LA CROSSE BOILING WATER REACTOR
LICENSE TERMINATION PLAN
CHAPTER 5, REVISION 1
REMEDIATION PLAN

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LIST OF ACRONYMS AND ABBREVIATIONS

AF	Area Factor
ALARA	As Low As Reasonably Achievable
AMCG	Average Member of the Critical Group
BFM	Basement Fill Model
CAQ	Conditions Adverse to Quality
CsI	Cesium Iodide
CoC	Chain of Custody
CVS	Contamination Verification Survey
DPC	Dairyland Power Cooperative
DCGL	Derived Concentration Guideline Levels
DQA	Data Quality Assessment
DQO	Data Quality Objectives
EMC	Elevated Measurement Comparison
ETD	Easy to Detect
FOV	Field of View

FSS	Final Status Survey
G-3	Genoa 3 Fossil Station
GPS	Global Positioning System
HPGe	High-Purity Germanium
HSA	Historical Site Assessment
HTD	Hard to Detect
ILAC	International Laboratory Accreditation Cooperation
ISFSI	Independent Spent Fuel Storage Installation
ISOCS	*in situ* Object Counting System
LACBWR	La Crosse Boiling Water Reactor
LBGR	Lower Bound of the Gray Region
LSE	LACBWR Site Enclosure
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
MRA	Mutual Recognition Arrangement
NAD	North American Datum
NaI	Sodium Iodide
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control
RA	Radiological Assessment
RASS	Remedial Action Support Survey
RESRAD	RESidual RADioactive Materials
ROC	Radionuclides of Concern
SOF	Sum of Fractions
SOP	Standard Operating Procedures
TEDE	Total Effective Dose Equivalent
TSD	Technical Support Document
UBGR	Upper Bound of the Gray Region
UCL	Upper Confidence Level
WGTV	Waste Gas Tank Vault
WTB	Waste Treatment Building

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5. Final Status Survey Plan

The purpose of the Final Status Survey (FSS) Plan is to describe the methods to be used in planning, designing, conducting, and evaluating the FSS at the La Crosse Boiling Water Reactor (LACBWR). The FSS Plan describes the final survey process used to demonstrate that the LACBWR facility and site comply with the radiological criteria for unrestricted use specified in 10 CFR 20.1402. Nuclear Regulatory Commission (NRC) regulations applicable to FSS are found in 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b).

The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 are; 1) 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 (AMCG) that does not exceed 25 millirem/year (mrem/yr), including that from groundwater sources of drinking water, and 2) the residual radioactivity has been reduced to levels that are As Low As Reasonably Achievable (ALARA).

Chapter 4 describes the methodologies and criteria that will be used to perform remediation activities and to demonstrate compliance with the ALARA criterion.

This FSS Plan has been developed using the guidance contained in the following documents:

- NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (1)
- NUREG-1505, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys* (2)
- NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions* (3)
- NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans* (4)
- NUREG-1757, Volume 2, Revision 1, *Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report* (5)
- Regulatory Guide 1.179, *Standard Format and Content of License Termination Plans for Nuclear Power Reactors* (6)

Dose modeling, as discussed in Chapter 6, was performed to develop the residual radioactivity levels that correspond to the 25 mrem/yr dose criteria. Site-specific, concentration-based Derived Concentration Guideline Levels (DCGL) were calculated for surface soils, buried pipe, and basement structures (i.e., basements to be backfilled). Default screening values from NUREG-1757, Appendix H, Table H-1 will be applied to above grade structures that will remain at the time of license termination.

It is anticipated that the NRC will choose to conduct confirmatory measurements during the implementation of FSS to assist the NRC in making a determination that the FSS was performed in accordance with this plan.

The FSS Plan includes the radiological assessment of all impacted backfilled structures, excavations created as a result of the removal of basement structures, buried piping, open land areas and above grade buildings that will remain following decommissioning. After successful implementation of this FSS

Plan, LaCrosse*Solutions* (Solutions) intends to release for unrestricted use the impacted open land areas, remaining backfilled structures, buried piping and above grade buildings from the 10 CFR 50 license, with the exception of the immediate area surrounding the Independent Spent Fuel Storage Installation (ISFSI). The ISFSI was established under the general license provisions of 10 CFR 72.210. This FSS Plan does not address non-impacted areas as identified in Chapter 2.

As indicated in Chapter 3 of this License Termination Plan (LTP), the Reactor Building and the Waste Gas Tank Vault (WGTV) will be demolished and removed to a depth of at least 3 feet below grade. All other impacted LACBWR buildings, structures and components, other than the following structures, will be demolished and removed in their entirety. The impacted above grade structures that will remain are:

- LACBWR Administration building
- G-3 Crib House
- LACBWR Crib House
- Transmission Sub-Station Switch House
- G-1 Crib House
- Barge Wash Break Room
- Back-up Control Center
- Security Station

None of the buildings and structures associated with the Genoa 3 Fossil Station (G-3) are expected to be radiologically impacted. Therefore, the structures associated with G-3 will remain intact and functional for G-3 power operations. The G-3 Crib House is classified as impacted due to its location in an impacted soil survey unit. The impacted above grade structures that will remain are not expected to contain residual radioactivity and will be subjected to FSS. The site and public roads and railways that traverse through the site will also remain.

The backfilled structures that will remain at license termination and be subjected to FSS include the basements of the Reactor Building and the WGTV. The Waste Treatment Building (WTB), Piping and Ventilation Tunnel, Reactor/Generator Plant, the Chimney Foundation, and the Turbine Building (including the sump and the Turbine pit) will be removed in their entirety. The resultant excavations will undergo FSS. All systems and components (associated with the structures) as well as all structures above the 636 foot elevation (with the exception of the minor buildings previously noted) will be removed during the decommissioning process and disposed of as a waste stream.

In the Reactor Building, all internal structural surfaces, systems and components will be removed. All internal concrete will be removed to expose the steel liner, which will also be removed, leaving only the remaining structural concrete outside the liner below the 636 foot elevation (i.e., concrete "bowl" below 636 foot elevation, concrete pile cap and piles). In the WGTV, the remaining structure will consist of the floors and foundation walls (including support columns), as well as concrete piling cap and piles below the 636 foot elevation.

The backfilled structures that will remain at LACBWR following the termination of the license are constructed of steel-reinforced concrete which will be covered by at least three (3) feet of soil and physically altered to a condition which would not allow the remaining backfilled structures to be plausibly occupied.

The End State will also include a range of buried piping as provided in Table 24 of EnergySolutions Technical Support Document (TSD) RS-TD-313196-004, *LACBWR Soil DCGL, Basement Concrete DCGL, and Buried Pipe DCGL* (7). The list of buried piping presented in RS-TD-313196-004 is intended to be a bounding end-state condition. No pipe that is not listed in Table 24 of RS-TD-313196-004 will be added to the end-state condition however, pipe can be removed from the list and disposed of as waste.

5.1. Radionuclides of Concern and Mixture Fractions

Energy*Solutions* TSD RS-TD-313196-001, *Radionuclides of Concern during LACBWR Decommissioning* (8) establishes the basis for an initial suite of potential Radionuclides of Concern (ROC) for decommissioning. Industry guidance was reviewed as well as the analytical results from the sampling of various media from past plant operations. Based on the elimination of some of the theoretical neutron activation products, noble gases and radionuclides with a half-life less than two years, an initial suite of potential ROC for the decommissioning of LACBWR was prepared. The initial suite is listed in Table 5-1.

LTP Chapter 2 provides detailed characterization data that describes current contamination levels in the basements and soils from the characterization campaign conducted from September 2014 through August 2015. The initial survey data for basements was based on core samples obtained from the walls and floors of the Reactor Building, WTB and the balance of the basement structures (primarily the Piping Tunnels) at biased locations with elevated contact dose rates, contamination levels, and/or evidence of leaks/spills. During subsequent characterizations, additional cores were obtained from the Reactor Building and the WGTV. Surface and subsurface soil samples were taken in each impacted open land survey unit (including soil beneath and adjacent to basements) and analyzed for the presence of plant-derived radionuclides. TSD RS-TD-313196-001 evaluates the results of the concrete core analysis data from the Reactor Building, WTB, Piping Tunnels and WGTV and refines the initial suite of potential ROC by evaluating the dose significance of each radionuclide.

Insignificant dose contributors were determined consistent with the guidance contained in section 3.3 of NUREG-1757. In all soil and concrete scenarios, Cs-137, Co-60, Sr-90, Eu-152 and Eu-154 contribute nearly 100% of the total dose. The remaining radionuclides were designated as insignificant dose contributors and are eliminated from further detailed evaluation. Therefore, the final ROCs for LACBWR soil, basement concrete and buried piping are Cs-137, Co-60, Sr-90, Eu-152 and Eu-154.

LTP Chapter 6, section 6.13.1 discusses the process used to derive the dose significant ROC for the decommissioning of LACBWR, including the elimination of insignificant dose contributors from the initial suite. Table 5-2 presents the ROC for the decommissioning of LACBWR and the normalized mixture fractions based on the radionuclide distribution from TSD RS-TD-313196-001.

The results of surface and subsurface soil characterization in the impacted area surrounding LACBWR indicate that there is minimal residual radioactivity in soil. Based on the characterization survey results to date, LACBWR does not anticipate the presence of significant soil contamination in any remaining subsurface soil that has not yet been characterized. In addition, minimal contamination is expected in the buried piping that LACBWR plans to leave in place (Section 5.7.1.8 provides additional information).

Table 5-1 Initial Suite of Radionuclides

Radionuclide	Half Life (Years)
H-3	1.24E+01
C-14	5.73E+03
Fe-55	2.70E+00
Ni-59	7.50E+04
Co-60	5.27E+00
Ni-63	9.60E+01
Sr-90	2.91E+01
Nb-94	2.03E+04
Tc-99	2.13E+05
Cs-137	3.00E+01
Eu-152	1.33E+01
Eu-154	8.80E+00
Eu-155	4.76E+00
Np-237	2.14+06
Pu-238	8.78E+01
Pu-239	2.41E+04
Pu-240	6.60E+03
Pu-241	1.44E+01
Am-241	4.32E+02
Cm-243/244	1.81E+01

Table 5-2 Dose Significant Radionuclides and Mixture[1]

Radionuclide	Soil/Pipe Mix Fraction[2]	Rx Bldg. Mix Fraction	WGTV Mix Fraction
Co-60	6.44E-02	7.41E-02	1.01E-02
Sr-90	9.81E-02	1.23E-01	1.94E-02
Cs-137	8.29E-01	7.96E-01	9.57E-01
Eu-152	5.49E-03	2.97E-03	9.56E-03
Eu-154	2.81E-03	4.04E-03	3.42E-03

1.) Based on maximum percent of total activity from Table 22 of RS-TD-313196-001, normalized to one for the dose significant radionuclides.

2.) The values for Soil/Pipe are intended to be applied to above grade buildings and other materials not associated with the Rx Bldg. or WGTV.

It is assumed that the contaminated water that caused concrete contamination would be similar to any potential source of soil contamination. Consequently, the ROC and radionuclide mixture derived for the concrete was considered to be reasonably representative of soils and buried piping for FSS planning and implementation. Note that due to the expectation of very low concentrations of soil and piping contamination, any uncertainties in the application of the concrete derived radionuclide mixture to soil and buried piping would be very unlikely to cause significant dose variability in relation to the 25 mrem/yr dose criteria. In addition, the FSS for soil and concrete will use gamma spectroscopy which directly measures gamma-beta emitters eliminating uncertainty related to beta-gamma mixture fractions. The uncertainty in mixture fractions of Hard-to-Detect (HTD) radionuclides, which will not be directly measured, corresponds to a very low uncertainty in dose.

Sufficient characterization samples have been taken of the Reactor Building, WTB, WGTV and Remaining Structures concrete to derive the radionuclide mixture and assess the dose impact of HTD radionuclides.

The previously inaccessible areas identified in LTP Chapter 2 will be characterized during the continuing characterization process. In order to verify that the insignificant contributor (IC) dose does not change prior to implementing the FSS, and to verify the HTD to surrogate radionuclide ratios used for the surrogate calculation are still valid, LACBWR will obtain and analyze concrete core and soil samples during continuing characterization (including radiological assessments) and FSS within each individual survey unit or area as described below.

For continuing characterization, 10% of all media samples collected in a survey unit during continuing characterization will be analyzed for HTD radionuclides, with a minimum of one sample analyzed for HTD radionuclides, whichever is greater. In addition, a minimum of one sample beyond the 10% minimum will be selected at random, also for HTD radionuclide analysis. All samples will first be analyzed by the on-site gamma spectroscopy system. In the absence of detectable gamma activity, locations will be selected based on the potential for the presence of activity using HSA information or other process knowledge data. All samples selected for HTD analysis during continuing characterization will be analyzed for the full suite of radionuclides from Table 5-1.

The actual IC dose will be calculated for each individual sample result using the DCGLs from TSD RS-TD-313196-004, Table 4 for soils and Table 35 for basement structures. If the IC dose calculated is less than the IC dose assigned for DCGL adjustment, then no further action will be taken. If the actual IC dose calculated from the sample result is greater than the IC dose assigned for DCGL adjustment, then a minimum of five (5) additional investigation samples will be taken around the original sample location. Each investigation sample will be analyzed by the on-site gamma spectroscopy system and sent for HTD analysis (full suite of radionuclides from Table 5-1). As with the original sample, the actual IC dose will be calculated for each investigation sample. In this case, the actual calculated maximum IC dose from an individual sample observed in the survey unit will be used to readjust the DCGLs in that survey unit. If the maximum IC dose exceeds 10%, then the additional radionuclides that were the cause of the IC dose exceeding 10% will be added as additional ROC for that survey unit. The survey unit-specific DCGLs used for compliance, the ROC for that survey unit and the survey data serving as the basis for the IC dose adjustment will be documented in the release record for the survey unit.

The final ROC for the decommissioning of LACBWR are Co-60, Cs-137, Eu-152 and Eu-154, which are gamma emitters and Sr-90 which is an HTD radionuclide. For sample(s) analyzed for HTD radionuclides during continuing characterization, if the analysis of the sample indicates positive results (greater than MDC) for both a HTD ROC (Sr-90) and the corresponding surrogate radionuclide (Cs-137), then the HTD to surrogate ratio will be derived. If the derived HTD to surrogate ratio is less than the applicable HTD to surrogate ratio from TSD RS-TD-313196-001, Table 40, then no further action is required. If the HTD to surrogate ratio exceeds the applicable ratio from TSD RS-TD-313196-001, Table 40, then a minimum of five (5) additional investigation samples will be taken around the original sample location. Each investigation sample will be analyzed by the on-site gamma spectroscopy system and then sent for HTD analysis. As with the original sample, the HTD to surrogate ratio will be calculated for each investigation sample. The actual maximum HTD to surrogate ratio observed in any individual sample will be used to infer HTD radionuclide concentrations in the survey units shown to be impacted by the investigation. The survey unit-specific HTD to surrogate ratio and the survey data serving as the basis for the ratio will be documented in the release record for the survey unit(s).

Survey unit-specific surrogate ratios, in lieu of the ratios from TSD RS-TD-313196-001, Table 40, may be used for compliance if sufficient radiological data exists to demonstrate that a different ratio is representative for the given survey unit. In these cases, the survey unit-specific radiological data and the derived surrogate ratios will be submitted to the NRC for approval. If approved, then the survey unit-specific ratios used and the survey data serving as the basis for the surrogate ratios will be documented in the release record for the survey unit.

Radiological Assessment (RA) surveys will be performed in currently inaccessible soil areas that are exposed after removal of asphalt or concrete roadways and parking lots, rail lines, or building foundation pads (slab-on-grade). A limited number of soil samples are typically collected as a part of the RA. Ten percent (10%) of any soil samples collected during an RA in a survey area, with a minimum of one sample, will be analyzed for the full initial suite of radionuclides. Additionally, if levels of residual radioactivity in an individual soil sample exceed the Sum-of-Fractions (SOF) of 0.1 (using the Operational DCGL) then the sample(s) will be analyzed for full suite of radionuclides from Table 5-1.

Soil samples and concrete cores will be collected during FSS to confirm the HTD to surrogate radionuclide ratios used for the surrogate calculation. Only Sr-90 will be analyzed in the FSS confirmatory samples. Concrete cores will be collected from the Waste Gas Tank Vault basement where concrete will remain. The number of cores collected and analyzed for ROC HTD will be ten percent (10%) of the number of FSS ISOCS measurements. The concrete core locations will be selected from the floor and lower walls in the survey unit to alleviate safety concerns from working at heights and to focus on the areas expected to contain the majority of residual radioactivity. For soil, ten percent (10%) of the FSS samples collected from open land survey units (including excavations where major sub-grade structures previously resided) will also be analyzed for ROC HTD radionuclides. Additionally, if levels of residual radioactivity in an individual soil sample exceed a SOF of 0.1 (using the Operational DCGL), then the sample(s) will be analyzed for ROC HTD radionuclides. For soil samples or concrete cores with positive results for both a Sr-90 and the corresponding surrogate radionuclide (Cs-137), the HTD to surrogate ratio will be derived. The applicable ratio from TSD RS-TD-313196-001, Table 40 will be used unless specific survey information from continuing characterization supports the use of a surrogate ratio that is specific to the area. In these cases, the area-specific ratios as determined by actual survey data will be used in lieu of the TSD RS-TD-313196-001, Table 40 ratios. The area-specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit.

For Quality Assurance (QA), 10% of the soil samples taken during FSS will also be analyzed for Sr-90 in addition to the gamma emitting radionuclides. For concrete structures, this requirement will be accomplished by taking concrete core samples. The number of concrete cores taken will be 10% of the number of direct measurements taken during FSS. The concrete core locations will be selected on the floor and lower walls in the survey unit to alleviate safety concerns from working at heights. In addition, the majority of the source term is expected in the lower walls and floors. For the analysis of FSS samples, if the sample has positive results (greater than MDC) for both Sr-90 and Cs-137, then the Sr-90/Cs-137 ratio will be compared to the Sr-90/Cs-137 ratio assigned for use in the surrogate calculation for Sr-90 (see section 5.2.4). If the Sr-90/Cs-137 ratio from the sample or core data exceeds the assigned ratio, then the ratio from the continuing characterization core will be applied to the FSS surrogate calculations for the survey unit.

5.2. <u>Release Criteria</u>

Before the FSS process can proceed, the DCGLs (referred to in this chapter as Base Case DCGLs to ensure a clear distinction from Operational DCGLs) that are used to demonstrate compliance with the 25 mrem/yr unrestricted release criterion must be established. The Base Case DCGLs are calculated by analysis of various pathways (direct radiation, inhalation, ingestion, etc.), media (concrete, soils, and groundwater) and scenarios through which exposures could occur. Chapter 6 of this LTP describes in detail the approach, modeling parameters and assumptions used to develop the Base Case DCGLs.

Each radionuclide-specific Base Case DCGL is equivalent to the level of residual radioactivity (above background levels) that could, when considered independently, result in a TEDE of 25 mrem per year to an AMCG. To ensure that the summation of dose from each source term is 25 mrem/yr or less after all FSS is completed, the Base Case DCGLs are reduced based on an expected, or a priori, fraction of the 25 mrem/yr dose limit from each source term. The reduced DCGLs, or "Operational" DCGLs can be related to the Base Case DCGLs as an expected fraction of dose based on an a priori assessment of what the expected dose should be based on the results of site characterization, process knowledge and the extent of planned remediation. The Operational DCGL is then used as the DCGL for the FSS design of the survey unit (calculation of surrogate DCGLs, investigations levels, etc.). Details of the Operational DCGLs derived for each dose component and the basis for the applied a priori dose fractions are provided in LC-FS-TSD-002, *Operational Derived Concentration Guideline Levels for Final Status Survey* (9).

