaluations may be performed to determine if the sampling density is sufficient for the survey unit. The characterization/remediation data for Class 1 or Class 2 survey units will be evaluated for the potential of elevated residual radioactivity.

This process begins with a statistical evaluation of the data to determine the concentration that represents the 95th percentile of the data. That concentration is compared to the $DCGL_{EMC}$ values in Table 5-6. The corresponding area associated with the $DCGL_{EMC}$ value is then compared to the sample density in the survey unit. If the sample density is greater than the $DCGL_{EMC}$ area representing the 95th percentile of the data, then the number of samples needs to be increased. The number of samples and grid size will be adjusted such that the sample density will be at least equal to the $DCGL_{EMC}$ area representing the 95th percentile of the data.

The adjustments to the sample grid can be estimated by dividing the area of the survey unit by the $DCGL_{EMC}$ area representing the 95th percentile of the data. The grid spacing is calculated by using this value for n in Equation 5-4. However, the shape of the survey unit or inaccessible areas within the survey unit may allow a larger area to go unsampled. Another option is to use the VSP software to determine the sample design that will have at least a 95% probability of placing at least one sample in an area corresponding to the $DCGL_{EMC}$ area. VSP computes the probability of success in locating hot spots based on the assumed size, shape, and orientation of the hot spots, and on the specified grid spacing.

As an example, consider the following survey unit. Statistical sampling design determined that 17 samples (including an extra 20%) will be needed to satisfy the $DCGL_W$ statistical test and the area of the survey unit is 2,000 m^2. The systematic triangular grid has a spacing of 11.7 meters, and a sample density of one per 117.6 m^2. Evaluation of the characterization/remediation data finds that the 95th percentile concentration is 1,311 pCi/g total uranium. This concentration corresponds to a $DCGL_{EMC}$ area of 100 m^2 from Table 5-6. Since this area is less than the sample density for the statistical based design, the grid size needs to be reduced. The 'locate a hotspot' sampling goal within VSP software was used to determine the least amount of samples in order to achieve a 95% probability of detection. For a hot spot with a concentration equal to the 95th percentile and with an area of 100 m^2, the VSP software determined that 20 samples on a triangular grid would be required. This sample pattern provides a triangular grid spacing of 10.9 meters and a sample density of one per 100 m^2.

5.2.4.3 Land Survey Investigation Levels

Another aspect of FSS is evaluating the preliminary results to investigation levels. Investigation levels are used to indicate when additional investigations may be necessary. When an investigation level is exceeded, the original measurement may be confirmed by reanalysis or collecting additional samples. Depending on the results of the investigation, the survey unit may require reclassification, remediation, and/or resurvey. The investigation levels are presented in Table 5-7.

Table 5-7 Final Status Survey Investigation Levels – Land Surveys

Survey Unit Classification	Sample Measurement Investigation Level	Scanning Measurement Investigation Level
Class 1	> $DCGL_W$	> $DCGL_{EMC}$
Class 2	> $DCGL_W$	> $DCGL_W$
Class 3	> 80% $DCGL_W$	> $DCGL_W$

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Any additional samples collected as part of an investigation or as judgmental (biased) samples will be presented in the FSS report and may aid in the elevated measurement comparison. However, these samples may not be included as part of the statistical evaluation unless it is obvious that they will not impart any bias to the randomly collected samples.

If it is determined that additional samples need to be collected in order to define the area of the locally elevated area, then the following process is planned. If scan survey data provides an indication of the boundaries of the locally elevated area, then the samples will be distributed within that area. The number of samples will be dependent on the actual size of the locally elevated area. If scan survey data does not provide any clear indication of the boundaries, then samples will be placed around the location of elevated activity as an initial assessment. The results of these samples may trigger additional samples in order to define the extent of the locally elevated area.

5.2.5 Investigation of Building Surfaces with Locally Elevated Concentrations

Additional surface survey measurements may be necessary in order to be assured that small surface areas having elevated surface activity are not missed during FSS. One approach to investigate for small areas of elevated surface activity is to perform scan surveys. Based upon the likelihood of finding elevated residual activity on building surfaces, the scan survey coverage recommended is presented in Table 5-8. In general, scan surveys will be performed along parallel transects of the survey unit along with select areas based on professional judgment (e.g., specific areas known to have been involved with radioactive material activities).

Table 5-8 Recommended Scan Survey Coverage – Building Surfaces

Survey Unit Classification	Scanning Coverage
Class 1	100 percent
Class 2	10 percent – 100 percent (10 percent to 50 percent for upper walls and ceilings)
Class 3	Judgmental (areas with greater potential for residual radioactivity)

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5.2.5.1 Determining Scan MDC to Detect Small Areas of Elevated Activity - Building Surfaces

Given the uncertainty associated with scan surveys of building surfaces, some additional evaluations may be performed to determine if the sampling density is sufficient for the survey unit. By deriving elevated measurement concentration values for building surfaces, scan MDCs can be evaluated for assessment of small areas of elevated activity. As necessary, the number of direct measurements in a survey unit will be increased from the calculated number to account for the scans ability to detect small areas of elevated activity.

One method for determining values for the $DCGL_{EMC}$ is to modify the $DCGL_W$ using a correction factor that accounts for the difference in area and the resulting change in dose or risk (Equation 4-6). The area factor is the magnitude by which the concentration within the small area of elevated activity can exceed $DCGL_W$ while maintaining compliance with the release criterion.

TheMDC of the scan procedure, needed to detect an area of elevated activity at the limit determined by the area factor (see Table 5-9), is calculated as follows:

$$\textit{Required Scan MDC} = (DCGL_W) \times (\textit{Area Factor}) \qquad (5\text{-}7)$$

The actual MDC of the selected scanning technique is compared to the required scan MDC. If the actual scan MDC is less than the required scan MDC, no additional sampling points are necessary for assessment of small areas of elevated activity. In other words, the scanning technique exhibits adequate sensitivity to detect small areas of elevated activity.

The area factors were derived by using the same RESRAD-BUILD case used for derivation of the site-specific DCGLs. In RESRAD-BUILD, the area of the contaminated area was sequentially reduced from 5,000 square meters (essentially infinite) to one square meter for both uranium and Co-60. The area factors were calculated such that multiplying the $DCGL_W$ by the area factor will result in the concentration in the smaller area that will deliver the same calculated dose ($DCGL_{EMC}$).

