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LACBWR Radiological Characterization Survey Report for June thru August 2015 Field Work
Genoa, Wisconsin

Project No. 164017 **Revision 0**

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List of Acronyms and Abbreviations

ALARA	As Low As Reasonably Achievable
AEC	U.S. Atomic Energy Commission
BWR	Boiling water reactor
Ci	Curie
COC	Chain-of-Custody
DAW	Dry Active Waste
DCGL	Derived Concentration Guideline Level
DPC	Dairyland Power Cooperative
DQA	Data Quality Assessment
DQO	Data Quality Objective
FESW	Fuel Element Storage Well
FSS	Final Status Survey
G-3	Genoa Station #3
GPR	Ground Penetrating Radar
GPS	Global Positioning System
HASP	Health and Safety Plan
HSA	Historical Site Assessment
HTD	Hard-To-Detect
ISFSI	Independent Spent Fuel Storage Installation
JHA	Job Hazard Analysis
LACBWR	La Crosse Boiling Water Reactor
LSE	LACBWR Site Enclosure Area
LTP	License Termination Plan
MARSAME	Multi-Agency Radiation Survey and Assessment of Materials and Equipment
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum detectable concentration
MDCR	Minimum Detectable Count Rate
MWe	Megawatts electric
NIST	National Institute of Standards and Technology
NRC	United States Nuclear Regulatory Commission
NORM	Naturally Occurring Radioactive Material
pCi/g	PicoCuries per gram
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control
RCA	Radiologically Controlled Area
ROC	Radionuclides of Concern
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
TEDE	Total Effective Dose Equivalent
TRU	Transuranic
μCi/g	MicroCuries per gram

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1.0 Introduction

This report presents radiological survey information gathered during the La Crosse Site Characterization Project on-site field work conducted from June 15, 2015 thru August 6, 2015. This data supports the License Termination Plan (LTP) development and overall decommissioning operations for the Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor (LACBWR), located near Genoa, Wisconsin. The LACBWR was permanently shut down in 1987. A 350-MWe fossil generating station, Genoa Station #3 (G-3), continues to operate on the same site. The LACBWR site is about one (1) mile south of the Village of Genoa, WI, and approximately 19 miles south of the city of La Crosse, WI. Figure 1-1, “La Crosse Boiling Water Reactor Site” shows the location of the site.

The licensed site area totals approximately 163.5 acres and includes land areas to the north of LACBWR, which includes the site switchyard and the site of the former G-1 facility (removed in 1989); areas south of LACBWR, which includes the area with the existing operational G-3 facility as well as the coal pile area and land surrounding the ISFSI; and a parcel of land to the east of Highway 35, across from LACBRW. Figure 1-1 shows the general layout of the site and surrounding areas.

The following buildings and structures are within the LACBWR site enclosure (LSE) fenced area:

- Reactor Building
- Turbine Building
- 1B Diesel Generator Building
- Waste Treatment Building
- Gas Storage Tank Vault
- Ventilation Stack
- LSA Storage Building
- Maintenance Eat Shack
- Pipe Tunnel

Other LACBWR Buildings nearby, but outside the LSE, include:

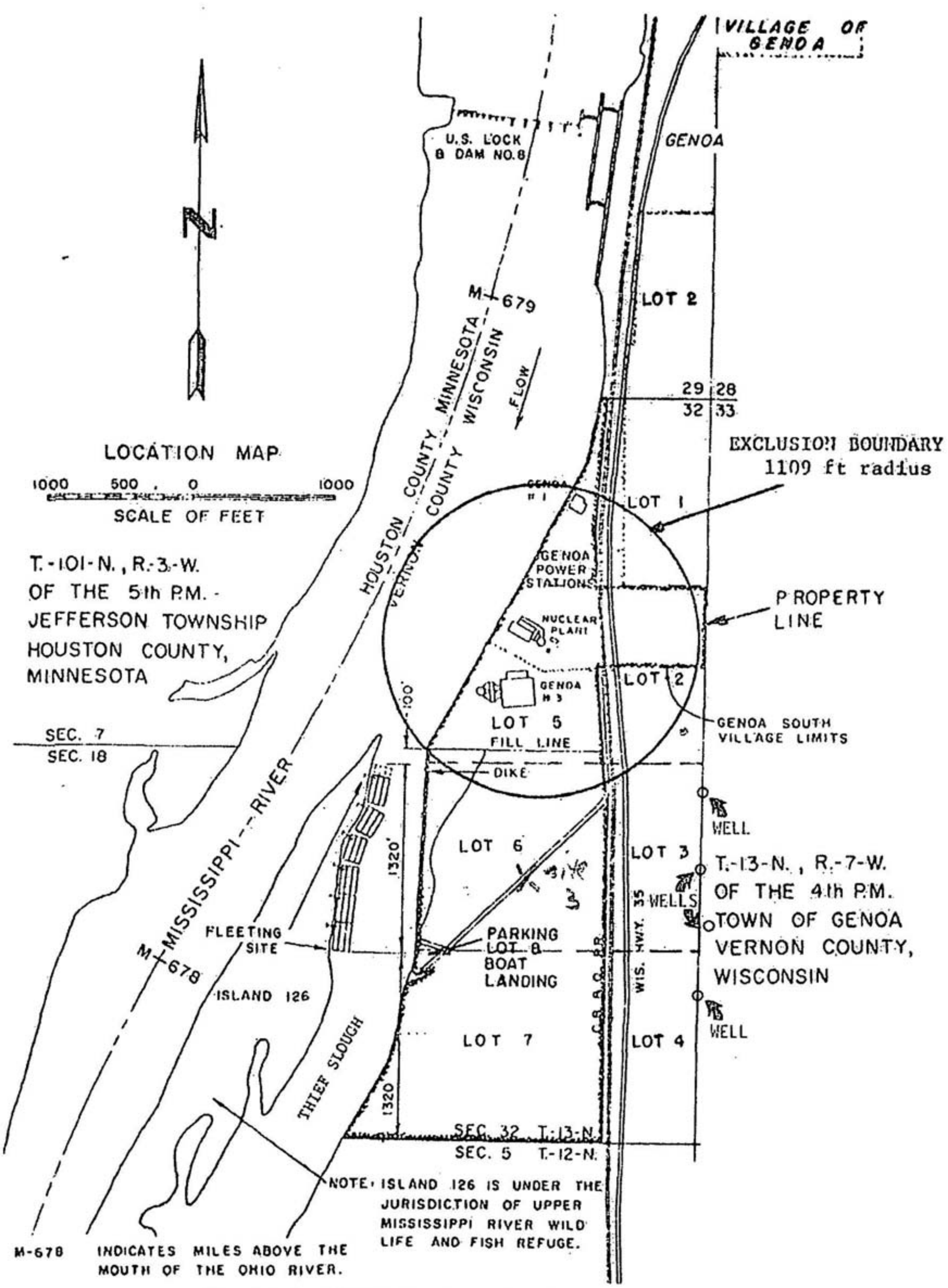
- Administrative Building
- Warehouses Nos. 1, 2, and 3
- LACBWR Crib House

Figure 3-2 “LACBWR Site Buildings Map” shows the respective location of the aforementioned facilities and structures.

As part of the LTP scope of work for DPC, Energy*Solutions* reviewed current and historical facility and radiological information as well as the radiological information from the 2014 on site characterization work [Reference 10-1]. This information has been used to develop the scope of the 2015 LACBWR on site characterization work and the goals of that site characterization work. With completion of this characterization effort, data is available to:

- Assist with determining the nature and extent of contamination under the Turbine Building from identified past Turbine Building drain piping integrity questions as discussed in the HSA;
- Evaluate Class 3 initial survey unit classifications for re-classification as Non- Impacted Area (Class 4) by performing additional soil sampling and gamma walkover surveys;
- Support the development of Derived Concentration Guideline Levels (DCGLs) and completion of LTP Chapters 2 (Site Characterization), 5 (Final Radiation Survey Plan), and 6 (Compliance with the Radiological Criteria for License Termination);
- Develop site specific Radionuclide of Concern (ROC) ratios based on the results of the concrete coring field work conducted in representative areas of the Class 1 facilities;
- Support planning for future unrestricted release surveys of materials in accordance with NUREG-1575, Supplement 1, "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME) [Reference 10-2] and;
- Develop decontamination and demolition options, and waste classification/segregation.
- Support the protection of workers, the general public, and the environment.

Figure 1-1, La Crosse Boiling Water Reactor Site



1.1. Definitions

Activity - Rate of disintegration (transformation) or decay of radioactive material. The units of activity are the curie (Ci).

Action Level – A derived media-specific, radionuclide-specific concentration or gross activity level of radioactivity that triggers a response, such as further measurements, investigation, or remediation, if exceeded.

Biased or Judgmental Measurements – Measurements performed at locations selected using professional judgment based on unusual appearance, location relative to known contamination areas, high potential for residual radioactivity or other general supplemental information.

Class 1 Area – An *impacted area* with the highest potential for contamination and has: (1) the potential for delivering a dose above the release criterion; (2) a potential for small areas of elevated activity; and (3) insufficient evidence to support reclassification as Class 2 or Class 3.

Class 2 Area – An *impacted area* that has: (1) a low potential for delivering a dose above the release criterion; (2) little or no potential for small areas of elevated activity; and (3) insufficient evidence to support reclassification as Class 3.

Class 3 Area – An *impacted area* with the lowest potential for contamination and has: (1) little or no potential for delivering a dose above the release criterion; and (2) little or no potential for small areas of elevated activity.

Class 4 Area – A *non- impacted area* (not impacted by site operations or decommissioning activities) where there is no reasonable possibility (extremely low probability) of residual contamination.

Data Quality Objectives (DQO) – Qualitative and quantitative statements derived from the DQO process that clarify technical and quality objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions.

Derived Concentration Guideline Levels (DCGL) – A derived, radionuclide-specific activity concentration within a survey unit corresponding to the release criterion. DCGLs are derived from activity/dose relationships through various exposure pathway scenarios.

Final Status Survey – Measurements and sampling to describe the radiological conditions of a site, following completion of decontamination and remediation activities, if any, in preparation for release of a survey area or unit(s) from a site license.

Impacted Area – An area with a possibility of containing residual radioactivity from licensed activities in excess of natural background or fallout levels.

Minimum Detectable Concentration (MDC) – The smallest amount of radioactive material in a sample that will yield a net count, above system background, that will be detected with a 95% probability with only 5% probability of falsely concluding that a blank observation represents a real signal. MDC depends upon the type of instrument, the counting geometry, and the radionuclide to be detected.

Minimum Detectable Count Rate (MDCR) -Is the minimum detectable count rate that a specific instrument and survey technique can be expected to detect.

2.0 Site Summary

2.1. Noted Radionuclides and Site Conditions

Historical site information has been reviewed at the site, including early initial site characterization data compiled in the DPC document LAC-TR-138, Initial Site Characterization Survey for SAFSTOR (issued October 1995 and revised up to December 2009) [Reference 10-3]. In addition, Energy*Solutions* has drafted a document, Technical Basis Document for Radionuclides of Concern During the Decommissioning of the La Crosse Boiling Water Reactor (July 2014) [Reference 10-4] which presents the expected radionuclides to be considered during characterization operations. The latest revision (March 2014) of the LACBWR Decommissioning Plan and Post-Shutdown Decommissioning Activities Report also provided a good update of ongoing material and metal removal operations.

2.1.1 Sources of Radioactivity

The operation of a boiling water reactor close to 19 years resulted in material activation and radionuclide distribution throughout plant systems and structures. In addition, the reactor experienced fuel cladding failures early in the plant life. These fuel element failures were severe enough to allow fission products to escape into the Fuel Element Storage Well and reactor coolant.

Assessments of radionuclide inventory in the reactor core, on plant system surfaces and internal locations, conducted in January 1988, indicated that the primary radionuclides were fission and activation products such as Co-60, Fe-55, Cs-137 and Mn-54. The review of 10 CFR 61 analysis results for various metal and structural material waste streams from 1998 to 2010 showed that the predominant radionuclides to be expected, after decay correction to January, 2015, would be Co-60 and to a lesser extent, Ni-63. Depending upon the type of sample, Co-60 concentrations, after decay correction, ranged from 6 to 45% of the total activity. Other radionuclides noted above 0.1% of the total activity in some of the samples were H-3, C-14, Fe-55, Ni-59, Cs-137, Eu-152, Pu-238, Pu-239/240, Am-241 and Pu-241. Additional 10 CFR 61 samples were recently obtained in July 2014 and focused on liquid waste from various tanks and sumps. The majority of samples were dominated by the presence of Cs-137, ranging from 71% to 92% of the total activity in a sample.

DPC has performed a number of soil surveys since the plant shutdown in 1987. These have involved soil sampling operations performed within the LACBWR RCA (e.g. LACBWR Site Enclosure Area, LSE, the fenced in area), the licensed site boundary and outside the site boundary. While most of the soil survey campaigns have been limited in scope, a review of the six soil sampling operations conducted between 1987 and 2008 indicated that the primary contaminant in the surface soils was Cs-137 at low levels. The last extensive soil sampling campaign conducted within the RCA was performed as part of an initial site characterization survey in 1995 and indicated the presence of Cs-137 with a maximum observed concentration of up to 1.30 pCi/g.

In the Fall of 2014 Energy*Solutions*, working with DPC site personnel, performed a characterization survey of the open land areas including: soils, asphalt, and concrete as well as the Administration Building in accordance with EnergySolutions PG-EO-313196-SV-PL-001, "Characterization Survey Plan for the La Crosse Boiling Water Reactor" [Reference 10-5]. The

result of that characterization work is presented in LACBWR Radiological Characterization Survey Report for October and November 2014 Field Work at Genoa, WI (GG-EO-313196-RS-RP-01) [Reference 10-6]. The surveys did not identify any different ROCs than past DPC assessments of radionuclide inventories or soil sampling campaigns.The initial suite of ROCs consists of 18 radionuclides with half-lives greater than 5 years, including gamma emitters, Hard-to-Detect (HTD) and Transuranic (TRU) alpha emitters. It was developed based upon the review of theoretical radionuclides noted in NUREG BWR studies, the specific engineering review of fuel inventory at LACBWR and other site-specific LACBWR sample results (e.g. the 10 CFR 61 reports, piping internal results, etc.). The list is shown in Table 2-2, "LACBWR Site-Specific Radionuclides of Concern." As site characterization progresses and additional survey and sample data is collected, this list will be reviewed and updated, if necessary.

Table 2-1, LACBWR Site-Specific Radionuclides of Concern

Radionuclide	Half Life (Years)
H-3	1.24E+01
C-14	5.73E+03
Fe-55 Ni-59	2.70E+00 7.50E+04
Co-60	5.27E+00
Ni-63	9.60E+01
Sr-90 Nb-94	2.91E+01 2.03E+04
Cs-137	3.00E+01
Eu-152 Eu-154	1.33E+01 8.80E+00
Np-237	2.14E+06
Pu-238	8.78E+01
Pu-239 Pu-240	2.41E+04 6.60E+03
Am-241	4.32E+02
Am-243 Cm-243/244*	7.37E+03 1.81E+01

*Listed half-life is the shortest half-life for the radionuclides in the pair

2.1.2 Preliminary Radionuclide Screening Criteria

As site specific DCGLs have not yet been established for the LACBWR decommissioning, alternative action levels were selected for characterization purposes. The concentration values associated with the interim screening DCGLs presented in NUREG-1757 (Reference 10-6) and the NUREG/CR-5512, Volume 3 (Reference 10-7), using Table 6.91 (P_{crit}) for soils were used as the alternate action level when assessing the characterization of impacted open land or soil survey units. The preliminary criteria are presented in Table 2-2, *"Preliminary* Criteria for Surfaces and Soils."

Table 2-2, Preliminary Criteria for Surfaces and Soils

Preliminary Criteria (Limits) for Surfaces, Structures, and Paved Areas			
Radionuclide	Total Activity (dpm/100cm^2)	Removable Activity (dpm/100cm^2)	Basis
Alpha emitters (Pu, U, and Am)	100	20	USNRC Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors
Beta-gamma emitters (with the exception of H-3)	7,100	710	7,100 dpm/100cm^2 for Co-60 in NUREG-1757 Volume 2 Table H.1 Screening Criteria. The removable value is based on the NRC default screening criterion that specifies 10% of the limit.

Preliminary Criteria (Limits) for Soils		
Radionuclide	(pCi/g)	Basis
H-3	110	NUREG-1757 Volume 2 Table H.2 Screening Criteria and concentration values found in NUREG/CR-5512 Volume 3, Residual Radioactive Contamination from Decommissioning Parameter Analysis, Table 6.91 (P_{crit} = 0.10) for soils.
C-14	12	
Fe-55	10,000	
Ni-59	5,500	
Co-60	3.8	
Ni-63	2,100	
Sr-90	1.7	
Nb-94	5.8	
Cs-137	11	
Eu-152	8.7	
Eu-154	8.0	
Np-237	0.09	
Pu-238	2.5	
Pu-239/240	2.3	
Am-241	2.1	
Am-243	2.0	
Cm-243	3.2	
Cm-244	4.2	

The preliminary criteria for the LACBWR ROCs for concentration (pCi/g) in soil represents the surface soil concentrations of individual radionuclides that would be deemed to be in compliance with the 25 mRem/year unrestricted release dose limit in 10 CFR 20.1402 [Reference 10-7]. If multiple ROCs are present, then the dose contribution from each ROC is accounted for using a sum-of-fractions (SOF) calculation to ensure that the total dose from all ROCs does not exceed the action level.

The FSS of backfilled below grade structures remaining following decommissioning will consist of a source term survey which will be conducted to demonstrate that the inventory of residual radioactivity in building basements is below a source term inventory commensurate with the dose criterion in 10 CFR 20.1402. Consequently, adjusted gross DCGL for direct measurements will not be used for the FSS of LACBWR backfilled structures. Some above grade structures will remain including the the Administration Building, G-3 Crib House, and LACBWR Crib House which will undergo either a MARSSIM Final Status Survey (FSS) or a MARSAME free release survey.

Therefore, radiological characterization of structures will provide the necessary data to derive defensible radionuclide distributions which would then allow for the derivation of an adjusted gross DCGL for direct measurements for the FSS. However, as a gross screening level that will will be used during the characterization, the nuclide-specific screening value of 7,100 dpm/100 cm^2 total gross beta-gamma surface activity for Co-60 from NUREG-1757, Appendix H will be used. Use of the Co-60 screening value is appropriate and conservative as it is anticipated that the radionuclides distribution for surface contamination will be principally Co-60 and Cs-137 and the more conservative approach is to assume a distribution of 100% Co-60 as the screening

value for Cs-137 is significantly greater. The removable activity (e.g. loose contamination) value is based upon a 10% removable fraction discussed in NUREG-1757.

For surface contamination by alpha emitters, the values from Regulatory Guide 1.86 [Reference 10-8] are the basis for characterization and assessment. The NUREG- 1757 screening tables do not contain alpha emitting radionuclides and therefore Regulatory Guide 1.86 values are used for screening purposes. The use of Regulatory Guide 1.86 values for screening had no impact on classifying land areas as Class 1, 2, 3 or 4.

3.0 Characterization Plan Scope and Survey Units

3.1. Scope of Characterization Plan

Characterization is an initial step in the decommissioning process and requires a logical approach in obtaining the necessary data required for planning decommissioning activities. Radiological characterization provides a reliable database of information showing the quantity and type of radionuclides, their distribution, and their physical state as it applies to facilities and/or areas of the LACBWR Site. The characterization process also incorporates previously recorded survey data which includes scan and wipe measurements as well as soil and structural sampling analyses to present a summary of radiological conditions of the survey unit as known. The characterization surveys were designed and executed using the guidance provided in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) [Reference 10-10] and NUREG-1757, Volume 2, Revision 1, "Consolidated Decommissioning Guidance, Characterization, Survey and Determination of Radiological Criteria," [Reference 10-5]. In addition, the guidance recommends development and use of a Quality Assurance Project Plan (QAPP) which describes policy, organization, functional activities, the Data Quality Objective (DQO) process, and measures necessary to achieve quality data [Reference 10-11]. Energy*Solutions* used subcontractor(s) to perform soil sampling as described in the LACBWR Site Characterization Plan, as well as obtained the necessary radiological surveys with hand-held instrumentation. The Energy*Solutions* related procedures were used for characterization work, along with any requirements in the LACBWR NRC Radioactive Material license and associated technical specifications. Energy*Solutions* and subcontractor personnel performed work under the requirements of the LACBWR Radiation Protection, Health and Safety, and Work Control Program.