At LACBWR, compliance is demonstrated through the summation of dose from five distinct source terms for the end-state (basements, soils, buried pipe, above ground structures, and groundwater). When applied to backfilled basements below 636 foot elevation, the DCGLs are expressed in units of activity per unit of area (pCi/m^2). When applied to soil, the DCGLs are expressed in units of activity per unit of mass (pCi/g). For above grade buildings that will remain and buried piping, DCGLs are calculated and expressed in units of activity per surface area (dpm/100 cm^2).

The dose contribution from each ROC is accounted for using the Sum-of-Fractions (SOF) to ensure that the total dose from all ROC does not exceed the dose criterion.

A Base Case DCGL that is established for the average residual radioactivity in a survey unit is called a $DCGL_W$. The $DCGL_W$ can be multiplied by Area Factors (AF) to obtain a Base Case DCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. The scaled value is called the $DCGL_{EMC}$, where EMC stands for Elevated Measurement Comparison. The $DCGL_{EMC}$ will only be applied to Class 1 open land (soil) survey units.

5.2.1. Base Case Derived Concentration Guideline Levels for Basement Surfaces

The Basement Fill Model (BFM) applies to the steel-reinforced concrete walls and floors of the backfilled Reactor Building and WGTV below the 636 foot elevation.

BFM DCGLs ($DCGL_B$) apply to basement concrete and are calculated in LTP Chapter 6, section 6.13. The insignificant dose contributor percentages for the most limiting basement scenario was used to adjust the $DCGL_B$ to account for the dose from the eliminated insignificant contributor radionuclides. The $DCGL_B$ values from LTP Chapter 6, section 6.13 are reproduced in Table 5-3.

Table 5-3 Base Case DCGLS for Basements ($DCGL_B$)

ROC	Rx Bldg $DCGL_B$ (pCi/m^2)	WGTV $DCGL_B$ (pCi/m^2)
Co-60	5.16E+06	4.10E+06
Sr-90	1.45E+07	6.40E+06
Cs-137	2.17E+07	1.76E+07
Eu-152	1.19E+07	9.69E+06
Eu-154	1.10E+07	8.97E+06

The $DCGL_B$ values in Table 5-3 correspond to summation of the total dose from the three BFM scenarios (Groundwater, Drilling Spoils, and Excavation) and are used for FSS. Using the summed dose is conservative because the *in situ* scenarios (groundwater and drilling spoils) cannot occur simultaneously with the Excavation scenario. Note that for the purpose of dose modelling nomenclature, *in situ* is designated as "Insitu" in LTP Chapter 6 and in this Chapter. Section 6.13 of LTP Chapter 6 also calculates DCGLs for each of the three scenarios separately.

5.2.2. Operational Derived Concentration Guideline Levels for Basement Surfaces

The Operational DCGLs for FSS of basement structural surfaces are shown in Table 5-4. Additional information pertaining to Operational DCGLs is provided in LC-FS-TSD-002.

Table 5-4 Operational DCGLS for Basements ($OpDCGL_B$)

ROC	Rx Bldg $OpDCGL_B$ (pCi/m^2)	WGTV $OpDCGL_B$ (pCi/m^2)
Co-60	3.61E+05	2.87E+05
Sr-90	1.02E+06	4.48E+05
Cs-137	1.52E+06	1.23E+06
Eu-152	8.33E+05	6.78E+05
Eu-154	7.71E+05	6.28E+05

5.2.3. Base Case Derived Concentration Guideline Levels for Soil

The results of surface and subsurface soil characterization in the impacted area surrounding LACBWR show that there is minimal residual radioactivity in soil. At this time, based on the characterization survey results to date, the presence of significant concentrations of soil contamination is not anticipated. Surface soil is usually defined as soil residing in the first 0.15 m layer of soil. For LACBWR, soils are defined as a layer of soil beginning at the surface but extending to a depth of 1 m to allow for flexibility in compliance demonstration if contamination deeper than 0.15 m is encountered. Based on characterization data and historical information, there are no expectations of encountering a source term geometry that is comprised of a clean surface layer of soil over a contaminated subsurface soil layer. Energy*Solutions* TSD RS-TD-313196-004 and LTP Chapter 6, section 6.8 provide the exposure scenarios and modeling parameters that were used to calculate the site-specific soil DCGLs. The adjusted soil DCGLs for the unrestricted release of open land survey units as provided in Chapter 6, section 6.13 are reproduced in Table 5-5. The insignificant dose contributor percentages for the most limiting basement scenario was used to adjust the DCGLs for soil to account for the dose from the eliminated insignificant contributor radionuclides. Dose assessment in soils will be performed using the soil DCGLs in Table 5-5. However, administrative action levels will be set for soils to ensure that the mean soil concentrations at license termination are less than the values in Table H-2 of NUREG-1757, Appendix H.

Table 5-5 Base Case DCGLs for Soil (DCGLs)

ROC	Soil DCGL (pCi/g)
Co-60	1.06E+01
Sr-90	5.47E+03
Cs-137	4.83E+01
Eu-152	2.36E+01
Eu-154	2.19E+01

5.2.4. Operational Derived Concentration Guideline Levels for Soil

The operational DCGLs for FSS of surface and subsurface soils are presented in Table 5-6. Once the FSS of structures is complete, the Operational DCGLs for soils may be revised by incorporating the difference between the *a priori* fraction of dose for the maximum basement and the actual fraction of dose for the maximum basement as measured by FSS results. Additional information pertaining to Operational DCGLs is provided in LC-FS-TSD-002.

Table 5-6 Operational DCGLs for Soil ($OpDCGL_S$)

ROC	Soil $OpDCGL_S$ (pCi/g)
Co-60	3.83
Sr-90	1970.45
Cs-137	17.39
Eu-152	8.51
Eu-154	7.89

5.2.5. Base Case Derived Concentration Guideline Levels for Buried Piping

Buried piping is defined as below ground pipe located outside of structures and basements. The dose assessment methods and resulting DCGLs for buried piping are described in detail in LTP Chapter 6, section 6.18. The buried piping was separated into two categories. The first category included the summation and grouping of all impacted buried pipe other than the Circulating Water Discharge Piping and is designated as the "Buried Pipe Group". The second category consisted of the Circulating Water Discharge Pipe only.

The final DCGLs to be used during FSS account for the fact that the dose from the *Insitu* and Excavation scenarios must be summed in the conceptual model for buried pipe dose assessment, i.e., the *Insitu* and Excavation scenarios occur in parallel. The summed Buried Pipe DCGLs are reproduced in Table 5-7 below. The insignificant dose contributor percentages for each of the buried pipe scenarios were used to adjust each buried pipe DCGL to account for the dose from the eliminated insignificant contributor radionuclides.

Table 5-7 Base Case DCGLs for Buried Piping ($DCGL_{BP}$)

ROC	Buried Pipe Group (dpm/100 cm^2)	Buried Pipe Circulating Water Discharge (dpm/100 cm^2)
Co-60	7.50E+04	7.75E+04
Sr-90	5.16E+05	7.55E+05
Cs-137	3.18E+05	3.30E+05
Eu-152	1.64E+05	1.67E+05
Eu-154	1.52E+05	1.56E+05

5.2.6. Operational Derived Concentration Guideline Levels for Buried Piping

The operational DCGLs for the FSS of buried piping are presented in Table 5-8. Once the FSS of structures is complete, the Operational DCGLs for buried piping may be revised by incorporating the difference between the *a priori* fraction of dose for the maximum basement and the actual fraction of dose for the maximum basement as measured by FSS results. Additional information pertaining to Operational DCGLs is provided in LC-FS-TSD-002.

Table 5-8 Operational DCGLs for Buried Piping ($OpDCGL_{BP}$)

ROC	Buried Pipe Group $OpDCGL_{BP}$ (dpm/100 cm^2)	Buried Pipe Circulating Water Discharge $OpDCGL_{BP}$ (dpm/100 cm^2)
Co-60	1.57E+04	1.63E+04
Sr-90	1.08E+05	1.58E+05
Cs-137	6.68E+04	6.94E+04
Eu-152	3.44E+04	3.51E+04
Eu-154	3.20E+04	3.27E+04

5.2.7. Base Case Derived Concentration Guideline Levels for Above Grade Buildings

The Base Case DCGLs for Above Grade Buildings are the screening values from US Nuclear Regulatory Commission, NUREG 1757, Vol. 2, Rev. 1, Consolidated Decommissioning Guidance, Final Report, September 2006 and are reproduced below in Table 5-9.

Table 5-9 Base Case DCGLs for Above Grade Buildings ($DCGL_{AGB}$)

ROC	Above Grade Building (dpm/100 cm^2)
Co-60	7100
Sr-90	8700
Cs-137	28000
Eu-152	12700
Eu-154	11500

5.2.8. Operational Derived Concentration Guideline Levels for Above Grade Buildings ($OpDCGL_{AGB}$)

The operational DCGLs for the FSS of buried piping are presented in Table 5-10. Additional information pertaining to Operational DCGLs is provided in LC-FS-TSD-002.

Table 5-10 Operational DCGLs for Above Grade Buildings ($OpDCGL_{AGB}$)

ROC	Above Grade Building (dpm/100 cm^2)
Co-60	1136
Sr-90	1392
Cs-137	4480
Eu-152	2032
Eu-154	1840

5.2.9. Surrogate Radionuclides

The instrumentation and methods used for FSS will be based on the measurement of beta-gamma emitting radionuclides by either gamma spectroscopy or gross counting. The option is available to use gross beta measurements for survey of piping, but this approach is not currently planned. Assuming gamma measurements are used for the survey, the concentrations of the HTD radionuclide(s) will be based on known ratio(s) of the HTD radionuclide(s) to beta-gamma radionuclide(s) when demonstrating compliance with the release criteria. This is accomplished through the application of a surrogate relationship.

As a general rule, surrogate ratio DCGLs are developed and applied to land areas and materials with residual radioactivity where fairly constant radionuclide concentration ratios can be demonstrated to exist. They are in most cases derived using pre-remediation site characterization data collected prior to the FSS. A surrogate ratio DCGL allows the DCGLs specific to HTD radionuclides in a mixture to be expressed in terms of a single radionuclide that is more readily measured or Easy-to-Detect (ETD). The ETD or measured radionuclide is typically a beta-gamma emitting radionuclide and is called the surrogate radionuclide.

As previously discussed in section 5.1, the radionuclide mixture for concrete developed in TSD RS-TD-313196-001 is listed in Table 5-2. Sr-90 is an HTD radionuclide. Cs-137 is the principle surrogate radionuclide (or ETD radionuclide) for the LACBWR site. The ratio of Sr-90 to Cs-137 is required to implement the surrogate approach. Both Sr-90 and Cs-137 was positively detected in all 30 concrete core samples assessed in the Reactor Building, Tunnel, and Waste Treatment Building. For the WGTV, Sr-90 was not detected in any of the 8 core samples. The 95% UCL of the Cs-137 fractions was chosen to represent the overall nuclide mix for soils/buried pipe, the Reactor Building, and the WGTV as reproduced in Table 5-11.

Table 5-11 Final Sr-90 to Cs-137 Surrogate Ratios

Building or Area	Sr-90/Cs-137 Surrogate Activity Ratio
WGTV	6.75E-02
Rx Bldg.	5.00E-01
Tunnel[1]	2.24E-02
WTB[1]	4.82E-01
Soils[2]	5.02E-01

1.) These buildings are not in the site end-state.
2.) The soil designation represents all concrete core bores and could also be used for other miscellaneous structures (e.g., above grade structures) to remain if needed.

Any future continuing characterization or FSS data that contains positive results for Sr-90 will be reviewed. In these cases, the area specific ratios as determined by actual survey data will be used in lieu of the maximum ratios presented in Table 5-11. The area-specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit. If the derived HTD to surrogate ratio is less than the applicable HTD to surrogate ratio from TSD RS-TD-313196-001, Table 40, then no further action is required.

Using the appropriate scaling factors, the DCGL of the measured radionuclide (Cs-137) will be modified to account for the inferred radionuclide (Sr-90) according to the following equation from section 4.3.2 of MARSSIM:

Equation 5-1

$$DCGL_{SUR} = DCGL_{ETD} \times \frac{DCGL_{HTD}}{\left[\left({Conc_{HTD}}/{Conc_{ETD}} \right)(DCGL_{ETD})\right] + DCGL_{HTD}}$$

where:

$DCGL_{SUR}$	=	modified DCGL (or Basement Dose Factor) for surrogate ratio,
$DCGL_{ETD}$	=	DCGL for easy-to-detect radionuclide,
$DCGL_{HTD}$	=	DCGL for the hard-to-detect radionuclide,
$Conc_{HTD}$	=	Ratio of the HTD or inferred radionuclide, and
$Conc_{ETD}$	=	Ratio of the ETD or surrogate radionuclide.

5.2.10. Sum-of-Fractions

The SOF or "unity rule" is applied to the data used for the survey planning, and data evaluation and statistical tests for soil sample analyses since multiple radionuclide-specific measurements will be performed or the concentrations inferred based on known relationships. The application of the unity rule serves to normalize the data to allow for an accurate comparison of the various data measurements to the release criteria. When the unity rule is applied, the $DCGL_w$ (used for the nonparametric statistical test) becomes one (1). The use and application of the unity rule is performed in accordance with section 4.3.3 of MARSSIM.

5.2.11. Dose from Groundwater

Based upon the results of groundwater monitoring performed on the LACBWR site since 1987, when the reactor was permanently shut down through the current period of active decommissioning, the dose from existing residual radioactivity in groundwater is expected to be low. However, if groundwater contamination is determined to be present at the time of license termination, the dose will be calculated using the Groundwater Exposure Factors presented in Chapter 6.

5.2.12. Demonstrating Compliance with Dose Criterion

The DCGLs for backfilled basements, soil, and buried piping for each ROC are presented in Tables 5-3, 5-5, and 5-7, respectively. These values are equivalent to the level of residual radioactivity in the media (above background) that could, when considered independently for each ROC, result in a TEDE of 25 mrem per year to the AMCG. For all media, the dose from the residual radioactivity from each ROC (radionuclide *i*) can be expressed as shown in the following equation:

Equation 5-2

$$Dose_{Media} = 25 mrem/yr \sum (i) \frac{Conc_{Radionuclide\ i}}{DCGL_{Radionuclide\ i}}$$

The final compliance dose will be calculated using Equation 5-3 after FSS has been demonstrated independently through FSS in all survey units. The results of the FSS performed for each FSS unit will be reviewed to determine the maximum dose from each of the five source terms (e.g., basement, soil, buried pipe, above grade building and existing groundwater if applicable) using the Base Case DCGLs to derive the mean SOF of FSS systematic results plus the dose from any identified elevated areas. For all media except soils, areas of elevated activity are defined in this context as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. The SOF (when using the Operational DCGL) for a systematic or judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign Test and, the mean SOF (when using the Operational DCGL) for the survey unit does not exceed one. For all media except soils, if the SOF for a sample/measurement(s) exceeds one when using Base Case DCGLs, then remediation is required. For soils, the EMC as described in section 5.10.4 of this Chapter will apply. Detailed information pertaining to the calculation of the compliance dose is provided in LC-FS-TSD-002 (see LTP Chapter 6, section 6.22 for additional discussion).

Equation 5-3

$$\text{Compliance Dose} = (\text{Max BcSOF}_{\text{BASEMENT}} + \text{Max BcSOF}_{\text{SOIL}} + \text{Max BcSOF}_{\text{BURIED PIPE}} + \text{BcSOF}_{\text{AG BUILDING}} + \text{GW BcSOF}_{\text{BS OB}} + \text{GW BcSOF}_{\text{BPS OBP}} + \text{Max SOF}_{\text{EGW}}) \times 25 \text{ mrem/yr}$$

where:

Compliance Dose	=	must be less than or equal to 25 mrem/yr,
Max $BcSOF_{BASEMENT}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for backfilled Basements,
Max $BcSOF_{SOIL}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for open land survey units,
Max $BcSOF_{BURIED\ PIPE}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) from buried piping survey units,
Max $BcSOF_{AG\ BUILDING}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) from above grade standing building survey units,
GW $BcSOF_{BS\ OB}$	=	Groundwater scenario dose from the “Other Basement” (OB) which is defined as the basement not used to generate the Max $BcSOF_{BASEMENT}$ term in Equation 1

$GW\ BcSOF_{BPS\ OBP}$ = Groundwater scenario dose from the "Other Buried Pipe" (OBP) which is defined as the buried pipe survey unit not used to generate the Max $BcSOF_{BURIED\ PIPE}$ term in Equation 1

Max SOF_{EGW} = Maximum SOF from existing groundwater (EGW)

5.2.13. Soil Area Factors

Section 2.5.1.1 and section 5.5.2.4 of MARSSIM address the concern of small areas of elevated radioactivity in a survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_w$ modified by an AF to account for the small area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_w$. Note that the consideration of small areas of elevated radioactivity applies only to Class 1 survey units, as Class 2 and Class 3 survey units by definition should not have contamination in excess of the $DCGL_w$.

The $DCGL_{EMC}$ is also referred to as the required MDC for scanning, as shown in Equation 5-3 of MARSSIM. The following equation defines the calculation of a $DCGL_{EMC}$.

Equation 5-4

$$DCGL_{EMC} = AF \times DCGL_w$$

AFs are calculated using RESidual RADioactive Materials (RESRAD) for each ROC and for source area sizes ranging from 1 m^2 up to the full source area of 100 m^2. The AFs for soils were calculated in TSD RS-TD-313196-004 and are provided in Table 5-12.

Table 5-12 Area Factors for Soils

Radionuclide	Area Factor				
	1 m^2	2 m^2	5 m^2	10 m^2	100 m^2
Co-60	9.44	5.56	3.07	2.04	1.19
Sr-90	11.22	6.66	3.69	2.45	1.41
Cs-137	9.11	5.42	3.01	2.00	1.18

5.3. <u>Summary of Characterization Survey Results</u>

Chapter 2 provides a description of the radiological status of the site including summary tables and figures that describe the characterization results. The following sections provide assessments of the characterization data to demonstrate the acceptability of the data for use in decommissioning planning, initial area classification, remediation planning, and FSS planning.

5.3.1. Field Instrumentation and Sensitivities

The field instrumentation for characterization was selected to provide both reliable operation and adequate sensitivity to detect the ROC identified for LACBWR at levels sufficiently below the established action levels. For characterization of impacted soils, the interim screening DCGLs presented in NUREG-1757, Appendix H, Table H.2 and NUREG/CR-5512 Volume 3, *Residual Radioactive*

Contamination from Decommissioning Parameter Analysis, (11) Table 6.91 ($P_{crit} = 0.10$) were used as the action levels to assess the correct classification of impacted open land or soil survey units. For impacted structures, the nuclide-specific screening value of 7,100 dpm/100cm^2 total gross beta-gamma surface activity based on Co-60 from NUREG-1757, Appendix H, Table H.1 was used as the action level to evaluate the classification of a structural survey unit. In all cases, the field instruments and detectors selected for static measurements and scanning were capable of detecting the initial suite of potential ROC at an MDC of 50% of the applicable action level.

Scanning was performed in order to locate areas of residual activity above the established action levels. Beta scans using hand-held beta scintillation and/or gas-flow proportional detectors (typically 126 cm^2) were performed over accessible structural surfaces including, but not limited to; floors, walls, ceilings, roofs, asphalt and concrete paved areas to identify locations for media sampling. Floor monitors using large area gas-flow proportional detectors (typically with 584 cm^2) were used to scan the basement floor in the WGTV.