Table 5-9 Area Factors for $DCGL_{EMC}$ – Building Surfaces

Surface Area (m^2)	Total U Area Factors	Total U Dose (mrem/y)	Total U $DCGL_{EMC}$ (dpm/100cm^2)	Co-60 Area Factors	Co-60 Dose (mrem/y)	Co-60 $DCGL_{EMC}$ (dpm/100cm^2)
1	6418.9	0.003	129,320,938	41.3	0.46	288,304
2	3380.8	0.006	68,112,095	23.2	0.82	161,929
3	2348.6	0.008	47,316,437	17.1	1.11	119,477
4	1826.9	0.010	36,806,728	13.9	1.37	96,803
5	1496.1	0.013	30,140,943	12.0	1.58	83,937
10	826.1	0.023	16,643,042	8.0	2.38	55,723
15	584.6	0.033	11,778,153	6.5	2.91	45,574
20	456.7	0.042	9,201,682	5.7	3.32	39,946
25	376.2	0.051	7,580,000	5.2	3.64	36,434
36	272.6	0.070	5,491,965	4.5	4.19	31,652
50	203.2	0.094	4,094,010	4.0	4.70	28,217
75	140.7	0.135	2,835,481	3.6	5.33	24,882

Surface Area (m^2)	Total U Area Factors	Total U Dose (mrem/y)	Total U $DCGL_{EMC}$ ($dpm/100cm^2$)	Co-60 Area Factors	Co-60 Dose (mrem/y)	Co-60 $DCGL_{EMC}$ ($dpm/100cm^2$)
100	107.3	0.177	2,162,655	3.3	5.79	22,905
125	87.2	0.218	1,755,917	3.1	6.15	21,564
250	45.1	0.421	909,240	2.6	7.27	18,242
500	23.0	0.826	463,426	2.3	8.40	15,788
1000	11.7	1.630	234,840	2.0	9.52	13,931
2000	5.9	3.240	118,145	1.8	10.60	12,511
3000	3.9	4.850	78,926	1.7	11.30	11,736
4000	2.9	6.460	59,255	1.6	11.70	11,335
5000	2.4	8.070	47,434	1.6	12.10	10,960

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5.2.5.2 Building Surface Survey Investigation Levels

Another aspect of FSS is evaluating the preliminary results to investigation levels. Investigation levels are used to indicate when additional investigations may be necessary. When an investigation level is exceeded, the original measurement may be confirmed by reanalysis or collecting additional measurements. Depending on the results of the investigation, the survey unit may require reclassification, remediation, and/or resurvey. The investigation levels for building surface contamination surveys are presented in Table 5-10.

Table 5-10 Final Status Survey Investigation Levels – Building Surfaces

Survey Unit Classification	Measurement Investigation Level	Scanning Measurement Investigation Level
Class 1	> DCGL	> $DCGL_{EMC}$
Class 2	> DCGL	> $DCGL_W$
Class 3	> 80% DCGL	> $DCGL_W$

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Any additional measurements collected as part of an investigation or as judgmental (biased) measurements will be presented in the FSS report and may aid in the elevated measurement comparison. However, these measurements may not be included as part of the statistical evaluation unless it is obvious that they will not impart any bias to the randomly collected measurements.

If it is determined that additional measurements need to be collected in order to define the locally elevated area, then the following process is planned:

> If scan survey data provides an indication of the boundaries of the locally elevated area, then the measurements will be distributed within that area. The number of measurements will be dependent on the actual size of the locally elevated area. If scan survey data does not provide any clear indication of the boundaries, then measurements will be placed around the location of elevated activity as an initial assessment. The results of these measurements may trigger additional measurements in order to define the extent of the locally elevated area.

6.0 SURVEY METHODS AND INSTRUMENTATION

This section will present details of the survey and sampling methods including measurement techniques, sampling procedures, and instrumentation.

Sample collection and measurement procedures are concerned mainly with ensuring that a sample is representative of the media and is consistent with assumptions used to develop the conceptual site model and DCGLs. Scanning is performed to identify areas of elevated activity of significance that may not be detected by other measurement methods (e.g., volumetric sampling or direct measurements).

The first portion of this section will address sampling methods, followed by survey methods. The last portion of this section will discuss instrumentation, calibration, and associated MDC.

6.1 VOLUMETRIC SAMPLING METHODS

Various sampling methods may be used to collect the soil and sediment samples in the survey units. These samples will be the basis for the statistical tests and comparison to the decision rules. Therefore these samples need to provide an estimate of the average residual radioactivity in each survey unit.

6.1.1 Soil/Sediment Sampling Methods

Most volumetric soil samples collected will be surface soil using hand collection techniques, but some samples may need to be collected at depth. A number of techniques have been developed to obtain samples from various depths below the ground surface. The techniques that have been selected provide a practical and efficient means of obtaining samples in a manner consistent with safety protocols and quality assurance/quality control (QA/QC) requirements. Additionally, they employ equipment that is normally available for use. Sampling procedures for obtaining subsurface soil samples for the different exploratory techniques include direct push and soil boring.

Volumetric samples collected during Site FSS activities shall be assigned unique sample identification numbers. These numbers are necessary to identify and track each of the samples collected for analysis during completion of the project. In addition, the sample identification numbers shall be used to identify and retrieve the analytical results received from the laboratory, as well as other data related to the sample.

Each sample shall be identified by a unique alpha-numeric code. To maintain consistency and comparability of sample location identification throughout the course of the FSS, samples shall be labeled by:

- a two-letter sample type designation (AA);
- a FSS designation (FSS);
- a two digit FSS area number (BB);
- a three digit exploration location designation (CCC); and
- a two digit depth layer interval number (##).

The format of the sample identification will be as follows:

AAFSSBBCCC##

The sample type designator (AA) identifies the specific sample type, such as sediment, soil or volumetric material from a building surface. The FSS area number (BB) identifies the specific FSS survey area, such as the building complexes, Building 3 High Bay, or other FSS environmental areas. The exploration location designation number (CCC) is a sequential number within a specific FSS area. The depth layer interval number (##) is a number that represents the incremental depth layer from where the sample was collected. For building volumetric samples where the volumetric sample is obtained from a shallow layer (i.e., paint, tile, asphalt roofing material, etc.), the depth layer interval number will normally be recorded as “01”. For surface volumetric sample locations where there is a significant thickness of material to be sampled, and multiple samples will be collected from the same location, the samples will be identified starting with “01” as the first layer sample.

Sample type designators and FSS area numbers are described below:

<u>Sample Type Designation (AA)</u>

SS	surface soil
SB	soil boring
SD	sediment
SC	soil beneath concrete
EX	excavated soil
SX	surface soil from trench or excavation
BV	building surface – volumetric

<u>FSS Area Numbers (BB)</u>

03	Building 3 Complex
06	Building 6 Complex
23	Equipment Storage Yard
25	Small Pond
26	Buffer/General Areas
30	Building 3 High Bay
32	Underground Utilities
33	Site Brook
34	Debris Piles
35	Clamshell Pile
36	Drum Burial Pit
38	Woods Area
39	Burning Grounds
40	B2 Sanitary Waste Line
41	Sanitary Waste Line
42	Industrial Waste Line
43	Former Waste Water Treatment Facility

For soil samples, the depth layer (sample interval layer) identifier, as measured from the ground surface, shall be included with the sample identification. For example, the soil sample collected from the survey unit at the second depth layer interval (e.g., the 2 to 4 foot depth level) in a soil boring within FSS Area Building 6 Complex would be labeled SBFSS0600102.