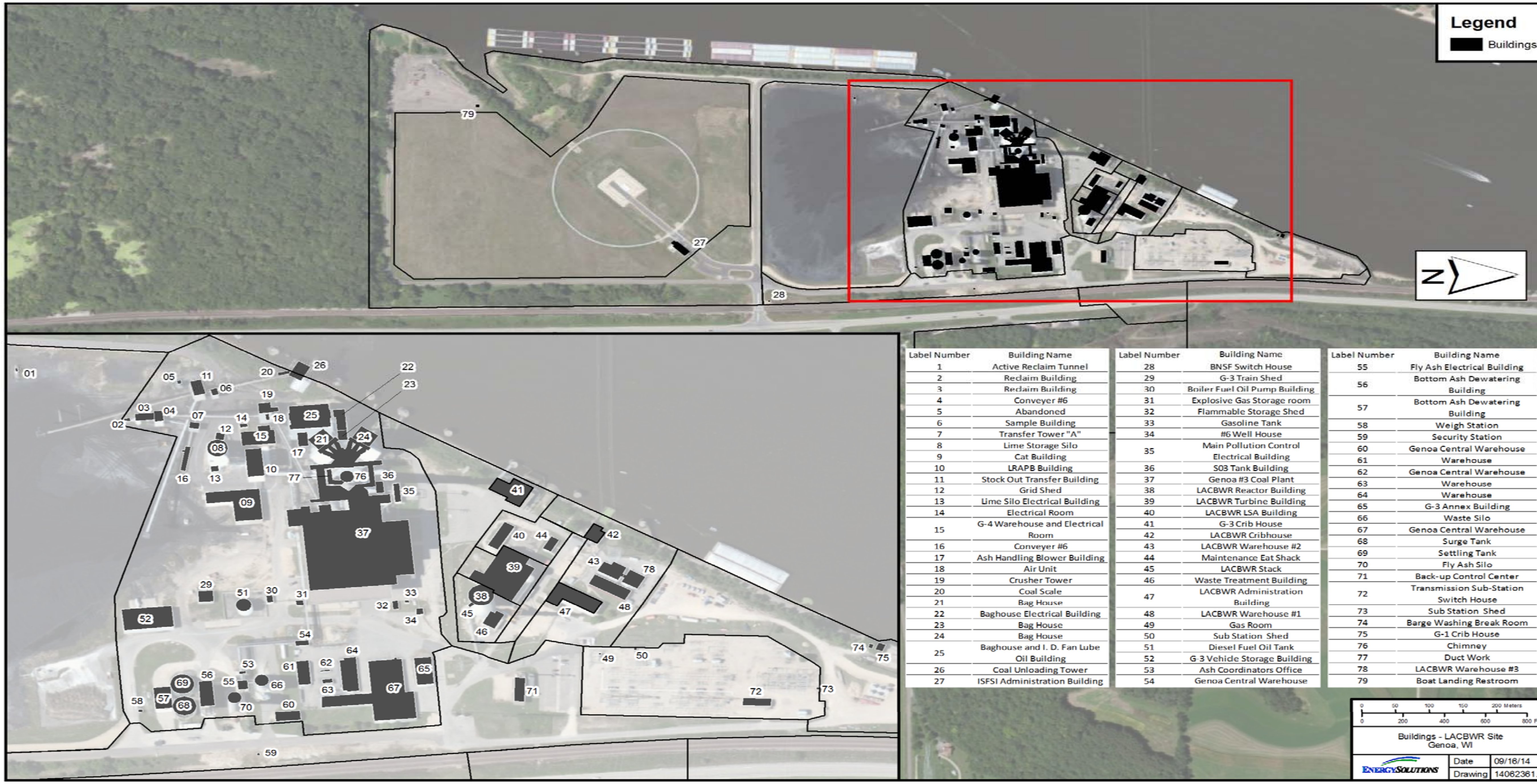
Using a combination of radiological scanning surveys, direct surveys, removable contamination measurements, exposure rate surveys and sampling of various media, the radiological conditions of the LACBWR Site open grounds areas, facility concrete, and the G-3 Coal Plant were determined and documented. The Data Quality Objectives of the survey were pre-defined to ensure sufficient reliable data was acquired to meet survey objectives.

3.2. Site Classification

Classification of survey units was based initially on historical information and available historical radiological survey data as discussed in the HSA. Classifying a survey unit has a minimum of two stages: (1) initial classification and (2) final classification. Initial classification is performed at the time of identification of the survey unit using the information available. This was completed and is summarized in [Reference 10-24]. The current classification of the LACBWR Site is based on the results of the surveys performed during this characterization field

work is presented in Figure 3-1, "LACBWR Site Survey Unit Map." A brief summary of the major findings and historical facts from the HSA, 2014 site radiological characterization work [Reference 10-1], and the recently completed 2015 site radiological characterization work that are relevant to the current classification of the areas, are presented as follows.

Figure 3-1, LACBWR Site Survey Unit Map



Label Number	Building Name
1	Active Reclaim Tunnel
2	Reclaim Building
3	Reclaim Building
4	Conveyer #6
5	Abandoned
6	Sample Building
7	Transfer Tower "A"
8	Lime Storage Silo
9	Cat Building
10	LRAPB Building
11	Stock Out Transfer Building
12	Grid Shed
13	Lime Silo Electrical Building
14	Electrical Room
15	G-4 Warehouse and Electrical Room
16	Conveyer #6
17	Ash Handling Blower Building
18	Air Unit
19	Crusher Tower
20	Coal Scale
21	Bag House
22	Baghouse Electrical Building
23	Bag House
24	Bag House
25	Baghouse and I. D. Fan Lube Oil Building
26	Coal Unloading Tower
27	ISFSI Administration Building
28	BNSF Switch House
29	G-3 Train Shed
30	Boiler Fuel Oil Pump Building
31	Explosive Gas Storage room
32	Flammable Storage Shed
33	Gasoline Tank
34	#6 Well House
35	Main Pollution Control Electrical Building
36	S03 Tank Building
37	Genoa #3 Coal Plant
38	LACBWR Reactor Building
39	LACBWR Turbine Building
40	LACBWR LSA Building
41	G-3 Crib House
42	LACBWR Cribhouse
43	LACBWR Warehouse #2
44	Maintenance Eat Shack
45	LACBWR Stack
46	Waste Treatment Building
47	LACBWR Administration Building
48	LACBWR Warehouse #1
49	Gas Room
50	Sub Station Shed
51	Diesel Fuel Oil Tank
52	G-3 Vehicle Storage Building
53	Ash Coordinators Office
54	Genoa Central Warehouse
55	Fly Ash Electrical Building
56	Bottom Ash Dewatering Building
57	Bottom Ash Dewatering Building
58	Weigh Station
59	Security Station
60	Genoa Central Warehouse
61	Warehouse
62	Genoa Central Warehouse
63	Warehouse
64	Warehouse
65	G-3 Annex Building
66	Waste Silo
67	Genoa Central Warehouse
68	Surge Tank
69	Settling Tank
70	Fly Ash Silo
71	Back-up Control Center
72	Transmission Sub-Station Switch House
73	Sub Station Shed
74	Barge Washing Break Room
75	G-1 Crib House
76	Chimney
77	Duct Work
78	LACBWR Warehouse #3
79	Boat Landing Restroom

Figure 3-2, LACBWR Site Buildings Map

3.2.1 Class 1 Structures

The following is a list of the buildings that are classified as impacted Class 1 structures in which, during operations, radioactive materials and wastes were routinely handled, transferred, and stored within these buildings. These structures contained the: nuclear reactor, primary reactor systems, reactor support systems, nuclear fuel handling and storage systems, turbine and turbine operating related systems, and radioactive waste systems. A vast majority of the current radioactive material inventory at the LACBWR Site continues to be found in these structures:

- LACBWR Reactor Building
- Waste Treatment Building
- LACBWR Stack
- Off Gas Retention Vault
- LACBWR Turbine Building
- LACBWR 1 B Diesel Generator Structure
- LSA Building
- Pipe Tunnel

Throughout facility operations, these structures were either subjected to or at risk of spills of radioactive liquids, the spread of loose surface contamination, and airborne radioactive material. The Class 1 structure survey areas are listed in Table 3-1.

3.2.2 Class 2 Structures

The following is a list of some of the major buildings that have been classified as impacted Class 2 structures and are found on the LACBWR Site. Their primary function is to house the raw water feed systems, or to act as office and/or warehouse space.

- LACBWR Crib House
- G-3 Crib House
- LACBWR Administration Building
- Warehouses 1, 2, and 3

The primary reasons these facilities are classified as potentially impacted facilities are based on the fact that these facilities are in open land buffer areas around the RCA with the potential for contamination translocation due to personnel and equipment movements as well as surface water runoff from the RCA. Additionally, the LACBWR Administration Building used radioactive check sources with radiological instrumentation and, according to former employee recollections; the warehouses stored other nuclear power plant-used equipment in the 1970s before the equipment was shipped off site.

The current decommissioning approach calls for the G-3 Crib House, LACBWR Crib House, and LACBWR Administration Building to remain on site following FSS surveys and the Warehouses 1, 2,and 3 to be surveyed in place for free release and then demolished and the rubble removed from site for disposal as radiologically clean debris.

3.2.3 Non Impacted Area Facilities

The following is a list of some of the major buildings that have been classified as Non- Impacted Class 4 structures and are found on the LACBWR Site. Their primary function is to serve as primary electrical distribution from the LACBWR Site and as facilities associated with the operation of the G-3 Coal Plant. The balance of the facilities is presented on Table 3-1 "Survey Areas for LACBWR Site Structures."

- G-3 Coal Plant
- G-3 Central Warehouse
- G-3 Bag House

The primary reasons these facilities and their associated support facilities are classified as non-impacted facilities are based on the fact that these facilities were not used for LACBWR-related operations. Additionally, the potential for having historically been impacted by windborne transmigration of LACBWR Stack released radioactivity due to predominant at-height meteorological conditions for the LACBWR Site and local area climatology has been evaluated by the 2014 site characterization evaluations of the LACBWR Administration Building Roof and the G-3 Coal Plant Roof as presented in the characterization report [Reference 10-24]. During the 2015 characterization field work the G-3 Coal Plant was also surveyed in areas of high foot traffic with building surfaces that were original construction. The results of these surveys are discussed later in this report.

The current decommissioning approach calls for all Non-Impacted Class 4 structures on LACBWR Site to remain following decommissioning activities. The Non-Impacted Class 4 facilities include all facilities on the Figure 3-2 other than the aforementioned Class 1 and Class 2 structures discussed in Section 3.2.1 and 3.2.2.

Table 3-1, Survey Areas for Class 1,2,and3 LACBWR Site Structures

Survey Unit	Survey Area Description	Classification
B1001101	Reactor Building	Class 1
B1002101	Waste Treatment Building	Class 1
B1003101	LACBWR Ventilation Stack	Class 1
B1004101	Off Gas Retention Tanks/Vault	Class 1
B1005101	LACBWR Turbine Bldg./Turbine Office Bldg.	Class 1
B1006101	LACBWR 1B Diesel Generator Structure	Class 1
B1007101	Low Specific Activity Storage Building	Class 1
B1008101	Pipe Tunnel	Class 1
B2009101	LACBWR Crib House	Class 2
B2009102	G-3 Cribhouse	Class 2
B2009103	LACBWR Admin Building	Class 2
B2009104	Warehouse #1	Class 2

Survey Unit	Survey Area Description	Classification
B2009105	Warehouse #2	Class 2
B2009106	Warehouse #3	Class 2
B3009129	Genoa Back Up Control Center	Class 3
B3009135	Genoa Security Station	Class 3

3.2.4 Class 1 Open Land Areas

Open land areas located inside the LACBWR (LSE) have been classified as impacted Class 1. The basis for this classification is due to either documented incidents of the contamination of surface or subsurface soil by radioactive material in these areas during facility operations or evidence demonstrating existing or likely soil contamination from past LSE area soil sampling studies as documented in the HSA and the 2014 and 2015 characterization field work [Reference 10-24 and this report]. These incidents include the spills of radioactive liquids/resins and radioactive system leakage. The storage of radioactive packages and containers was also part of the operational history of the LSE area.

Figure 3-1, "Survey Units for LACBWR Site Open Land Areas," lays out the open land survey unit areas in their entirety on the LACBWR Site.

3.2.5 Class 2 Open Land Areas

The open land areas as shown on Figure 3-1 and identified as L2011102 and L2011101 have been classified as impacted Class 2. This classification is selected based on the fact that this survey unit is a buffer area around the LSE with the potential for contamination translocation due to personnel and equipment movements as well as surface water runoff from the LSE.

3.2.6 Class 3 Open Land Areas

The open land areas as shown on Figure 3-1 and identified as L3012109 and L3012110 have been classified as Class 3. Historical information contained in the HSA as well as the 2014 and 2015 characterization field work [Reference 10-24 and this report] indicates that the presence of residual radioactivity in concentrations in excess of the unrestricted release criteria is not expected.

3.3. Class 4 Open Land Areas

The open land areas as shown on Figure 3-1 and identified as L4012101 thru L4012108 have been classified as Non-Impacted areas (Class 4) with no potential for residual contamination in excess of the unrestricted release criteria. This classification is based on the HSA and characterization surveys conducted during the 2014 and 2015 characterization field work [Reference 10-24 and this report]. The Non-Impacted Class 4 survey areas are listed in Table 3-2.

Table 3-2, Survey Units for LACBWR Site Open Land Areas

Survey Unit	Survey Area Description	Classification	Area in m^2
L1010101	Reactor Building/WTB Area	Class 1	1,992
L1010102	Turbine Building Area	Class 1	2,315
L1010103	LSA Building/Start Up Switchyard Area	Class 1	1,749
L1010104	North End of LSE Area	Class 1	2,387
L2011101	Grounds North of LSE Fence	Class 2	7,211
L2011102	Grounds South of LSE Fence	Class 2	6,785
L3012109	Grounds Area East of LSE Fence	Class 3	27,500
L3012110	Grounds Area Surrounding L2011101 and L2011102 Area	Class 3	4,505
L4012101	North end of site and outside of switchyard	Class 4	18,246
L4012102	Switchyard	Class 4	11,711
L4012103	G-3 Coal Plant and related facilities area grounds	Class 4	66,869
L4012104	Grounds west of railroad right of way and east of survey units	Class 4	4,804
L4012105	Coal Pile area grounds	Class 4	82,894
L4012106	Capped ash impoundment ground area w/o ISFSI controlled area	Class 4	111,899
L4012107	Grounds across Highway 35 to east	Class 4	81,254
L4012108	Right of Ways -Hwy 35/Railroad	Class 4	9,444

4.0 Data Quality Objectives

4.1. Radiological Data Quality Objectives

The characterization surveys at LACBWR were designed to gather the appropriate data using the DQO process as outlined in NUREG-1575 (MARSSIM), Appendix D – The Planning Phase of the Data Life Cycle. This process is an integral part of the planning and design steps for the characterization survey. The DQO process involves a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. It is flexible such that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. It is also iterative, allowing for the incorporation of new data and modification of the output of previous steps to act as input in subsequent steps.

The specific objectives for the characterization surveys were defined for each Survey Unit and addressed in the survey packages and survey and sampling instructions. To support further development of DCGLs and FSS planning, additional information is needed and the DQO process will ensure that appropriate, valid radiological data is obtained. Characterization data collection and evaluation included measurements of radiation exposure rates, direct surface contamination, removable surface contamination, volumetric contamination levels (for certain structural components and soil) and radionuclide analysis at on-site and off-site laboratories.

The seven steps of the DQO process are outlined in the following sections and described in more detail in the LACBWR Site Characterization Plan, PG-EO-313196-SV-PL-001 [Reference 10-23].

4.1.1 Problem Identification

Based on previous operations at LACBWR and storage of radioactive materials it has been determined that radiological contamination exists and that the facility may require remediation in order to meet the unrestricted release criteria of 10 CFR 20.1402, "*Radiological Criteria for Unrestricted Use"* [Reference 10-7]. Based upon this criteria, the site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mRem per year, including from groundwater sources of drinking water.

The problem associated with radiological characterization is to perform characterization inspections and surveys of sufficient quality and quantity to determine the nature, extent and range of radioactive contamination in each survey unit. According to U.S. NRC Regulatory Guide 1.179, "*Standard Format and Content of License Termination Plans for Nuclear Power Reactors*" [Reference 10-12] the site characterization should be sufficiently detailed to allow the NRC to determine the extent and range of radiological contamination of structures, systems (including sewer systems, waste plumbing systems, floor drains, ventilation ducts, and piping and embedded piping), rubble, and paved parking lots (both on and beneath the site)." For land areas, radiological data is also obtained for soil, concrete, and asphalt areas.

Characterization data is required to evaluate the radiological contaminants present and the extent of the radiological contamination for structures, systems, and open land areas which will remain following license termination. The data may be used to develop a conceptual site model for use in deriving site specific DCGLs. The data will also likely be useful in defining future measurement and sampling protocols that will be employed for remedial action surveys and FSS.

The approach for demonstrating that the site meets the criteria for unrestricted release will depend on some factors that have yet to be fully investigated including final DCGLs, the defined End State condition of the site at the time of the FSS, the statistical tests to be employed, etc. The approach for demonstrating that the site meets the criteria for unrestricted release will be discussed in detail in the LTP Chapter 5 – Final Radiation Survey Plan and Chapter 6 – Compliance with the Radiological Criteria for License Termination.

4.1.2 Decision Identification

During radiological characterization, an important step in the DQO process is decision identification. This step consists of developing a decision statement, or in most cases, several decision statements, based on a principal study question (e.g. the stated problem) and determining alternative actions that may be taken based on the answers. For each survey unit, each of the characterization objectives must be assessed with regards to their applicability to the end state of each specific survey unit. These objectives included:

- Providing a basis for the survey unit classification;
- Confirming the expected ROCs and determining the relative distribution fractions;

- Providing a basis for surrogate relationships for HTD radionuclides;
- Providing a basis for the extent of remediation of surface and subsurface soils;
- Collecting data to support future planning of remediation, decontamination, and waste management operations and,
- Support the protection of workers, the general public, and the environment.

4.1.3 Inputs to the Decision

This step in the DQO process identifies the types and quantity of information necessary to address the different decisions which are identified in the previous steps. The information required depends on the type of media under consideration (e.g. soil, sludge, concrete, asphalt, etc.) and the adequacy of existing data. If new data is need, then the type of measurement (e.g. scan measurement, direct measurement, and sampling) will be determined in the next step. Initial sources of information that has been utilized for determination of the necessary inputs include:

- Historical site information, including incidents and evidence of previous radioactive material storage;
- ROC assessments;
- Survey unit classification and basis;
- Action Levels associated with decision objectives;
- Instrumentation and MDC values;
- Laboratory counting and analytical requirements; and,
- QC sample requirements.

4.1.4 Boundaries of the Study

This step of the DQO process typically includes identification of the target population and material of interest, the spacial boundaries, and other constraints for collecting the data (e.g. weather conditions and impact on personnel and instrumentation; physical obstacles or work interferences, etc.). The target population for characterization tasks for this Plan is the set of scan, direct, or sample measurements from the Survey Unit. The media of interest is the type of materials that will be surveyed or sampled (e.g. soil, sludge, and sediment, asphalt, concrete). The spacial boundaries to be defined include the entire area of interest within Survey Units, including area dimensions and depth of soil, depth of concrete, etc.