Gamma scans were performed over open land surfaces to identify locations of residual surface activity. Sodium iodide (NaI) gamma scintillation detectors (typically 2" x 2") were typically used for these scans. Energy*Solutions* TSD RS-TD-313196-006, *Ludlum Model 44-10 Detector Sensitivity* (12) examines the response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils.

5.3.2. Laboratory Instrument Methods and Sensitivities

Gamma spectroscopy was primarily performed by the on-site radiological laboratory. Gas proportional counting and liquid scintillation analysis was performed by an approved vendor laboratory in accordance with approved laboratory procedures. The quality programs of the contracted off-site vendor laboratories that were used for the receipt, preparation and analysis of characterization samples provided the same level of quality as the on-site laboratory under Energy*Solutions* GP-EO-313196-QA-PL-001, *Quality Assurance Project Plan LACBWR Site Characterization Project* (Characterization QAPP) (13). In all cases, analytical methods were established to ensure that required MDC values are achieved. The analysis of radiological contaminants used standard approved and generally accepted methodologies or other comparable methodologies.

5.3.3. Summary of Survey Results

A detailed discussion of the results of site characterization at LACBWR is presented in Chapter 2.

5.3.3.1. Impacted and Non-Impacted Areas

The approximate area of the licensed LACBWR site is 163.5 acres. The licensed area includes land areas to the north of the LACBWR facility, including the site switchyard and the site of the former G-1 facility (removed in 1989); land to the south, which includes the existing operational G-3 facility, the coal pile area, the closed coal ash landfill surrounding the ISFSI; and a parcel of land to the east of Highway 35, across from the site. Structures and open land classified in accordance with MARSSIM as "impacted Class 1" are delineated by a surrounding single-security fence line that has been designated as the "LACBWR Site Enclosure" (LSE). The approximate area is 1.5 acres and has been segregated into five Class 1 survey units. An approximately 3.46 acre area that surrounds the LSE has also been identified as "impacted" by reactor operations. This open land area has been segregated into three Class 2 survey units and one Class 1 survey unit. Three impacted Class 3 survey units were designated for the area north of the LSE, the transmission switchyard and the area encompassing the site access off Highway 35 and the haul road used to transport dry fuel casks to the ISFSI. The area of the three Class 3 survey units combined is approximately 16.5 acres.

Characterization of the impacted and non-impacted open land survey units, as designated in RS-TD-313196-003, *La Crosse Boiling Water Reactor Historical Site Assessment* (HSA) (10), as well as the building basements that were originally planned to remain and be subjected to FSS before backfilling was performed from October 2014 to August 2015. During this 11-month period, approximately 11,072 m^2 of surface soil was scanned, 85 surface soil samples were acquired and analyzed, 126 subsurface samples were acquired and analyzed, 31 samples of asphalt were acquired and analyzed, and 15 concrete core samples were acquired from subsurface basement structures. Additionally, between September 2017 and February 2018, continuing characterization was performed in previously inaccessible areas. An additional 18 concrete core samples were obtained in the Reactor Building and WGTV basements, and approximately 25 soil samples were obtained from locations beneath and adjacent to structure basements.

5.3.3.2. Justification for Non-Impacted Areas

MARSSIM defines non-impacted areas as those areas where there is no reasonable possibility of residual contamination. A review of the operating history of the facility, historical incidents, interviews with station personnel and operational radiological surveys was conducted as documented in the HSA.

Based upon the information compiled in the HSA, a large portion of the open land areas on the 163.5 acre licensed site surrounding the LSE and ISFSI received a classification as "non-impacted." The determination that the contiguous open land areas surrounding the LSE and ISFSI were not impacted by licensed operations was based on the location(s) of licensed operations (i.e., within the LSE), site use, topography, site discharge pathways, and other site physical characteristics. The non-impacted classification was supported by the Cs-137 results from characterization in the area which were all within the range of natural background. See LTP Chapter 2 for a detailed discussion of characterization results.

The non-impacted open land area is approximately 88 acres in size. This area was segregated into five survey units. The G-3 station and the surrounding land have also been designated as not impacted by licensed activities or materials. In June 2016, LACBWR submitted a request to the NRC for approval of a partial site release of the non-impacted open land areas in accordance with 10 CFR 50.83. In April 2017, the NRC approved the release of the 88 acres from the LACBWR 10 CFR Part 50 license.

5.3.3.3. Adequacy of the Characterization

The site characterization of LACBWR included the information that should be collected per the guidance in NUREG-1700 and is discussed in detail in Chapter 2. Extensive characterization and monitoring have been performed. Measurements and samples taken in each area, along with the historical information, provide a clear picture of the residual radioactive materials and its vertical and lateral extent at the site. Using appropriate Data Quality Objectives (DQO), monitoring well water samples, surface soil, sediment, and sub-surface soil have been collected to provide the profile of residual radioactivity at the site. Samples have been analyzed for the applicable radionuclides with detection limits that provide the level of detail necessary for decommissioning planning. Based upon the volume of characterization data collected and an assessment of the characterization results, the characterization survey is considered adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected.

The soil (i.e., open land) survey units and survey unit classifications that will be used for the FSS of open land at LACBWR are presented in LTP Chapter 2, section 2.1.6 and Table 2-1. Basements that will be subjected to FSS and backfilled have also received characterization sufficient to understand the nature and extent of contamination. The initial survey units and survey unit classifications for structures, both above and below 636 foot elevation that were developed for characterization and decommissioning planning purposes are presented in LTP Chapter 2, section 2.1.6 and Table 2-2. However, the FSS that will be applied to structures below 636 foot elevation uses a different design criterion that is not directly driven by the preliminary classifications selected for characterization. Therefore, the preliminary survey unit boundaries and classifications will not apply to the FSS of structures (basements) below 636 foot elevation. See section 5.5 for the FSS design criteria for basement structure survey unit boundaries and the approach to determining survey area coverage.

5.3.3.4. <u>Inaccessible or Not Readily Accessible Areas</u>

Section 2.4 describes areas where characterization surveys were deferred including soils under structures (to be surveyed as access is achieved), soils under concrete or asphalt coverings (to be surveyed when covering is removed), currently inaccessible concrete basement surfaces (WGTV interior surfaces and the underlying concrete beneath the Reactor Building liner) and the interiors of buried pipe that will remain. A majority of the areas where the initial characterization was deferred have since been surveyed and is discussed in detail below. As access is gained to areas that were previously inaccessible, additional characterization data will be collected as necessary, evaluated and stored with-other radiological survey data in a survey history file for the survey unit. In addition, as the decommissioning progresses, data from operational events caused by equipment failures or personnel errors which may affect the radiological status of a survey unit(s) will be captured. These events will be evaluated and, when appropriate, stored in the appropriate characterization survey package. This additional characterization data will be used in validating the initial classification and in planning for the FSS.

Areas where characterization surveys were deferred will be surveyed in accordance with LC-FS-PN-002, *Characterization Survey Plan* (14) as they become accessible or in some instances the areas will be incorporated into the FSS plan (such as soil beneath or adjacent to the WTB). In other areas (e.g., soil beneath slab-on-grade structures after the slab is removed), continuing characterization may be performed in accordance with LC-FS-PR-003, *Radiological Assessments and Remedial Action Support Surveys* (15). The scope, methods and adequacy of the surveys are summarized as follows:

<u>WGTV interior structural surfaces:</u>

Continuing characterization of the Waste Gas Tank Vault was performed in September of 2017. The scope of the survey was the interior concrete surfaces as well as the soil adjacent to and beneath the structure.

The continuing characterization of the structure interior concrete surfaces consisted of a beta scan of 100% of all accessible surfaces augmented with a minimum of 30 loose surface contamination samples. Five (5) concrete core samples were obtained on the floor and wall surfaces and an additional three (3) concrete cores were biased towards areas of elevated activity identified during the scan survey (including two cores in the sump). The cores were 3" diameter to a depth of 6" and all cores were sliced into ½" pucks to ascertain the depth of contamination. All core samples underwent gamma spectroscopy using the on-site laboratory and all were sent to the off-site laboratory for HTD analysis of the full suite of radionuclides. An assessment of the results confirmed the calculated IC dose is unchanged (less than 10% of the dose limit) prior to FSS.

The soil adjacent to and beneath the WGTV was also characterized as part of the sample plan. A gamma scan was performed over 100% of the safely accessible topside soil adjacent to the structure and four (4) surface soil samples were obtained. Additionally, four (4) soil samples were obtained beneath the WGTV floor at the locations of highest activity identified during the scan survey, at low points (e.g., sump) or areas that could act as conduits for contamination migration such as cracks. This was accomplished by coring through the concrete until soil was encountered. Like the concrete core samples, all eight of the soil samples underwent gamma spectroscopy using the on-site laboratory and all were sent to the off-site laboratory for HTD analysis of the full suite of radionuclides. An assessment of the results confirmed the calculated IC dose is unchanged prior to FSS and there is no change to the surrogate ratio.

<u>Underlying concrete in the Reactor Building basement:</u>

Continuing characterization of the underlying concrete in the Reactor Building basement will be performed once all interior demolition is complete. The characterization survey will consist of a Radiological Assessment (considered a form of continuing characterization) of the liner and underlying concrete to ensure that any individual ISOCS measurements will not exceed the Operational $DCGL_B$ during FSS. The RA will consist of a beta-gamma scan over 100% of all accessible surfaces of the liner and a minimum of 30 loose surface contamination samples will be obtained. Six (6) concrete core samples will be obtained at evenly distributed locations and an additional four (4) cores will be obtained at any areas of elevated activity identified during the scan survey. If no areas of elevated activity are identified during the scan, then the four core samples will be obtained at biased locations such as low points, cracks, or areas of discoloration. The core samples will be obtained by drilling through the liner and coring into the concrete to a depth of 6". All core samples will first be analyzed by the on-site gamma spectroscopy system. Because an RA is a form of continuing characterization, 10% of all media samples collected in this survey unit, with a minimum of one sample, will be analyzed for HTD radionuclides. In addition, a minimum of one sample beyond the 10% minimum will be selected at random, also for HTD radionuclide analysis. Additionally, if levels of residual radioactivity in an individual sample exceed the Sum-of-Fractions (SOF) of 0.1 then the sample(s) will be analyzed for HTD radionuclides. All samples selected for HTD analysis during the RA will be analyzed for the full initial suite of radionuclides from Table 5-1 in the LTP.

<u>Soil under the Turbine Building (suspect broken drain line):</u>

On June 25, 2015, as part of a broader site characterization, five (5) locations were selected for angled coring to obtain soil from beneath the Turbine Building at the location of the broken drain lines. GeoProbe® technology was used to obtain the samples. At each of the 5 locations, samples were collected from the 10', 15' and 20' depths, for a total of fifteen (15) soil samples. The results are provided in LC-RS-PN-164017-001, *2015 Characterization Survey Report* (16).

In February of 2018, the Turbine Building foundation was removed in its entirety, including all broken drain lines and adjacent soil. In the eastern portion of the excavation, a total of eight (8) soil samples were collected from the region beneath the broken drain lines, turbine sump, turbine pit, and condenser pit. Although four (4) of the samples were required to be sent off-site for Sr-90 analysis, as a conservative measure, all were sent off-site for Sr-90 analysis. Additionally, 7 of the 8 samples were sent off-site for HTD analyses for the full initial suite of ROC. An assessment of the results confirmed the calculated IC dose is unchanged prior to FSS and there is no change to the surrogate ratio. An additional seven (7) soil samples will be obtained from the western region beneath the broken drain lines when that area becomes accessible.

<u>Soils adjacent to and beneath basement structures</u>:

The soil adjacent to and beneath the WGTV was characterized in September of 2017.

Continuing characterization of the soil beneath and adjacent to the Reactor Building is estimated to start in June of 2018 and will consist of soil borings (approximately six) at the nearest locations along the foundation walls that can be feasibly accessed and the acquisition of angled soil borings (four) to assess migration potential from building interiors to soils under basement concrete. The number of angled soil borings is limited to four due to the presence of deep concrete pilings. Angled soil bores will be performed via GeoProbe®.

For the Waste Treatment Building (WTB), Stack (Chimney Slab), Piping and Ventilation Tunnels and Reactor/Generator Plant (Turbine Building), which have been removed in their entirety, after concrete removal, the resultant excavations will undergo FSS in accordance with LC-FS-PR-002, *Final Status Survey Package Development* (17). The excavations traverse two Class 1 open land survey units and FSS was performed in these areas as follows:

After total removal of the WTB, continuing characterization samples were collected during the FSS of the resultant excavation. The sample plan specified a gamma scan over 100% of the survey unit including sloped walls. In addition to the systematic samples collected during FSS, two (2) additional samples were collected for continuing characterization. These 2 samples were sent off-site for HTD analysis of the full suite.

After total removal of the Stack Slab, Piping and Ventilation Tunnels (and a small portion of the Reactor/Generator Plant), continuing characterization samples were collected during the FSS of the resultant excavation. The sample plan specified a gamma scan over 100% of the survey unit including sloped walls. In addition to the systematic samples collected during FSS (minimum of 14), five (5) additional samples were collected for continuing characterization. These 5 samples were sent off-site for HTD analysis of the full suite.

After total removal of the Turbine Building (including the suspect broken drain lines) and the remaining portion of the Reactor/Generator Plant), continuing characterization samples were collected during the FSS of the resultant excavation. As previously discussed, the sample plan specified that four soil samples be taken for continuing characterization; however, eight soil samples were collected and sent off-site for HTD analysis (one for Sr-90 and seven for the full initial suite of ROC). An additional seven (7) judgmental samples will be obtained during the FSS of the western portion of the excavation.

<u>Soils under concrete or asphalt coverings</u>:

Continuing characterization of the soil beneath concrete slabs or asphalt will consist of a RA in accordance with LC-FS-PR-003 when the soil beneath is exposed. An RA was performed in February of

2018 on the sub-slab soil beneath Warehouses 1, 2 and 3 after removal of the structures. Because the structures were designated as Class 2, a minimum of 25% of the soil area underwent a gamma scan. A total of fourteen (14) surface soil samples were systematically obtained along with one (1) judgmental sample. Two (2) soil samples were sent off-site for HTD analysis for the full initial suite of ROC. An assessment of the results confirmed the calculated IC dose is unchanged prior to FSS and there is no change to the surrogate ratio.

Interiors of buried pipe:

When the interior surfaces become accessible, several potentially contaminated buried pipe systems to be abandoned in place will be characterized. The objective of the continuing characterization survey is to assess the potential radiological classification in the pipe in cases which the HSA or process knowledge has been determined to be insufficient. Continuing characterization will consist of direct measurements on pipe openings and the acquisition of sediment and/or debris samples (if available) for analysis. If necessary (as part of an investigation) the radiological survey may be expanded further into the pipe. Any sediment or debris samples will be analyzed by the on-site gamma spectroscopy system and 10%, with a minimum of one, will be sent off-site for HTD analysis for the full initial suite of ROC.

There are several previously inaccessible soils and buried pipe where historical information, process knowledge or operational survey data indicate that no significant concentrations of residual radioactivity is identified or anticipated and, that the soil or pipe is classified correctly. In these cases, survey design for FSS will be use a coefficient of variation of 30% as a reasonable value for sigma (σ) in accordance with the guidance in MARSSIM, section 5.5.2.2. All continuing characterization sample plans will be provided to the NRC for information and the results will be provided to the NRC for evaluation.

5.4. Decommissioning Support Surveys

5.4.1. Radiological Assessment (RA)

A Radiological Assessment (RA) is performed to characterize soil in areas that were previously inaccessible and have been exposed due to decommissioning and demolition activities (e.g., removal of slab-on-grade foundations, asphalt parking surfaces and excavations due to buried system removal, installation or reconfiguration).

The RA of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-18. In addition to direct and scan radiation measurements, the RA will include the collection of potentially impacted soil, sediment and/or surface residue samples for laboratory analysis.

An RA can also be performed on structures to determine when an area or survey unit has been adequately prepared for FSS. This form of RA will principally rely on direct and scan radiation measurements as well as samples for loose surface contamination.

5.4.2. Remedial Action Support (In-Process) Surveys

Remedial Action Support Surveys (RASS) are performed while remediation is being conducted, and guides cleanup in a real-time mode. RASS are conducted to: 1) guide remediation activities, and 2) provide updated estimates of the FSS design parameters (such as standard deviation) to be used for planning the FSS.

RASS of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-18. In addition to direct and scan radiation measurements, the RASS will include the collection of samples of potentially impacted soil, sediment and/or surface residue for laboratory analysis as appropriate.

RASS of structural surfaces and systems that undergo remediation will be performed using surface contamination monitors, augmented with sampling for removable surface contamination. RASS surveys may also be performed using the *in situ* Object Counting System (ISOCS), especially where personnel safety is of concern. Examples include: overhead ceilings, upper walls and cavity locations where the use of scaffolding and areal lifts is impractical.

5.4.3. Instrumentation for RA and RASS

Table 5-18 shows typical field instruments that will be used for performing FSS. The same or similar instruments will be used during the performance of the RA and RASS. The typical MDCs for field instruments used for scanning are provided in Table 5-19 and are sufficient to measure concentrations at the same action levels used during characterization as specified in section 5.3.2.

Analytical capability for soil sample analysis will supplement field scanning techniques to provide radionuclide-specific quantification, achieve lower MDCs, and provide timely analytical results. The on-site laboratory will include a gamma spectroscopy system calibrated for various sample geometries. The system will be calibrated using mixed gamma standards traceable to the National Institute of Standards and Technology (NIST) and intrinsic calibration routines. Count times will be established such that the DQOs for MDC will be achieved. Gas proportional counting and liquid scintillation analysis will be performed by an approved vendor laboratory in accordance with approved laboratory procedures. The quality programs of any contracted off-site vendor laboratory that is used for the receipt, preparation and analysis of RA and RASS samples will be confirmed to ensure the same level of quality as the on-site laboratory under LC QA-LTP-PL-001, *Quality Assurance Project Plan LACBWR License Termination Plan (LTP) Development, Site Characterization and Final Radiation Survey Projects* (QAPP) (18)

5.4.4. Field Screening Methods for RASS During the Excavation of Soils

A gamma walk-over survey will be performed over the exposed excavated surface, typically using a 2 inch by 2 inch NaI gamma scintillation detector. Appropriate scanning speed and scanning distance will be implemented to ensure that an MDC of 50% of the Operational DCGL for soil ($OpDCGL_S$) is achieved. Locations of elevated count rate will be identified for additional scanning and/or the collection of biased soil samples to determine if the elevated count rate indicates the presence of soil concentration in excess of the $OpDCGL_S$. The information obtained during the RA and RASS (scan results and the analytical data from any associated soil samples) will be used to determine if the remaining exposed soils:

- contain radioactivity concentrations above the applicable $OpDCGL_S$ and require further excavation;
- contain radioactivity concentrations that are less than the $OpDCGL_S$, but require removal in order to access additional soil/debris that potentially contains radioactivity concentrations above the applicable DCGL; or,

- contain radioactivity concentrations that are less than the $OpDCGL_S$, and not requiring removal.

If pilings are identified in areas of contaminated soil and, the pilings are also found to be contaminated, the contamination will be evaluated volumetrically considering the entire mass of the concrete piling. The resulting volumetrically contaminated volume will be assessed against the soil DCGL in the same manner as the surrounding soil.

TSD RS-TD-313196-006 examines the response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. If the survey instrument scan MDC is less than the $OpDCGL_S$, then scanning will be the primary method for guiding the remediation. Once the scan surveys and the laboratory data obtained from any biased soil samples that may have been collected indicate residual concentrations are less than the $OpDCGL_S$, the area will be considered suitable for FSS.