To permit proper evaluation of the sample analysis results, it is important that the actual location of the samples be properly documented. Sample locations will be identified in the field with pin flags, stakes, or other markers. Sample locations will also have GPS performed to record their position.

Split or duplicate samples shall be provided as requested to NRC, the State and/or their authorized representatives of samples collected. Similarly, split or duplicate samples may be taken by NRC, the State and/or their authorized representatives. Identical procedures shall be used to collect all samples unless otherwise specified by NRC or CTDEP.

Sample tracking and custody procedures will be followed for samples collected as part of the FSS. The sample tracking and sample custody procedures will be followed to assure that each sample is accounted for at all times. To maintain this level of sample monitoring, sample container labels and chain of custody (COC) records will be used to track samples collected for analysis.

6.1.2 Building Sampling Methods

A number of techniques have been developed to obtain volumetric samples from various building surfaces. The techniques that have been selected (hole saw, knife, scraper, etc.) provide a practical and efficient means of obtaining samples in a manner consistent with safety protocols and QA/QC requirements. Additionally, they employ equipment that is available for use at this project site.

Building material volumetric samples collected during Site FSS activities shall be assigned unique sample identification numbers. These numbers are necessary to identify and track each of the samples collected for analysis during completion of the project. In addition, the sample identification numbers shall be used to identify and retrieve the analytical results received from the laboratory, as well as other data related to the sample.

Each sample shall be identified by a unique alpha-numeric code as identified in Section 6.1.1 above.

6.2 SURVEY METHODS

Volumetric sampling of soil/sediment and building surface materials may have a low probability of identifying small areas of elevated radioactivity. Scanning surveys may be performed when needed to locate radiation anomalies indicating residual radioactivity that may require further investigation or action. Since both source terms emit gamma radiation, gamma scan surveys can be performed to investigate for localized areas of elevated radioactivity in surface soils and on building surfaces.

Collecting direct measurements on building surfaces, using hand-held portable survey instrumentation, is a survey methodology used to obtain radiological data (surface area radioactivity) specific to a single small area within a survey unit, analogous to collecting a volumetric soil sample for a land area survey. Because it is a timed measurement, instrument MDCs are usually lower than those of the same instrument used for a scan survey and are normally included as a population of data for the survey unit.

6.2.1 Land Survey Methods

Gamma walkover surveys or other appropriate land area survey methods are specified for this FSS. Gamma walkover surveys are performed by holding the detector close to the ground surface and moving it in a pendulum (back-and-forth) motion while walking at a speed that allows the investigator to detect the desired investigation level. Discernable increases in the count rate (meter or audible) to the investigation level or greater will trigger a more focused survey of the area. This may include allowing the survey meter response to stabilize at the location or a time-integrated direct reading. Locations that exceed the investigation level will be marked (flag or stake) for additional investigation as an elevated area. Land area surveys using multiple detector array system methodology, along with GPS and data logging capabilities, may be used to perform this survey so long as radiation emission type and scan MDCs are suitable for the detector system.

Characterization data used in creating the survey units should be reviewed prior to performing scan surveys so that the surveyor is aware of the levels of contamination that existed in the area. Class 1 survey units are likely to receive the most focused scan survey and Class 2 survey units may have focused areas of scan surveys within the survey unit. Class 3 survey units may not have a scan survey unless it is determined to be needed based upon professional judgment. The investigation levels for scan surveys are presented in Section 4.2.4.3, Table 4-7. The detection sensitivity for scan surveys will be discussed in Section 6.3.1.2.

6.2.2 Building Surface Scan Methods

Gamma surface scan surveys are performed by holding the detector close to the building surface and moving it over the surface at a rate that allows the investigator to detect the desired investigation level. Discernable increases in the count rate (meter or audible) to the investigation level or greater will trigger a more focused survey of the area. This may include allowing the survey meter response to stabilize at the location or performing a direct reading. Locations that exceed the investigation level will be marked (flag or stake) for additional investigation as an elevated area.

Characterization data used in creating the survey units should be reviewed prior to performing scan surveys so that the surveyor is aware of the levels of contamination that existed in the area. Class 1 survey units are likely to receive the most focused scan survey and Class 2 survey units may have focused areas of scan surveys within the survey unit. Class 3 survey units may not have a scan survey unless it is determined to be needed based upon professional judgment. The investigation levels for building scan surveys are presented in Section 5.2.5.2, Table 5-10. The detection sensitivity for scan surveys is discussed in Section 5.2.5.1.

6.2.3 Direct Measurement Methods

The type of instrument and method of performing the direct measurement are selected as dictated by the type of potential contamination present, the measurement sensitivity requirements, and the objectives of the radiological survey. Direct measurements are taken by placing the instrument at the appropriate distance above the surface, taking a discrete measurement for a pre-determined time interval, and recording the reading. A one-minute integrated count technique is a practical field survey procedure for most equipment and provides detection sensitivities that are below most DCGLs. However, longer or shorter integrating times may be warranted. Direct measurements may be collected at random locations in the survey unit. Alternatively, direct measurements may be collected at systematic locations and supplement scanning surveys for the identification of small areas of elevated activity. Direct measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify location for direct measurements to further define the areal extent of contamination. All direct measurement locations and results should be documented.

6.3 INSTRUMENTATION

Several different types of instrumentation may be used for performing surveys or analyzing samples. This section will provide details about the instrumentation that may be used during FSS including calibration, operational checks, and minimum detectable concentrations. The first part of this section will discuss instruments for scan and direct surveys and the second part will present laboratory instruments for analyzing samples.

6.3.1 Field Instruments

Since a variety of instruments may be used during final status survey, this section will focus on the pertinent issues generically. Instruments that may be used as part of FSS will be calibrated and have the MDC evaluated prior to actual use.

Land area scan survey instrumentation may consist of a survey meter and sodium iodide (NaI) probe or a sophisticated drive-over multiple detector array system with GPS and data logging capabilities. While a wide variety of portable instruments and detectors are readily available the drive-over systems are usually a one-of-a-kind instrument system that provide a greater amount of survey data in a shorter timeframe. While drive-over systems are currently not as common as the portable instruments, the drive-over multiple detector array systems are being used with a greater frequency for FSS. Instruments used for direct surface measurements may consist of portable beta/gamma probe and appropriate instrument for building surface direct measurements.

The rest of this section regarding portable instruments will use NaI probes and associated instruments as an example of how calibration, minimum detectable concentration, and reporting results will be performed. Similar processes will be utilized for any other portable survey instruments and detectors used during FSS.