The characterization work at LACBWR did involve Open Land Class 1,2,3,and 4 Survey Units as applied to the nearby paved and unpaved grounds as well as the Administration Building in the Class 2 Survey Unit. The focus for the Class 3 and 4 Survey Units was various paved and unpaved grounds that are considered part of the LACBWR licensed area. Additionally, concrete coring was performed in Class 1 facilities to obtain radiological information (by radionuclide and depth) to assist with developing ratios of ROCs.

4.1.5 Decision Rule

This step of the DQO process develops the binary type statement that presents a logical process for choosing among alternative actions. Making decisions is facilitated by developing a clear statement using the “If…then…else” format. For characterization surveys, this process often

involves selecting a workable Action Level (e.g. some level of contamination defined in dpm/100 cm^2 or pCi/g, typically associated with meeting desired MDC criteria) and developing a decision rule which involves comparison with the Action Level. Depending upon the objectives of a specific characterization survey task, there could be a number of decision statements.

One of the key evaluations involved comparison of the surface measurements and soil sampling results from within the impacted areas to the preliminary criteria noted in Table 2-3. If radionuclides other than those shown in Table 2-1, were identified they were compared with the default screening values in NUREG 1757, Volume 2, Appendix H [Reference 10-5].

When appropriate, material specific background measurements or activity concentrations were obtained to aid in the evaluation of material specific measurements and sample analysis results.

4.1.6 Limits on Decision Error

This step of the DQO process often involves statistical hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis when characterization data is to be considered for FSS purposes. However, site characterization surveys are more of an exploratory nature versus the verification phase of the FSS. Therefore, decision errors are more subjective during the characterization process and the use of descriptive statistics is more appropriate.

The decision errors were limited by performing measurements, smears, and sampling activities in accordance with the LACBWR Site Characterization Plans [Reference 10-23] and this characterization report] and the corresponding survey packages which specified the number of measurements and samples to be collected, the sample locations, the amount of sample to be collected, chain-of-custody requirements for each sample, sample preparation, the type of analyses, and the MDC for each of the analyses.

Split samples were collected to monitor the accuracy of the on-site laboratory, with designated QC samples also sent to a qualified, licensed off-site laboratory.

In addition to the samples specified in the survey package, additional biased measurements and samples, as dictated by the professional judgment of the Project Health Physicist, were collected to support the characterization of specific areas.

4.1.7 Design for Data Collection

The first six steps of the DQO process provide information that supports optimizing the plan for data collection. The final step is to use this information to establish an adequate survey design.

Both random and biased measurements and samples were collected as part of the Characterization Plan. Random samples were located based at pre-defined locations. The biased samples or judgemental samples were based in part on the results of gamma/beta scans or direct readings performed in the area to be sampled and/or based on the judgment of the Project Health Physicist.

The characterization surveys performed complied with the LACBWR Site Characterization Plan for sufficient sensitivity, accuracy, reproducibility, and are well documented. Implementation of the LACBWR Site Characterization Plan and associated procedures, including the preparation of characterization survey packages, ensured characterization survey quality.

To ensure data collection was optimized, all areas to be surveyed were walked down as part of the characterization survey package development. Minimum data requirements were defined, special situations identified, specific instructions provided, etc.

Quality control of instrumentation included efficiency checks, source checks, and background checks.

Where possible, the MDC for each measurement or sample was less than 50% of the Action levels for the expected ROCs.

Chain-of-Custody was maintained for all samples that were analyzed off-site. All collected samples were archived following analysis.

4.1.8 Project Organization

Energy Solutions established the LACBWR Characterization Project organization with sufficient management and technical resources to fulfill all project objectives and goals. The LACBWR Characterization Project organization was responsible for LACBWR Site characterization.

Characterization encompassed all survey and sampling activities related to the characterization plan. This included characterization surveys and release type surveys. The duties and responsibilities of key Energy*Solutions* managers, as well as the various key positions within the characterization group as they pertained to the project implementation, are described below. Responsibilities for each of the positions were assigned to a designee as appropriate.

<u>Project Manager</u>:

The EnergySolutions Project Manager reports to the EnergySolutions EVP Reactor D&D Projects for:

- The overall responsibilities for all on site activities.
- The management of personnel assigned to the LACBWR Site Characterization Project.
- Ensuring all contractual obligations as they pertain to characterization is satisfied.
- The review and approval of project plans and procedures.
- All supporting documents that are subject to controlled distribution requirements are maintained properly.
- Ensure that activities conducted in accordance with the QAPP.
- Approving personnel access to characterization file cabinets and computer data bases.

<u>Project Health Physicist</u>:

The Energy*Solutions* Project Health Physicist reports to the Energy*Solutions* Project Manager and Energy*Solutions* Characterization/License Termination Manager for:

- Development and approval of characterization survey packages and sample plans.
- Development and approval of characterization final reports.
- Resolving and documenting any survey design, instructions, or performance discrepancies.
- Control and implementation of survey packages and sample plans as received and to ensure that all quality objectives are met/documented.

- Perform data review, verification, and validation.

Health Physics Technicians:

The Energy*Solutions* and contractor Health Physics Technicians report directly to the Characterization/License Termination Manager and are responsible for understanding and implementing the requirements of the surveys as prescribed in the LACBWR Characterization Survey Plan. The technicians are responsible for the acquisition and documentation of survey data and collected samples.

Quality Engineer (off site position):

The Energy*Solutions* Quality Engineer reports to the Energy*Solutions* Project Manager and ensures activities affecting quality are performed satisfactorily and to provide support as needed.

5.0 Radiological Instrumentation and Laboratory Analysis

5.1. Instrumentation

The selection and use of survey instrumentation ensured that their sensitivities were sufficient to detect the identified radionuclides of concern at the desired (MDC) requirements. Table 5-2 provides a list of the instruments used during the characterization survey.

The Ludlum Model 2350 Data Logger is a portable microprocessor computer-based counting instrument. The data logger is designed to operate with a wide variety of detectors. It was used in combination with 2" x 2" sodium iodide detectors for obtaining count rate or exposure rate measurements on soils.

The Ludlum Model 2360 Data Logger was used with the dual phosphor scintillation detectors to obtain direct (static) measurements of alpha and beta activity and for performing beta scans on surfaces.

Analysis for removable alpha and beta activity was performed using the DPC Protean alpha/beta counter.

An Exploranium GR-135 portable multi-channel analyzer was used to screen soil samples in the land areas for the presence of natural radionuclides or radionuclides of interest.

An on-site calibrated gamma spectroscopy system was used for soil and concrete sample analysis.

An off-site lab (Test America) was used to conduct gamma spectroscopy analyses of split soil samples as well as concrete cores analyses. The off-site lab also analyzed submitted soil and concrete samples for (HTD) nuclides and transuranic radionuclides.

As presented in the Energy*Solutions* Technical Basis Document for the Radionuclides of Concern [Reference 10-4], the ROCs remaining at the LACBWR Site is as shown on Table 5-1. The table below lists these radionuclides including half-lives, major radiations and intensities.

Table 5-1, Radionuclides of Concern

Radionuclide	Half Life (Years)	Major Radiations Energies and Intensities		
		Alpha	Beta (average)	Gamma
H-3	1.24E01		5.685 keV	
C-14	5.73E03		49.47 keV	
Fe-55	2.70E0			Low energy x-rays
Co-60	5.27E0		95.79 keV	1173 keV 100% 1332 keV 100%
Ni-59	7.50E04			Low energy x-rays
Ni-63	9.60 E01		17.13 keV	
Sr-90/Y-90	2.91 E01		195.8 keV /934.8keV	
Nb-94	2.03E04		145.9 keV	702 keV 100% 871 keV 100%
Cs-137	3.0E01		156.8 keV 94.6% 415.2 keV 5.4%	
Eu-152	1.33E01		300.8 keV	121.8 keV 28.4% 964 keV 14.4% 1085.8 keV 10% 1112 keV 13.3% 1407 keV 20.7% 344 keV 26.5% 778.9 keV 12.7%
Eu-154	8.80E0		225.4 keV	123.1 keV 40% 1274 keV 35.5%
Np-237	2.140E06	4.8 MeV (average)	70keV	35keV (average)
Pu-238	8.78E01	5500keV 72% 5460keV 28%		
Pu-239	2.41E4	5104 keV 11.5% 5142 keV 15.1% 5155 keV 73.3%		
Pu-240	6.60E03	5170keV 76% 5120keV 24%		
Am-241	4.32E02	5443 keV 13% 5486 keV 85%		59.5 keV 35.9%

Radionuclide	Half Life (Years)	Major Radiations Energies and Intensities		
Am-243	7.37E03	5280 keV 87% 5230 keV 12%		55 keV (average)
Cm-243/244	1.81E01 (shortest half-life)	5810keV 77% 5770 keV 23%		130keV (average)

Table 5-2, Survey Instrumentation

Detector Model [2]	Meter Model	Application	Nominal Detection Sensitivity	
			MDC_{scan} (dpm/100cm^2)	MDCdirect[1] (dpm/100cm^2)
Exploranium	\ GR-135 Plus	Gamma Isotope ID		
Ludlum 43-93 β mode	Ludlum 2360	β direct & scan	1800-2000 2000-2200	800-900 800-900
Ludlum 43-93 α mode	Ludlum 2360	α direct	N/A	90-100 90-100
Ludlum 44-10	Ludlum 2350-1	γ scan	3.5 pCi/g ^{60}Co 6.5 pCi/g ^{137}Cs	N/A
Protean Counter		α and/or β smear analysis	N/A	α –10-20 β – 80-90
Gamma Spectroscopy System	N/A	γ Analysis	N/A	~0.10 pCi/g for Cs-137 for soils

1. Based on 1-minute count time and use of Cs-137 energy max for surface efficiencies, $\in_{s,}$ as specified in International Standard, ISO 7503-1 [Reference 10-13].
2. Functional equivalent instrumentation may be used.

NOTE: Based on the current version of Energy*Solutions* Procedure CS-FO-PR-001, "Performance of Radiological Surveys" [Reference 10-14] and NUREG 1507 "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions" [Reference 10-15].

5.2. Calibration

The Data Loggers, associated detectors and all other portable instrumentation are calibrated on an annual basis using National Institute of Technology (NIST) traceable sources and calibration equipment. Procedures for calibration, maintenance, operation and quality control implement the appropriate guidance established in American National Standard Institute (ANSI) standards. Calibration typically includes:

- High voltage calibration
- Discriminator/threshold calibration
- Window calibration
- Alarm operation verification
- Scalar calibration verification

Detector calibration includes:

- Operating voltage determination
- Calibration constant determination
- Dead time correction determination

Calibration labels showing the instrument identification number, calibration date, and calibration due date were attached to all instruments. All instrumentation was inspected and source response checked daily before use to verify calibration status and proper operation. Control charts and/or source check criteria were established prior to the initial use of the instrument including the calculation of instrument efficiencies and surface efficiencies for portable instruments based on the requirements in the Energy*Solution*s procedure CS-FO-PR-001, "Performance of Radiological Surveys" [Reference 10-14].

The receipt, inspection, issue, control, and accountability of portable radiological instrumentation used for characterization or release surveys was performed in accordance with issue and control procedures as addressed in Section 5.0 of the Quality Assurance Project Plan-LACBWR LTP Development Site Characterization and Final Radiation Survey Projects (QAPP) [Reference 10-11].

5.3. On-Site Sample Analysis

At the LACBWR site, a DPC Canberra gamma spectroscopy system with a High Purity Germanium (HPGe) detector was set up and calibrated to analyze soil, asphalt samples, concrete pucks, sediment, and liquid/sludge type samples. The analysis library included Co-60, Cs-137, and other fission and activation products to ensure the ROCs [Reference 10-4] could be flagged, if present, during analysis. All samples collected were uniquely numbered and tracked through the analysis process. All soil, asphalt, and sediment samples were dried prior to analysis. Analytical results and daily QC runs of the gamma spectroscopy system were reviewed by system operator and the Radiological Protection Supervisor daily to ensure system was operating satisfactorily and that the Cs-137 MDCs were being maintained at </= to 1.0 pCi/g for soils and 2.0 pCi/g for concrete.

On-site sample types and quantities are discussed in greater detail in Chapter 7 of this document.

5.4. Off-Site Sample Analysis

A representative population of soil and concrete was sent to an off-site lab for analysis for (HTD) and transuranic analyses based on the ROCs. In addition, a representative population of samples analyzed on-site by gamma spectroscopy was sent to the off-site lab for duplicate analysis by gamma spectroscopy. Off-site sample types and quantities are discussed in greater detail in Chapter 7 of this document.

All samples were uniquely numbered. All soil samples were dried prior to analysis excepting those samples for C-14 or tritium analysis. Offsite samples were tracked using chain of custody records, packing lists for transportation, laboratory verification of receipt, and laboratory tracking during analyses. A summary of lab methods used for analyses and MDCs met for those analyses is included in Table 5-3 Analytical Methods and Typical MDCs.

Table 5-3, Analytical Methods and Typical MDCs

Analysis	Technique	(Soils)/[Concrete] Method/MDC(pCi/g)	Water Method/MDC(pCi/L)
Gamma Radionuclides Cs-137	Gamma Spectroscopy	DOE EML HASL 300/(<0.1)/[<0.1]	EPA Method 901.1/10.0
Isotopic Neptunium (Np-237)	Alpha Spectrometry	DOE EML HASL 300/(<0.4)/[<0.4]	
Isotopic Plutonium (Pu-238/239/240)	Alpha Spectrometry	DOE EML HASL 300/(<0.4)/[<0.4]	EML Pu-02 Modified/ 1.0
Isotopic Thorium (Th-228/230/232)	Alpha Spectrometry	DOE EML HASL 300/(<0.4)	
Isotopic Uranium (U-234/235/238)	Alpha Spectrometry	DOE EML HASL 300/(<0.4)	
Isotopic Curium (Cm-243/244)	Alpha Spectrometry	DOE EML HASL 300/(<0.4)/[<0.4]	
Isotopic Americium (Am-241/243)	Alpha Spectrometry	DOE EML HASL 300/(<0.4)/[<0.4]	
C-14	LSC	EPA EERF C-01/(<5.0)/[<5.0]	ENIC Modified/50.0
Tritium	LSC	EPA 906.0/(<10.0)/[<10.0]	EPA 906.0/300.0
(Sr-90) [Beta Strontium]	(GFPC) [Extraction Chromatography-GFPC]	DOE EML HASL 300/(<1.0)[<1.0]	EICHROM Modified Method /2.0
Pu-241	LSC	DOE EML HASL 300/(<10.0)/[<10.0]	EML Pu-01/50.0

Analysis	Technique	(Soils)/[Concrete] Method/MDC(pCi/g)	Water Method/MDC(pCi/L)
Tc-99	LSC	DOE EML HASL 300/(<2.0)/[<2.0]	EICHROM Tc-01/15.0
Gross Alpha and Beta	Gas Flow Proportional		EPA 900.0/5.0
Fe-55	LSC	DOE EML HASL 300/(<10.0)/[<10.0]	EML Fe-01 Modified/100.0
Ni-63	LSC	DOE EML HASL 300 (SRW01)/(<2.0)/[<2.0]	EML Ni-01 Modified/15.0
Ni-59	LSC	DOE EML HASL 300 (SRW01)/(<50.0)/[<50.0]	EML Ni-01 Modified/100.0

6.0 Minimum Detectable Concentration

The MDC is dependent on count times, geometry, sample size, media type, detector efficiency, background, and for scanning, the scanning rate and the efficiency of the surveyor.

Typically, the MDCs were to set to see less than 50% of the preliminary limits specified in Table 2-2, where possible for hand held instrument surveys. The desired MDCs were determined prior to analysis of soil, asphalt, concrete, sediment, and sludge/liquid samples and were included in the purchase order requirements for the offsite samples. The desired MDCs were verified as obtained on site during the review of the individual sample gamma spectroscopy results with the onsite equipment.

The equations used for calculating the MDC for direct measurements and smears are obtained from the formulas noted in procedure CS-FO-PR-001, "Performance of Radiological Surveys" [Reference 10-14].

7.0 Survey Design and Implementation

To facilitate the characterization survey, the areas to be surveyed were divided into survey unitsas outlined in the respective survey package. The characterization surveys included beta scans, measurements for total alpha and total beta activity, smears for determining the presence of removable alpha and removable beta activity, gamma scans and exposure rate measurements. In addition, representative structural material and soil samples were collected for on-site analysis by gamma spectroscopy and for off-site analysis. The areas and materials to be surveyed or sampled in each survey unit were determined by the professional judgment of the Project Health Physicist during the survey design process based upon the review of historical and current radiological data at LACBWR, process knowledge, and goals of the characterization surveys. The HP technician also made judgmental decisions on sampling and surveying based on field indications and observations during the actual field work.

The areas and materials surveyed and/or sampled in and around the LACBWR Site as part of the characterization survey included:

- Concrete in the Reactor Building, Waste Treatment Building, and Pipe Tunnel;

- Soils under the Turbine Building
- Soils for tritium activity in survey units L4012101 and L4012106
- Gamma walk over scans of all Non Impacted Class 4 areas as shown on Figure 3-2 other than L4012102
- Direct and removable contamination surveys of representative high personnel traffic areas of the G-3 Coal Plant which were original construction facility areas.

7.1. Survey Unit Preparation for Characterization

Preparation for characterization was performed in all survey units as deemed appropriate and practical. Prior to performing characterization surveys on structural surfaces, the areas were cleared of all loose equipment and materials to the extent possible. All physical hazards in the survey unit were either identified and removed or marked as appropriate.

The measurement locations were marked on the building surface or equipment and/or on a map or drawing prepared to document the characterization measurement locations.

For open land survey units, reference coordinates were established using a Global Positioning System (GPS) coupled with a standard topographical grid coordinate system such as the North American Datum (NAD) system. Non Impacted (Class 4) survey units and the sampling locations were located in the field with the GPS and marked for survey work.

7.2. Survey Package Development

A Characterization Survey Package was developed for each survey area to be included in the characterization survey. A total of five survey packages were generated for the work performed in the 2015 characterization site work.

Specific survey instructions/work sheets were prepared for each survey unit detailing the survey requirements and providing instructions for completing the survey. The survey instructions described the number, type and location of scan and direct measurements, smears, and samples to be collected as well as the type of analyses to be performed. Direction was also provided for selection of instruments, count times, instrument modes, survey methods, required documentation, actions levels, investigation actions, background requirements and other appropriate instructions. In conjunction with the survey instructions, survey data forms, indicating desired measurements, were prepared to assist in survey documentation.

Each individual survey plan contained the following types of information:

- Detailed description of the survey unit,
- Photographs, maps, and/or drawings of the survey unit,
- A summary of the operational history pertinent to the survey unit and summary data from any previous radiological surveys if available,
- The specific DQO(s) for the survey unit,
- Types and number of survey measurements and/or samples prescribed for the survey,
- Specific survey instructions,
- Survey measurement and sample designation codes and locations,

- Quality Assurance measures in accordance with the QAPP requirements, and,
- Any pertinent information such as support from others, health and safety information and necessary Work Orders (e.g. for coring, drilling, access, lifting, etc.) and permits (e.g. Excavation Permit, Radiation Work Permit (RWP), etc.).

The survey packages were the primary method of controlling and tracking the survey results. Survey records, including the sample analysis results are maintained in the survey packages.

7.3. Survey Unit Walk Down

Survey plan development began with the performance of a walk down of the survey unit. During the walk down, details regarding the physical survey area were compiled such as the surface(s) in the unit (location, size and material makeup of wall, floor, ceiling, surface soils, etc.). Data from available operational surveys were reviewed and utilized as appropriate.