If the scan MDC is greater than the $OpDCGL_S$, the gamma walk-over survey will still be used to initially guide remediation however, as the levels are reduced to the range of the $OpDCGL_S$ an additional number of biased soil samples will be taken to ensure that the area can be released as suitable for FSS.

5.4.5. Contamination Verification Surveys (CVS) of Basement Structural Surfaces

All remaining structural surfaces will be surveyed to meet the criteria for open air demolition specified in Energy*Solutions* TSD RS-TD-313196-005, *La Crosse Open Air Demolition Limits* (19). These criteria are the acceptable removable contamination and contact exposure rate levels that are allowable for open air demolition. A contamination verification survey (CVS) will be performed to identify areas requiring remediation to meet the open-air demolition limits. A CVS will be performed within any structure that contains, or previously contained, radiological controlled areas. The CVS will be performed using hand-held beta-gamma instrumentation as presented in Table 5-18 in typical scanning and measurement modes

The CVS will include extensive scan surveys on the structural surfaces (walls and floors and miscellaneous equipment) that will be subject to open air demolition, regardless of elevation. The scan coverage is dependent on the contamination potential of the structural surface being surveyed. Class 1 survey units will require 100% scan coverage of all accessible surface area. Any areas identified in excess of the open-air demolition limits will be earmarked for remediation.

For structural surfaces below the 636 foot elevation that will remain and be subject to an FSS (primarily any basement floor and outer walls), additional remediation will be performed to ensure that any individual ISOCS measurement will not exceed the Operational $DCGL_B$ from Table 5-4 during FSS. Any areas identified that have the potential to exceed the Operational $DCGL_B$ by ISOCS measurement during the performance of CVS in these areas will be remediated. Any areas of elevated activity that could potentially approach the Operational $DCGL_B$ will be identified as a location for a judgmental ISOCS measurement during FSS.

5.4.6. Post-Demolition Survey

Following demolition, after all debris is removed and the floors cleaned, an additional scan survey will be performed to ensure that any individual ISOCS measurement will not exceed the $OpDCGL_B$ from

Table 5-4 during FSS. The survey will be performed using hand-held beta-gamma instrumentation as presented in Table 5-18 in typical scanning and measurement modes.

5.5. Final Status Survey of Basement Structures

As indicated in Chapter 3 of this LTP, the Reactor Building and the WGTV will be demolished and removed to a depth of at least 3 feet below grade and are the only basements that will remain and be backfilled at the time of license termination. As described in section 5.4.5, all remaining floor and wall concrete surfaces will be remediated to levels below the $OpDCGL_B$. When a basement structure has been successfully remediated, an FSS will be conducted to demonstrate that the residual radioactivity in building basements is below the dose corresponding to the Operational DCGL which is well below the 25 mrem/yr criterion.

5.5.1. Instruments Selected for Performing FSS of Basement Structures

The Canberra ISOCS has been selected as the primary instrument that will be used to perform FSS of basement structures. LC-FS-TSD-001, *Use of ISOCS for FSS of End State Sub Structures at LACBWR* (20) has been developed to describe the method and source term geometry assumption that will be used to determine the ISOCS efficiency calibration. Direct beta measurements taken on the concrete surface will not provide the data necessary to determine the residual radioactivity at depth in concrete and therefore, would have to be augmented with core sampling. The ISOCS was selected as the instrument of choice to perform the FSS of basement structures for the following reasons:

- The surface area covered by a single ISOCS measurement is large (a nominal range of 10-30 m^2) which essentially eliminates the need for scan surveys.
- Access for ISOCS measurements can be more readily accomplished remotely and does not require extensive and prolonged contact with structural surfaces that would be necessary to perform scan surveys using beta instrumentation.
- ISOCS measurements will provide results that can be used directly to determine total activity with depth in concrete.
- One of the most significant advantages of the ISOCS system in the FSS application is the analytical capability to perform comprehensive uncertainty analysis of various potential source term geometries (depth and areal distribution). After an ISOCS measurement is collected, it can be tested against a variety of geometry assumptions to address uncertainty in the source term geometry if necessary. This uncertainty analysis could potentially be used to generate a clearly conservative result using an efficiency based on a clearly conservative geometry to resolve questions without additional core samples measurements.

The analysis of concrete core samples taken during the initial characterization indicate that there is minimal variability in the geometry of residual radioactivity detected at depth. Additional concrete core samples were taken during continuing characterization in the WGTV and Reactor Building and have been evaluated to ensure that the ISOCS geometry used for efficiency calculations is sufficiently conservative. In addition, as discussed in section 5.1, concrete core samples were taken at 10% of measurement locations on the floor and lower walls of the WGTV basement for the analysis of Sr-90 (the only HTD ROC).

5.5.2. Basement Structure FSS Units

The FSS of basement structures will be performed in accordance with approved procedures and in compliance with FSS quality requirements in the QAPP.

The survey units designated for structures below 636 foot elevation from the HSA that were presented in LTP Chapter 2, Table 2-2 were based on screening values and source term assumptions that are significantly different from the BFM and are therefore not applicable.

The basement structure FSS units will be comprised of the combined wall and floor surfaces of each remaining building basement, i.e., the Reactor Building and the WGTV. Contamination potential is the prime consideration for grouping these FSS units. Contiguous surface areas with the same contamination potential will minimize uncertainty in the estimate of the mean concentration and ensure the appropriate level of areal coverage. Based on the results of concrete core sample analysis, the basements of the Reactor Building (612 and 621 foot elevations), and WGTV (618 and 621 foot elevations) were identified as being unique FSS units. Characterization data, radiological surveys performed to support commodity removal and surveys performed to support structural remediation for open-air demolition will continue to be used to verify that the contamination potential within each FSS unit is reasonably uniform throughout all walls and floor surfaces.

5.5.2.1. Classification and Areal Coverage for FSS of Basement Structures

The primary consideration for determining FSS classification and areal coverage in basement structures is the potential for and individual ISOCS measurement in a FSS unit to exceed the $OpDCGL_B$. The continuing characterization plans will be provided to the NRC for information and reports provided for evaluation. The Reactor Building and WGTV basement FSS units are designated as Class 1 and the FSS areal coverage will be 100% which is consistent with MARSSIM, Table 5.9.

5.5.2.2. Sample Size Determination for FSS of Basement Structures

Based on the contamination potential of each FSS unit that was determined in the previous section, along with the corresponding areal coverage, the number of ISOCS measurements required in each FSS unit can be calculated as the quotient of the ISOCS FOV divided into the surface area required for areal coverage. Table 5-13 presents the basement FSS units, the classification based on contamination potential, the surface area to be surveyed and the minimum number of ISOCS measurements that will be required based on a measurement FOV of 28 m^2.

Table 5-13 Number of ISOCS Measurements per FSS Unit based on Areal Coverage

FSS Unit	Classification	Area (m^2)	Minimum Areal Coverage (% of Area)	# of ISOCS Measurements (based on 28 m^2 FOV)
Reactor Building basement	Class 1	512	100%	19
WGTV basement	Class 1	311	100%	11

As shown above, a sufficient number of ISOCS measurements will be obtained in the two Class 1 sub-grade basement survey units to meet the 100% scan requirement. Additionally, the ISOCS FOV will be overlapped to ensure that there are no un-surveyed corners and gaps. The adjusted minimum number of

ISOCS measurements in each of the basement FSS units to account for the overlap is provided in Table 5-14.

Table 5-14 Adjusted Minimum Number of ISOCS Measurements per FSS Unit

FSS Unit	Classification	Required Areal Coverage (m^2)	Adjusted # of ISOCS Measurements (based on FOV-~16 m^2)	Adjusted Areal Coverage (m^2)	Adjusted Areal Coverage (% of Area)
Reactor Building basement	Class 1	512	43	512	100%
WGTV basement	Class 1	311	22[1]	311	100%

(1) Includes 18 floor/wall measurements plus 4 measurements to account for interior concrete column.

5.5.3. Survey Approach for FSS of Basement Structures

The FSS of basement structure at LACBWR will be planned, designed, implemented and assessed using the same process used for FSS as specified in MARSSIM and section 5.6. A survey package will be generated for each FSS unit. The same area preparation, area turnover and control measures specified in section 5.6.3 will also apply to FSS units. The QA requirements specified in section 5.9 will also apply to the acquisition of FSS measurements.

As previously stated, the ISOCS was selected as the instrument of choice to perform FSS in basement structures. In summary, the ISOCS detector will be oriented perpendicular to the surface of interest. In most cases, the exposed face of the detector will be positioned at a distance of 3 meters above the surface. A plumb or stand-off guide attached to the detector will be used to establish a consistent source to detector distance and center the detector over the area of interest. With the 90-degree collimation shield installed, this orientation corresponds to a nominal FOV of 28 m^2.

For survey units where physical constraints prevent a FOV of 28 m^2, the detector to source distance can be reduced, thereby reducing the FOV, which will increase the number of measurements to ensure that the required FSS coverage as presented in Table 5-14 is achieved.

If during the course of performing a FSS, measurement results are encountered that are not as expected for the surface undergoing survey, an investigation will be performed to determine the cause of the discrepancy. Investigations will also be performed if the SOF for an individual measurement exceeds one.

5.5.4. Basement Structure FSS Data Assessment

After a sufficient number of ISOCS measurements are taken in an FSS unit to achieve 100% coverage as specified in Table 5-14, the data will be summarized, including any judgmental or investigation measurements. The measured activity for each gamma-emitting ROC (and any other gamma emitting radionuclide identified at levels greater than the ISOCs MDC) will be recorded (in units of pCi/m^2). Background will not be subtracted from any measurement. Using the radionuclide mixture fractions applicable to the survey unit, an inferred activity will be derived for Sr-90 using the surrogate approach specified in section 5.2.9. The scaling factor that will be used is presented in Table 5-2. A sum of fractions (SOF) calculation will be performed for each measurement by dividing the reported concentration by the $DCGL_B$. The individual ROC fractions will then be summed to provide a total SOF value for the measurement.

As described in section 5.10.3, the Sign Test will be used to evaluate the remaining residual radioactivity in each Class 1 survey unit against the dose criterion. The SOF for each measurement will be used as the sum value for the Sign Test. If the Sign Test demonstrates that the mean activity for each ROC is less than the Operational $DCGL_B$ at a Type 1 decision error of 0.05, then the mean of all the total SOFs for each measurement in a given survey unit is calculated. If the Sign Test fails, or if the mean of the total SOFs in a basement exceeds one (using Operational DCGLs), then the survey unit will fail FSS. If a survey unit fails FSS, then an investigation will be implemented in accordance with section 5.6.4.6.

For building surfaces, areas of elevated activity are defined as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. Any area that exceeds the Base Case DCGL will be remediated. The SOF (based on the Operational DCGL) for a systematic or a judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign Test and, the mean SOF (based on the Operational DCGL) for the survey unit does not exceed one. Once the survey data set passes the Sign Test (using Operational DCGLs), the mean radionuclide activity (pCi/m^2) for each ROC from systematic measurements along with any identified elevated areas from systematic and judgmental measurements will be used with the Base Case DCGLs to perform a SOF calculation for each surface FSS unit in a basement in accordance with the following equation. The dose from residual radioactivity assigned to the FSS unit is the SOF_B multiplied by 25 mrem/yr.

Equation 5-5

$$SOF_B = \sum_{i=1}^{n} \frac{Mean\ Conc_{B_{ROC_i}}}{Base\ Case\ DCGL_{B_{ROC_i}}} + \frac{\left(Elev\ Conc_{B_{ROC_i}} - Mean\ Conc_{B_{ROC_i}}\right)}{\left[Base\ Case\ DCGL_{B_{ROC_i}} \times \left(\frac{SA_{SU}}{SA_{Elev}}\right)\right]}$$

where:

SOF_B	=	SOF for structural surface survey unit within a Basement using Base Case DCGLs
$Mean\ Conc_{B\ ROCi}$	=	Mean concentration for the systematic measurements taken during the FSS of structural surface in survey unit for each ROC_i
$Base\ Case\ DCGL_{B\ ROCi}$	=	Base Case DCGL for structural surfaces ($DCGL_B$) for each ROC_i
$Elev\ Conc_{B\ ROCi}$	=	Concentration for ROC_i in any identified elevated area (systematic or judgmental)
SA_{Elev}	=	surface area of the elevated area
SA_{SU}	=	adjusted surface area of FSS unit for DCGL calculation

5.6. Final Status Survey (FSS) Design

FSS design includes the development of sample plans that when implemented, demonstrate compliance with the dose-based unrestricted release criteria at LACBWR. This process will pertain to open land survey units, above grade structures and buried pipe at LACBWR.

5.6.1. Survey Planning

FSS provides data to demonstrate that all radiological parameters in a specific survey unit satisfy the conditions for unrestricted release. The primary objectives of the FSS are to:

- verify survey unit classification;
- demonstrate that the potential dose from residual radioactivity is below the release criterion for each survey unit; and,
- demonstrate that the potential dose from small areas of elevated radioactivity is below the release criterion for each survey unit.

The FSS process consists of four principal elements:

- Planning;
- Design;
- Implementation; and,
- Data Assessment

The DQO and Data Quality Assessment (DQA) processes are applied to these four principal elements. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions (as is the case in FSS). The DQA process is an evaluation method used during the assessment phase of the FSS to ensure the validity of survey results and demonstrate achievement of the sampling plan objectives (e.g., to demonstrate compliance with the release criteria in a survey unit).

Survey planning includes review of the HSA, the results of the site characterization, and other pertinent radiological survey information to establish the ROC and survey unit classifications. Survey units are fundamental elements for which FSS are designed and executed. The classification of a survey unit determines how large it can be in terms of surface area.

Before the FSS process can proceed to the implementation phase, turnover and control measures will be implemented for an area or survey unit as appropriate. A formal turnover process will ensure that decommissioning activities have been completed and that the area or survey unit is in a suitable physical condition for FSS implementation. Isolation and control measures are primarily used to limit the potential for cross-contamination from other decommissioning activities and to maintain the final configuration of the area or survey unit.

Survey implementation is the process of carrying out the survey plan for a given survey unit. This consists of scan measurements, total surface contamination measurements, and collection and analysis of samples. Quality assurance and control measures are employed throughout the FSS process to ensure that subsequent decisions are made on the basis that data is of acceptable quality. Quality assurance and control measures are applied to ensure:

- DQOs are properly defined and derived;
- the plan is correctly implemented as prescribed;
- data and samples are collected by individuals with the proper training using approved procedures;
- instruments are properly calibrated and source checked;
- collected data are validated, recorded, and stored in accordance with approved procedures;
- documents are properly maintained; and,
- corrective actions are prescribed, implemented and followed up, if necessary.

The initial survey units for open land areas and above grade structures and the survey unit classifications that will be used for the FSS are presented in LTP Chapter 2, section 2.1.6 and Table 2-1 and shown on Figure 2-1. The initial survey units for basement structures and the survey unit classifications that will be used for the FSS are presented in section 5.5.2 of this Chapter. A FSS Package will be prepared for each applicable survey unit. The survey package is a collection of documentation detailing FSS Sample Plan survey design, survey implementation and data evaluation. A FSS Package will contain one or more FSS Sample Plans. FSS Packages shall be controlled in accordance with the record quality requirements of LACBWR QAPP.

5.6.2. Data Quality Objectives

The DQO process will be incorporated as an integral component of the data life cycle, and is used in the planning phase for scoping, characterization, remediation and FSS plan development using a graded approach. Survey plans that are complex or that have a higher level of risk associated with an incorrect decision (such as FSS) require significantly more effort than a survey plan used to obtain data relative to the extent and variability of a contaminant. The DQO process entails a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions. Furthermore, the DQO process is flexible in that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. The DQO process is iterative allowing the survey planning team to incorporate new knowledge and modify the output of previous steps to act as input to subsequent steps. The appropriate design for a given survey will be developed using the DQO process as outlined in Appendix D of MARSSIM. The seven steps of the DQO process are outlined in the following sections.

5.6.2.1. State the Problem

The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated and the estimated resources. The problem associated with FSS is to determine whether a given survey unit meets the radiological release criterion of 10 CFR 20.1402.

5.6.2.2. Identify the Decision

This step of the DQO process consists of developing a decision statement based on a principal study question (i.e., the stated problem) and determining alternative actions that may be taken based on the answer to the principal study question. Alternative actions identify those measures to resolve the problem. The decision statement combines the principal study question and alternative actions into an expression of choice among multiple actions. For the FSS, the principal study question is "does residual radioactive contamination that is present in the survey unit exceed the established $DCGL_w$ values?" The alternative actions can include no action, investigation, resurvey, remediation and/or reclassification.

Based on the principal study question and alternative actions listed above, the decision statement for the FSS is to determine whether or not the average radioactivity concentration for a survey unit results in a SOF less than unity.

5.6.2.3. Identify Inputs to the Decision

The information required depends on the type of media under consideration (e.g., soil, water) and whether existing data are sufficient or new data are needed to make the decision. If the decision can be based on existing data, then the source(s) will be documented and evaluated to ensure reasonable confidence that the data are acceptable. If new data are needed, then the type of measurement (e.g., scan, direct measurement and sampling) will need to be determined.

Sampling methods, sample quantity, sample matrix, type(s) of analyses and analytic and measurement process performance criteria, including detection limits, are established to ensure adequate sensitivity relative to the release criteria.

The following information will be utilized to support the decision:

- ROC;
- use of surrogate relationships to infer HTD ROC;
- minimum detectable concentrations; and,
- measurement and sampling results.

5.6.2.4. Define the Study Boundaries

This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame for collecting the data, practical constraints and the scale of decision making. In FSS, the target population is the set of samples or direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest (e.g., soil, water) is specified during the planning process. The spatial boundaries include the entire area of interest including soil depth, area dimensions, contained water bodies and natural boundaries, as needed. Temporal boundaries include those activities impacted by time-related events including weather conditions, seasons, operation of equipment under different environmental conditions, resource loading and work schedule.

5.6.2.5. Develop a Decision Rule

This step of the DQO process develops the binary statement that defines a logical process for choosing among alternative actions. The decision rule is a clear statement using the "If...then..." format and includes action level conditions and the statistical parameter of interest (e.g., mean of data). Decision statements can become complex depending on the objectives of the survey and the radiological characteristics of the affected area.

For FSS, the decision rule will be based on the question pertaining to whether or not the radioactivity concentration of residual radioactivity in a survey unit exceeds the operational $DCGL_w$ value.

- If the SOF is less than unity (1), then no additional investigation will be performed and the survey unit meets the criteria for unrestricted release.
- If the SOF is greater than or equal to unity (1), then the survey unit does not meet the criteria for unrestricted release. Additional remediation followed by FSS redesign and resurvey will be performed.

5.6.2.6. Specify Limits on Decision Errors

This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided for rejection.

The primary consideration during FSS will be demonstrating compliance with the release criterion. For FSS, the null hypothesis is expressed as "the survey unit exceeds the criteria for unrestricted release".

Decision errors occur when the data set leads the decision-maker to make false rejections or false acceptances during hypothesis testing. For the design of FSS at LACBWR, the α error (Type I error) will always be set at 0.05 (5 percent) unless prior NRC approval is granted for using a less restrictive value. The β error (Type II error) will also be initially set at 0.05 (5 percent). However, the Type II error may be adjusted with the concurrence of the Characterization/License Termination Manager, after weighing the resulting change in the number of required sample or measurement locations against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criterion.

Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors are considered acceptable. The upper bound corresponds to the release criteria. The LBGR is determined as another limit on decision error. LBGR is influenced by a parameter known as the relative shift. The relative shift is the $DCGL_W$ minus the LBGR (i.e., the width of the Gray Region) divided by the standard deviation of the data set used to design the survey. In accordance with NUREG-1757, Appendix A, the LBGR should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit. However, if no other information is available regarding the survey unit, the LBGR may be initially set equal to 0.5 times the operational $DCGL_W$. However, if the relative shift exceeds a value of 3, then the LBGR should be adjusted until the relative shift value is equal to 3. The adjustment of decision errors is discussed in more detail in section 5.6.4.1.6.

Analytical uncertainty is controlled by using appropriate instrumentation, methods, techniques, training, and Quality Control (QC). The MDC values for individual radionuclides using specific analytical methods will be established. Uncertainty in the decision to release areas for unrestricted use is controlled by the number of samples and/or measurement points in each survey unit and the uncertainty in the estimate of the mean radionuclide or gross radioactivity concentrations. The specific types of instruments that can be used for the FSS of LACBWR and their respective MDC values are presented in section 5.8 and Tables 5-18 and 5-19.

Graphing the probability that a survey unit does not meet the release criteria can be used during FSS. This graph, known as a power curve, can be performed retrospectively (i.e., after FSS) using actual measurement data. This retrospective power curve is a tool that can be used to demonstrate that the DQOs are met when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria).

5.6.2.7. Optimize the Design for Obtaining Data

The first six steps of the DQO process develop the performance goals of the survey. This final step in the DQO process leads to the development of an adequate survey design.

By using an on-site analytical laboratory, sampling and analyses processes are designed to provide near real-time data assessment during implementation of field activities and FSS. Gamma scans provide information on soil areas that have residual radioactivity greater than background and allow appropriate selection of biased sampling and measurement locations. This data will be evaluated and used to refine the scope of field activities to optimize implementation of the FSS design and ensure the DQOs are met.

5.6.3. Area Preparation: Turnover and Control Measures

Following the conclusion of remediation activities and prior to initiating FSS, isolation and control measures will be implemented. The determination of readiness for controls and the preparation for FSS will be based on the results of characterization, Radiological Assessments (RA), and/or RASS that indicate residual radioactivity is unlikely to exceed the applicable DCGLs in the respective survey unit. The control measures will be implemented to ensure the final radiological condition is not compromised by the potential for re-contamination as result of access by personnel or equipment.

These measures will consist of both physical and administrative controls. Examples of the physical controls include rope boundaries and postings indicating that access is restricted to only those persons authorized to enter by the Characterization/License Termination group. Administrative controls include approved procedures and personnel training on the limitations and requirements for access to areas under these controls. In the event that additional remediation is required in an area following the implementation of isolation and control measures, local contamination control measures will be employed to prevent any potential cross-contamination.

Prior to transitioning an area from decommissioning activities to isolation and control, a walk down may be performed to identify access requirements and to specify the required isolation and control measures. The physical condition of the area will also be assessed, with any conditions that could interfere with FSS activities identified and addressed. If any support equipment is needed for FSS activities, it will be evaluated to ensure that it does not pose the potential for introducing radioactive material into the area. Industrial safety and work practice issues, such as access to high areas or confined spaces, will also be identified during the pre-survey evaluation.

Open land areas, access roads and boundaries will be posted (as well as informational notices) with signs instructing individuals to contact Characterization/License Termination group personnel prior to conducting work activities in the area. For open land areas that do not have positive access control (i.e., areas that have passed FSS but are not surrounded by a fence), the area will be inspected periodically and any material or equipment that has been introduced into the area since the last inspection will be investigated (i.e., scanned and/or sampled).

Isolation and control measures will be implemented through approved plant procedures and will remain in force throughout FSS activities and until there is no risk of recontamination from decommissioning or the survey area has been released from the license.

5.6.4. Final Status Survey Design Process

The general approach prescribed by MARSSIM for FSS requires that at least a minimum number of measurements or samples be taken within a survey unit, so that the non-parametric statistical tests used for data assessment can be applied with adequate confidence. Decisions regarding whether a given survey unit meets the applicable release criterion are made based on the results of these tests. Scanning measurements are used to confirm the design basis for the survey by evaluating if any small areas of elevated radioactivity exist that would require reclassification, tighter grid spacing for the total surface contamination measurements, or both.

The level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Class 3 survey units receive judgmental (biased) scanning and randomly located measurements or samples. Class 2 survey units receive scanning over a portion of the survey unit based on the potential for contamination, combined with total surface contamination measurements or sampling performed on a systematic grid. Class 1 survey units receive scanning over 100 percent of the survey unit combined with total surface contamination measurements or sampling performed on a systematic grid. Depending on the sensitivity of the scanning method, the grid spacing may need to be adjusted to ensure that small areas of elevated radioactivity are detected.

5.6.4.1. <u>Sample Size Determination</u>

Section 5.5 of MARSSIM and Appendix A of NUREG-1757 both describe the process for determining the number of sampling and measurement locations (sample size) necessary to ensure an adequate set of data that are sufficient for statistical analysis such that there is reasonable assurance that the survey unit will pass the requirements for release. The number of sampling and measurement locations is dependent upon the anticipated statistical variation of the final data set such as the standard deviation, the decision errors, and a function of the gray region as well as the statistical tests to be applied.

5.6.4.1.1. Decision Errors

The probability of making decision errors is established as part of the DQO process in establishing performance goals for the data collection design and can be controlled by adopting a scientific approach through hypothesis testing. In this approach, the survey results will be used to select between the null hypothesis or the alternate condition (the alternative hypothesis) as defined and shown below.

- Null Hypothesis (H_0) – The survey unit does not meet the release criterion; and,
- Alternate Hypothesis (H_a) – The survey unit does meet the release criterion.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criterion, or false negative. This occurs when the null hypothesis is rejected when in fact it is true. The probability of making this error is designated as "α".

A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criterion, or false positive. This occurs when the null hypothesis is accepted when it is in fact not true. The probability of making this error is designated as "β".

Appendix E of NUREG-1757 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the guidance in NUREG-1757, the decision error(s) that will be used for the FSS at LACBWR are:

- the α value will always be set at 0.05 (5 percent) unless prior NRC approval is granted for using a less restrictive value; and,
- the β value will also be initially set at 0.05 (5 percent), but may be modified, as necessary, after weighing the resulting change in the number of required sampling and measurement locations against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criterion.

5.6.4.1.2. Unity Rule

The unity rule or SOF, as discussed in section 5.2.5, will be used for the survey planning and data evaluations for soil sample analyses since multiple radionuclide-specific measurements will be performed. As a result, the evaluation criteria and data must be normalized in order to accurately compare and relate the various data measurements to the release criteria.

5.6.4.1.3. Gray Region

The gray region is defined in MARSSIM as the range of values for the specified parameter of interest for the survey unit in which the consequences of making a decision error is relatively minor. This can be explained as the range of values for which there is a potential of making a decision error; however, there is reasonable assurance that the parameters will meet the specified criteria for the rejection of the null hypothesis.

The gray region is established by setting an upper and lower boundary. Values for the specified parameter above and below these boundaries usually result in a "black and white" or "go no go" decision. Values between the upper and lower boundary are within the "gray region" where decision errors apply most.

5.6.4.1.4. Upper Bound of the Gray Region (UBGR)

For the purposes of the FSS, release parameters at or near the release guidelines will typically result in a decision that the survey unit will not meet the requirements for release, with the exception of evaluating elevated areas. As a result, the upper boundary of the gray region is typically set as the Operational DCGL.

5.6.4.1.5. Lower Bound of the Gray Region (LBGR)

The LBGR is the point at which the Type II error (β), or false positive, applies. In accordance with NUREG-1757, Appendix A, the LBGR should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit. However, if no other information is available regarding the survey unit, the LBGR may be initially set equal to 0.5 times the applicable Operational DCGL and may be set as low as the MDC for the specific analytical technique. This will help in maximizing the relative shift and effectively reduce the number of required sampling and measurement locations based upon acceptable risks and decision errors.

5.6.4.1.6. Relative Shift

The relative shift (Δ/σ) for the survey unit data set is defined as shift (Δ), which is the upper boundary of the gray region, or $DCGL_w$, minus the LBGR, divided by sigma (σ), which is the standard deviation of the data set used for survey design. For survey design purposes, sigma values in a survey unit and/or reference area will be initially be calculated from characterization survey and/or investigation data to assess the readiness of a survey area for FSS. For survey unit where no significant concentrations of residual radioactivity is identified or anticipated, then survey design for FSS will be use a coefficient of variation of 30% as a reasonable value for sigma (σ) in accordance with the guidance in MARSSIM, section 5.5.2.2. Standard deviation values, as determined from the characterization data are generally not recommended for Class 1 areas as this will typically contain values in excess of the guidelines and have excessive variability which will not be representative of the conditions at the time of the FSS. The standard deviation at the time of the FSS will be approximated as best as possible to ensure the FSS requirements are not too restrictive. This can be accomplished by taking additional measurements in a survey unit prior to performing FSS to establish an acceptable standard deviation. The optimal value for the relative shift should range between (and including) 1 and 3.

5.6.4.2. Statistical Test

At LACBWR, the Sign Test will be used for the statistical evaluation of the survey data. The Sign Test will be implemented using the unity rule, surrogate methodologies, or combinations thereof as described in MARSSIM and Chapters 11 and 12 of NUREG-1505.

The Sign Test is the most appropriate test for FSS at LACBWR, as background is expected to constitute a small fraction of the $DCGL_w$ based on the results of characterization surveys. Consequently, the Sign Test will be applied when demonstrating compliance with the unrestricted release criteria without subtracting background.

The number of sampling and measurement locations (N) that will be collected from the survey unit will be determined by establishing the acceptable decision errors, calculating the relative shift, and using Table 5-5 of MARSSIM. As stated in section 5.6.4.1.6, optimal values for the relative shift are between (and including) 1 to 3. Smaller values for relative shift substantially increase the number of required sampling and measurement locations, while larger values do little to reduce the required number.

By reading the relative shift from the left side of the Table 5-5 of MARSSIM and cross referencing to the specified decision errors, the number of sampling and measurement locations can be determined. The specified number within the table includes the recommended 20 percent adjustment or increase to ensure an adequate set of data is collected for statistical purposes. MARSSIM Equation 5-2 may alternatively be used to calculate the number of sampling and measurement locations. The result will be rounded up by 20 percent. The sample size calculations will be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or spreadsheets.

5.6.4.3. Small Areas of Elevated Activity

Section 2.5.1.1 of MARSSIM addresses the concern of small areas of elevated radioactivity in the survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. This is referred to as the EMC. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_w$ modified by an AF to account for the small area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_w$. Note that at LACBWR, the consideration of small areas of elevated radioactivity will only be applied to Class 1 open land (soil) survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_w$. For basement structures, any residual radioactivity identified by a FSS measurement at concentrations in excess of the respective Base Case DCGL will be remediated.

The statistical tests that determine if the residual radioactivity exceeds the $DCGL_w$ are not adequate for providing assurance that small areas of elevated radioactivity are successfully detected, as discussed in section 5.5.2.4 of MARSSIM. Systematic sampling and measurement locations in conjunction with surface scanning are used to obtain adequate assurance that small elevated areas comply with the $DCGL_{EMC}$; however, the number of statistical systematic sampling and measurement locations must be compared to the scan sensitivity to determine the adequacy of the sampling density. The calculation of the $DCGL_{EMC}$ is detailed in section 5.10.4.

The comparison begins by determining the area bounded by the statistical systematic sampling and measurement locations. This value is calculated by dividing the area of the survey unit (A_{SU}) by N for the Sign Test.

Equation 5-6

$$A = \frac{A_{SU}}{n}$$

where:

A	=	Area bounded by samples;
A_{SU}	=	Area of the survey unit; and
n	=	number of samples (N).

The AF is selected from Table 5-12 for soils corresponding to the bounded area (A) calculated. If the calculated bounded area (A) falls between two area categories on Tables 5-12, then the larger of the two areas will be selected along with the corresponding AF. $DCGL_{EMC}$ is then derived by multiplying the selected AF by the applicable $DCGL_w$.

The required scan MDC, which is equal to the $DCGL_{EMC}$, is then compared to the actual scan MDC. If the actual scan MDC is less than or equal to the required scan MDC, then the spacing of the statistical systematic sampling and measurement locations is adequate to detect small areas of elevated radioactivity. If the actual scan MDC is greater than the required scan MDC, then the spacing between locations needs to be reduced due to the lack of scanning sensitivity.

To reduce the spacing, a new number of sampling and measurement locations must be calculated. First, a new AF that corresponds to the actual scan MDC is calculated as follows;

Equation 5-7

$$\text{Adjusted AF}=\frac{Actual\ Scan\ MDC}{DCGL_W}$$

Next, the adjusted AF is used to look up a new adjusted area (A') from Table 5-12. Finally, using the adjusted area (A'), an adjusted number of statistical systematic sampling and measurement locations (n_{EMC}) is calculated as follows:

Equation 5-8

$$n_{EMC}=\frac{A_{SU}}{A'}$$

Therefore, the number of systematic sampling and measurement locations in the survey unit will be adjusted to equal to the value derived for n_{EMC}. When multiple measured radionuclides are present, this process is repeated for each measured radionuclide or the surrogate radionuclide, if a surrogate radionuclide is used. The greatest number of systematic sampling and measurement locations determined from the radionuclides will be used for the survey design.

5.6.4.4. Scan Coverage

The purpose of scan measurements is to confirm that the area was properly classified and that any small areas of elevated radioactivity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, the number of total surface contamination measurement locations may need to be increased so the spacing between measurements is reduced.

The amount of area to be covered by scan measurements is presented in Table 5-15, which is reproduced from the portion of Table 5.9 from MARSSIM. As intended by the guidance, the emphasis will be placed on a higher frequency of scans in areas of higher risk. The scan coverage requirements that will be applied for scans performed in support of the FSS of open land and above grade structure survey units are:

- For Class 1 survey units, 100 percent of the accessible soil or structure surface will be scanned;
- For Class 2 survey units, between 10 percent and 100 percent of the accessible soil or structural surface will be scanned, depending upon the potential of contamination. The amount of scan coverage for Class 2 survey units will be proportional to the potential for finding areas of elevated radioactivity or areas close to the release criterion. Accordingly, the site will use the results of individual measurements collected during characterization to correlate this radioactivity potential to scan coverage levels; and,
- For Class 3 survey units, judgmental (biased) surface scans will typically be performed on areas with the greatest potential of contamination. For open land areas, this will include surface drainage areas and collection points. In the absence of these features the locations of these judgmental scans will at the discretion of the survey designer.

5.6.4.5. Reference Grid, Sampling and Measurement Locations

The survey sampling and measurement locations are a function of the sample size and the survey unit size. The guidance provided in section 4.8.5 and section 5.5.2.5 of MARSSIM has been incorporated in this section. For the FSS open land survey units, reference coordinates will be acquired using a Global Positioning System (GPS) coupled with the North American Datum (NAD) standard topographical grid coordinate system. For the FSS of above grade structures, reference coordinates will be determined based on the reference grid.

Table 5-15 Recommended Survey Coverage for Open Land Areas and Structures

Area Classification	Surface Scans	Soil Samples/Static Measurements
Class 1	100%	Number of sample/measurement locations for statistical test, additional sample/measurements to investigate areas of elevated activity
Class 2	10% to 100%, Systematic and Judgmental	Number of sample/measurement locations for statistical test
Class 3	Judgmental	Number of sample/measurement locations for statistical test

5.6.4.5.1. Reference Grid

A reference grid will be used for reference purposes and to locate the sampling and measurement locations. The reference grid will be physically marked during the survey to aid in the collection of samples and measurements. At a minimum, each survey unit will have a benchmark defined that will serve as an origin for documenting survey efforts and results. This benchmark (origin) will be provided on the map or plot included in the FSS package.

5.6.4.5.2. Systematic Sampling and Measurement Locations

Systematic sampling and measurement locations for Class 1 and Class 2 survey units will be located in a systematic pattern or grid. The grid spacing (L), will be determined using a triangular or square grid. Where in most cases, a triangular grid will be preferred, a square grid can be used if the physical dimensions of a survey unit are conducive to the square grid approach. The equations used to determine the grid spacing for systematic measurement locations in Class 1 and Class 2 open land survey units are as follows:

Equation 5-9

$$L = \sqrt{\frac{A}{0.866N}} \text{ (for a triangular grid or),}$$

$$L = \sqrt{\frac{A}{N}} \text{ (for a square grid)}$$

where:

L	=	grid spacing (dimension is square root of the area);
A	=	the total area of the survey unit; and,
N	=	the desired number of measurements.

Once the grid spacing is established, a random starting point will be established for the survey pattern using a random number generator. Starting from this randomly-selected location, a row of points will then be established parallel to one of the survey unit axes at intervals of L. Additional rows will then be added parallel to the first row. For a triangular grid, additional rows will be added at a spacing of $0.866L$ from the first row, with points on alternate rows spaced mid-way between the points from the previous row. For a square grid, points and rows will be spaced at intervals of L.

The grid spacing can be rounded down for ease of locating sampling and measurement locations on the reference grid. The number of sampling and measurements locations identified will be counted to ensure the appropriate number of locations has been identified. Depending upon the configuration and layout of the survey unit and the starting grid location, the minimum number of sampling and measurement locations could fall outside of the survey unit boundary. In this event, either a new random starting location will be specified or the grid spacing adjusted downward until the appropriate number of locations is reached.

Software tools that accomplish the necessary grid spacing, including random starting points and triangular or square shape, will be employed during FSS design. When available, this software will be used with suitable mapping programs to determine coordinates for a GPS. The use of these tools will provide a reliable process for determining, locating and mapping measurement locations in open land areas separated by large distances and will be helpful during independent verification.

For Class 3 survey units, each sampling and measurement location will be randomly selected using a random number generator.

The systematic sampling and measurement locations within each survey unit will be clearly identified and documented for the purposes of reproducibility. Actual measurement locations will be marked and identified by tags, labels, flags, stakes, paint marks, GPS location, photographic record, or equivalent.

5.6.4.6. Investigation Process

During the FSS, any areas of concern will be identified and investigated. This will include any areas as identified by the surveyor in real-time during the scanning of surface soils or structural surfaces, any areas identified during post-processing and reviewing of scan survey data, and any results of soil or bulk material analyses that exceed the DCGL. Based on this review, the suspect areas will be addressed by further biased surveys and sampling as necessary. The applicable investigation levels are provided in Table 5-16.

Table 5-16 FSS Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	>Operation DCGL or >MDC_{scan} if MDC_{scan} is greater than Operational DCGL	>Operational DCGL
Class 2	>Operational DCGL or >MDC_{scan} if MDC_{scan} is greater than Operational DCGL	>Operational DCGL
Class 3	>Operational DCGL or >MDC_{scan} if MDC_{scan} is greater than Operational DCGL	>0.5 Operational DCGL

5.6.4.6.1. Remediation, Reclassification and Resurvey

In Class 1 survey units, any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated to reduce the residual radioactivity to acceptable levels. In Class 1 survey units for media other than soil (structural surfaces, above-grade structures and buried pipe), any areas of elevated residual radioactivity above the Base Case DCGL will be remediated. If an area is remediated, then a RASS will be performed to ensure that the remediation was sufficient.

If an individual FSS survey measurement (ISOCS for basements, sample for soil, and instrument reading for buried pipe) in a Class 2 survey unit exceeds the Operational DCGL, the survey unit, or portion of the survey unit will be investigated. If small areas of elevated activity exceeding the Operational DCGL are confirmed by this investigation or, if the investigation suggests that there is a reasonable potential that contamination is present in excess of the Operational DCGL, then all or part of the survey unit will be reclassified as Class 1 and the survey strategy for that survey unit will be re-designed and re-performed as discussed above for Class 1 or Class2.