6.3.1.1 Calibration

Calibration of portable instruments will conform to the manufacturer's recommendations as well as established standards (American National Standards Institute [ANSI], 1997). In general, this will entail verifying that the electronics of the meter are working properly, performing a voltage plateau, and establishing the efficiency of the probe. For land area gamma scans, the efficiency needs to be determined in terms of counts per minute (cpm) as a function of exposure rate (cpm per micro-Roentgen per hour [μR/h]). This can be accomplished by measuring the count rate produced by a source at a set distance with the NaI probe and then measuring the exposure rate with a micro range exposure rate instrument or by using a calibration source that has a certified exposure rate at a set distance and recording the count rate produced by the NaI probe. Since the NaI probe response rate is gamma energy dependent, the results of this efficiency calibration will need to be adjusted for the gamma energies of concern for this Site. This process involves using the physical characteristics of NaI and will be discussed as part of the MDC calculation in the next part of this section.

6.3.1.2 Minimum Detectable Concentration

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

Derivation of the MDC_{SCAN} for soil is a four step process. First, the relationship between the NaI detectors counting rate to exposure rate (cpm per μR/h) as a function of gamma energy is determined. Second, the relationship between radionuclide concentration in soil and exposure (pCi/g per μR/h) is established. Next, the $MDCR_{SURVEYOR}$ is calculated, and finally all three parameters are utilized to calculate the MDC_{SCAN}. This is an *a priori* determination of MDC_{SCAN} that can be used for planning FSS. The actual MDC_{SCAN} will be determined with the operating parameters (background, efficiency) of the actual instruments that will be used in the FSS.

Several factors need to be determined in order to establish the relationship between the detector's count rate to exposure rate. The response of the NaI detector is relative to the gamma energy interacting with the detector. Therefore the cpm produced by the detector will be a function of the probability of interaction for a gamma of particular energy. This parameter is determined by taking a known detector response (calibration) and applying it to the relative response of the detector at different gamma energies. For this *a priori* determination of MDC_{SCAN}, the manufacturer provided a valuesof 900 cpm per μR/h (Ludlum) for Cs-137 will be used in lieu of actual calibration efficiency. The relative response of the detector is calculated by multiplying the probability of interaction by the relative fluence rate for a given gamma energy. The probability of interaction is determined from the mass attenuation coefficients (μ/ρ) for NaI and the fluence rate is determined from the mass energy-absorption coefficients (μ_{en}/ρ) for air (Hubbel and Seltzer 1997).

The second phase of this process is to determine the relationship between the radionuclide concentration in the soil and exposure rate. The best way to accomplish this is to model the soil with a code such as Microshield™ in order to determine the exposure rate. The geometry used for this modeling was input as a cylindrical volume with a radius of 28.2 centimeters (area of 0.25 m^2) and a thickness of 7.5 centimeters (based on the most likely thickness of the contaminated layer used in RESRAD to derive the DCGLs). The dose point was located 10 centimeters directly above the center of the cylinder to represent the typical height above the surface during scanning. The soil was input into Microshield as the standard material concrete with a density of 1.6 g/cm^3 (to represent typical soil). The byproduct and uranium source terms were input at the DCGL concentration and the uranium source was decayed for fifty years in Microshield in order for all the decay products to be present in the modeling. The results are 309 pCi/g per μR/h for uranium (557 pCi/g divided by 1.801 μR/h) and 1.41 pCi/g per μR/h for Co-60 (5 pCi/g divided by 3.549 μR/h). The thorium and radium source terms were input at the DCGL concentration and the source was decayed for fifty years in Microshield in order for all the decay products to be present in the modeling. The results are 1.37 pCi/g per μR/h for thorium (4.0 pCi/g divided by 2.927 μR/h) and 1.60 pCi/g per μR/h for Ra-226 (4.5 pCi/g divided by 2.821 μR/h).

The first step in determining the MDC_{SCAN} is to calculate the minimum detectable count rate for the surveyor ($MDCR_{SURVEYOR}$). $MDCR_{SURVEYOR}$ is a function of the background count rate, the length of the counting interval, surveyor efficiency, and the index of sensitivity (statistical) as shown in Equation 6-1. Background for a 2” x 2” NaI detector is approximately 10,000 cpm, and the index of sensitivity (*d’*) will be based upon a 95% true positive rate and a rate of 60% false positive, which yields a value of 1.38. The surveyor efficiency has a value of 0.5 and the length of the counting interval will be 1 second. The results of this evaluation is shown in Table 6-1 and indicate that 1,513 cpm above background (11,513 cpm with background) is the minimum value for 95% true positive detection.

$$MDCR_{surveyor} = \frac{d' * \sqrt{b_i} * (60/i)}{\sqrt{p}} \tag{6-1}$$

$MDCR_{surveyor}$ = surveyor minimum detectable count rate (above background)

d' = the index of sensitivity (the number of standard deviations between the means of background and radioactivity above background).

b_i = the number of background counts in the counting interval, i.

i = the length of the counting interval in seconds.

p = surveyor efficiency

Table 6-1 MDCR$_{SURVEYOR}$ Values

Parameter		Value
i	The length of the counting interval (seconds)	1
d'	Index of sensitivity	1.38
C_b	Background count rate (cpm)	10,000
b_i	Number of background counts in counting interval i	167
s_i	Minimum detectable net counts in counting interval i	17.8
MDCR	Minimum detectable count rate (cpm)	1,070
p	Surveyor efficiency	0.5
$MDCR_{surveyor}$	Surveyor minimum detectable count rate (cpm)	1,513

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The minimum detectable exposure rate in μR/h is calculated by dividing the MDCR$_{SURVEYOR}$ by the detector efficiency in cpm per μR/h. Multiplying the minimum detectable exposure rate by the soil concentration exposure rate factor in pCi/g per μR/h will yield the MDC$_{SCAN}$ as shown in Equation 6-2. The parameters for calculating MDC$_{SCAN}$ for a 0.25 m^2 (radius of 28.2 cm) circular hot spot with a depth of 7.5 cm and the dose point located 10 cm directly above the center of the circle are shown in Table 6-2.

$$MDC_{SCAN} = \frac{\mathrm{MDCR}_{\mathrm{surveyor}}}{\varepsilon_t} * S_c \qquad (6\text{-}2)$$

Where:

MDC_{SCAN} = the minimum radioactivity concentration in soil above background radioactivity (in pCi/g) that can be reliably detected.

$MDCR_{surveyor}$ = surveyor minimum detectable count rate (above background)

ε_t = Counting system efficiency in cpm per μR/h.

S_c = Soil concentration exposure rate factor in pCi/g per μR/h

Table 6-2 MDC_{SCAN} Values For 2 X 2 NaI Detectors

Parameter		Byproduct	Uranium	Thorium	Radium
$MDCR_{surveyor}$	Surveyor minimum detectable count rate (cpm)	1,513	1,513	1,513	1,513
ε_t	Counting system efficiency (cpm per μR/h)	424	4,582	830	760
S_c	Soil concentration exposure rate factor (pCi/g per μR/h)	1.41	309	1.37	1.60
MDC_{SCAN}	Scan minimum detectable concentration (pCi/g)	5.0	102	1.8	2.8

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It should be noted that for the additional contaminants of concern (Th-232 and Ra-226) listed in Table 6-2, the weighted efficiency (ε_t) and the Scan MDC (MDC_{SCAN}) were taken from the 2” x 2” NaI Detector column from Table 6.4 of NUREG 1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, The values used above are appropriate since the detector volume used for the site is identical to the detector volume presented in the reference.