Significant health and safety concerns include the potential industrial hazards commonly found at a construction site, such as exposed electrical lines, excavations, enclosed work spaces, hazardous atmospheres, insects, unstable or sharp surfaces, hot and cold temperature extremes, tripping hazards, vehicle traffic, and working at heights. The pre-survey walk down identified potential industrial safety hazards specific to the survey unit.

Each hazard was evaluated to determine if the hazard could be eliminated, avoided, or minimized, as well as to determine if the need for additional outside support/expertise was necessary to complete further evaluation or mitigation.

7.4. Survey Design and Protocols

Characterization surveys were designed and performed in accordance with the LACBWR Site Characterization Survey Plan [Reference 10-23] as well as the QAPP [Reference 10-11] and applicable approved procedures (e.g. procedures CS-FO-PR-001, CS-FO-PR-002, CS-FO-PR-003, and CS-FO-PR-004 [Reference 10-14, 10-16, 10-17 and 10-18] as well as the Energy*Solutions* instrument-specific operating procedures).

7.4.1 Determine the Number of Direct Measurements and/or Samples

The number of measurements and/or samples that were taken in each survey unit was based upon the LACBWR Site Characterization Plan [Reference 10-23] and the respective survey package requirements. The number of measurements and/or samples that will be taken in each survey unit will be based upon this Characterization Plan as well as reflecting the population size necessary to satisfy the specific DQOs for each survey unit.

For the characterization of impacted Class 1, 2 and 3 structures, the number of static measurements and/or samples will be based upon the DQO (e.g. confirming extent of radiological contamination that may need remediation; obtaining additional radiological information to confirm the presence and ratios of ROC nuclides; etc.) and the professional judgment of the Radiological Protection Supervisor and/or the Project Health Physicist, dependent upon existing radiological conditions and background levels, accessibility to structures expected to be remaining and subject to FSS, radiological/ALARA exposure and industrial hazard considerations, and other factors.

For Class 4 open land areas and facility areas, sufficient random or biased measurements or samples, at the professional judgment of the Radiological Protection Supervisor and/or the Project Health Physicist will be taken to assess the initial classification and determination that no licensed radioactive material is present in these areas.

7.4.2 Determine the Scan Coverage

Survey units were scanned in accordance with their classification. The area to be scanned in each survey unit was determined by the professional judgment of the Characterization/License Termination Manager and/or the Project Health Physicist during the survey design process, including information obtained during the initial survey unit walk downs and on site ambient radiation conditions.

7.4.3 Performance of Direct Measurements

Direct measurements were performed to locate areas of residual radioactivity by detecting total levels of contamination on structural surfaces of the G-3 Coal Plant. These measurements were performed using the Model 2360 with 43-93 detector.

Direct measurements were conducted by placing the detector within one quarter inch of the surface to be counted and acquiring data over a one minute count time for both beta/gamma and alpha contamination. A total of (10) direct measurements were taken in the G-3 Coal Plant Areas based on the results of the beta scan surveys. .

Health Physics Technicians monitored the visual and audible responses to identify locations of elevated activity that required further investigation and/or evaluation.

7.4.4 Performance of Beta Surface Scans

Scanning was performed in order to locate elevated areas of residual activity. Beta scans were performed over concrete surfaces of the G-3 Coal Plant including floors and walls up to six feet as well as tools and equipment.

Beta scanning was performed with the detector positioned within one half inch of the surface and with a scanning speed of one detector active window width per second. Approximately 5% of the G-3 Coal Plant areas of interest concrete surfaces were beta scanned.

Health Physics Technicians monitored the visual and audible responses to identify locations of elevated activity that required further investigation and/or evaluation.

7.4.5 Performance of Gamma Surface Scans

Gamma scans were performed over grounds areas in the Non-Impacted Areas (Class 4) as depicted in Figure 3-1 to identify locations of residual surface activity. Sodium iodide (NaI) gamma scintillation detectors (2” x 2”) coupled to the Model 2350 was used for these scans.

Scanning was performed by moving the detector in a serpentine pattern, while advancing at a rate of about 0.5 m per second with the distance between the detector end cap and the surface maintained at less than six inches. Visual and audible signals were monitored and locations of elevated levels were flagged for further investigation and/or sampling. In each Non-Impacted survey unit the goal was to gamma scan at least 1% of the total survey unit area.

7.4.6 Removable Surface Contamination

Removable beta and alpha contamination surveys were performed in the G-3 Coal Plant to evaluate the presence of residual loose surface contamination. A smear for removable activity was taken at each direct measurement location on structural surfaces. A 100 cm^2 surface area was wiped with a circular cloth or paper filter, using moderate pressure. Smears were analyzed for the presence of gross beta and/or gross alpha activity. This was done using a dual channel gas proportional system.

7.4.7 Concrete and Core Sampling

Concrete core boring and sample collection were conducted to gather information to support the development of ROC ratios and to evaluate contamination in depth in structural surfaces. Concrete coring and associated puck cutting were conducted in the representative areas of the Reactor Building, Waste Treatment Building, and Pipe Tunnel. Coring sampling of concrete involved the use of a three inch diameter diamond bit core drill.

The nominally 4-6 inch long concrete core produced by the three inch diameter concrete coring machine was cut with a concrete saw to gather one half inch thick pucks . Beta scan measurements were performed on the top and bottom of the pucks with a Model 2360 with 43-93 to determine contamination intrusion depth and/or the activation of the material. The concrete pucks were transported to the onsite laboratory and had gamma spectroscopy measurements taken on the first puck at surface level on both sides and continued sequentially until the gamma spectroscopy unit demonstrated that the puck had results that were less than <MDC. Pucks that exceeded MDC were then selected by the Project Health Physicist and the Characterization/License Termination Manager for offsite radiological analysis for gamma spectroscopy, HTD, and transuranic isotopes.

7.4.8 Soil Sampling

Samples of soil were obtained from underneath the Turbine Building by use of a Geoprobe operating with angled boring and selecting five individual locations either on the north or south side of the facility. Samples were taken at depths of ten, fifteen, and twenty feet below the facility at each individual location.

A sufficient amount of material was collected at each location based on the requested analyses. The amount of material needed was also defined by the requirement for split sampling and the volumes needed for onsite gamma spectroscopy and the off-site laboratory for the analyses including gamma spectroscopy, HTD, and transuranic isotopes. Sample preparation included the removal of extraneous material and the homogenization and drying of the sample for onsite analysis. It should be noted that drying was not performed for collected off site samples for C-14/H-3 analysis. Separate plastic containers were used for each sample and each container was accounted for throughout the sampling and analysis process. Samples were split as acquired which included one soil sample from each individual sample location with the sample showing the highest gamma scan result.

In survey units L4012101 and L4012106 three additional soil samples at depths of surface and one meter each were collected by Geoprobe operations for each survey unit.

The total of (12) collected soil samples were submitted off site for tritium analysis only. These samples were collected in areas of the respective survey unit where during the 2014 site

characterization work [Reference 10-24] slightly elevated tritium levels in soil were identified. It should be noted that drying was not performed for collected off site samples for C-14/H-3 analysis.

7.5. Survey Implementation

When a survey package was approved and prior to implementation, the Project Manager performed a pre-survey briefing with the Health Physics Technicians who performed the survey. During the briefing, the survey package and its specific survey instructions were reviewed. The survey package typically addressed the following tasks:

- Survey instrument set up,
- Check source and background radiation evaluation before and after each shift to ensure proper instrument operation,
- Performance of preliminary inspections of the survey area to identify any additional pre-survey support requirements or any additional specific survey instructions,
- Locating and marking all direct measurement and/or sample locations using the coordinates or directions provided in the survey instructions,
- Acquisition of survey measurements and analyses of samples using appropriate calibrated instruments,
- Documentation of survey measurements and sample analysis data collected,
- Review of the survey results to identify any areas exceeding the specific action levels.
- Review of the completed survey packages and ensuring all required surveys have been performed and samples collected.

8.0 Quality Assurance and Quality Control

Energy*Solutions* possesses a USNRC-approved QA Program; the EnergySolutions QA Program meets the requirements of 10CFR50 Appendix B [Reference 10-19], ASME NQA-1[Reference 10-20], 10 CFR 71 Subpart H [Reference 10-21], and 10 CFR 72 subpart G [Reference 10-22]. Energy*Solutions*' QA/QC Programs ensure that all quality and regulatory requirements are satisfied. All activities affecting quality were controlled by written plans and procedures, including a (QAPP) [Reference 10-11]. The following QA measures were an integral part of the characterization survey.

8.1. Selection of Personnel

All management and supervisory personnel involved with the acquisition of characterization data have had experience in performing characterization surveys at nuclear or reactor facilities, and in the implementation of NUREG-1575 (MARSSIM). All personnel were familiar with the requirements of the characterization survey plan, instrumentation to be used and all implementing procedures. All project personnel have received satisfactory and documented radiation worker training by the DPC Radiation Protection Manager prior to commencement of field characterization work at LACBWR Site.

8.2. Training

In addition to radiation worker training, all on-site project personnel received site-specific training that included applicable site emergency procedures and safety orientation. All personnel reviewed all plans and procedures to be used in implementing the characterization survey. Acknowledgement sheets were used to document the review.

8.3. Written Plans and Procedures

All activities affecting quality were controlled by written plans and procedures. This included the LACBWR Site Characterization Plan, a project-specific (QAPP) and key implementing procedures such as the following:

- CS-FO-PR-001, Performance of Radiological Surveys [Reference 10-14],
- CS-FO-PR-002, Calibration and Maintenance of Radiological Survey Instruments[Reference 10-16],
- CS-FO-PR-003, Soil Surveys; Collection of Water, Sediment, Vegetation and Soil Samples; and Chain-of-Custody Procedure[Reference 10-17], and,
- CS-FO-PR-004, QA/QC of Portable Radiological Survey Instruments [Reference 10-18].

8.4. Survey Documentation

Hard copies of all survey results were maintained in survey packages. A separate survey package was maintained for each survey area of interest. As applicable, each measurement and sample result was identified by date, technician, instrument type and serial number, detector type and serial number, location code, etc. All completed survey packages were reviewed by at least two individuals including the Characterization/License Termination Manager and the Project Health Physicist to ensure the package was complete and that the results adequately characterized the survey area. All onsite and off-site laboratory analysis results were reported in appropriate units and electronic records of results were maintained on a dedicated internal SharePoint website with a hard copy of results kept with survey package records.

8.5. Chain-of-Custody

All samples sent off-site for analysis were accompanied by a chain-of-custody (COC) record to track the location of the sample and ensure that each sample received the appropriate analysis. Upon receipt of sample analysis results, the results were compared to the COC records to ensure all samples were analyzed and that the correct analyses were performed. All analyzed samples were archived for long term retention at a location on the LACBWR Site.

8.6. Data Validation

Characterization survey measurement and/or analysis results were reviewed to ensure that the survey was complete, fully documented, and technically acceptable. The QAPP provides the directions for conducting the data validation [Reference 10-11].Validation ensured that the data set was comprised of qualified measurement results collected in accordance with the survey design, which accurately reflected the radiological status of the survey unit. The review criteria for data acceptability included the following items:

- Verifying compliance with survey instructions as specified in the survey package or noting the basis for any changes in implementation.
- Verification that the MDCs were appropriate for the instruments and techniques used to perform the survey.
- Verification that the instrument calibration was current and traceable to NIST standards.
- Verification that the survey methods used to collect data was proper for the types of radiation involved and for the media being surveyed.
- Verification that the data set was comprised of qualified measurement results collected in accordance with the survey design, which accurately reflected the radiological status of the survey unit.
- Verification that the data has been properly recorded.

If the data review criteria was not met, then the Project Health Physicist was informed. The discrepancy was reviewed and the decision to accept or reject the data was documented in the survey package.

8.7. Data Evaluation and Review

Direct beta-gamma and alpha measurements and removable contamination samples collected during the characterization surveys were compared against the Table 2-2 criteria for surfaces, structures, and paved areas. Soil sample analysis results were compared against the Table 2-2 criteria for soils. If soil ROCs were identified in concentrations greater than MDC, then the sum of the fraction rule was applied to the sample results. As per the QAPP, split samples were used to determine the acceptability of the analyses by use of criteria in NRC Inspection Procedure No. 84750 “Radioactive Waste Treatment, and Effluent and Environmental Monitoring.” Any results exceeding this analysis were evaluated.

At the completion of the surveys conducted in each survey unit, measurement results were assessed and evaluated according to the DQOs. Data evaluation utilized guidance from MARSSIM [Reference 10-10] and encompassed a review of the survey package data sheets, technician notes, on-site and offsite gamma spectroscopy analytical reports. Once all DQOs had been achieved, the characterization was considered complete.

9.0 Characterization Survey Results

The on site characterization work was conducted at LACBWR beginning on June 15, 2015 and ending on August 6, 2015. The following is a summary by survey area of sampling results including data analysis.

9.1. Concrete Core Sampling

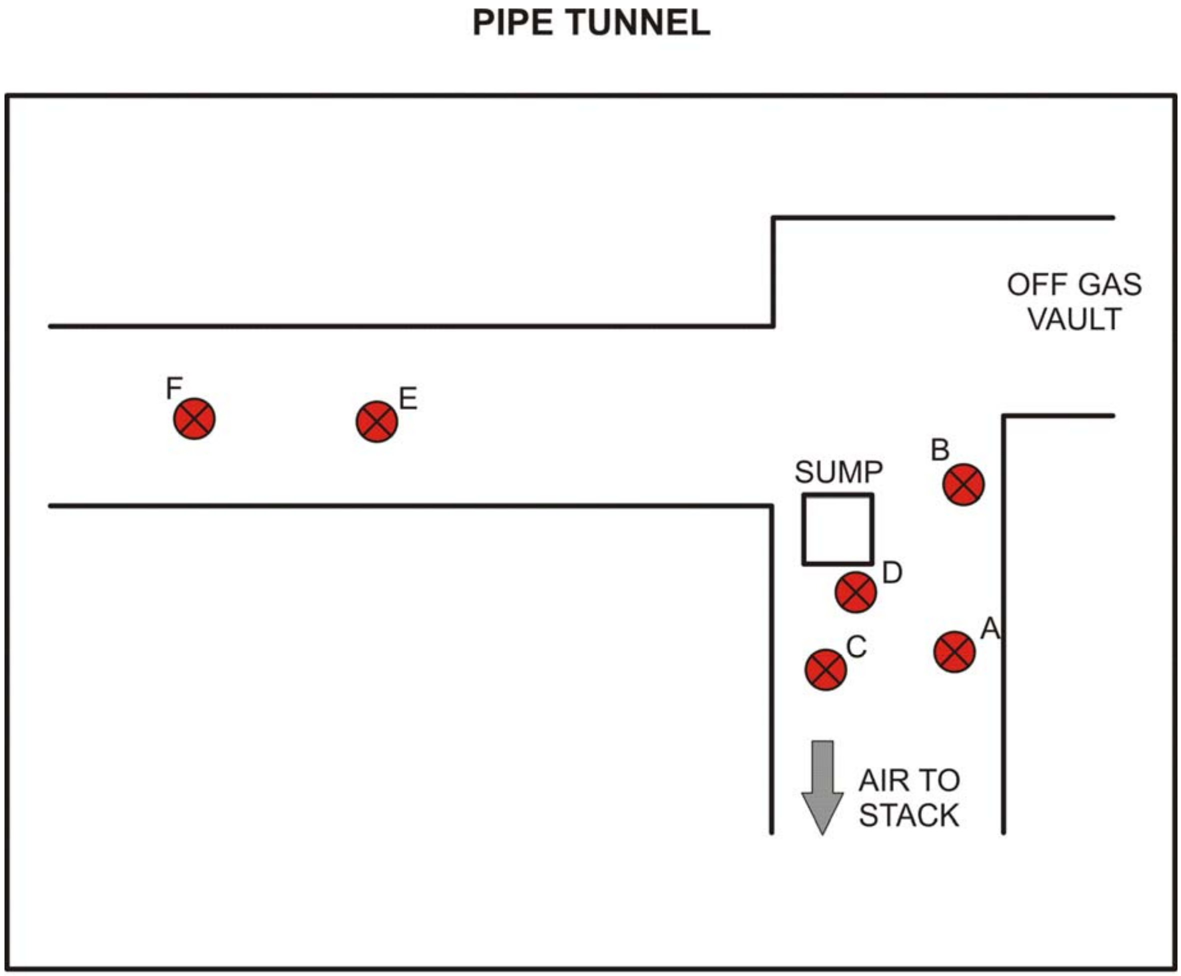
Concrete Core Sampling started on June 17, 2015 and was completed on June 24, 2015. Three separate facilities were selected for the concrete sampling as being representative of the LACBWR Site for the purposes of evaluating the ROCs and ratios of radionuclides present. All three facilities had a past history of operational use and spills during operation. The three facilities selected were the Reactor Building, including the basement and sub-basement levels, the Waste Treatment Building, including the basement, and the Pipe Tunnel. In total fifteen (15)

separate locations in the facilities had concrete cores taken including three (3) in the Waste Treatment Building, six (6) in the Reactor Building, and six (6) in the Pipe Tunnel. All three inch diameter cores were taken of floor areas excepting one wall core taken in the Reactor Building. Per the direction of the respective survey package, actual concrete coring locations were selected by the HP Technician based on elevated count rates or dose rates being present during the scanning of the concrete surfaces as well as visual indications and process knowledge. Radiological surveys were documented of the respective evaluation of areas by the HP Technicians. The goal of the concrete coring was to identify the areas with the highest potential for significant contamination for the purpose of bounding the radiological condition by facility. This judgmental sampling approach also ensured there was sufficient source term in the cores to achieve the sensitivities required to determine the radionuclide distributions of gamma emitters as well as HTD and transuranic radionuclides. The concrete cores were taken with thicknesses of approximately 6 inches and prior to sample preparation, cores were wiped down and smear surveyed to ensure that any activity identified was not from cross contamination. The concrete core removal location was also identified as to its number and floor surface scan rate. The core was then sliced into concrete pucks approximately one half inch in thickness and field scanned on both sides by the HP Technicians and readings documented. Following the scanning of the pucks the sliced core pucks were submitted to the onsite lab for one hour gamma spectroscopy counting. The pucks were counted individually on both sides from the first one in contact with the facility top surface until such time that the activity reported were < MDC. Pucks with gamma spectroscopy results >MDC were selected by facility by the Project Health Physicist and Characterization/License Termination Manager for shipment to the off-site laboratory for gamma spectroscopy, HTD, and transuranic radionuclide analyses. The off-site laboratory crushed the concrete prior to the analyses performed at their facility. The following sections discuss the sampling results by facility.

9.1.1 Pipe Tunnel Concrete Cores Analysis

A total of (6) concrete cores of floor surfaces were taken with core closeout depths of approximately four to six inches. All cores were removed complete without significant cracking problems. In total (49) concrete pucks were created from the (6) concrete cores. The onsite lab identified (11) of the concrete pucks with gamma spec results >MDC with the highest result being 19.5 pCi/g Cs-137. The only isotopes detected on the concrete pucks were Cs-137 and Co-60. There was no instance where concrete pucks past the first one half inch puck had activity >MDC on the onsite gamma spec equipment. The gamma spec MDC for Cs-137 was maintained at < 2.0 pCi/g. The results are presented in Table 9-1 A total of (3) concrete pucks were sent off site for radiological analysis. The results showed that: Cs-137, Co-60, Sr-90, Ni-63, Am-241, and Tritium were present in the samples. The results of the analyses are shown in Table 9-12.

The figure below shows the location of the samples taken in the Pipe Tunnel.