If an individual survey measurement in a Class 3 survey unit exceeds 50 percent of the Operational DCGL, the survey unit, or portion of a survey unit, will be investigated. If the investigation confirms residual radioactivity in excess of 50 percent of the Operational DCGL, then the survey unit, or the impacted portion of the survey unit will be reclassified to a Class 1 or a Class 2 survey unit and the survey will be re-designed and re-performed as discussed above for Class 1 or Class 2.

The DQO process will be used to evaluate the remediation, reclassification and/or resurvey actions to be taken if an investigation level is exceeded. Based upon the failure of the statistical test or the results of an investigation, Table 5-17 presents actions that will be required.

Re-classification of a survey unit from a less restrictive classification to a more restrictive classification may be done without prior NRC approval. However, reclassification to a less restrictive classification requires prior NRC approval.

Table 5-17 Remediation, Reclassification and Resurvey Actions

<table>
<tr><td colspan="4">REMEDIATION</td></tr>
<tr><td colspan="3">Remediation Criteria</td><td>Proposed Remediation</td></tr>
<tr><td rowspan="3">Class 1 FSS Survey Unit</td><td colspan="2">1) Passes Sign Test and the mean SOF for survey unit is less than or equal to unity (1) (SOF EMC for open land survey units or Equation 5-5 for structural survey units)</td><td>None</td></tr>
<tr><td colspan="2">2) Passes Sign Test and the mean SOF for survey unit is less than or equal to unity (1) with several elevated areas present that require remediation (>$DCGL_{EMC}$ for soils or Base Case DCGL for other media)</td><td>Spot Remediation & Resurvey under Existing Survey Design</td></tr>
<tr><td colspan="2">3) Does not pass Sign Test, or the mean SOF is greater than unity</td><td>General Remediation and Restart FSS under new Survey Design</td></tr>
<tr><td rowspan="2">Class 1 Basement FSS Unit</td><td colspan="2">1) The mean inventory fraction (total mean dose for the survey unit divided by the dose criterion of 25 mrem/yr) is greater than or equal to one.</td><td rowspan="2">General Remediation and Restart FSS under new Survey Design</td></tr>
<tr><td colspan="2">2) The sum of the mean inventory fractions for each FSS unit contained within a building basement is greater than or equal to one.</td></tr>
<tr><td colspan="4">RECLASSIFICATION</td></tr>
<tr><td colspan="3">Reclassification Criteria</td><td>Proposed Action</td></tr>
<tr><td rowspan="3">Class 2 Survey Unit</td><td rowspan="3">One or several survey measurements (scan, sample or direct measurement) exceed the Operational DCGL or a portion of the survey unit is remediated.</td><td>The extent of the elevated area relative to the total area of the survey unit is minimal and the source of the residual radioactivity is known</td><td>Reclassify only the bounded discrete area of elevated activity to Class 1.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is minimal and the source of the residual radioactivity is unknown</td><td>Reclassify 2,000 m^2 for soils or 100 m^2 for structures around the area of elevated activity as Class 1.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is significant.</td><td>Reclassify the entire survey unit as Class 1.</td></tr>
<tr><td rowspan="4">Class 3 Survey Unit</td><td rowspan="2">One or several survey measurements (scan, sample or direct measurement) exceed 50% of the Operational DCGL or a portion of the survey unit is remediated.</td><td>The extent of the elevated area relative to the total area of the survey unit is minimal</td><td>Reclassify the area of elevated activity to Class 1 and create a Class 2 buffer zone of appropriate size around the area.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is significant.</td><td>Reclassify the area of elevated activity to Class 1 and create a Class 2 buffer zone of appropriate size around the area.</td></tr>
<tr><td rowspan="2">One or several survey measurements (scan, sample or direct measurement) exceed 1% of the Operational DCGL</td><td>The extent of the elevated area relative to the total area of the survey unit is minimal</td><td>Reclassify the area of elevated activity to Class 2.</td></tr>
<tr><td>The extent of the elevated area relative to the total area of the survey unit is significant.</td><td>For soils, reclassify 10,000 m^2 around the area of elevated activity to Class 2. For structures, reclassify 1,000 m^2 around the area of elevated activity to Class 2.</td></tr>
</table>

Table 5-17 (continued) Remediation, Reclassification and Resurvey Actions

RESURVEY			
Resurvey Criteria			Proposed Action
Class 1 Survey Unit	The survey unit has been remediated.	Survey unit passed Sign Test and the mean SOF for survey unit was less than unity with several elevated areas present that required remediation. The power of the original survey is unchanged.	Re-scan remediated area; collect samples/measurements within the remediated area to demonstrate that remediation was successful.
		Survey unit did not pass Sign Test, or mean SOF exceeded unity	Resurvey entire survey unit using a new survey design.
	Survey unit has been reclassified from a Class 2 survey unit.	No remediation was performed.	Increase scan or areal coverage to 100%. Additional statistical samples are not required.
Class 2 Survey Unit	Survey unit has been divided to accommodate a new Class 1 survey unit.	The area of the new Class 1 survey unit relative to the area of the initial Class 2 survey unit is minimal and no statistical samples were affected.	Increase scan or areal coverage in Class 2 survey unit.
		Statistical sample population was affected by the reclassification.	Increase scan or areal coverage in Class 2 survey unit and resurvey entire survey unit using a new survey design.
Class 3 Survey Unit	Survey unit has been divided to accommodate a new Class 2 survey unit.	The area of the new Class 2 survey unit relative to the area of the initial Class 3 survey unit is minimal and the power of the original Class 3 survey is unchanged.	Increase scan or areal coverage in Class 3 survey unit.
		The area of the new Class 2 survey unit relative to the area of the initial Class 3 survey unit is significant.	Resurvey entire survey unit using a new survey design.

5.7. Final Status Survey Implementation

Trained and qualified personnel will perform survey measurements and collect samples. FSS measurements include surface scans, static measurements, gamma spectroscopy of volumetric materials, and in-situ gamma spectroscopy. The surveying and sampling techniques are specified in approved procedures.

5.7.1. Survey Methods

The survey methods to be employed for FSS will consist of combinations of gamma scans and static measurements, soil and sediment sampling and in-situ gamma spectroscopy. Additional specialized methods may be identified as necessary between the time this plan is approved and the completion of FSS activities. Any new technologies will meet the applicable DQOs of this plan, and the technical approach will be documented for subsequent regulator review.

5.7.1.1. Scanning

Scanning is performed in order to locate small, elevated areas of residual activity above the investigation level. It is the process by which a surveyor passes a portable radiation detector within close proximity of a surface with the intent of identifying residual radioactivity. Scan surveys that identify locations where the magnitude of the detector response exceeds an investigation level indicate that further investigation is warranted to determine the amount of residual radioactivity. The investigation levels will be based on the Operational DCGL, a fraction of the Operational DCGL, or the $DCGL_{EMC}$ for Class 1 soils.

One of the most important elements of a scan survey is defining the limit of detection in terms of the *a priori* scanning MDC in order to gauge the ability of the field measurement system to confirm that the unit is properly classified, and to identify any areas where residual radioactivity levels are elevated relative to the Operational DCGL. If the scanning indicates that the survey unit or a portion of the survey unit has been improperly classified, then the survey design process must be evaluated to either assess the effect of reclassification on the survey unit as a whole (if the whole unit requires reclassification) or a new design must be established for the new unit(s) (in the case of sub-division). A new survey design will require a re-evaluation of the survey strategy to decide if it can meet the requirements of the revised survey design. If not, the survey strategy must be revised based on the available instrumentation and methods.

Technicians will respond to indications of elevated areas while surveying. Upon detecting an increase in visual or audible response, the technician will reduce the scan speed or pause and attempt to isolate the elevated area. If the elevated activity is verified to exceed the established investigation level, the area will be bounded (e.g., marked and measured to obtain an estimated affected surface area).

If surface conditions prevent scanning at the specified distance, the detection sensitivity for an alternate distance will be determined and the scanning technique adjusted accordingly. Whenever possible, surveyors will monitor the visual and audible responses to identify locations of elevated activity that require further investigation and/or evaluation.

For the FSS of basement structures, the surface area covered by a single ISOCS measurement is large (a nominal range of 10-30 m^2) which eliminates the need for traditional scan surveys.

5.7.1.2. Volumetric Sampling

Volumetric sampling is the process of collecting a portion of a media as a representation of the locally remaining media. The collected portion of the medium is then analyzed to determine the radionuclide concentration. Examples of materials that will be sampled include soil, sediments, concrete and groundwater for open land areas. Bulk material samples will be analyzed via gamma spectroscopy, alpha spectroscopy and/or liquid scintillation counting as appropriate.

Trained and qualified individuals will collect and control samples. All sampling activities will be performed under approved procedures. A Chain-of-Custody (CoC) process will be utilized to ensure sample integrity.

QA requirements for FSS activities that apply to sample collection (e.g., split samples, duplicates, etc.) and onsite and offsite laboratories employed to analyze samples as a part of the FSS process will be controlled by approved procedures, in conformance with the QAPP and is further described in section 5.9. Performance of laboratories will be verified periodically in accordance with the QAPP.

5.7.1.3. Fixed Measurements

Fixed measurements are taken by placing a detector at a defined distance above a surface, taking a discrete measurement for a pre-determined time interval, and recording the reading. Fixed measurements are collected at random or systematic locations in a survey unit and supplement scanning surveys for the identification of small areas of elevated activity. Fixed measurements are also collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment can also be used to identify locations for fixed measurements to further define the areal extent of contamination.

5.7.1.4. Surface Soils

In this context, surface soil refers to soil located from the surface down to a depth of 1 meter. These areas will be surveyed through combinations of sampling and scanning as appropriate.

5.7.1.4.1. Gamma Scans of Surface Soils

Gamma scans will be performed over open land surfaces to identify locations of residual surface activity. NaI gamma scintillation detectors (typically 2” x 2”) will be used for these scans. TSD RS-TD-313196-006 presents the response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. Cs-137 will be used as the surrogate to infer Sr-90 in accordance with section 5.2.9.

When using hand-held detectors, gamma scanning is generally performed by moving the detector in a serpentine pattern, usually within 15 cm (6 in) from the surface, while advancing at a rate of approximately 0.5 m (20 in) per second. Audible and visual signals will be monitored.

Surveyors will respond to indications of elevated areas while surveying. Upon detecting an increase in visual or audible response, the surveyor will reduce the scan speed or pause and attempt to isolate the elevated area. If the elevated activity is verified to exceed the established investigation level, the area is bounded (e.g., marked or flagged and measured to obtain an estimated affected surface area).

5.7.1.4.2. Sampling of Surface Soils

Samples of surface soil (including sediment or sludge) will be obtained from designated random or systematic locations and at areas of elevated activity identified by gamma scans. An appropriate volume of soil (typically 0.5-1 liter) will be collected at each sampling location using hand trowels, bucket augers, or other suitable sampling tools. A GPS reading will be obtained at each surface soil location and a pinned flag or similar marker will be placed in the ground to mark the location.

Sample preparation includes removing extraneous material and homogenizing and drying the soil for analysis. Separate containers are used for each sample and each container is tracked through the analysis process using a chain-of-custody process.

All surface soil samples taken during continuing characterization and FSS will be analyzed by gamma spectrometry. Ten percent (10%) of the FSS samples collected from open land survey units (including excavations where major sub-grade structures previously resided) will also be analyzed for ROC HTD radionuclides (Sr-90). Additionally, if levels of residual radioactivity in an individual soil sample exceed a SOF of 0.1, then the sample(s) will be analyzed for ROC HTD radionuclides.

5.7.1.5. <u>Subsurface Soils</u>

Subsurface soil refers to soil that resides at a depth greater than 1 m below the final configuration of the ground surface or soil that will remain beneath structures such as basement floors/foundations or pavement at the time of license termination.

Any soil excavation created to expose or remove a potentially contaminated subgrade basement structure will be subjected to FSS prior to backfill. The FSS will be designed as an open land survey using the classification of the removed structure in accordance with section 5.6.4 of the LTP using the Operational DCGLs for soils as the release criteria.

During site characterization, the HSA was consulted to identify those survey areas where the potential existed for subsurface radioactivity. Such areas included, but were not limited to, areas under buildings, building floors/foundations, or outside components where leakage was known or suspected to have occurred in the past and on-site storage areas where radioactive materials have been identified. Soil data from both the HSA and any pertinent surface characterization data were used to establish locations and potential depth for any potential sub-surface radioactivity. During site characterization, a total of 126 composited subsurface samples were collected in impacted open land survey units to depths ranging from 1 m below grade to approximately 3 m below grade and analyzed for the potential ROC. To date, only Cs-137 and Co-60 have been identified at concentrations greater than the analytic MDC of the instrument used and no residual radioactivity was identified at concentrations greater than the generic screening values ($DCGL_w$) from NUREG-1757, Appendix H for each of the potential ROC.

During the decommissioning of LACBWR, any subsurface soil contamination that is identified by continuing characterization or operational radiological surveys at concentrations exceeding the site-specific Base Case DCGLs for each potential ROC presented in Table 5-2 will be remediated. The remediation process will include performing RASS of the open excavations in accordance with section 5.4.2 of this FSS Plan. The RASS will include scan surveys and the collection of soil samples during excavation to gauge the effectiveness of remediation, and to identify locations requiring additional excavation. The scan surveys and the collection of and subsequent laboratory analysis of soil samples will be performed in a manner that is intended to meet the DQOs of FSS. The data obtained during the RASS is expected to provide a high degree of confidence that the excavation, or portion of the excavation, meets the criterion for the unrestricted release of open land survey units. Soil samples will be collected to depths at which there is high confidence that deeper samples will not result in higher concentrations. Alternatively, a NaI detector or intrinsic germanium detector of sufficient sensitivity to detect residual radioactivity at the Operational DCGL can be used to scan the exposed soils in an open excavation to identify the presence or absence of soil contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the Operational DCGL, additional confirmatory investigation and analyses of soil samples of the suspect areas will be performed.

5.7.1.5.1. Scanning of Subsurface Soils during FSS

Per NUREG-1757, scanning is not applicable to subsurface soils during the performance of FSS. Scanning will be performed during the RASS of excavations resulting from any remediation of subsurface soil contamination. The scanning of exposed subsurface soils during the RASS, where accessible as an excavated surface, will be used with the analysis of soil samples to demonstrate compliance with site release criteria.

5.7.1.5.2. Sampling of Subsurface Soils during FSS

In accordance with NUREG-1757, Appendix G, if the HSA indicates that there is no likelihood of substantial subsurface residual radioactivity then subsurface surveys are not necessary. The HSA as well as the results of the extensive characterization of subsurface soils in the impacted area surrounding the LACBWR facility have shown that there is minimal residual radioactivity in subsurface soil. Consequently, minimal subsurface sampling will be performed during FSS.

In Class 1 open land survey units, a subsurface soil sample will be taken at 10% of the systematic surface soil sample locations in the survey unit with the location(s) selected at random. In addition, if during the performance of FSS, the analysis of a surface soil sample, or the results of a surface gamma scan indicates the potential presence of residual radioactivity at a concentration of 75% of the soil Operational DCGL, then additional biased subsurface soil sample(s) will be taken within the area of concern as part of the investigation.

In Class 2 and Class 3 open land survey units, no subsurface soil sample(s) will be taken as part of the survey design. However, as with the Class 1 open land survey units, if during the performance of FSS, the analysis of a surface soil sample, or the results of a surface gamma scan indicates the potential presence of residual radioactivity at a concentration of 75% of the soil Operational DCGL, then biased subsurface soil sample(s) will be taken to the appropriate depth within the area of concern as part of the investigation.

GeoProbe®, split spoon sampling or other methods will be used to acquire subsurface soil samples. Subsurface soil samples will be obtained to a depth of at least 1 meter or refusal, whichever is reached first. In cases where refusal is met because of bedrock, the sample will be used “as is”. In cases where a non-bedrock refusal is met prior to the 1 meter depth, the available sample will be used to represent the 1 meter sample. If residual radioactivity is detected in the 1 meter sample, an additional meter of depth will be sampled and analyzed.

Subsurface soil samples will be segmented and homogenized over each 1 meter of depth. Extraneous material will be removed from each segment and the sample will be adequately dried. The material will then be placed into a clean sample container and properly labeled. All samples will be tracked from time of collection through the final analysis in accordance with procedure and survey package instructions.

All subsurface soil samples will be analyzed by gamma spectrometry. Ten percent (10%) of subsurface soil samples collected for FSS will also be analyzed for ROC HTD radionuclides (Sr-90). Additionally, if levels of residual radioactivity in an individual subsurface soil sample exceed a SOF of 0.1, then the sample(s) will be analyzed for ROC HTD radionuclides.

5.7.1.5.3. Sampling of Subsurface Soils below Structure Basement Foundations

The foundation walls and basement floors below the 636 foot elevation (3 feet below grade) of the Reactor Building and WGTV and the concrete piles and piling caps supporting the Reactor and Turbine Buildings will remain at the time of license termination. Based on the results of subsurface soil sampling performed during site characterization, it is not likely that the residual radioactivity concentrations in soil beneath these building foundations exceed the site-specific Base Case soil DCGLs as presented in Table 5-5. However, prior to license termination, continuing characterization surveys will be performed as necessary to ascertain the radiological conditions of these sub-slab soils. The presence of significant concentrations of residual radioactivity in these samples may correlate to a loss of integrity of structural basements and will be further investigated. The Continuing Characterization Plans and Reports will be provided to NRC for information and results for evaluation.

There is one unique location where subsurface contamination may currently be present which is the soil under the Turbine Building floor in the vicinity of suspect broken drain lines. As stated in section 5.3.4.4 this area will be addressed after the Turbine Building floor is removed and during the subsequent FSS in the form of additional judgmental samples obtained in areas specifically under the locations of the suspect drain lines. Note that borehole samples were collected under the Turbine Building in the vicinity of the suspect broken drain lines in 2015. No plant derived radionuclides were identified at concentrations exceeding background levels, which supports the current assumption that the extent of subsurface contamination in the area is limited.

Samples of building basement sub-slab soils may be obtained by coring through concrete slabs and foundations to facilitate the collection of soil samples. Additionally, GeoProbe® technology may be employed to access the sub-slab soils from outside of the building footprint by drilling at an angle under the building. Locations selected for sampling will be biased to locations having a high potential for the accumulation and migration of radioactive contamination to sub-surface soil. The biased locations for sub-slab soil and concrete assessment could include stress cracks, floor and wall interfaces, penetrations through walls and floors for piping, run-off from exterior walls, and leaks or spills in adjacent outside areas, etc. All samples taken from sub-slab soils will be analyzed by gamma spectrometry. Ten percent (10%) of any sub slab soil samples taken will be analyzed for the initial suite of HTD radionuclides as well as any individual sample where analysis indicates gamma activity in excess of a SOF of 0.1. See section 5.1 for additional details.

5.7.1.6. <u>Excavated Soils and Clean Fill</u>

LACBWR will not stockpile and store excavated soil for reuse as backfill. However, overburden soils may be excavated to expose buried components (e.g. concrete pads, buried pipe, buried conduit, etc.) that will be removed and disposed of as waste or, to install a new buried system. In these cases, the overburden soil will be removed, the component will be removed or installed, and the overburden soil will be replaced back into the excavation. In these cases, an RA will be performed. The footprint of the excavation will be scanned prior to the excavation. In addition, periodic scans will be performed on the soil as it is excavated, and the exposed surfaces of the excavated soil will be scanned after it is piled next to the excavation for reuse. Scanning will be performed in accordance with section 5.7.1.4.1. A soil sample will be acquired at any scan location that indicates activity in excess of 50% of the soil Operational DCGL. Any soil confirmed as containing residual radioactivity at concentrations exceeding 50% of the soil Operational DCGL will not be used to backfill the excavation and will be disposed of as waste.