Since land area scan surveys will be used as part of the investigation of areas with locally elevated concentrations, the MDC_{SCAN} values should be evaluated with the investigation levels (DCGLs). In Table 6-3 the byproduct MDC_{SCAN} and DCGLs are compared and in Table 6-4 the uranium MDC_{SCAN} and DCGLs are compared. In Table 6-5 the Ra-226 MDC_{SCAN} and DCGLs are compared and in Table 6-6 the Th-232 MDC_{SCAN} and DCGLs are compared. From review of these tables it is clear that the MDC_{SCAN} values are less than or equal to the DCGLs and therefore land area scan sensitivity is acceptable for this FSS. However, if the actual scan sensitivity is not adequate (i.e., the available scan sensitivity is not sufficient to detect small layers of elevated activity), then it will be necessary to increase sample size as discussed in MARSSIM (NRC, 2000).

Table 6-3 Comparison to Byproduct DCGLs

Parameter	Value						
Area (m^2)	0.25	1	10	100	300	500	1000
Area Factor	13.38	13.38	2.68	1.35	1.19	1.14	1.09
$DCGL_W$ (pCi/g)	5	5	5	5	5	5	5
$DCGL_{EMC}$ (pCi/g)	67	67	13	7	6	6	5
MDC_{SCAN} (pCi/g)	5.0	3.2	2.3	2.1	2.1	2.1	2.1

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Table 6-4 Comparison to Uranium DCGLs

Parameter	Value						
Area (m^2)	0.25	1	10	100	300	500	1000
Area Factor	20.17	19.61	4.60	2.35	1.94	1.73	1.42
$DCGL_W$ (pCi/g)	557	557	557	557	557	557	557
$DCGL_{EMC}$ (pCi/g)	11,235	10,922	2,562	1,311	1,082	962	790
MDC_{SCAN} (pCi/g)	102	70.5	57.1	54.4	54.0	53.8	53.7

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Table 6-5 Comparison to Radium DCGLs

Parameter	Value						
Area (m^2)	0.25	1	10	100	300	500	1000
Area Factor	19.4	19.4	3.88	1.88	1.54	1.39	1.16
$DCGL_W$ (pCi/g)	4.5	4.5	4.5	4.5	4.5	4.5	4.5
$DCGL_{EMC}$ (pCi/g)	87.6	87.3	17.5	8.5	7.0	6.3	5.2
MDC_{SCAN} (pCi/g)	2.9	1.8	1.3	1.2	1.2	1.2	1.2

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Table 6-6 Comparison to Thorium DCGLs

Parameter	Value						
Area (m^2)	0.25	1	10	100	300	500	1000
Area Factor	58.3	15.1	3.07	1.52	1.31	1.22	1.11
$DCGL_W$ (pCi/g)	4.0	4.0	4.0	4.0	4.0	4.0	4.0
$DCGL_{EMC}$ (pCi/g)	233.1	60.3	12.3	6.1	5.2	4.9	4.4
MDC_{SCAN} (pCi/g)	2.7	1.7	1.2	1.1	1.1	1.1	1.1

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6.3.1.3 Reporting Results.

The results from surveys will be reported as raw results on the survey data sheets. This allows each step of the conversion to the concentration to be reviewed and will allow for any changes that might be necessary in the background or efficiency of the detector. In addition, the uncertainty associated with results and associated MDC will also be reported with the final results.

6.3.2 Laboratory Instruments

Samples analyzed on-site will be performed by gamma spectroscopy. The gamma spectroscopy system accurately identifies and quantifies the concentrations of multiple gamma-emitting radionuclides in soil with minimum sample preparation. This system consists of a high-purity germanium detector connected to a dewar of liquid nitrogen, high voltage power supply, spectroscopy grade amplifier, analog to digital converter, and a multichannel analyzer (MCA). The system is energy calibrated so the MCA data channels are given an energy equivalence. The MCA's display then becomes a display of intensity versus energy. Efficiency calibration is performed so that a curve of gamma ray energy versus counting efficiency is generated. Since the counting efficiency depends on the distance from the sample to the detector, each geometry must be given a separate efficiency calibration curve. Each peak is identified manually or by gamma spectroscopy analysis software. The counts in each peak or energy band, the sample weight, the efficiency calibration curve, and the isotope's decay scheme are factored together to give the sample concentration.

The gamma spectroscopy system is planned to be operated with Canberra's Genie 2000 software or comparable (Canberra 2002a). Genie 2000 is a comprehensive set of tools for acquiring and analyzing spectra from MCAs. Its functions include MCA control, spectral display and manipulation, comprehensive spectrum analysis for gamma spectroscopy, and quality assurance (Canberra 2002b).

In addition to gamma spectroscopy, alpha spectroscopy or other laboratory analytical methods may also be used on samples sent to an off-site laboratory for analysis

6.3.2.1 Calibration

The gamma spectroscopy system is calibrated for energy and efficiency (ANSI, 1991). This is achieved by using a calibration source in the same geometry (with a volumetric equivalent density) as the samples to be counted. A National Institute of Standards and Technology (NIST) traceable multi-line standard will be utilized to achieve calibration of the gamma spectroscopy system. The spectrum is analyzed to determine the energy vs. channel calibration curve. Then the efficiency for each of the gamma peaks is calculated and an efficiency calibration curve (efficiency vs. energy) is determined which can be a dual polynomial, linear polynomial, or empirical polynomial. This information is stored as a calibration file and will be used for analysis of spectra from samples

The calibration source for the gamma spectroscopy system will be NIST traceable. The source will be fabricated with a density equivalent material in the same container that the samples will be collected. The calibration source will consist of a mixture of radionuclides in order to cover a large range of gamma energy, and may include radionuclides as shown in Table 6-7.

Table 6-7 Typical Gamma Spectroscopy Calibration Source

Radionuclide	Gamma Energy (keV)
Am-241	60
Cd-109	88
Co-57	122
Ce-139	166
Hg-203	279
Sn-113	392
Cs-137	662
Y-88	898
Co-60	1173
Co-60	1332
Y-88	1836

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6.3.2.2 Minimum Detectable Concentration

The MDC for samples analyzed by gamma spectroscopy is calculated by the analysis software. MDC for gamma spectroscopy is calculated as shown Equation 6-3. For radionuclides with multiple gamma energies, a separate MDC value is calculated for each energy. The lowest of the values will be assigned as the radionuclide MDC. It is not uncommon for soil sample MDCs to be less than 1 pCi/g by gamma spectroscopy. After calibration of the gamma spectroscopy system, MDC values will be compared for various count times in order to determine an optimum count time for the radionuclides of interest.

$$MDC = \frac{L_D}{T_1 * \varepsilon * y * V * K_c * K_w * U_f} \quad (6\text{-}3)$$

Where:

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

6.3.2.3 Reporting Results

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

The results of gamma spectroscopy analysis will be reported by radionuclide as the actual concentration (pCi/g), the uncertainty associated with that result, and the MDC. The actual result may be less than the MDC or even a negative number. Statistical evaluations of the data will be performed on the actual results.