A	B1008101-CJ-FC-001-CV
B	B1008101-CJ-FC-002-CV
C	B1008101-CJ-FC-003-CV
D	B1008101-CJ-FC-004-CV
E	B1008101-CJ-FC-005-CV
F	B1008101-CJ-FC-006-CV

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Figure 9-1, Pipe Tunnel Scanning Locations

Table 9-1, Pipe Tunnel Concrete Coring Sample Results

Pipe Tunnel Concrete Coring Sample Results					
Sample ID	Sample Date	Analysis Date	Depth (inches)	Co-60 [a] (pCi/g)	Cs-137 [a] (pCi/g)
B1008101CJFC001-CV	6/19/2015	6/22/2015	0.0-0.5 IN BOTTOM	**4.23E-01**	1.54E+00
			0.0-0.5 IN TOP	**4.98E-01**	2.74E+00
			0.5-1.0 IN TOP	**3.78E-01**	**4.33E-01**
B1008101CJFC002-CV	6/18/2015	6/23/2015	0.0-0.5 IN TOP	**4.75E-01**	1.06E+01
			0.0-0.5 IN BOTTOM	**4.62E-01**	5.81E+00
	6/19/2015		0.5-1.0 IN TOP	**3.46E-01**	**3.22E-01**
			0.5-1.0 IN BOTTOM	**3.45E-01**	**3.23E-01**
B1008101CJFC003-CV	6/19/2015	6/23/2015	0.0-0.5 IN TOP	1.31E+00	9.75E+00
			0.0-0.5 IN BOTTOM	**5.41E-01**	4.99E+00
			0.5-1.0 IN TOP	**3.63E-01**	**4.10E-01**
B1008101CJFC004-CV	6/19/2015	6/23/2015	0.0-0.5 IN TOP	**6.57E-01**	1.95E+01
			0.0-0.5 IN BOTTOM	**5.42E-01**	1.05E+01
			0.5-1.0 IN TOP	**4.01E-01**	**4.44E-01**
B1008101CJFC005-CV	6/19/2015	6/23/2015	0.0-0.5 IN TOP	**3.73E-01**	1.09E+00
			0.0-0.5 IN BOTTOM	**3.88E-01**	**4.97E-01**
B1008101CJFC006-CV	6/19/2015	6/23/2015	0.0-0.5 IN TOP	**4.51E-01**	9.69E-01
		6/24/2015	0.0-0.5 IN BOTTOM	**4.76E-01**	5.29E-01
			0.5-1.0 IN TOP	**4.11E-01**	**4.15E-01**

(a) MDC data is shown in bold

9.1.2 Waste Treatment Building Concrete Cores Analysis

A total of three (3) concrete cores of floor surfaces were taken with core closeout depths of approximately six inches. All cores were removed complete without significant cracking problems. In total twenty-seven (27) concrete pucks were created from the three (3) concrete cores. The onsite lab identified twenty-three (23) of the concrete pucks with gamma spec results >MDC with the highest result being 20,200 pCi/g Cs-137. The only isotopes detected on the concert pucks were Cs-137 and Co-60. Concrete pucks had Cs-137 activity >MDC down to three inches based on the gamma spectroscopy measurements. The gamma spectroscopy MDC for Cs-137 was maintained at < 2.0 pCi/g. The results are presented in Table 9-2. A total of seven (7) concrete pucks were sent off-site for radiological analysis. The off-site lab results showed that: Cs-137, Co-60, Sr-90, Fe-55, Ni-59, Ni-63, Tc-99, Pu-238, Pu-239/240, Am-241, Am-243, C-14, and Tritium were present in the samples. The results of the analyses are shown in Table 9-11.

The figure below shows the location of the samples taken in the Waste Treatment Building.

A	B1002101-CJ-FC-001-CV
B	B1002101-CJ-FC-002-CV
C	B1002101-CJ-FC-003-CV

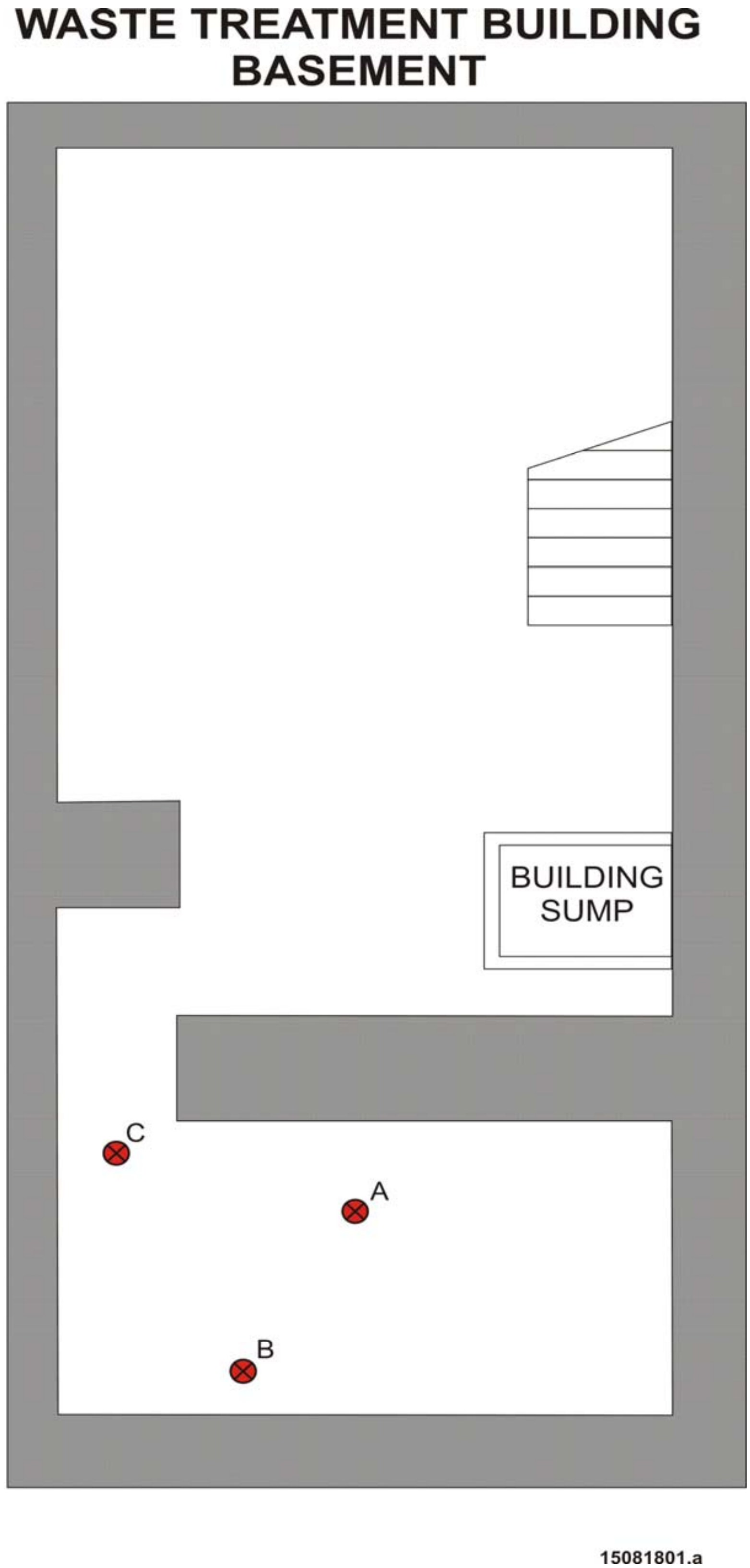


Figure 9-2, Waste Treatment Building Basement Scanning Locations

Table 9-2, Waste Treatment Building Concrete Coring Sample Results

Waste Treatment Building Concrete Coring Sample Results					
Sample ID	Sample Date	Analysis Date	Depth (inches)	Co-60 [a] (pCi/g)	Cs-137 [a] (pCi/g)
B1002101CJFC001-CV	6/17/2015	6/24/2015	0.0-0.5 IN TOP	1.01E+02	4.48E+03
			0.0-0.5 IN BOTTOM	4.64E+01	1.95E+03
			0.5-1.0 IN TOP	**6.53E-01**	3.69E+00
			0.5-1.0 IN BOTTOM	8.07E-01	3.52E+00
			1.0-1.5 IN TOP	**5.20E-01**	3.55E+00
			1.0-1.5 IN BOTTOM	**5.20E-01**	4.26E+00
			1.5-2.0 IN TOP	3.32E-01	2.45E+00
			1.5-2.0 IN BOTTOM	**5.41E-01**	2.13E+00
		6/27/2015	2.0-2.5 IN TOP	5.24E-01	1.68E+00
B1002101CJFC002-CV		6/24/2015	0.0-0.5 IN TOP	2.56E+02	2.02E+04
			0.0-0.5 IN BOTTOM	1.29E+02	9.48E+03
			0.5-1.0 IN TOP	6.19E+00	1.58E+02
			0.5-1.0 IN BOTTOM	5.40E+00	1.15E+02
			1.0-1.5 IN TOP	**4.64E-01**	1.10E+00
		6/25/2015	1.0-1.5 IN BOTTOM	**4.85E-01**	9.25E-01
			1.5-2.0 IN TOP	**3.87E-01**	1.28E+00
			1.5-2.0 IN BOTTOM	**3.76E-01**	1.23E+00
			2.0-2.5 IN TOP	**3.95E-01**	1.05E+00
			2.0-2.5 IN BOTTOM	**4.26E-01**	1.18E+00
			2.5-3.0 IN TOP	**3.76E-01**	4.73E-01
			2.5-3.0 IN BOTTOM	**3.59E-01**	7.16E-01
B1002101CJFC003-CV			0.0-0.5 IN TOP	8.41E+00	1.14E+03
			0.0-0.5 IN BOTTOM	4.60E+00	5.44E+02
			0.5-1.0 IN TOP	**4.04E-01**	**3.51E-01**

(a) MDC data is shown in bold

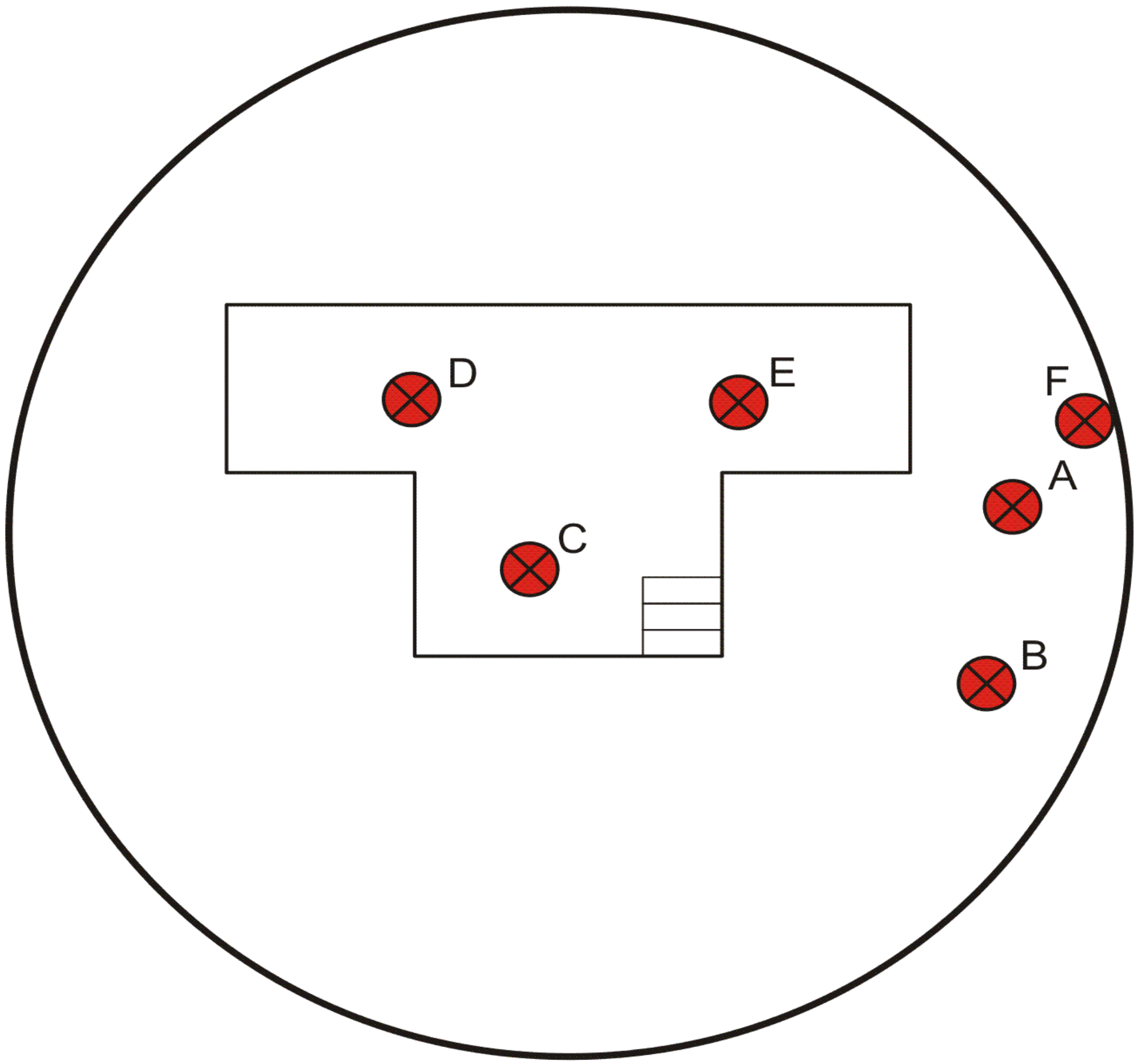
9.1.3 Reactor Building Concrete Cores Analysis

A total of five (5) concrete cores of floor surfaces and one (1) wall concrete core were taken with core closeout depths of approximately six inches. All cores were removed complete without significant cracking problems. In total fifty-four (54) concrete pucks were created from the six (6) concrete cores. The onsite lab identified twenty-eight (28) of the concrete pucks with gamma spectroscopy results >MDC with the highest result being 7540 pCi/g Cs-137. The only isotopes detected on the concrete pucks were Cs-137 and Co-60. Concrete pucks had Cs-137 activity >MDC down to two inches based on the gamma spec equipment measurements. The gamma spec MDC for Cs-137 was maintained at < 2.0 pCi/g. The results are presented in Table 9-3. A total of ten (10) concrete pucks were sent off-site for radiological analysis. The off-site laboratory results showed that: Cs-137, Co-60, Sr-90, Fe-55, Ni-59, Ni-63, Pu-238, Pu-239/240,

Am-241, Cm-243/244, Am-243, and Tritium were present in the samples. The results of the analyses are shown in Table 9-10.

The figure below shows the location of the samples taken in the Reactor Building.

REACTOR BUILDING BASEMENT AND SUB-BASEMENT



A	B1001101-CJ-FC-001-CV
B	B1001101-CJ-FC-002-CV
C	B1001101-CJ-FC-003-CV
D	B1001101-CJ-FC-004-CV
E	B1001101-CJ-FC-005-CV
F	B1001101-CJ-WC-006-CV

15081802.a

Figure 9-3, Reactor Building Basement and Sub-basement Scanning Locations

Table 9-3, Reactor Building Concrete Coring Sample Results

Reactor Building Concrete Coring Sample Results					
Sample ID	Sample Date	Analysis Date	Depth (inches)	Co-60 [a] (pCi/g)	Cs-137 [a] (pCi/g)
B1001101CJFC001-CV	6/23/2015	6/25/2015	0.0-0.5 IN TOP	1.50E+01	7.54E+03
			0.0-0.5 IN BOTTOM	7.36E+00	3.86E+03
			0.5-1.0 IN TOP	**3.53E-01**	7.74E+00
	6/17/2015		0.5-1.0 IN BOTTOM	**3.62E-01**	4.94E+00
	6/23/2015	6/26/2015	1.0-1.5 IN TOP	**3.59E-01**	4.43E-01
			1.0-1.5 IN BOTTOM	**3.86E-01**	**4.66E-01**
B1001101CJFC002-CV			0.0-0.5 IN TOP	3.78E-01	3.99E+02
			0.0-0.5 IN BOTTOM	**4.46E-01**	2.03E+02
			0.5-1.0 IN TOP	**3.64E-01**	1.67E+00
			0.5-1.0 IN BOTTOM	**3.68E-01**	1.70E+00
			1.0-1.5 IN TOP	**3.66E-01**	1.04E+00
		6/27/2015	1.0-1.5 IN BOTTOM	**3.56E-01**	1.07E+00
B1001101CJFC003-CV		6/26/2015	0.0-0.5 IN TOP	6.94E+01	6.51E+02
			0.0-0.5 IN BOTTOM	3.46E+01	3.07E+02
			0.5-1.0 IN TOP	**5.57E-01**	9.19E-01
			0.5-1.0 IN BOTTOM	**5.07E-01**	5.80E-01
			1.0-1.5 IN TOP	**4.06E-01**	1.45E+00
			1.0-1.5 IN BOTTOM	**4.59E-01**	1.22E+00
			1.5-2.0 IN TOP	**4.08E-01**	1.18E+00
			1.5-2.0 IN BOTTOM	**3.91E-01**	9.66E-01
B1001101CJFC004-CV			0.0-0.5 IN TOP	3.08E+01	2.02E+02
		6/27/2015	0.0-0.5 IN BOTTOM	1.49E+01	9.28E+01
			0.5-1.0 IN TOP	**3.86E-01**	3.57E-01
B1001101CJFC005-CV			0.0-0.5 IN TOP	1.08E+02	7.68E+01
			0.0-0.5 IN BOTTOM	4.83E+01	3.19E+01
			0.5-1.0 IN TOP	**3.78E-01**	4.50E-01
B1001101CJFC006-CV			0.0-0.5 IN TOP	2.58E+00	1.36E+03
			0.0-0.5 IN BOTTOM	1.36E+00	6.50E+02
			0.5-1.0 IN TOP	**4.05E-01**	7.65E-01

(a) MDC data is shown in bold

9.2. Under Turbine Building Geoprobe Soil Sampling

On June 25, 2015 a total of five (5) individual locations were selected on the north and south side of the Turbine Building to conduct angled Geoprobe soil sampling under the Turbine Building. The original plan was to Geoprobe under the south and west side of the facility but the startup switch yard on the west side of the facility was still active so for safety reasons the west side did

not have locations selected for geoprobing. The figure below shows the individual locations where the soil sampling was conducted.