For any soil excavation created to remove a potentially contaminated subgrade basement structure, the excavation will be subject to a FSS prior to backfill. The FSS will be designed as an open land survey unit using the most limiting classification of the removed structure and using the Operational DCGLs for soils as the release criteria. Scanning and sampling requirements of the overburden soil will be performed using a RA sample plan as described above and implemented prior to backfill.

A radiological assessment is performed prior to introducing off-site material to LACBWR for use as backfill in a basement, or for any other use from a barrow pit, landfill, or other location. The radiological assessment consists of a gamma scans and material sampling. Gamma scans are performed *in situ*, or by package (using a hand-held instrument or through the use of a truck monitor). Material samples are analyzed by gamma spectroscopy.

5.7.1.7. Pavement Covered Areas

Paved surfaces that remain at the site following decommissioning activities will require surveys for residual radioactivity. Paved areas will be incorporated into the larger open land survey units in which they reside. This is appropriate as the pavement is outdoors where the exposure scenario is most similar to direct radiation from surface soil. Pavement will be released as a surface soil and surveyed accordingly in accordance with the classification of the open land survey unit in which it resides. Samples of the pavement will be acquired at each systematic sample location. The sample media will be pulverized, analyzed by gamma spectrometry and compared with the Operational DCGL for soil for each of the potential ROC. If pavement exhibits residual radioactivity in excess of the Base Case DCGL for soil, then the pavement will be removed and disposed of as radioactive waste and the soil beneath will be investigated.

5.7.1.8. Buried Piping

Designated sections of buried piping will be remediated in place and undergo FSS. The inventory of buried piping located below the 636 foot grade that will remain and be subjected to FSS is provided in Table 24 of RS-TD-313196-004. Compliance with the Operational DCGL values, as presented in Table 5-8, will be primarily demonstrated by measurements of total surface contamination and by the collection of sediment samples when available. The acquisition of direct measurements involves the insertion of appropriately sized detectors into the pipe interior by a simple “push-pull” methodology, whereby the position of the detector in the piping system can be easily determined in a reproducible manner.. The detectors are configured in a fixed geometry relative to the surveyed surface, thus creating a situation where an appropriate efficiency can be calculated. The detectors are then deployed into the actual pipe and timed measurements are acquired at intervals commensurate with the contamination potential of the pipe. A conservative "area of detection" is assumed for each pipe size. It is also conservatively assumed that any activity is uniformly distributed in the area of detection.

A static measurement is acquired at a pre-determined interval for the areal coverage to be achieved. The measurement output represents the gamma activity in gross cpm for each foot of piping traversed. This measurement value in cpm is then converted to dpm using the efficiency of the detector. The total activity in dpm is then converted to activity per unit area commensurate with the pipe diameter, resulting in measurement results in units of dpm/100 cm^2. A surrogate correction based upon the radionuclide distribution present in the pipe is then applied to the gamma emission to account for the presence of other non-gamma emitting radionuclides in the mixture. During data assessment, the measurement results are compared to the buried pipe Operational DCGLs.

Radiological evaluations for piping or drains that cannot be accessed directly will be performed via measurements made at traps and other appropriate access points where the radioactivity levels are deemed to either bound or be representative of the interior surface radioactivity levels providing that the conditions within the balance of the piping can be reasonably inferred based on those data.

5.7.1.9. Groundwater

Assessments of any residual radioactivity in groundwater at the site will be via groundwater monitoring wells installed at LACBWR. This is further described in Chapter 2, section 2.3.7.

5.7.1.10. Sediments and Surface Water

Sediments will be assessed by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments, as appropriate. Such samples will be collected using approved procedures based on accepted methods for sampling of this nature.

Sediment samples will be evaluated against the site-specific soil Operational DCGLs for each of the potential ROC as presented in Table 5-2. The assessment of residual radioactivity levels in surface water drainage systems will be made through the sampling of sediments, total surface contamination measurements, or both, as appropriate, making measurements at traps and other appropriate access points where it is expected that radioactivity levels will be representative or bounding of the residual radioactivity on the interior surfaces.

5.7.1.11. Survey Considerations for Buildings, Structures and Equipment

Static measurements for total surface contamination and removable surface contamination (smears) primarily apply to the radiological assessment of solid media such as structures, systems and/or equipment. The impacted above grade structures that will remain are:

- LACBWR Administration building
- G-3 Crib House
- LACBWR Crib House
- Transmission Sub-Station Switch House
- G-1 Crib House
- Barge Wash Break Room

- Back-up Control Center
- Security Station

The above grade structures listed above will be subjected to FSS using the screening values for building surface contamination from Table H.1 of NUREG-1757, Vol. 2, Appendix H. The survey approach that will be used to radiologically assess the residual radioactivity in these above grade structures is presented in section 5.6.

5.8. Final Status Survey Instrumentation

Radiation detection and measurement instrumentation for performing FSS is selected to provide both reliable operation and adequate sensitivity to detect the ROC identified at the site at levels sufficiently below the $DCGL_w$. Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field.

The DQO process includes the selection of instrumentation appropriate for the type of measurement to be performed (i.e., scan, static measurement) that are calibrated to respond to a radiation field under controlled circumstances; evaluated periodically for adequate performance to established quality standards; and sensitive enough to detect the ROC with a sufficient degree of confidence.

When possible, instrumentation selection will be made to identify the ROC at levels sufficiently below the $DCGL_w$. Detector selection will be based upon detection sensitivity, operating characteristics, and expected performance in the field. The instrumentation will, to the extent practicable, use data logging to automatically record measurements to minimize transcription errors. Commercially available portable and laboratory instruments and detectors are typically used to perform the three basic survey measurements: 1) surface scanning; 2) static measurements; and 3) radionuclide specific analysis of media samples such as soil and other bulk materials.

Specific implementing procedures will control the issuance, use, and calibration of instrumentation used for FSS. The specific DQOs for instruments are established early in the planning phase for FSS activities, implemented by Standard Operating Procedures (SOP) and executed in the survey plan. Further discussion of the DQOs for instruments is provided below.

5.8.1. Instrument Selection

The selection and proper use of appropriate instruments for both total surface contamination measurements and laboratory analyses is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The survey plan design must establish acceptable measurement techniques for scanning and direct measurements. The DQO process must include consideration as to the type of radiation, energy spectrum and spatial distribution of radioactivity as well as the characteristics of the medium to be surveyed.

For direct measurements and sample analyses, MDCs less than 10% of the Operational DCGL are preferable while MDCs up to 50% of the Operational DCGL are acceptable. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the Base Case DCGL. The target MDC for measurements obtained using laboratory instruments will be 10 percent of the applicable Operational DCGL. Measurement results with associated MDC that exceed these values may be accepted as valid data after evaluation by health physics supervision. The evaluation will consider the actual MDC, the reported value for the measurement result, the reported uncertainty and the fraction of the Operational DCGL identified in the sample. Measurement results with associated MDC that exceed these values may be accepted as valid data after evaluation by health physics supervision. The evaluation will consider the actual MDC, the reported value for the measurement result, and the fraction of the DCGL identified in the sample.

Table 5-18 Typical FSS Survey Instrumentation

Measurement Type	Detector Type	Effective Detector Area & Window Density	Instrument Model	Detector Model
Beta Static/Scan Measurement	Gas-Flow Proportional	126 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68
Beta Static/Scan Measurement	Scintillation	1.2 mg/cm^2 0.01” Plastic Scintillation 125 cm^2	Ludlum 2350-1	Ludlum 44-116
Beta Scan Measurement	Gas-Flow Proportional	584 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-37
Gamma Scan Measurement	Scintillation	2” diameter x 2” length NaI	Ludlum 2350-1	Ludlum 44-10
Gamma Static/Scan Measurement	High-purity Germanium	N/A	Canberra *in situ* Object Counting System (ISOCS)	
Gamma Pipe Static Measurement	CsI NaI NaI	0.75” x 0.75” 2” x 2” 3” x 3”	Ludlum 2350-1	Ludlum 44-159 Ludlum 44-157 Ludlum 44-162
Surface and Volumetric Material (soil, etc.)	High-purity Germanium	N/A	Canberra Lab or *in situ* Detector	N/A

Table 5-19 Typical FSS Instrument Detection Sensitivities

Instruments and Detectors [a]	Radiation	Background Count Time (minutes)	Typical Background (cpm)	Typical Instrument Efficiency [b] (ε_t)	Count Time (minutes)	Static MDC (dpm/100 cm^2)	Scan MDC
Model 43-68	Beta-Gamma	1.0	300	0.258	1.0	256	612 [c]
Model 44-116	Beta	1.0	200	0.124	1.0	539	1990 [c]
Model 43-51	Beta	1.0	40	0.126		810	2782 [c]
Model 43-37	Beta-Gamma	1.0	1,200	0.236	1.0	119	372 [c]
Model 44-10	Gamma	1.0	8,000	N/A	0.02	N/A	5.2 pCi/g [d]
HPGe	Gamma	Up to 60	N/A	60% relative	10-60	0.05 pCi/g volumetric	0.15-0.30 pCi/g [e] volumetric
Model 44-159 [f]	Gamma	1.0	700	0.024	1	5,250	N/A
Model 44-157 [f]	Gamma	1.0	6,300	0.212	1	1,750	N/A
Model 44-162 [f]	Gamma	1.0	16,000	0.510	1	1,150	N/A

[a] Detector models listed are used with the Ludlum 2350-1 Data Logger

[b] Typical calibration source used is Cs-137. The efficiency is determined by counting the source with the detector in a fixed position from the source (reproducible geometry). The ε_t value is based on ISO-7503-1 and conditions noted for each detector.

[c] Scan MDC, in dpm/100 cm^2, for the 43-68 was calculated assuming a scan rate of 5.08 cm/sec, which is equivalent to a count time of 1.73 seconds (0.028 minutes) using a detector width of 8.8 cm. The 43-37 detector assumes a scan rate of 12.7 cm/s and results in a count time of 1.05 seconds (0.018 minutes) for a detector width of 13.34 cm. The 44-116 detector width is 2.54 cm and results in a count time of 1.00 seconds at 2.54 cm/s scan speed. The 43-51 detector's width is 3.81 cm and at a scan rate of 5.08 cm/s results in a count time of 0.75 seconds.

[d] Scan MDC in pCi/g is calculated using the approach described in section 6.7.2.1 of MARSSIM for a Cs-137 nuclide fraction of 0.95 and a Co-60 fraction of 0.05 with a determined detector sensitivity of 1000 and 430 cpm per uR/hr for each radionuclide respectively. The weighted MicroShield-determined conversion factor was 0.282 pCi/g per uR/hr.

[e] *in situ* spectroscopy HPGe uses the "count to MDA" function in order to achieve the required MDC.

[f] The efficiency varies for the pipe detectors depending on the pipe diameter used. The efficiency used for the table is the averaged efficiency value for the pipe diameters. The detectors and diameters are: model 44-159: 2-4 in. dia., model 44-157: 4-8 in. dia., model 44-162: 8-12 in. dia.

Measurement results with associated MDC that exceed these values may be accepted as valid data after evaluation by health physics supervision. The evaluation will consider the actual MDC, the reported value for the measurement result, and the fraction of the DCGL identified in the sample. Other measurement instruments or techniques may be utilized. The acceptability of additional or alternate instruments or technologies for use in the FSS will be justified in a technical basis evaluation document prior to use. Technical basis evaluations for alternate FSS instruments or techniques will be provided for NRC review 30 days prior to use. This evaluation will include the following:

- Description of the conditions under which the method would be used;
- Description of the measurement method, instrumentation and criteria;
- Justification that the technique would provide the required sensitivity for the given survey unit classification; and,
- Demonstration that the instrument provides sufficient sensitivity for measurement.

Instrumentation currently proposed for use in the FSS is listed in Table 5-18. Instrument MDCs are discussed in section 5.8.4 and nominal MDC values for the proposed instrumentation are presented in Table 5-19.

5.8.2. Calibration and Maintenance

Instruments and detectors will be calibrated for the radiation types and energies of interest or to a conservative energy source. Instrument calibrations will be documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels will also be attached to all portable survey instruments. Prior to using any survey instrument, the current calibration will be verified and all operational checks will be performed.

Instrumentation used for FSS will be calibrated and maintained in accordance with approved site calibration procedures. Radioactive sources used for calibration will be traceable to the NIST and have been obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector is used, suitable NIST-traceable sources will be used for calibration, and the software set up appropriately for the desired geometry. If vendor services are used, these will be obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality.

5.8.3. Response Checks

Prior to use on-site, all project instrument calibrations will be verified and initial response data collected. These initial measurements will be used to establish performance standards (response ranges) in which the instruments will be tested against on a daily basis when in use. An acceptable response for field instrumentation is an instrument reading within ±20% of the established check source value. Laboratory instrumentation standards will be within ±3 sigma as documented on a control chart.

Instrumentation will be response checked in accordance with approved procedures for instrumentation use. Response checks will be performed daily before instrument use and again at the end of use. The check sources used for response checks will emit the same type of radiation as that being measured in the field and will be held in fixed geometry jigs for reproducibility. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that data are discarded, replacement data will be collected at the original locations.

5.8.4. Measurement Sensitivity

The measurement sensitivity or MDC will be determined *a priori* for the instruments and techniques that will be used for FSS. The MDC is defined as the *a priori* activity level that a specific instrument and technique can be expected to detect 95% of the time. When stating the detection capability of an instrument, this value should be used. The MDC is the detection limit, (*LD*), multiplied by an appropriate conversion factor to give units of activity. The critical level, (*LC*), is the lower bound on the 95% detection interval defined for *LD* and is the level at which there is a 5% chance of calling a background value "greater than background. The MDC is dependent upon the counting time, geometry, sample size, detector efficiency and background count rate.

5.8.4.1. Total Efficiency

Instrument efficiencies (ε_i) are derived from the surface emission rate of the radioactive source(s) used during the instrument calibration. Total efficiency (ε_t) is calculated by multiplying the instrument efficiency (ε_i) by the surface efficiency (ε_s) commensurate with the radionuclide's alpha or beta energy using the guidance provided in ISO 7503-1, Part 1, *Evaluation of Surface Contamination, Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters* (21).

5.8.4.2. Static Minimum Detectable Concentration

For static (direct) surface measurements with conventional detectors, such as those listed in Table 5-13, the MDC is calculated using the following equation:

Equation 5-10

$$MDC_{static} = \frac{\frac{3.0}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{\varepsilon_t\left(\frac{A}{100cm^2}\right)}$$

where:

MDC_{static}	=	Minimum Detectable Concentration in dpm/100cm^2;
t_s	=	sample count time,
t_b	=	background count time,
R_b	=	background count rate (cpm),
ε_t	=	total efficiency, and
A	=	detector window area (cm^2).

5.8.4.3. Beta-Gamma Scan Measurement Minimum Detectable Concentration

Following the guidance of sections 6.7 and 6.8 of NUREG-1507, MDCs for surface scans of surfaces for beta and gamma emitters will be computed in accordance with the following equation. For determining scan MDCs, a rate of 95% of correct detections is required and a rate of 60% of false positives is determined to be acceptable. Consequently, a sensitivity index value of 1.38 was selected from Table 6.1 of NUREG-1507. The formula used to determine the scanning MDC at the 95% confidence level is:

Equation 5-11

$$MDC_{scan} = \frac{d'\left(\sqrt{b_i} \times \frac{60}{i}\right)}{\varepsilon_t \sqrt{p}\left(\frac{A}{100}\right)}$$

where:

MDC_{scan}	=	Minimum Detectable Concentration in dpm/100cm^2;
d'	=	index of sensitivity (1.38),
i	=	observation interval (seconds),
b_i	=	background counts per observation interval,
ε_t	=	total efficiency,
p	=	surveyor efficiency (0.5), and
A	=	detector window area (cm^2).

The numerator in the beta-gamma scan MDC equation represents the Minimum Detectable Count Rate (MDCR) that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) variable is set at 0.5, as recommended by section 6.7.1 of NUREG-1507. The factor of 100 corrects for probe areas that are not 100 cm^2. The observation interval (i) is considered to be the amount of time required for the detector field of view to pass over the area of concern. This time depends upon the scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source. The scan speed is based on approximately one detector window width per second. For the Ludlum Model 43-68 gas flow proportional detector, the window width is 8.8 cm resulting in a scan speed of ~3.5 inches per second. The floor monitor detector is the Ludlum Model 43-37 with a window width of 13.35 cm which results in a scan speed of 5.25 inches per second. The source efficiency term (ε_s) will be selected to account for effects such as self-absorption, using the values found in Tables 2 and 3 in ISO 7503-1.

5.8.4.4. Gamma Scan Measurement Minimum Detectable Concentration

In addition to the MDCR and detector characteristics, the scan MDC (in pCi/g) for land areas is based on the areal extent of the hot spot, depth of the hot spot, and the radionuclide (i.e., energy and yield of gamma emissions). If one assumes constant parameters for each of the above variables, with the exception of the specific radionuclide in question, the scan MDC may be reduced to a function of the radionuclide alone.

The evaluation of open land areas requires a detection methodology of sufficient sensitivity for the identification of small areas of potentially elevated activity. Scanning measurements are performed by passing a hand-held detector, typically 2" x 2" NaI gamma scintillation detector, in gross count rate mode across the land surface under investigation. The centerline of the detector is maintained at a source-to-detector distance within 15 cm (6 in) and moved from side to side in a 1-meter wide pattern at a rate of 0.5 m/sec. This serpentine scan pattern is designed to cross each survey cell (one square meter) five times in approximately ten seconds. The audible and visuals signals are monitored for detectable increases in count rate. An observed count rate increase results in further investigation to verify findings and define the level and extent of residual radioactivity.

An *a priori* determination of scanning sensitivity is performed to ensure that the measurement system is able to detect concentrations of radioactivity at levels below the regulatory release limit. Expressed in terms of scan MDC, this sensitivity is the lowest concentration of radioactivity for a given background that the measurement system is able to detect at a specified performance level and surveyor efficiency.

This method represents the surface scanning process for land areas defined in NUREG-1507 and is the basis for calculation of the scanning detection sensitivity (scan MDC). The gamma scan MDC is discussed in detail in TSD RS-TD-313196-006, which examines the gamma sensitivity for 5.08 cm by 5.08 cm (2" x 2") NaI detectors to several radionuclide mixtures of Co-60 and Cs-137 using sand (SiO_2) as the soil base. TSD RS-TD-313196-006 derives the MDC for the radionuclide mixtures at various detector distances and scan speeds. The model in TSD RS-TD-313196-006 uses essentially the same geometry configuration as the model used in MARSSIM. TSD RS-TD-313196-006 provides MDC values for the expected LACBWR soil mixture based on detector background condition, scan speed, soil depth (15 cm), soil density (1.6 g/cm^3) and detector distance to the suspect surface.

5.8.4.5. HPGe Spectrometer Analysis

The onsite laboratory at LACBWR maintains gamma isotopic spectrometers that are calibrated to various sample geometries, including a one-liter Marinelli geometry for soil analysis. The geometries are created using the Canberra LABSOCS software. These systems are calibrated using a NIST-traceable mixed gamma source. A typical expression of MDC is presented in the following equation although this expression will vary under certain applications:

Equation 5-12

$$MDC_{(pCi/g)} = \frac{3 + 4.65\sqrt{B}}{K \times V \times t}$$

where:

B = number of background counts during the count interval t;

K = proportionality constant that relates the detector response to the activity level in a sample for a given set of measurement conditions,

V = mass of sample (g), and

t = count time (minutes).