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

An evaluation of the multiplier of 31 can be accomplished by comparing the actual total uranium to the calculated total uranium (using multiplier of 31) for variations of the three uranium isotopes in 3.5% enriched uranium and 90% enriched uranium. Two samples are based on the NRC enrichment formula (specific activity) as shown in Equation 3-1, and two additional samples are variations based on typical enrichment results from the gaseous diffusion process. The NRC equation generally under predicts the specific activity for enrichments between 1% and 80% and the error may exceed 40%. The actual specific activity of any enriched uranium will vary depending on what plant produced the enriched uranium and when it was produced due to variations in the gaseous diffusion process. The two variation samples are based on a specific activity of approximately 30% and 40% greater than the NRC calculated value to represent likely variations. Using the NRC equation derived mixture of uranium isotopes for 3.5% and 90% enriched uranium, a multiplier range from 23 to 32 is appropriate to determine the total amount of uranium in a sample from the U-235 activity. These hypothetical samples and the comparison of the multipliers of 23 and 31 are shown in Table 6-8. The results show that the NRC equation based multiplier of 23 is not as conservative as the historical Site multiplier of 31 in determining a reasonable approximation of the overall total uranium, regardless of the percent enrichment of the sample being analyzed. Therefore, the historical Site multiplier of 31 times the U-235 concentration will be used to determine the total concentration of uranium in samples from the Site.

Table 6-8 Evaluation of Total Uranium Calculation

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

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For any samples that have both gamma spectroscopy and alpha spectroscopy results (or other laboratory analytical methods), the gamma spectroscopy results will be used unless there is a significantly large discrepancy between the results that could affect the statistical tests. Samples that are determined to contain HEU may have total uranium determined by alpha spectroscopy (or other laboratory analytical methods) results or a more appropriate multiplier for the U-235 may be used for the gamma spectroscopy results. In addition, FSS samples that have concentrations near the uranium $DCGL_W$ may have alpha spectroscopy (or other laboratory analytical methods) performed to provide a more accurate determination of total uranium.

7.0 QUALITY CONTROL AND DATA ASSESSMENT

In addition to the samples and surveys designed to demonstrate compliance with the approved DCGLs, additional samples and parameters will need to be collected in order to validate the accuracy of the data. QC activities are necessary to obtain additional quantitative information to demonstrate that measurement results have the required precision and are sufficiently free of errors to accurately represent the site being investigated. Furthermore, as data is gathered it will need to be reviewed and evaluated for use in demonstrating compliance with the release criterion.

The first part of this section will focus on methods to be integrated into the FSS process in order to ensure results that are accurate and usable. The second part of this section will present the DQIs that will establish the criteria for evaluation of the data in order to provide quantitative and qualitative measures of data quality and usability. The third part of this section will outline the process of reviewing the FSS data in order to verify that all parameters have been met and present the statistical evaluations and tests to be used for comparison to the release criterion.

7.1 QUALITY CONTROL

The goal of QC is to identify and implement sampling and analytical methodologies that limit the introduction of error into analytical data. This plan serves to provide the necessary control for providing sufficient data of adequate quality and usability for the purpose of confirming that the project's release levels have been met. It also serves to ensure that such data are authentic, appropriately documented, and technically defensible. QC will be achieved through three primary approaches: data management, sample custody, and QC measurements.

7.1.1 Data Management

Sample collection, field surveys and direct measurement, and laboratory analytical result data will, to the extent practicable, be recorded both electronically and on paper. Records of field-generated data will be reviewed by supervisory personnel knowledgeable in the measurement method for completeness, consistency, and accuracy. Electronic copies of original electronic data sets will be preserved on a retrievable data storage device. No data reduction, filtering, or manipulation will be performed on the original electronic versions of data sets.

Record copies of surveys, sampling, and analytical data (and their supporting data) will be located appropriately in project record files.

7.1.2 Sample Custody

Sample quality is potentially impacted by sample collection methods and by preservation of sample quality after the sample is collected and through the analytical process.

Sample quality related to sample collection is controlled through the use of trained sample personnel implementing approved standard operating procedures (SOPs) referenced throughout this FSSP. Methods employed in SOPs take into account the need to prevent sample contamination through the use of techniques such as disposable sampling apparatus and materials, decontamination of equipment between sample collection, and isolation of samples in discrete sample containers.

Once the sample has been collected and isolated, sample quality related to the processing and subsequent handling and analysis of the sample is controlled by maintaining appropriate sample custody.

Sample custody and control will be effected by:

- Assigning unique sample identification numbers to samples collected expressly for FSS in accordance with this plan.
- Recording the date, time, sample type, and location and linking that information with the sample identification number and the required analysis.
- Requiring that trained and qualified personnel be permitted to possess samples while they are on-site.
- Requiring that a Chain-of-Sample-Custody protocol be employed for volumetric sample materials.

7.1.3 Quality Control Measurements

The final status survey relies on in situ measurements using conventional health physics techniques and methods and upon media samples measured with either gamma or alpha spectroscopy measurement methods. Both will require additional steps in order to ensure the accuracy of the sampling techniques and analysis methodologies. The QC techniques will be subdivided into those for field activities and laboratory activities.

<u>7.1.3.1 Field Activities/Instruments</u>

QC activities performed during field survey and sampling will consist of field replicate (split) volumetric sampling (*i.e.*, a single sample that is collected, homogenized, and split into equivalent fractions in the field) and replicate (duplicate) direct measurements for direct surveys. Split samples are designed to assess the consistency and precision of the overall sampling and analytical system while duplicate direct measurements are designed to provide an estimate of overall precision of the field survey measurement system.

Split samples will be collected at a frequency of 5 percent (1:20), with at least one for each survey unit. Split samples will be collected from the same sampling location, depth, or interval as the original field sample after field homogenization. Duplicate measurements will be collected at a frequency of 5 percent (1:20), with at least one duplicate measurement from each survey unit.

QC activities performed during radiological scan surveys will consist of survey instrument checks and biased replicate measurements. The survey instruments utilized for performing scan surveys will have periodic source response checks and background checks performed (at least twice per day). This data will be used to generate control charts to demonstrate that the instruments have been performing accurately and consistently. Biased replicate measurements will be collected at a frequency of 5 percent (1:20), with at least one sample from each survey unit. This will be accomplished by performing replicate static surface measurements in areas where elevated radioactivity has been identified. This is primarily due to the fact that there is a necessity to evaluate whether the same number of locations was identified by both replicates, as well as if the identified locations are the same, which creates a difficulty in developing precision as a DQO that can be evaluated for replicate scanning. For that reason, only biased-direct replicate measurements taken in areas where elevated residual radioactivity has been identified will be evaluated for QC for the scan survey instrument. If there are no elevated radioactivity areas located within the survey unit during the FSS, then biased-replicate measurements will not be collected for the scanning instrument.