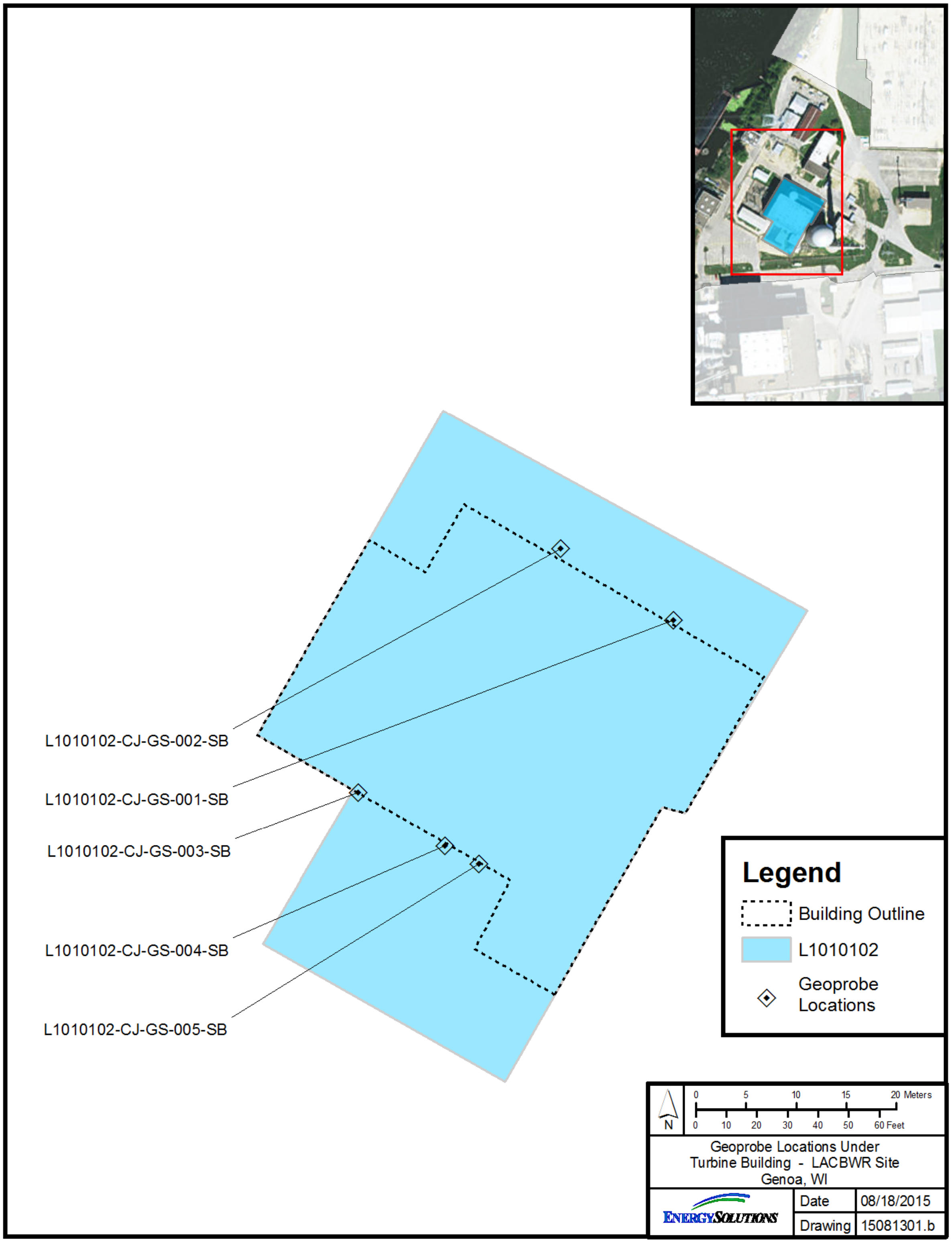


Figure 9-4, Turbine Building Geoprobe Soil Scanning Locations

All five (5) individual sampling locations were completed with soil sampling on June 25, 2015. At each location the angled Geoprobing collected soil samples at depths of ten, fifteen, and twenty feet under the Turbine Building with direction of the angled Geoprobe either due north if accessing on the south side of the facility or due south if accessing from the north side of the facility. The samples collected per each Geoprobe location are listed on the following table. It is noted that the sample locations identified as L1010102-CJ-GS-001,007,008,009,011,012,and 013-SB are duplicative of the same number used last year's characterization field work at LACBWR Site.

Table 9-4, Sample Numbers For Sample Locations

Sample Location	Sample Number
L1010102-CJ-GS-001-SB	L1010102-CJ-GS-001-SB L1010102-CJ-GS-002-SB L1010102-CJ-GS-003-SB
L1010102-CJ-GS-002-SB	L1010102-CJ-GS-004-SB L1010102-CJ-GS-005-SB L1010102-CJ-GS-006-SB
L1010102-CJ-GS-003-SB	L1010102-CJ-GS-007-SB L1010102-CJ-GS-008-SB L1010102-CJ-GS-009-SB
L1010102-CJ-GS-004-SB	L1010102-CJ-GS-010-SB L1010102-CJ-GS-011-SB L1010102-CJ-GS-012-SB
L1010102-CJ-GS-005-SB	L1010102-CJ-GS-013-SB L1010102-CJ-GS-014-SB L1010102-CJ-GS-015-SB

During the collection of the three soil samples at each Geoprobe sampling location, gamma scans were conducted of the collected soil samples. The soil sample exhibiting the highest gamma scan reading had a split sample taken for off-site laboratory analyses of gamma spectroscopy, HTD, and transuranic radionuclides. All collected soil samples were analyzed for one hour counts on the onsite gamma spectroscopy system following drying. The results of the analyses are presented on Table 9-6. The off-site split samples were dried except for the Tritium and C-14 analyses. The results of the analyses are presented in Table 9-13. The results only showed detection above MDC level of Cs-137 at site environmental levels and Ni-63 at levels just above MDC. The off-site laboratory reported that there were high levels of mineral interferences in the columns used for the analyses of Ni-63 that likely account for the trace levels of Ni-63 reported.

The relationship of the split samples to the original sample location is as follows.

Table 9-5, Split Sample Cross Reference Location

Split QC Sample Number	Site Sample Location
L1010102-QQ-GS-001-SB	L1010102-CJ-GS-002-SB
L1010102-QQ-GS-002-SB	L1010102-CJ-GS-005-SB
L1010102-QQ-GS-003-SB	L1010102-CJ-GS-009-SB
L1010102-QQ-GS-004-SB	L1010102-CJ-GS-012-SB
L1010102-QQ-GS-005-SB	L1010102-CJ-GS-013-SB

Table 9-6, Subsurface Soil Sample Analysis L1010102

Characterization Survey Results - Survey Unit # L1010102 Geoprobe Underneath Turbine Building

Subsurface Soil Sample Analysis

Sample ID	Sample Depth (feet)	Date	Weight (g)	Wet/Dry	Co-60			Cs-137		
					Activity (pCi/g)	1σ (pCi/g)	MDC (pCi/g)	Activity (pCi/g)	1σ (pCi/g)	MDC (pCi/g)
L1010102-CJ-GS-001-SB	10	6/25/2015	630	DRY			8.04E-02	1.30E-01	2.22E-02	7.67E-02
L1010102-CJ-GS-002-SB	15	6/25/2015	745	DRY			6.62E-02	6.91E-02	1.74E-02	6.26E-02
L1010102-CJ-GS-003-SB	20	6/25/2015	850	DRY			5.61E-02			5.92E-02
L1010102-CJ-GS-004-SB	10	6/25/2015	730	DRY			6.42E-02	8.10E-02	1.84E-02	7.26E-02
L1010102-CJ-GS-005-SB	15	6/25/2015	790	DRY			5.71E-02			6.50E-02
L1010102-CJ-GS-006-SB	20	6/25/2015	750	DRY			5.38E-02			6.40E-02
L1010102-CJ-GS-007-SB	10	6/25/2015	750	DRY			6.05E-02			6.79E-02
L1010102-CJ-GS-008-SB	15	6/25/2015	760	DRY			6.46E-02			6.68E-02
L1010102-CJ-GS-009-SB	20	6/25/2015	760	DRY			6.38E-02			6.48E-02
L1010102-CJ-GS-010-SB	10	6/25/2015	705	DRY			6.18E-02			6.76E-02
L1010102-CJ-GS-011-SB	15	6/25/2015	710	DRY			6.79E-02			6.18E-02
L1010102-CJ-GS-012-SB	20	6/25/2015	755	DRY			6.39E-02			6.81E-02
L1010102-CJ-GS-013-SB	10	6/25/2015	715	DRY			7.20E-02			6.88E-02
L1010102-CJ-GS-014-SB	15	6/25/2015	765	DRY			5.66E-02			6.46E-02
L1010102-CJ-GS-015-SB	20	6/25/2015	760	DRY			5.82E-02	6.68E-02	1.68E-02	6.07E-02

	Co-60		Cs-137	
# of Measurements	15		15	
# >MDA	0		4	
Mean	6.31E-02	pCi/g	7.10E-02	pCi/g
Median	6.38E-02	pCi/g	6.68E-02	pCi/g
Max	8.04E-02	pCi/g	1.30E-01	pCi/g
Min	5.38E-02	pCi/g	5.92E-02	pCi/g
Standard Deviation	6.88E-03	pCi/g	1.70E-02	pCi/g

9.3. Non Impacted Area (Class 4) Open Land Walk Over Surveys and Soil Sampling.

In an effort to continue with the work in the open land areas conducted during the 2014 site characterization work [Reference 10-23] to provide due diligence that the Class 4 areas as presented on Figure 3-1 are non impacted land areas gamma walk over scans and additional soil sampling were conducted. As in the 2014 characterization work, L4012102 Survey Unit was excluded from the characterization work for safety reasons due to it being an active main switchyard for the Genoa DPC Site. In Survey Unit L4012107, the HP Technicians were unable to meet the 1% coverage requirement for gamma scanning due to accessibility and safety issues because of the presence of tall, thick vegetation and steep terrain in the survey unit. During the time period of June 27, 2015 thru August 6, 2015 the walk over surveys and sampling were conducted in the respective Class 4 survey units.

The gamma walk over surveys were conducted with a Model 2350-1 data logger with a 2” by 2” NaI detector (Model 44-10) at a pace of 0.5 meters/second with detector end cap at a height of less than six inches from the land surface area. The Model 2350-1 instrument was set up to alarm based on the established area background as per procedure and survey package requirements and adding the MDCR for that established background. Additionally, the HP Technician listened for audible indication of elevated count rates. If the instrument alarmed the HP Technician flagged the area of interest for further study with the Exploranium GR-135 (a portable multi-channel analyzer).

The survey methodology was to survey a minimum of 1% of the open land area per survey unit. HP Technicians and the Project Health Physicist walked down the individual survey units looking for areas where radioactive material could concentrate such as low points, depressions, drainage ditches, or areas where soil or snow was or may have been piled. If such areas were not available in the survey unit, or just a few of such areas were identified, the rest of the individual areas were distributed throughout the survey unit. The maximum individual area which was then marked for walk over scan was either 25 M^2 or 50 M^2 depending on the geography, topography and size of the survey unit. Requiring that a number of individual scan locations per survey unit be established (because of size limitation) ensured that scan surveys were evenly distributed accross the survey unit. Established individual scan locations were identified with a GPS unit so they could be placed on a map with coordinates by survey unit.

The results of the gamma scan walkover surveys are presented in Figures 9-5 to 9-11 for survey units L4012101 and L4012103 thru L4012108. A total of four alarms were evaluated during the performance of the gamma walk over surveys which included:

- One (1) in survey unit L4012106 survey area L4012106-09,
- One (1) in survey unit L4012108 survey area L4012108-01, and
- Two (2) in survey unit L4012107: survey areas L4012107-03 and L4012107-04.

All alarmed areas were evaluated with the Exploranium GR-135 which indicated the radionuclides detected were natural background radionuclide activity. The results of the gamma scan surveys in the Non Impacted (Class 4) areas are evaluated as normal environmental background conditions for the types of terrain and vegetation cover encountered during the course of the surveys.

The soil sampling conducted consisted of continuing with the evaluation of slightly elevated Tritium levels in soils during the 2014 site characterization work [Reference 10-23] in L4012101 and L4012106 survey unit areas. Three additional soil samples at surface and one meter depths were collected by use of the Geoprobe in both survey units of interest in the areas where the 2014 characterization work identified the elevated soil Tritium levels. See attached Figures (L4012101 Figure 9-5 and L4012106 Figure 9-9). The collected soil samples were sent to the off-site laboratory for analysis. The results of the analyses are reported in Table 9-7. All reported Tritium analytical results are at MDC levels which reflect background conditions.