7.1.3.2 Laboratory Instruments

QC for laboratory instruments will consist of instrument checks and replicate measurements. Germanium detectors will have periodic energy calibration checks, efficiency calibration checks, and background checks. This data will be used to generate control charts to demonstrate that the instruments have been performing accurately and consistently. Replicate measurements will be accomplished by repeating the analysis of a randomly chosen sample at a frequency of 5 percent (1:20).

Another QC method to assess the potential error that might occur with laboratory measurements of soil is to perform secondary measurements of the sample using independent counting equipment. Secondary counting (analysis) of samples is scheduled to be performed by an off-site laboratory at a frequency of 5 percent (1:20) of the samples collected (at least one per survey unit). In addition, samples will be made available upon request to the NRC and CTDEP.

Analytical quality control for samples submitted to an off-site laboratory for analysis will be specified by contract and are designed to ensure that the detection confidence levels are adequate to demonstrate compliance with the decision criterion for a given sample or sample set. An upper confidence level of 95% will be specified (UCL_{95}).

7.2 MEASUREMENT UNCERTAINTY AND DATA QUALITY INDICATORS

Measurement uncertainty in the techniques prescribed in the FSS arises from two principal sources: field-sampling variation and instrument measurement variation. Of the two sources, field-sampling variation will likely be the greatest contributor to overall uncertainty because of the inherent logistics of sample collection activities. To the extent practicable, field operations will be governed by procedures and survey personnel will be trained. Additionally, individuals who are well versed in the overall survey approach and its data quality objectives will be available to guide and referee unclear situations that may arise. The measurement methods, on the other hand, employ standard instruments and laboratory procedures whose aspects and nuances are well understood through many years of application. Procedures and their associated rigor will also govern instrument calibrations, source response checks, and operations.

A major activity in determining the usability of the sampling and survey data is assessing the effectiveness of the sampling program. DQIs listed in Table 7-1 will be used in the field and in the data quality analysis (DQA) process to provide quantitative and qualitative measures of overall data quality and usability. Key points evidenced by Table 7-1 include:

- **Completeness -** The project is striving for a 90-percent completeness objective. Attaining or not attaining the objective does not necessarily authenticate or compromise the study. However, a 90-percent completion goal is a desirable performance metric that indicates that nearly all of the specified data has been acquired. Completeness is a measure of the amount of valid data obtained from a measurement system compared to the amount that was expected under correct, normal conditions.

- **Comparability -** Comparability expresses the confidence with which one data set can be compared to another. When comparing data, it is important to compare data collected under the same set of conditions. Seasonal trends, depth of sample collection, analytical protocol, method detection limits, and any other sampling/analytical variables must be taken into account when comparing data sets.
- Comparability of data has been "designed-in" to the random spatial sampling approach. The same instrument types and measurement techniques will be used in comparable areas subject to FSS. As a result, inter-area comparability should be assured for randomly selected (unbiased) locations. The nature of a FSS is to collect not only data from randomly selected sample locations, but also to augment the data set with data collected based upon professional judgment and knowledge of the history and processes that occurred in specific areas of the site. Data from judgment-based sampling will naturally have some selection bias and may have unique characteristics that will not permit comparability with data collected from randomly selected locations. Still, if data from one strata or sample allocation type can be shown to be sufficiently comparable to data from a related strata or sample allocation type, the data sets may be pooled to extend the capability to make decisions about broader areas or media types and to improve the statistical power for making a decision.
- **Representativeness -** Representativeness is a measure of the degree to which the measured results accurately reflect the medium being sampled and the overall situation at the site. It is a qualitative parameter which is addressed through the proper design of the sampling program in terms of sample location, number of samples, and actual material collected as a sample of the whole.
- The random sample design with its unbiased allocation and preference for spatial distribution within the survey unit was intended to ensure representativeness to the extent practicable. Deviations from the unbiased allocation (e.g., selecting locations based on prior knowledge as might be gained from scanning for elevated concentrations of residual radioactivity) will indicate bias and compromise the ability to defend representativeness as a DQI. For this reason, data acquired through judgment sampling will be representative only of the immediate locality from which it was collected. Thus, a posteri data analysis will not include aggregation of randomly acquired data with data acquired through judgment sampling without a suitable treatment to correct the bias.
- **Precision -** Precision refers to the level of agreement among repeated measurements of the same parameter. The overall precision of a piece of data is a mixture of its parts (e.g., sampling and analytical factors for volumetric samples and instrument operation and measurement technique for measurements). The analytical precision and instrument operation is much easier to control and quantify because the laboratory is a controlled and

therefore measurable environment and field instrumentation is calibrated and checked daily. Sampling precision (volumetric collection or direct measurement) is unique to each site, making it much harder to control and quantify, while measurement technique varies between instrument type and technician performing the measurement.

- In general, sampling and measurement techniques specified for obtaining FSS can be controlled such that data has a high degree of reproducibility. Consequently, good precision between replicate measurements of a specific activity radioactive check source is usually achieved. However, it is expected that the FSS at the CE site will encounter a significant proportion of measurements that are representative of background. Thus, in cases where a replicate measurement is made at a location that does not have an appreciable concentration of residual radioactivity (clearly elevated above background), the instrument response will express 1) variation stemming from the instrument's inherent detection and counting capability and 2) the variation from background. Experience indicates that, frequently, the later component, background variability, obscures the ability to gauge instrument precision between replicate measurements when replicate measurements are made at locations where significant detectable radioactivity is not present.
- **Accuracy -** Accuracy refers to the difference between a measured value for a parameter and the true value for the parameter. It is an indicator of the bias in the measurement system. Field instrument accuracy will be evaluated by implementing instrument source response checks and the charting of the results (control charts). In addition, some comparison of a direct measurement of a soil sample location with the laboratory result may be performed as well. On-site laboratory analysis accuracy will be evaluated by source response check control charts. Samples may be sent to an off-site laboratory for comparison. Off-site laboratory accuracy will be evaluated by the analysis of one method blank per sample batch and one spiked sample per sample batch as applicable for radionuclides. Generally, the accuracy of analyses must be within historically derived, method-specific criteria.

Table 7-1 Target Data Quality Indicators

DQI	Significance	Action/Remark
Completeness	Less than complete data set could decrease confidence in supporting information	Objective of 90-percent completeness.
Comparability	Affects ability to combine analytical results	Data collected from randomly selected locations within a survey area are unbiased and comparable by design and can be combined. Combining of other data sets will be subject to appropriate two-sample statistical test methods designed to detect significant differences between samples or populations.
Representativeness	Non-representativeness increases or decreases Type I error depending on the bias.	Sample allocation will include a minimum number of unbiased, randomly distributed sample locations based on survey design.
Precision	Measurement variability, due to techniques and/or technology, may increase uncertainty.	Field sampling and instrument operation will be governed by procedures. Replicate measurements, background measurements, and source response check measurements will be used to gauge reproducibility. The number of field replicate measurements specified meets or exceeds MARSSIM guidance (NRC, 2000).
Accuracy	Sampling and data handling can introduce bias and affect Type I and Type II errors.	Field measurement will be governed by procedures. Instruments will be calibrated with NIST traceable sources.

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7.3 DATA QUALITY ASSESSMENT

Assessment of environmental and analytical data is used to evaluate whether the data meet the objectives of the sampling event, and whether the data are sufficient to determine compliance with release levels. The assessment phase of the data life cycle consists of three phases: data verification, data validation, and statistical evaluations.

7.3.1 Data validation/verification

Data verification compares the collected data to the DQOs documented in this plan. Data verification ensures that the requirements stated in the FSSP and SOPs are implemented as prescribed. The data and documentation used for release will be verified (100%). Data verification will include:

- assessment of activities performed during implementation by means such as inspections, QC checks, or surveillance;
- documentation of deficiencies or problems encountered during implementation; and
- review of corrective actions to ensure adequacy and appropriateness.

Data validation activities ensure that the results of data collection activities and analytical results support the objectives of the sampling event as documented in this plan, or support a determination that these objectives should be modified. MARSSIM defines data validation by six data descriptors. The six data descriptors for this FSSP are summarized in

Table 7-2. The data and documentation used for verification of release criteria will be validated (100%).

Table 7-2 Data Validation Parameters

Data descriptor	Consideration	Impact if not met	Corrective action
Reports to decision maker	• Sample design with measurement locations • Analytical method and detection limit • Background radiation data • Field reports	• Unable to perform a quantitative survey or analysis	• Request missing information
Documentation	• Chain-of-custody records • SOPs • Field and analytical records	• Unable to have adequate assurance of results • Unable to verify survey results or sample results	• Resurveying or re-sampling • Correct deficiencies
Data sources	• Data used meet DQOs	• Inadequate sample design • Lower confidence of data quality	• Resurveying, re-sampling, or reanalysis for unsuitable or questionable measurements
Analytical method and detection limit	• Routine methods used to analyze contaminants of concern	• Unquantified precision and accuracy • Potential for Type I and Type II decision errors	• Reanalysis, resurveying, or re-sampling • Documented statements of limitation
Data review	• Defined level of data review	• Potential for Type I and Type II decision errors • Increased variability and bias due to analytical process, calculation errors, or transcription errors	• Perform data review
Data quality indicators	• Completeness of survey and sampling • Representativeness of sampling locations • Precision of analytical methods • Accuracy of sampling and surveying	• Insufficient power to defend decision • Increases or decreases Type I decision error depending on the bias • Increase in data variability	• Conduct additional surveying or sampling • Use proper analytical instrumentation and approved protocol • Follow FSSP

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The data analysis framework will incorporate DQA components discussed in MARSSIM to assess the overall usability of the data for its intended use. The data evaluation process will be validated, and statistical analysis methods will be used, to assess whether variability and bias in the data are small enough to allow the data to support release of the Site from radiological controls with acceptable confidence.

Missing data may reduce the precision of estimates or introduce bias, thus lowering the confidence level of the conclusions. The importance of lost or suspect data will be evaluated in terms of the sample location, analytical parameter, nature of the problem, decision to be made, and the consequence of an erroneous decision. Critical locations or parameters for which data are determined to be inadequate may be resampled.

7.3.2 Statistical Evaluation

After data validation and verification is complete, the next step in DQA is to interpret the data. This is accomplished by performing a data review, testing the data, and drawing conclusions from the data.

7.3.2.1 Data Review

The data review entails calculating various statistical parameters (mean, median, standard deviation, etc.) in order to evaluate the data. In addition, data review will include qualitative visual analysis (e.g., histograms, scatter diagrams, box and whisker plots, etc.). Some additional analytical methods (e.g., spatial correlation) as well as spatial analysis (e.g., probability plots) not required to support the decision rule are not explicitly planned for but could be employed on an ad-hoc basis to gain insight. The intent of the data review is to evaluate the data for skewness or spatial trends that might be indicators of levels of residual radioactivity that exceed the release criterion.

7.3.2.2 Statistical Test

The Sign Test will be used to evaluate compliance with $DCGL_W$s. The Sign Test is a one-sample, non-parametric test that can be used to evaluate compliance with the DCGL. The Sign Test is the recommended compliance evaluation procedure when the contaminant(s) under evaluation are not present at significant levels in background. While the uranium series radionuclides clearly exist in nature, because background concentrations are appreciably lower than the DCGL(s), ABB does not feel that it is necessary to utilize the background reference area and distinguish measurements from background.

The actual results will then be utilized to evaluate the power of the sign test. A retrospective power curve will be generated from the actual results and the critical sample size will be determined for comparison to the actual number of samples.

7.3.3 Decision Rule

The combination of sample data and scan data will be used to demonstrate compliance with the release criterion. In addition to the Sign Test, some additional comparisons are needed, including the elevated measurement comparison (EMC). The decision to release the survey unit will be based upon the comparisons in Table 7-3.

Table 7-3 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
Any measurement greater than 2 times the $DCGL_W$	Survey unit does not meet release criteria (CTDEP RSR)
Any measurement greater than $DCGL_W$ and the 95% UCL of the mean is greater than $DCGL_W$	Survey unit does not meet release criteria (CTDEP RSR)

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Given that there are two different source terms that are unrelated, the unity rule will need to be used. The unity rule is to ensure 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 7-1.

$$\frac{C_U}{DCGL_U} + \frac{C_B}{DCGL_B} \leq 1 \qquad (7\text{-}1)$$

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

Another factor in the decision rule is the EMC. Each measurement in the survey unit (systematic and scan) is compared to the investigation levels. Any measurement that is equal to or greater than the investigation level should be investigated. The EMC is intended to flag potential failures in the remediation process, not to demonstrate compliance with the release criterion. The derived concentration guideline level for the EMC is shown in equation 7-2.

$$DCGL_{EMC} = A_m * DCGL_W \qquad (7\text{-}2)$$

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 or 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 7-3.

$$\frac{\delta_U}{DCGL_U} + \frac{\delta_B}{DCGL_B} + \frac{\overline{\chi}_U - \delta_U}{A_m * DCGL_U} + \frac{\overline{\chi}_B - \delta_B}{A_m * DCGL_B} < 1 \qquad (7\text{-}3)$$

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 uranium

$DCGL_B$ = derived concentration guideline level for byproduct

If there is more than one area of elevated residual radioactivity in a survey unit then additional terms can be added to equation 7-3. An alternative is to use the actual results as input into RESRAD and calculate the dose for each area of elevated residual radioactivity in order to show that the total dose is within the release criterion.