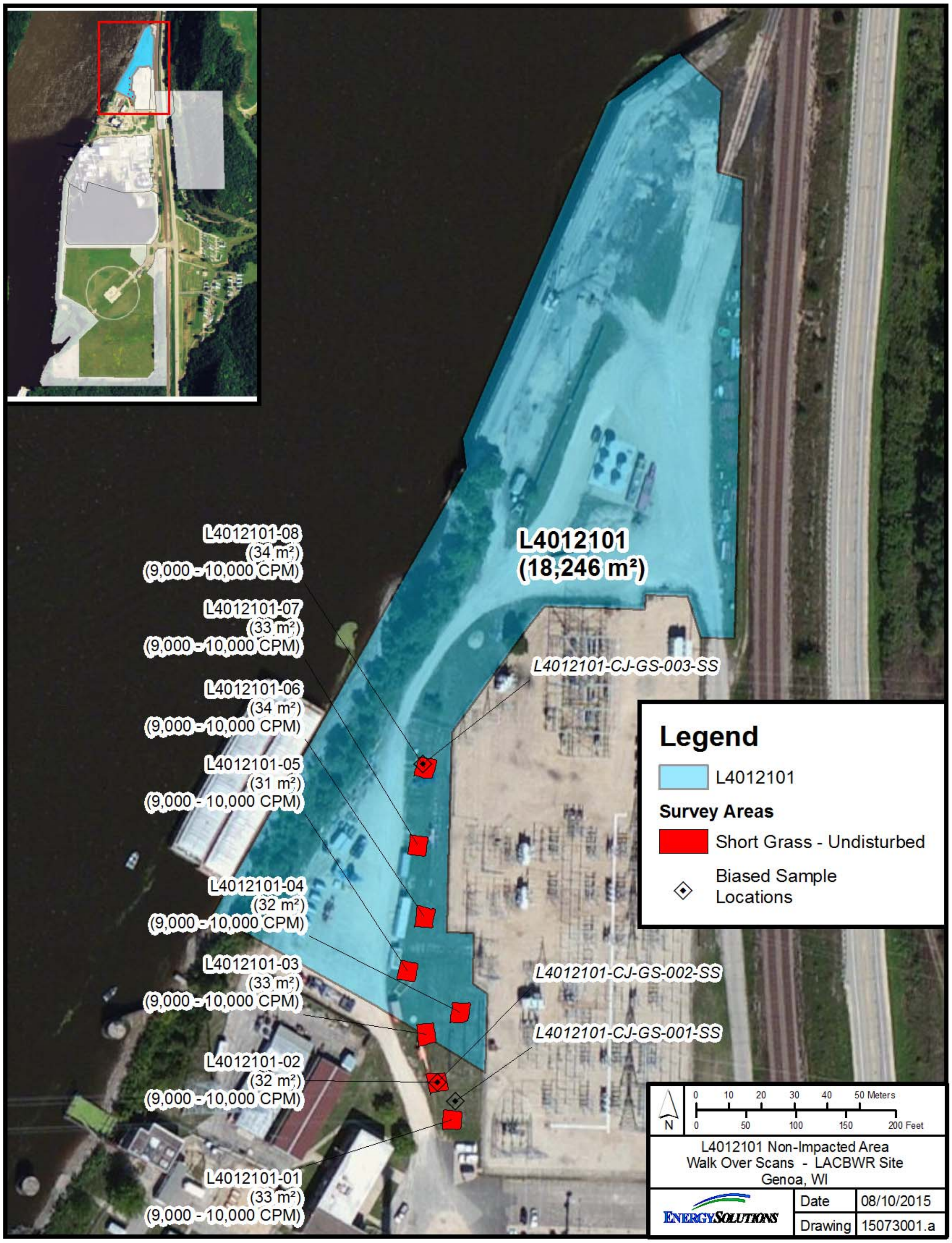


Figure 9-5, L4012101 Scanning Locations

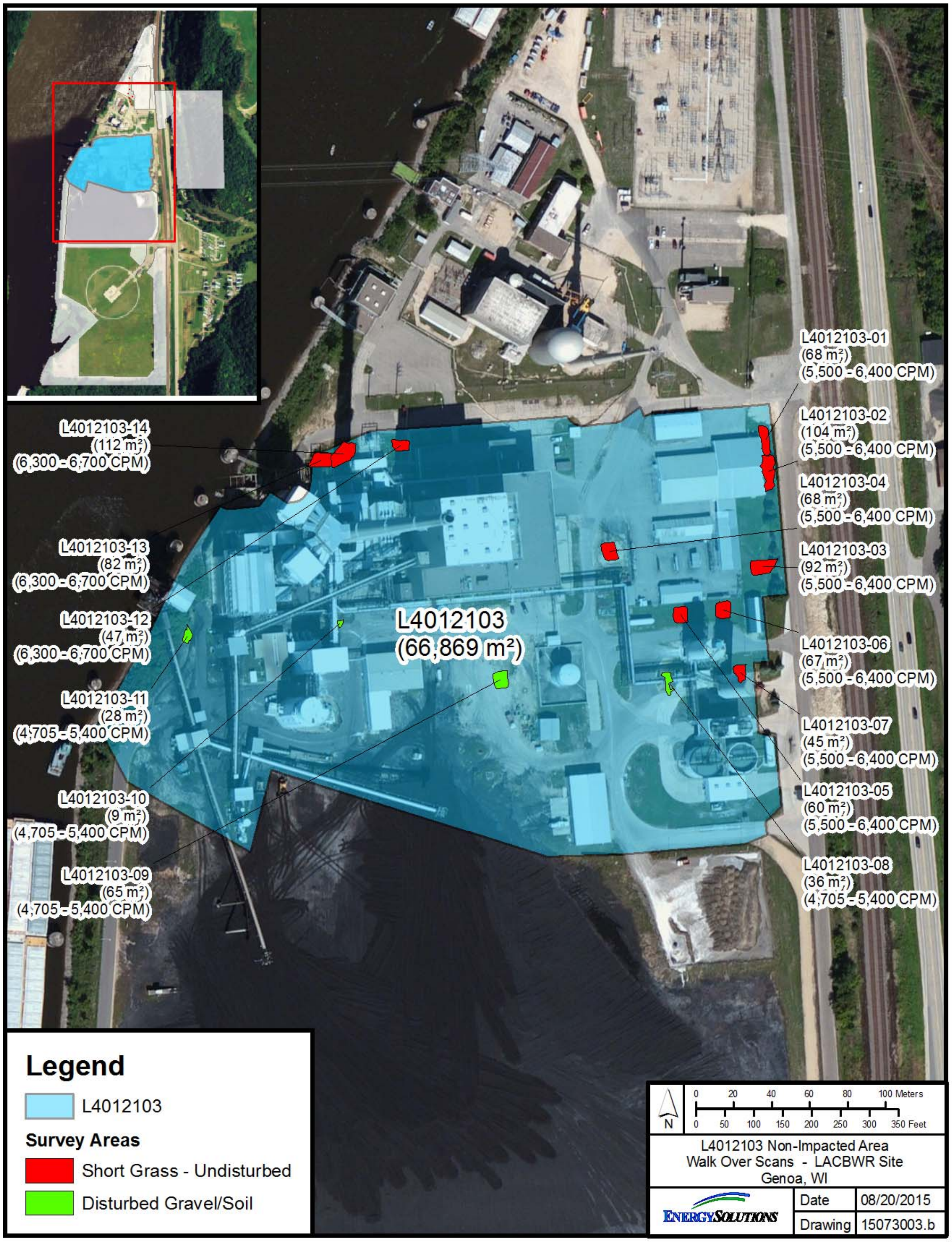


Figure 9-6, L4012103 Scanning Locations

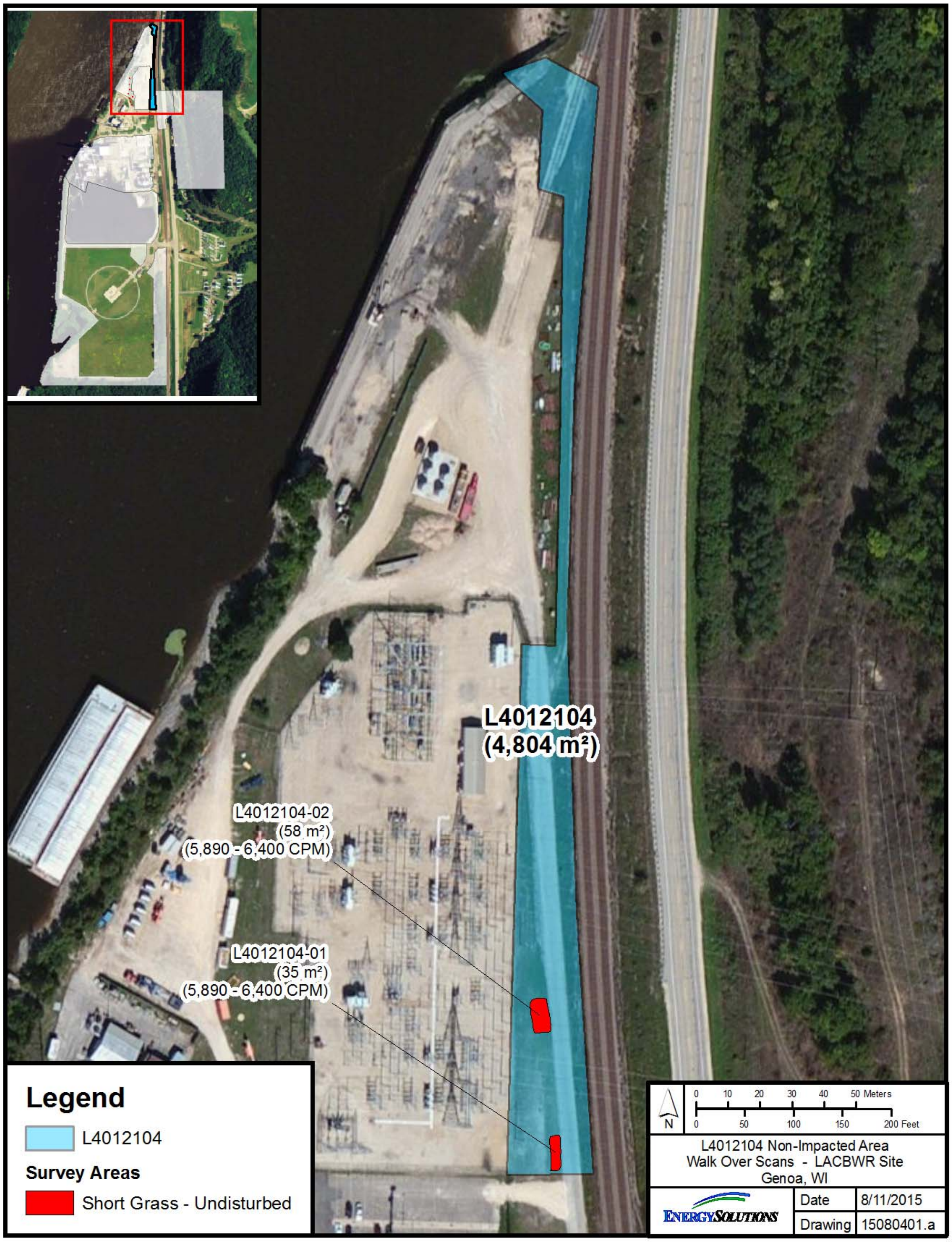


Figure 9-7, L4012104 Scanning Locations

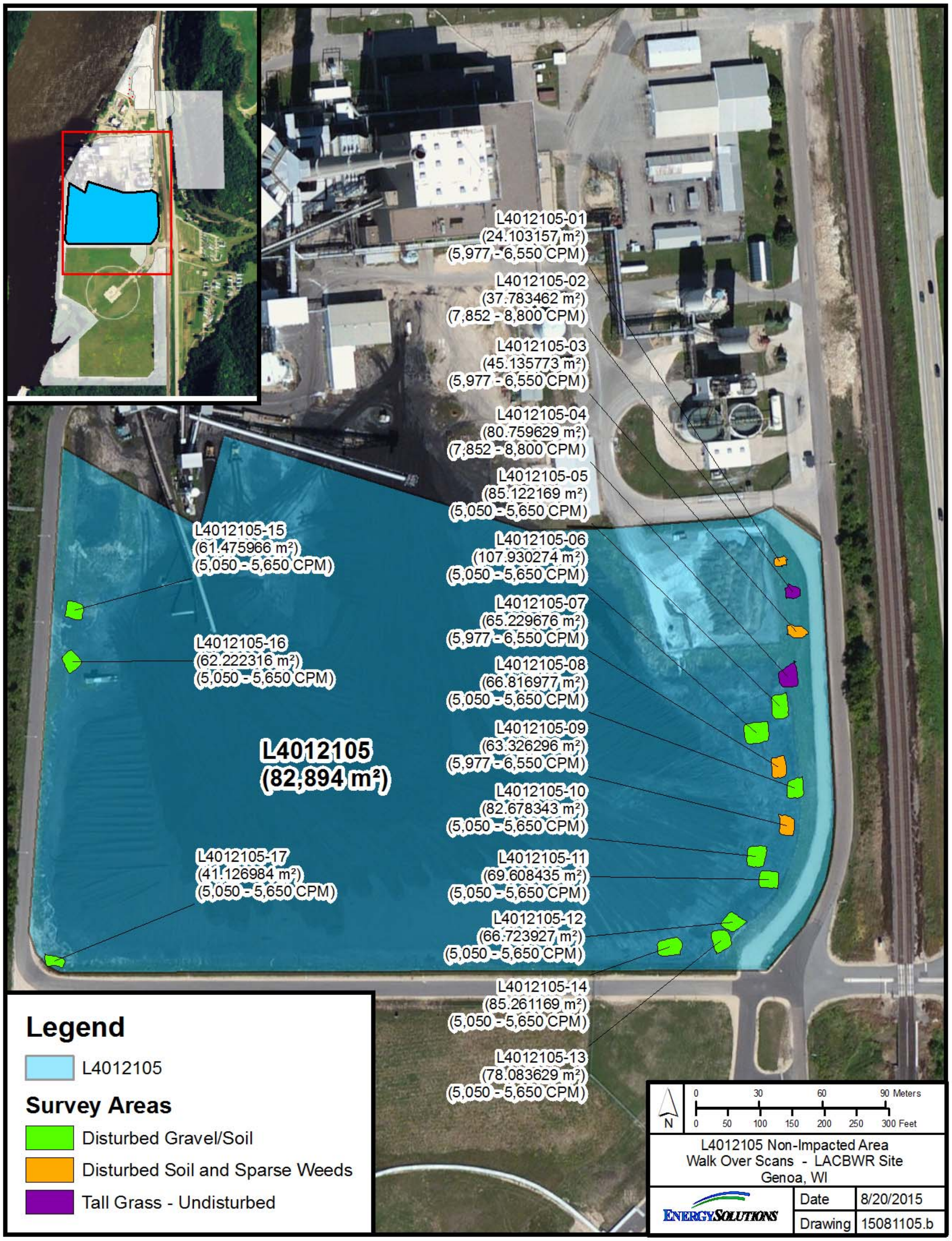


Figure 9-8, L4012105 Scanning Locations

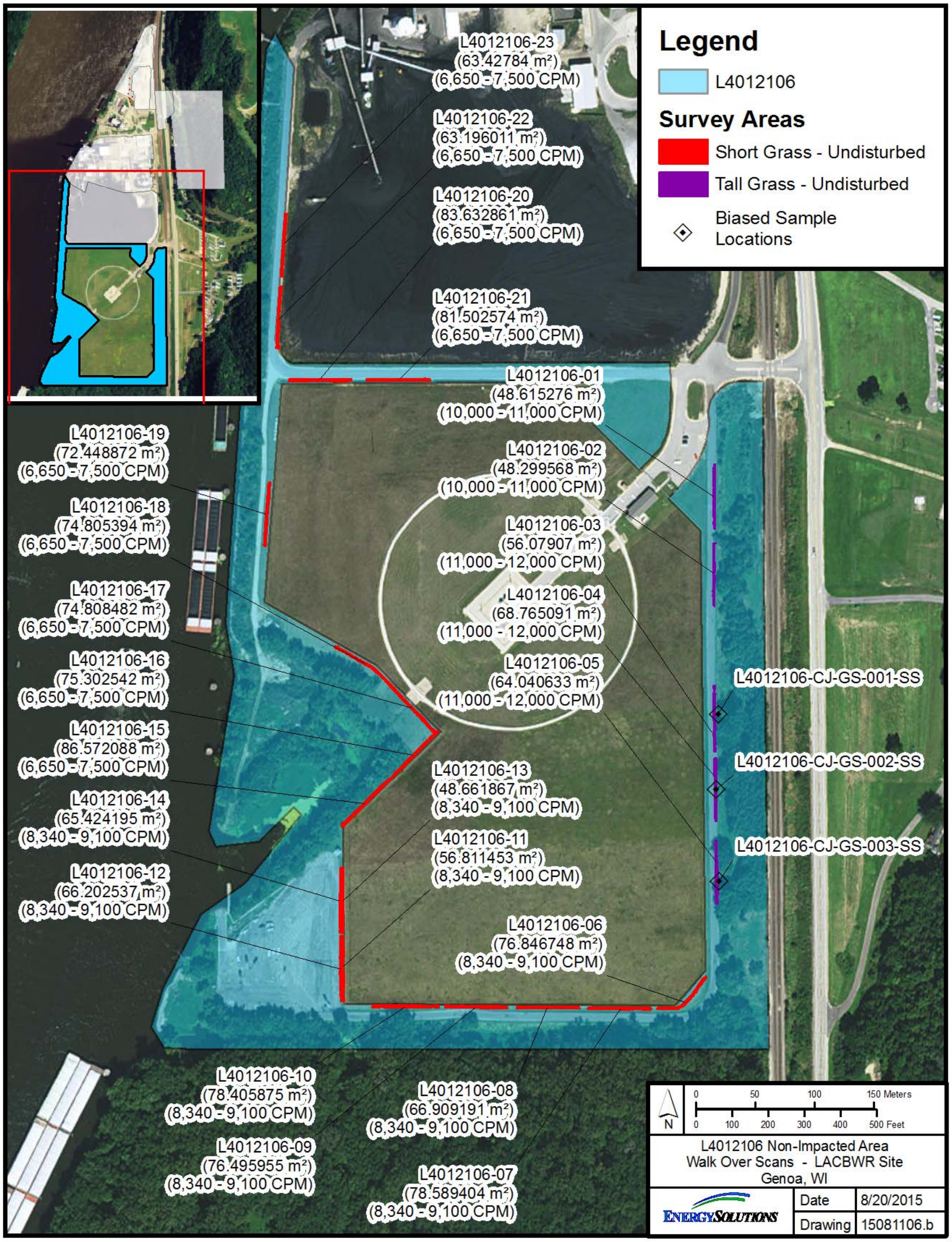


Figure 9-9, L4012106 Scanning Locations

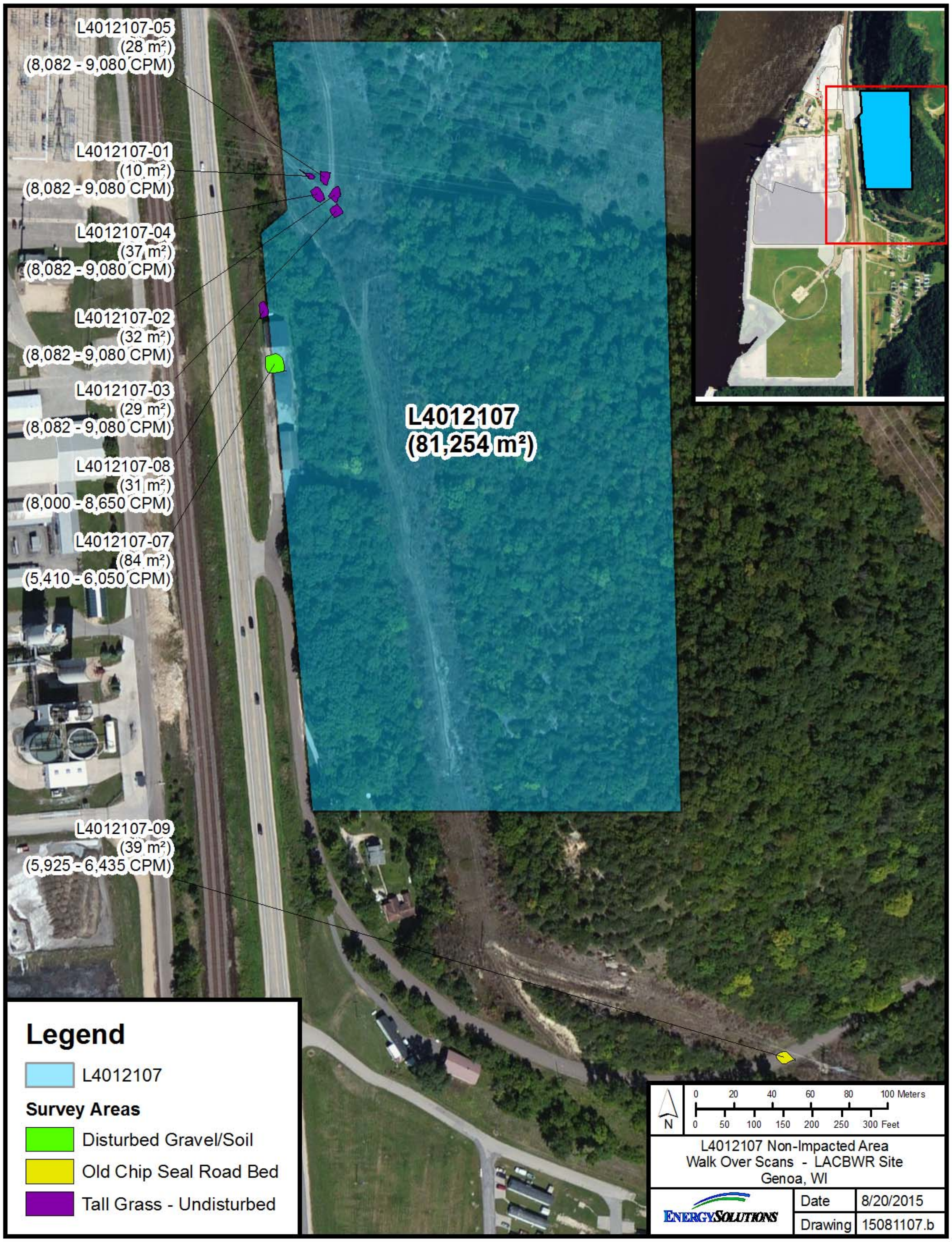


Figure 9-10, L4012107 Scanning Locations

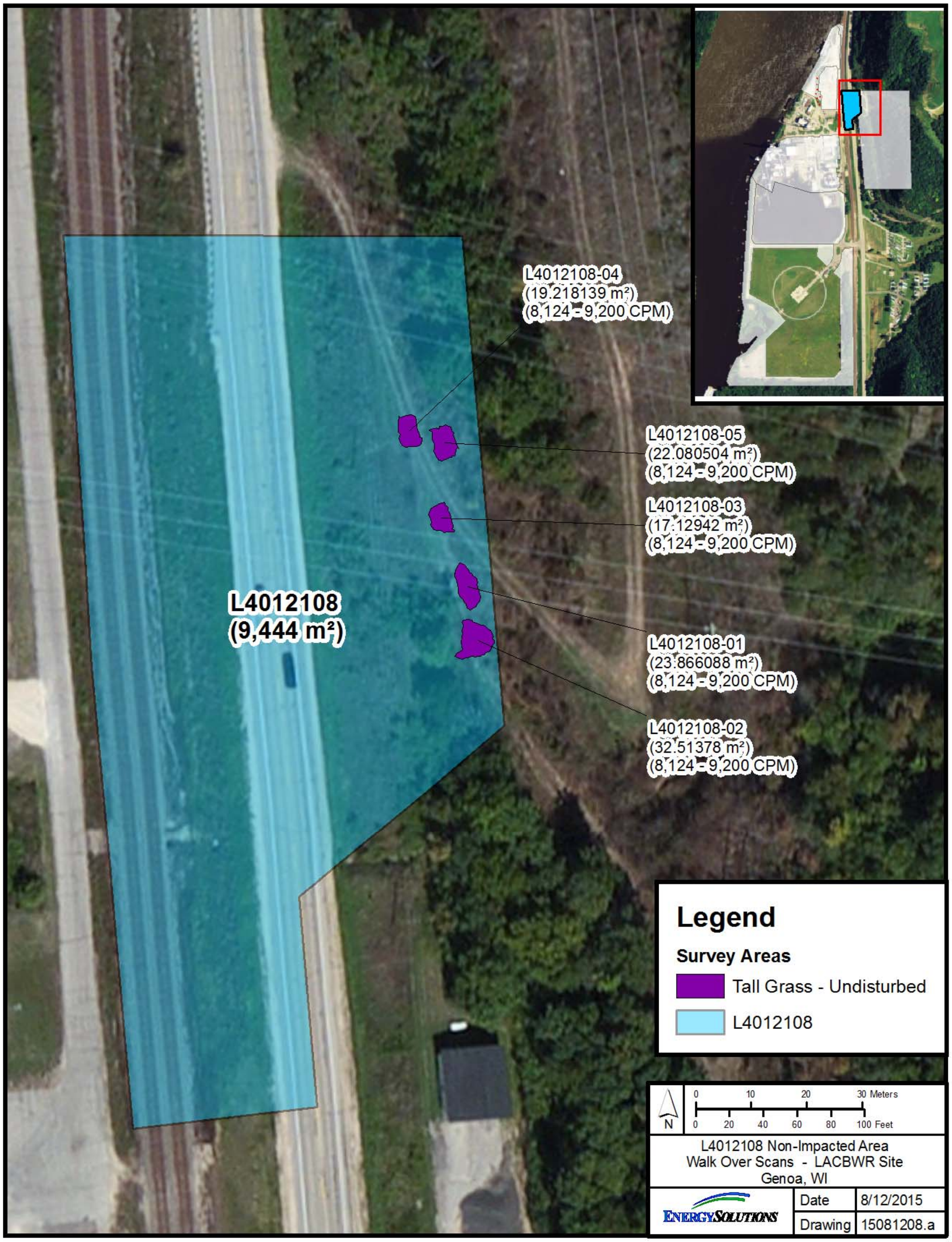


Figure 9-11, L4012108 Scanning Locations

Table 9-7, Class 4 Tritium Sample Results

SU L4012101 and L4012106 Tritium Soil Sample Results

Sample ID	Depth	H-3
L4012101-CJ-GS-001-SS	surface	**9.52E-01**
L4012101-CJ-GS-001-SB	one meter	**1.18E+00**
L4012101-CJ-GS-002-SS	surface	**1.85E+00**
L4012101-CJ-GS-002-SB	one meter	**1.00E+00**
L4012101-CJ-GS-003-SS	surface	**1.17E+00**
L4012101-CJ-GS-003-SB	one meter	**1.14E+00**
L4012106-CJ-GS-001-SS	surface	**1.15E+00**
L4012106-CJ-GS-001-SB	one meter	**1.16E+00**
L4012106-CJ-GS-002-SS	surface	**1.07E+00**
L4012106-CJ-GS-002-SB	one meter	**1.06E+00**
L4012106-CJ-GS-003-SS	surface	**1.20E+00**
L4012106-CJ-GS-003-SB	one meter	**1.62E+00**

LEGEND
SS= surface soil
SB=subsurface soil

NOTES:
Bold = MDC
Italicized= activity result>MDC

9.4. G-3 Coal Plant Surveys

Land Survey Unit L4012103 contains many facilities of which exist to support the G-3 Coal Plant facility and did not have a past mission in supporting LACBWR Operations other than the Central Warehouse which serves as a receipt facility for non-radiological items coming into the Genoa Site destined for LACBWR Facilities. The intent of the surveys conducted with the Model 2360 with 43-93 detector was to verify that in areas of the original facilities that have had high personnel presence and pass through traffic that there had not been a buildup of residual radiological contamination that exceeds Table 2-3 criteria for surfaces. For the purposes of this survey the G-3 Coal Plant Service Building was selected and in particular the Main Lobby Area and the Maintenance Shop Area were established as survey areas and surveyed on July 23-24, 2015.

The focus of the survey was to conduct beta scans and alpha as well as beta/gamma direct surveys in elevated beta scan areas or if no elevated areas were found perform (5) direct surveys of surfaces per survey area. Additionally, removable contamination surveys were conducted for alpha and beta/gamma contamination wherever direct surveys were performed. The surfaces

surveyed were concrete floors and walls up to six feet in height. Floor drain accesses areas were beta scanned and tools as well as fixed work stations in the Maintenance Shop Area were spot checked with beta scans. Floors and walls were approximately 5% beta scanned. The average beta scan MDC was 2540-2617 dpm/100cm2 and during the course of the survey no structural areas, tools, drains, or work stations were noted that exceeded the beta scan MDC. The results of the Direct Surveys are shown on Table 9-8 and the Removable Contamination Surveys on Table 9-9. The locations of the direct and removable contamination surveys is presented on Figure 9-11. The results of the surveys of the Maintenance Shop Area and Main Lobby did not indicate any indication of a buildup of residual radiological contamination from years of use supporting the G-3 Coal Plant mission.

During the course of the 2014 site characterization surveys[Reference 10-23] the G-3 Coal Plant mid-level roof had beta scanning as well as direct and removable contamination surveys for alpha and beta/gamma contamination conducted on it. As a follow up to that survey the roof material itself was sampled on June 30, 2015 and the sample counted on the onsite gamma spectroscopy unit on the same day. The gamma spectroscopy results showed that only Cs-137 was detected at levels of 0.061 pCi/g which was just slightly above the MDC of the gamma spec equipment for a one hour count.

Table 9-8, G-3 Coal Plant Direct Surveys

Description	G-3 Coal Plant Service Building Survey

	Alpha Directs				Beta Directs			
Sample Number	gross cpm	background cpm	net cpm	dpm/100cm^2	gross cpm	background cpm	net cpm	dpm/100cm^2
1- Main Lobby	1.5	0.5	1	22.5	297	255	42	456.5
2- Main Lobby	1.5	0.5	1	22.5	280	255	25	271.7
3- Main Lobby	1	0.5	0.5	11.25	258	255	3	32.6
4-Main Lobby	2	0.5	1.5	33.7	291	255	36	391.3
5-Main Lobby	1.5	0.5	1	22.5	260	255	5	54.35
6- Maintenance Area	1.5	0.5	1	22.5	301	261	40	434.8
7- Maintenance Area	2	0.5	1.5	33.7	332	261	71	771.7
8- Maintenance Area	1.5	0.5	1	22.5	326	261	65	706.5
9- Maintenance Area	1.5	0.5	1	22.5	270	261	9	97.9
10- Maintenance Area	2.5	0.5	2	44.9	278	261	17	184.8
	MDC-74 dpm/100cm2				MDC-875 dpm/100cm2			

Table 9-9, G-3 Coal Plant Removable Surveys

Description	G-3 Coal Plant Service Building Surveys

	Alpha Removable Contamination				Beta Removable Contamination			
Sample Number	gross cpm	background cpm	net cpm	dpm/100cm^2	gross cpm	background cpm	net cpm	dpm/100cm^2
1- Main Lobby	0.1	0.2	-0.1	-0.75	67	71	-4	-23.3
2- Main Lobby	0.1	0.2	-0.1	-0.75	59	71	-12	-70.2
3- Main Lobby	0.6	0.2	0.4	3.01	72	71	1	5.2
4- Main Lobby	0.1	0.2	-0.1	-0.75	61	71	-10	-58.4
5- Main Lobby	0.1	0.2	-0.1	-0.75	65	71	-6	-35.1
6- Maintenance Area	0.1	0.2	-0.1	-0.75	63	71	-8	-46.8
7- Maintenance Area	0.1	0.2	-0.1	-0.75	51	71	-20	-118.1
8- Maintenance Area	0.1	0.2	-0.1	-0.75	53	71	-18	-105.1
9- Maintenance Area	0.1	0.2	-0.1	-0.75	55	71	-16	-93.7
10- Maintenance Area	0.6	0.2	0.4	3.01	67	71	-4	-23.3
	Alpha MDC: 15 dpm/100cm2				Beta MDC: 89 dpm/100cm2			

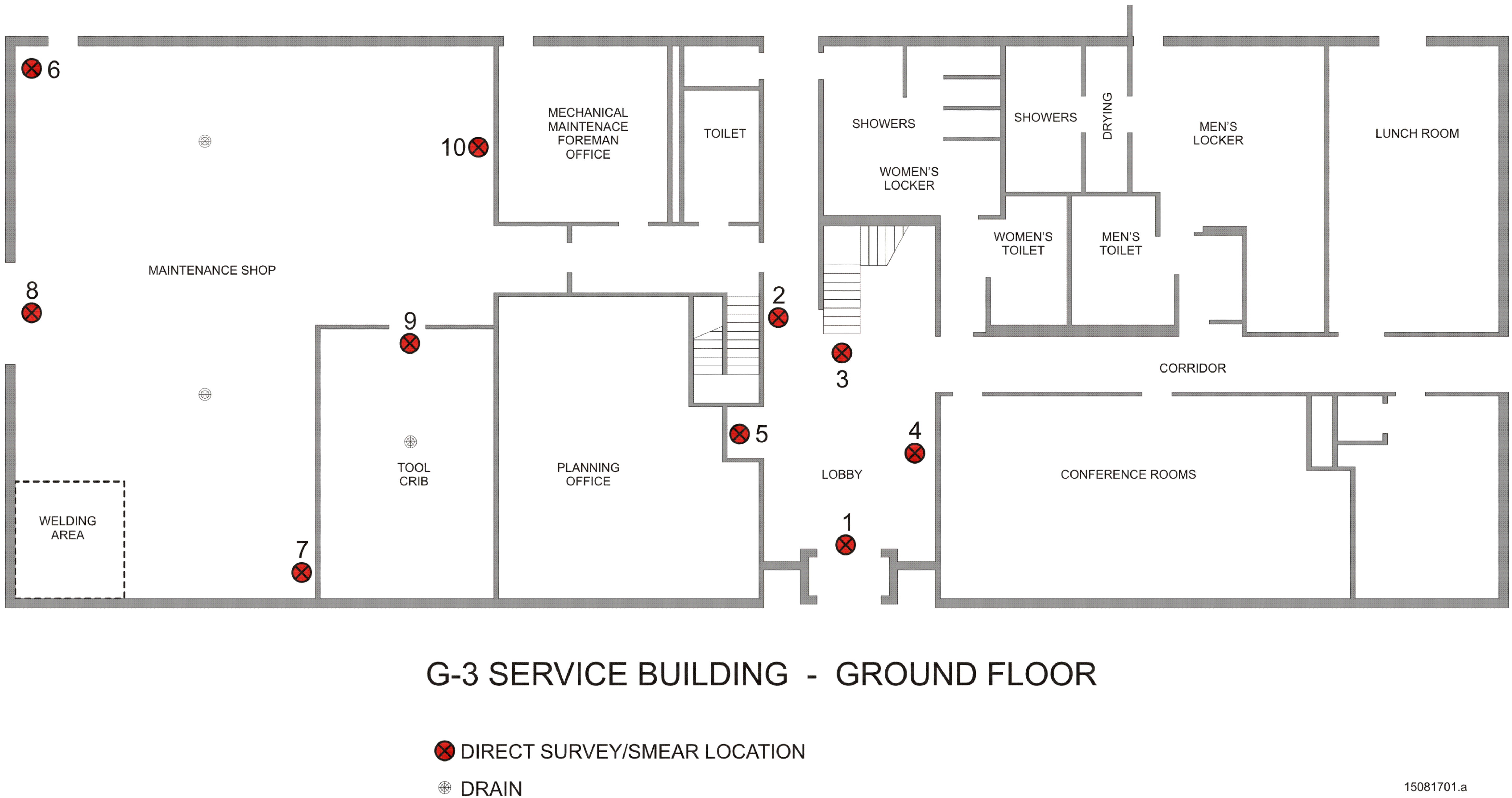


Figure 9-12 G-3 Service Building-Ground Floor Scanning Locations

Table 9-10, Reactor Building Off Site Concrete Coring Results

Reactor Building Concrete Coring Results

Sample ID	Depth (inches)	H-3	C-14	Fe-55	Ni-59	Co-60	Ni-63	Sr-90	Nb-94	Tc-99	Cs-137	Eu-152	Eu-154	Eu-155	Np-237	Pu-238	Pu-239/240	Pu-241	Am-241	Am-243	Cm-243/244
B1001101-CJ-FC-001-CV	0 - 1/2	3.95E+00	1.40E+00	1.60E+01	2.54E-01	1.15E+01	3.93E+01	6.91E+00	3.12E-01	5.66E-01	7.50E+03	9.02E+00	2.41E+00	4.44E+00	4.19E-02	1.76E-01	1.32E-01	2.70E+00	4.07E-01	4.60E-02	5.79E-02
B1001101-CJ-FC-001-CV	1/2 - 1	9.72E+00	1.44E+00	2.13E+01	2.64E-01	1.68E-01	3.85E+00	1.62E-01	1.39E-01	5.26E-01	9.66E+00	5.11E-01	1.26E+00	3.76E-01	4.50E-02	6.55E-02	4.85E-02	2.50E+00	3.66E-02	2.96E-02	4.78E-02
B1001101-CJ-FC-002-CV	0 - 1/2	3.64E+00	1.41E+00	1.13E+01	2.89E-01	8.00E-01	4.86E+00	1.16E+01	1.65E-01	5.64E-01	4.50E+02	2.15E+00	1.46E+00	8.37E-01	4.24E-02	6.17E-01	7.15E-01	2.35E+01	5.50E-01	4.33E-01	4.67E-01
B1001101-CJ-FC-003-CV	0 - 1/2	7.24E+00	1.46E+00	9.45E+00	3.65E-01	5.77E+01	2.02E+02	3.45E+01	3.92E-01	5.86E-01	6.14E+02	2.87E+00	5.46E+00	1.51E+00	4.11E-02	5.28E-01	4.91E-01	7.47E+00	1.45E+00	5.19E-02	8.79E-02
B1001101-CJ-FC-003-CV	1/2 - 1	1.26E+01	1.39E+00	1.43E+01	2.57E-01	4.35E-01	2.99E+00	3.04E-01	1.53E-01	5.96E-01	2.45E+00	3.46E-01	1.02E+00	2.82E-01	5.13E-02	6.44E-02	4.65E-02	2.66E+00	3.76E-02	3.04E-02	1.68E-02
B1001101-CJ-FC-004-CV	0 - 1/2	4.49E+00	1.45E+00	7.60E+00	2.76E-01	2.81E+01	8.83E+01	1.10E+01	2.64E-01	5.92E-01	2.13E+02	1.57E+00	3.07E+00	8.50E-01	5.98E-02	4.26E-01	3.36E-01	2.60E+01	6.14E-01	2.92E-01	4.32E-01
B1001101-CJ-FC-005-CV	0 - 1/2	2.96E+00	1.41E+00	9.77E+00	2.59E+00	7.39E+01	2.21E+02	3.28E+01	5.31E-01	5.62E-01	6.57E+01	1.24E+00	5.65E+00	6.99E-01	1.46E-02	7.06E-01	4.89E-01	9.09E+00	1.59E+00	4.75E-02	1.16E-01
B1001101-CJ-FC-005-CV	1/2 - 1	4.18E+00	1.35E+00	9.87E+00	2.70E-01	1.71E-01	4.87E+00	4.23E-01	1.28E-01	5.40E-01	8.01E-01	3.73E-01	9.23E-01	2.55E-01	5.41E-02	5.30E-02	3.75E-02	2.53E+00	1.80E-02	5.44E-02	1.76E-02
B1001101-CJ-WC-006-CV	0 - 1/2	4.21E+00	1.44E+00	1.17E+01	2.63E-01	2.04E+00	1.34E+01	7.15E+00	1.54E-01	5.80E-01	1.49E+03	3.66E+00	1.19E+00	1.47E+00	4.44E-02	3.67E-02	3.46E-02	2.65E+00	5.24E-02	2.85E-02	4.19E-02
B1001101-CJ-WC-006-CV	1/2 - 1	3.59E+00	1.49E+00	1.45E+01	2.57E-01	2.07E-01	2.31E+00	1.46E-01	1.41E-01	5.56E-01	1.41E+00	4.34E-01	1.27E+00	3.36E-01	4.49E-02	4.32E-02	3.46E-02	2.36E+00	3.58E-02	3.05E-02	2.87E-02

Table 9-11, Waste Treatment Building Off Site Concrete Coring Results

Waste Treatment Building Concrete Coring Results

Sample ID	Depth (inches)	H-3	C-14	Fe-55	Ni-59	Co-60	Ni-63	Sr-90	Nb-94	Tc-99	Cs-137	Eu-152	Eu-154	Eu-155	Np-237	Pu-238	Pu-239/240	Pu-241	Am-241	Am-243	Cm-243/244
B1002101-CJ-FC-001-CV	0 - 1/2	9.85E-01	1.41E+00	1.54E+01	6.55E+01	9.50E+01	8.17E+02	9.98E+01	4.75E-01	5.64E-01	4.71E+03	5.94E+00	5.46E+00	2.12E+00	5.99E-02	9.20E-01	1.28E+00	2.67E+01	3.20E+00	4.06E-01	4.20E-01
B1002101-CJ-FC-001-CV	1/2 - 1	9.31E-01	1.44E+00	5.51E+00	2.45E-01	1.01E+00	5.64E+00	2.53E-01	1.02E-01	5.07E-01	3.87E+00	4.44E-01	1.48E+00	2.72E-01	6.35E-02	5.88E-02	4.20E-02	2.68E+00	3.74E-02	5.44E-02	3.66E-02
B1002101-CJ-FC-001-CV	1 - 1 1/2	1.00E+00	1.41E+00	5.67E+00	2.28E-01	7.42E-01	3.36E+00	2.41E+00	1.49E-01	5.64E-01	3.63E+00	4.20E-01	1.40E+00	3.35E-01	4.47E-02	6.06E-02	4.85E-02	2.82E+00	1.46E-02	2.85E-02	1.43E-02
B1002101-CJ-FC-001-CV	1 1/2 - 2	9.17E-01	1.38E+00	2.50E+01	2.59E-01	8.72E-01	5.73E+00	1.06E-01	1.42E-01	5.28E-01	2.44E+00	3.83E-01	9.56E-01	3.26E-01	5.24E-02	6.11E-02	4.14E-02	2.70E+00	4.14E-02	2.89E-02	3.31E-02
B1002101-CJ-FC-002-CV	0 - 1/2	1.71E+00	1.54E+01	8.96E+00	4.23E+02	3.00E+02	2.36E+03	2.20E+02	9.35E-01	7.92E-01	2.54E+04	1.37E+01	7.17E+00	7.37E+00	4.15E-02	3.20E+00	2.58E+00	3.69E+01	9.08E+00	3.88E-01	5.45E-01
B1002101-CJ-FC-002-CV	1/2 - 1	1.12E+00	6.18E+00	2.53E+01	2.69E-01	8.27E+00	1.75E+02	3.52E+00	2.30E-01	5.22E-01	2.08E+02	1.58E+00	1.90E+00	8.64E-01	5.57E-02	6.42E-02	5.57E-02	2.97E+00	2.38E-01	5.98E-02	4.38E-02
B1002101-CJ-FC-003-CV	0 - 1/2	9.62E-01	1.44E+00	1.70E+01	3.34E-01	8.57E+00	7.06E+01	7.00E+00	2.11E-01	5.43E-01	1.24E+03	2.60E+00	2.32E+00	1.71E+00	4.38E-02	1.05E-01	1.26E-01	2.56E+00	3.39E-01	3.65E-02	1.75E-02

Table 9-12, Pipe Tunnel Off Site Concrete Coring Results

Pipe Tunnel Concrete Coring Results

Sample ID	Depth (inches)	H-3	C-14	Fe-55	Ni-59	Co-60	Ni-63	Sr-90	Nb-94	Tc-99	Cs-137	Eu-152	Eu-154	Eu-155	Np-237	Pu-238	Pu-239/240	Pu-241	Am-241	Am-243	Cm-243/244
B1008101-CJ-FC-002-CV	0 - 1/2	9.40E-01	1.37E+00	6.55E+00	2.34E-01	6.62E-01	2.73E+00	1.54E-01	2.11E-01	5.79E-01	1.60E+01	6.33E-01	1.20E+00	3.57E-01	4.77E-02	5.97E-02	5.43E-02	2.71E+00	4.17E-02	4.36E-02	1.87E-02
B1008101-CJ-FC-003-CV	0 - 1/2	1.30E+00	1.44E+00	1.45E+01	2.41E-01	9.74E-01	4.40E+00	2.35E-01	1.41E-01	5.92E-01	1.00E+01	4.70E-01	1.41E+00	3.48E-01	6.82E-02	5.67E-02	3.36E-02	2.55E+00	5.08E-02	5.24E-02	4.44E-02
B1008101-CJ-FC-004-CV	0 - 1/2	1.20E+00	1.43E+00	1.15E+01	2.29E-01	1.39E+00	4.26E+00	2.16E-01	1.43E-01	5.99E-01	1.98E+01	5.80E-01	1.37E+00	3.76E-01	6.53E-02	5.20E-02	4.21E-02	2.61E+00	5.19E-02	3.18E-02	1.50E-02

Table 9-13, Off Site Soil Sample Results

L1010102 Soil Sample Results

Sample ID	Depth (feet)	H-3	C-14	Fe-55	Ni-59	Co-60	Ni-63	Pm-147	Sr-90	Nb-94	Tc-99	Cs-137	Eu-152	Eu-154	Eu-155	Np-237	Pu-238	Pu-239/240	Pu-241	Am-241	Am-243	Cm-243/244
L1010102-QQ-GS-001-SB	15	**7.37E-01**	**1.39E+00**	**1.96E+00**	**1.73E+00**	**9.21E-02**	*3.15E+00*	**9.68E-01**	**2.93E-01**	**9.64E-02**	**4.87E-01**	**9.56E-02**	**1.78E-01**	**5.37E-01**	**1.50E-01**	**4.00E-02**	**5.44E-02**	**3.78E-02**	**2.06E+00**	**2.90E-02**	**2.29E-02**	**1.30E-02**
L1010102-QQ-GS-002-SB	15	**6.56E-01**	**1.40E+00**	**1.93E+00**	**1.83E+00**	**6.46E-02**	*3.98E+00*	**9.67E-01**	**3.35E-01**	**6.61E-02**	**4.99E-01**	**6.74E-02**	**1.49E-01**	**1.86E-01**	**1.39E-01**	**4.36E-02**	**5.38E-02**	**3.74E-02**	**2.09E+00**	**3.36E-02**	**5.23E-02**	**2.94E-02**
L1010102-QQ-GS-003-SB	20	**6.53E-01**	**1.38E+00**	**1.95E+00**	**1.77E+00**	**8.83E-02**	*2.78E+00*	**7.90E-01**	**2.78E-01**	**6.18E-02**	**5.49E-01**	**5.20E-02**	**1.64E-01**	**2.31E-01**	**1.13E-01**	**5.50E-02**	**5.15E-02**	**3.34E-02**	**2.32E+00**	**2.80E-02**	**3.49E-02**	**1.25E-02**
L1010102-QQ-GS-004-SB	20	**6.39E-01**	**1.38E+00**	**1.87E+00**	**1.72E+00**	**1.04E-01**	*2.78E+00*	**9.74E-01**	**2.95E-01**	**7.16E-02**	**6.12E-01**	**4.89E-02**	**1.87E-01**	**4.40E-01**	**1.60E-01**	**4.28E-02**	**6.89E-02**	**3.49E-02**	**2.46E+00**	**3.13E-02**	**4.13E-02**	**3.06E-02**
L1010102-QQ-GS-005-SB	10	**6.53E-01**	**1.40E+00**	**2.01E+00**	**1.76E+00**	**8.55E-02**	*3.61E+00*	**1.00E+00**	**4.96E-01**	**4.69E-02**	**6.00E-01**	**6.05E-02**	**1.57E-01**	**5.40E-01**	**1.59E-01**	**6.17E-02**	**5.36E-02**	**4.78E-02**	**2.27E+00**	**3.25E-02**	**2.97E-02**	**3.19E-02**

LEGEND
SB=subsurface soil
FC= Concrete Floor
WC= Concrete Wall

NOTES:
Bold = MDC
Italicized= activity result>MDC

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