5.8.4.6. Pipe Survey Instrumentation

Pipe survey instruments proposed for use with pipe having diameters between 0.75 and 18 inches have been shown to have efficiencies ranging from approximately 0.02 to 0.5. This equates to detection sensitivities of approximately 350 dpm/100 cm^2 to 5,200 dpm/100 cm^2. This level of sensitivity is adequate to detect residual radioactivity below the Operational DCGLs derived for the unrestricted release of buried pipe as presented in Table 5-8.

5.9. Quality Assurance

The licensee is responsible for the overall execution of the decommissioning of LACBWR including; all licensing activities, safety, radiation protection, environmental safety and health, engineering and design, quality assurance, construction management, environmental management, waste management and financial management. The licensee interfaces directly with the NRC and other stakeholders on all issues pertaining to decommissioning project activities at LACBWR.

A comprehensive QA Program has been developed to assure conformance with established regulatory requirements. The quality requirements and quality concepts are presented in the Quality Assurance Project Plan (QAPP) which adequately encompasses all risk-significant decommissioning activities. The participants in the QA Program assure that the design, procurement, construction, testing, operation, maintenance, repair, modification, dismantlement and remediation of nuclear reactor components are performed in a safe and effective manner.

The QA Program complies with the requirements set forth in Appendix B of 10 CFR 50, Appendix H of 10 CFR 71, Appendix G of 10 CFR 72. References to specific industry standards for QA and QC measures governing FSS activities are reflected in the QAPP as well as all applicable supporting procedures, plans, and instructions. Effective implementation of QA and QC measures will be verified through audit activities, with corrective actions being prescribed, implemented and verified in the event any deficiencies are identified. These measures will also apply to the any FSS related services provided by off-site vendors, in addition to on-site sub-contractors.

The QAPP has been prepared to ensure the adequacy of data being developed and used during FSS. Compliance with the QAPP will serve to ensure that FSS surveys are performed by trained individuals using approved written procedures and properly calibrated instruments that are sensitive to the suspected ROC. In addition, QC measures will be taken to obtain quantitative information to demonstrate that measurement results have the required precision and are sufficiently free of errors to accurately represent the area being investigated. QC checks will be performed as prescribed by the QAPP for both field measurements and laboratory analysis. Effective implementation of FSS operations will be verified through periodic audit and surveillance activities, including field walk-downs by Characterization/License Termination group management and program self-assessments, as appropriate. Corrective actions will be prescribed, implemented, and verified in the event any deficiencies are identified. These measures will apply to any applicable services provided by off-site vendors, as well as on-site sub-contractors.

Audit and surveillance of off-site vendors may be satisfied by International Laboratory Accreditation Cooperation (ILAC) Mutual Recognition Arrangement (MRA) accreditation as described in the NRC endorsed NEI 14-05 guidance.

5.9.1. Project Management and Organization

The Characterization/License Termination Group has been established (within the Radiation Protection and Environmental organization) with sufficient management and technical resources to fulfill project objectives and goals. The Characterization/License Termination Group is responsible for:

- Site characterization;
- LTP development and implementation; and
- The performance of FSS.

Characterization and FSS encompasses all survey and sampling activities related to the LTP. This includes site characterization surveys, RASS, RA and FSS. The duties and responsibilities of key managers as well as the various key positions within the Characterization/License Termination Group are provided in section 2.3 of the QAPP. Responsibilities for each of the positions described may be assigned to a designee. An organizational chart is provided as Figure 5-1.

5.9.2. Quality Objectives and Measurement Criteria

The QA objectives for FSS is to ensure the survey data collected are of the type and quality needed to demonstrate, with sufficient confidence, that the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. Compliance with the QAPP ensures that the following items are accomplished:

- The elements of the FSS plan are implemented in accordance with the approved procedures,
- Surveys are conducted by trained personnel using calibrated instrumentation,
- The quality of the data collected is adequate,
- All phases of package design and survey are properly reviewed, with QC and management oversight provided, and
- Corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

The following describe the basic elements of the QAPP.

5.9.2.1. Written Procedures

Sampling and survey tasks will be performed properly and consistently in order to assure the quality of FSS results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for FSS measurements.

5.9.2.2. Training and Qualifications

Personnel performing FSS measurements will be trained and qualified. Training will include the following topics:

- Procedures governing the conduct of the FSS,
- Operation of field and laboratory instrumentation used in the FSS, and
- Collection of FSS measurements and samples.

Qualification is obtained upon satisfactory demonstration of proficiency in implementation of procedural requirements. The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity required to be performed by that individual. Records of training and qualification will be maintained in accordance with approved training procedures.

5.9.2.3. Measurement and Data Acquisitions

The FSS records will be designated as quality documents and will be governed by site quality programs and procedures. Generation, handling and storage of the original FSS design and data packages will be controlled by site procedures. Each FSS measurement will be identified by individual, date, instrument, location, type of measurement, and mode of operation.

5.9.2.4. Instrument Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the required *a priori* MDC as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the approved procedures.

5.9.2.5. Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the FSS results is established by procedure. When custody is transferred outside of the organization, a CoC form will accompany the sample for tracking purposes. Secure storage will be provided for archived samples.

5.9.2.6. Control of Consumables

In order to ensure the quality of data obtained from FSS surveys and samples, new sample containers will be used for each sample taken. Tools used to collect samples will be cleaned to remove contamination prior to taking additional samples. Tools will be decontaminated after each sample collection and surveyed for contamination.

5.9.2.7. Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, will be procured from appropriate vendors in accordance with approved quality and procurement procedures.

5.9.2.8. Database Control

Software used for data reduction, storage or evaluation will be fully documented. The software will be tested and validated prior to use by an appropriate test data set.

5.9.2.9. Data Management

Survey data control from the time of collection through evaluation will be specified by procedure and survey package instructions. Manual data entries will be verified by a second individual.

5.9.3. Measurement/Data Acquisition

QC surveys and samples will be performed primarily as verification that the original FSS results are valid. QC surveys may include replicate surveys, field blanks and spiked samples, split samples, third party analysis and sample recounts. Replicate surveys apply to scan and static direct measurements. Field blanks and sample recounts apply to loose surface and material sampling surveys. Spiked samples and split samples apply to material sampling surveys. Third party analysis applies to material samples counted by a different laboratory than normally used. QC survey results will be evaluated and compared to the original FSS survey results in accordance with the appropriate acceptance criteria.

5.9.3.1. Replicate Measurements and Surveys

Replicate measurements will be performed on 5% of the static and scan locations in each applicable FSS survey package in locations chosen at random. QC replicate surveys, conducted during the FSS will be performed at the discretion of the Characterization/License Termination Manager.

Replicate static and scan measurement results will be compared to the original measurement results to determine if the acceptance criteria are met. The acceptance criteria for static measurements and scan surveys are that the same conclusion is reached for each survey unit and no other locations, greater than the scan investigation level for the area classification, are found. If the same conclusion is not reached or any exceptions are reported that were not reported in the original survey, further evaluations will be performed.

The acceptance criteria for QC replicate surveys is that both data sets either pass or fail the appropriate statistical test (i.e. Sign Test) for that survey unit. Agreement is ultimately determined that the same conclusion is reached for each data set. If the same conclusion is not reached or any exceptions are reported that were not reported in the original survey, further evaluations will be performed.

5.9.3.2. Duplicate and Split Samples

A split sample is when the original sample aliquot is separated into two aliquots and analyzed as separate samples. A duplicate sample is a second complete sample taken at the same location and same time as the original. For the FSS of surface and subsurface soils, asphalt, and sediment, a split sample analysis will be performed on 5% of the soil samples taken in a survey unit with the locations selected at random. Duplicate samples will be acquired in accordance with the direction in the specific survey package or sample plan. In addition, approximately 5% of the total number of split samples taken will be sent for analysis by a qualified off-site laboratory or separate sample analysis by the on-site laboratory using a separate detector.

The NRC Inspection Procedure No. 84750 *Radioactive Waste Treatment, and Effluent and Environmental Monitoring* (22) will be used to determine the acceptability of split and duplicate sample analyses. The sample results will be compared to determine accuracy and precision. Agreement is ultimately determined when the same conclusion is reached for each compared result. If the split sample or duplicate sample results do not agree, then further evaluations will be performed.

5.9.3.3. Field Blanks and Spiked Samples

Field blanks and spiked samples will not be performed on a routine basis. Field blanks and spiked samples will only be performed when directed by the Characterization/License Termination Manager.

The acceptance criteria for field blank samples are that no plant derived radionuclides above background are detected. If the analysis of the field blank shows the presence of plant derived radionuclides, then further evaluations will be performed.

Spiked sample results will be compared with the expected results to determine accuracy and precision in the same manner as duplicate or split samples. Agreement is ultimately determined that the same conclusion is reached for each compared result. If the spiked sample results do not agree with the expected results, further evaluations will be performed.

5.9.3.4. QC Investigations

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be performed in accordance with approved procedures; and if necessary, shall warrant a condition report in accordance with approved corrective action procedures. The investigation will include verification that the proper data sets were compared, the relevant instruments were operating properly and the survey/sample points were properly identified and located. Relevant personnel will be interviewed, as appropriate, to determine if proper instructions and procedures were followed and proper measurement and handling techniques were used including CoC, where applicable. If the investigation reveals that the data is suspect and may not represent the actual conditions, additional measurements will be taken. Following the investigation, a documented determination is made regarding the usability of the survey data and if the impact of the discrepancy adversely affects the decision on the radiological status of the survey unit.

5.9.4. Assessment and Oversight

5.9.4.1. Assessments

Focused self-assessments of FSS activities will be performed in accordance with applicable guidence. The findings will be tracked and trended.

5.9.4.2. Independent Review of Survey Results

Randomly selected survey packages (approximately 5%) from survey units will be independently reviewed to ensure that the survey measurements have been taken and documented in accordance with approved procedures.

5.9.4.3. Corrective Action Process

The corrective action process, already established as part of the site QA Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. The process will be conducted in accordance with approved corrective action procedures, which describes the methods used to identify potential Conditions Adverse to Quality (CAQ), condition reporting, self-assessment resolution and corrective action issues related to FSS. The CAQ evaluation effort is commensurate with the classification of the CAQ and could include root cause determination, extent of condition reviews, and preventive and remedial actions. Reports of audits and trend data will be reported to management in accordance with the QAPP and approved procedures.

5.9.5. Data Validation

Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels will be made and measurements exceeding the investigation levels will be evaluated. Procedurally verified data will be subjected to the Sign test and unity rule.

5.9.6. NRC and State Confirmatory Measurements

The NRC may take confirmatory measurements to assist in making a determination in accordance with 10 CFR 50.82(a)(11) that the FSS and associated documentation, demonstrate the site is suitable for release in accordance with the criteria for decommissioning in 10 CFR 20.1402. Confirmatory measurements may include collecting radiological measurements for the purpose of confirming and verifying the adequacy of the LACBWR FSS measurements. Timely and frequent communications with the NRC will ensure it is afforded sufficient opportunity for these confirmatory measurements prior to implementing any irreversible decommissioning actions.

5.10. Final Status Survey Data Assessment

The DQA approach being implemented at LACBWR is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process will include a review of the DQOs and survey plan design, will include a review of preliminary data, will use appropriate statistical testing, will verify the assumptions of the statistical tests, and will draw conclusions from the data. The DQA includes:

- verification that the measurements were obtained using approved methods;
- verification that the quality requirements were met;
- verification that the appropriate corrections were made to any gross measurements and that the data is expressed in the correct reporting units;
- verification that the measurements required by the survey design, and any measurements required to support investigation(s) have been included;
- verification that the classification and associated survey unit design remain appropriate based on a preliminary review of the data;
- subjecting the measurement results to the appropriate statistical tests;
- determining if the residual radioactivity levels in the survey unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

Once the FSS data is collected, the data for each survey unit will be assessed and evaluated to ensure that it is adequate to support the release of the survey unit. Simple assessment methods such as comparing the survey data mean result to the appropriate Operational DCGL will be performed first. The SOF will be calculated to ensure a value less than unity to demonstrate compliance with the TEDE criterion, as several radioisotopes are measured. The specific non-parametric statistical evaluations will then be applied to the final data set as necessary including the EMC (if applicable) and the verification of the initial data set assumptions. Once the assessment and evaluation is complete, any conclusions will be made as to whether the survey unit actually meets the site release criteria or whether additional actions will be required.

In some cases, data evaluation will show that all of the measurements made in a given survey unit were below the applicable Operational DCGL. If so, demonstrating compliance with the release criterion is simple and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the Operational DCGL are observed. In these cases, statistical tests must be performed to determine whether the survey unit meets the release criterion. The statistical tests must also be used in the survey design to ensure that a sufficient number of measurements are collected.

For LACBWR, the Sign Test is the most appropriate test for FSS. Characterization surveys indicate that Cs-137 found in background due to global fallout constitutes a small fraction of the DCGL. Consequently, the Sign Test will be applied to open land survey units, structural survey units, concrete basements and buried piping when demonstrating compliance with the unrestricted release criteria without subtracting background.

Survey results will be converted to appropriate units of measure (e.g., dpm/100 cm^2, pCi/g, pCi/m^2) and compared to investigation levels to determine if the action levels for investigation have been exceeded. The mean values from FSS will include the results of judgmental samples based on an area-weighted average approach. Measurements exceeding investigation action levels will be investigated. If confirmed within a Class 1 open land (soil) survey unit, the location of elevated concentration will be evaluated using the EMC, or the location can be remediated and re-surveyed. If measurements exceeding investigation action levels are confirmed within a Class 2 or 3 survey unit, the affected portions, up to the entire survey unit will be reclassified and a re-survey performed consistent with the change in classification.

5.10.1. Review of DQOs and Survey Plan Design

Prior to evaluating the data collected from a survey unit against the release criterion, the data are first confirmed to have been acquired in accordance with all applicable procedures and QA/QC requirements.

The DQO outputs will be reviewed to ensure that they are still applicable. The data collection documentation will be reviewed for consistency with the DQOs, such as ensuring the appropriate number of measurements or samples were obtained at the correct locations and that they were analyzed with measurement systems with appropriate sensitivity. A checklist will be incorporated into the approved procedure for FSS data assessment and this checklist will be used in the review. Any discrepancies between the data quality or the data collection process and the applicable requirements will be resolved and documented prior to proceeding with data analysis. Data assessment will be performed by trained personnel using the approved procedure.

5.10.2. Preliminary Data Review

The first step in the data review process is to convert all of the survey results to the appropriate units. Basic statistical quantities are then calculated for the sample data set (e.g., mean, standard deviation, and median). An initial assessment of the sample and measurement results will be used to quickly determine whether the survey unit passes or fails the release criterion or whether one of the specified non-parametric statistical analyses must be performed.

Individual measurements and sample concentrations will be compared to the Operational DCGL for evidence of small areas of elevated radioactivity or results that are statistical outliers relative to the rest of the measurements. For most FSS, interpreting the results from a survey is most straightforward when all measurements are higher or lower than the Operational DCGL. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the Operational DCGL (or an SOF of one).

5.10.2.1. Data Validation

The initial step in the preliminary review of the FSS data is a validation of the data to ensure that the data is complete, fully documented and technically acceptable. At a minimum, data validation should include the following actions:

- Ensure that the instrumentation MDC for fixed or volumetric measurements was less than 10% of the Operational DCGL, which is preferable. MDCs up to 50% of the Operational DCGL are acceptable,
- Ensure that the instrument calibration was current and traceable to NIST standards,
- Ensure that the field instruments used for FSS were source checked with satisfactory results before and after use each day that data were collected,
- Ensure that the MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey,
- Ensure that the survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed,
- Ensure that the sample was controlled from the point of sample collection to the point of obtaining results,
- Ensure that the data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility, and
- Ensure that the data have been properly recorded.

If the data review criteria are not met, the discrepancy(s) will be evaluated and the decision to accept or reject the data will be documented in accordance with approved procedures. A condition report will be used to document and resolve discrepancies as applicable.

5.10.2.2. Graphical Data Review

Graphical analyses of survey data that depict the spatial correlation of the measurements are especially useful for such assessments and will be used to the extent practical. At a minimum, a graphical review will consist of a posting plot and a frequency plot or histogram. Additional data review methodologies can be used and are detailed in section 8.2.2 of MARSSIM.

5.10.2.2.1. Posting Plot

Posting plots can be used to identify spatial patterns in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal patches of elevated radioactivity or local areas in which the Operational DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Incongruities in the background data may be the result of residual, undetected activity, or they may just reflect background variability.

5.10.2.2.2. Frequency Plot

Frequency plots can be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity. Multiple peaks in the data may indicate the presence of isolated areas of residual radioactivity or background variability due to soil types or differing materials of construction. Variability may also indicate the need to more carefully match background reference areas to survey units or to subdivide the survey unit by material or soil type.

5.10.3. Applying Statistical Test

The statistical evaluations that will be performed will test the null hypothesis (H_o) that the residual radioactivity within the survey unit exceeds the Operational DCGL. There must be sufficient survey data at or below the Operational DCGL to statistically reject the null hypothesis and conclude the survey unit meets the site release criteria. These statistical analyses can be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or electronic spreadsheets and/or databases.

5.10.3.1. Sum-of-Fractions

The SOF or "unity rule" will be applied to FSS data in accordance with the guidance provided in section 2.7 of NUREG-1757. This will be accomplished by calculating a fraction of the Operational DCGL for each sample or measurement by dividing the reported concentration by the Operational DCGL. If a sample has multiple ROC, then the fraction of the Operational DCGL for each ROC will be summed to provide a SOF for the sample.

If a surrogate Operational DCGL was calculated as part of the survey design for the FSS, then the surrogate Operational DCGL calculated will be used for the selected surrogate radionuclide. Unity rule equivalents will be calculated for each measurement result, using the surrogate adjusted Operational DCGL (Cs-137), as shown in the following equation:

Equation 5-13

$$SOF = \frac{Conc_{Cs\text{-}137}}{DCGL_{Cs\text{-}137_s}} + \frac{Conc_{Co\text{-}60}}{DCGL_{Co\text{-}60}}$$

where:

$Conc_{Cs\text{-}137}$ = measured mean concentration for Cs-137,

$DCGL_{Cs\text{-}137s}$ = Surrogate Operational DCGL for Cs-137 (inferring Sr-90),

$Conc_{Co\text{-}60}$ = measured mean concentration for Co-60,

$DCGL_{Co\text{-}60}$ = Operational DCGL for Co-60,

The unity rule equivalent results will be used to demonstrate compliance assuming the Operational DCGL is equal to one.

5.10.3.2. Sign Test

The Sign Test is a non-parametric statistical evaluation used to evaluate sample analyses where the ROC is not present in background or, present at acceptably low fractions as compared to the Operational DCGL. The Sign Test will be applied using the guidance in section 8.3 of MARSSIM.

In the event that the Sign Test fails, the survey unit will be re-evaluated to determine whether additional remediation will be required or the FSS re-designed to collect more data (i.e., a higher frequency of measurements and samples).

5.10.4. Elevated Measurement Comparison Evaluation

During FSS, areas of identified elevated activity (hot spots) may be detected and they must be evaluated both individually and in total to ensure compliance with the release criteria. The EMC is only applicable to Class 1 open land (soil) survey units when an elevated area is identified by surface scans and/or biased and systematic samples or measurements. The application of the $DCGL_{EMC}$ does not apply to basement structures, above-grade structures or buried pipe.

The investigation level for the EMC is the $DCGL_{EMC}$, which is the Base Case DCGL modified by an AF. Locations identified by surface scans or sample analyses which exceed the Base Case DCGL are subject to additional surveys to determine compliance with the elevated measurement criteria. Based upon the size of the elevated measurement area, the corresponding AF will be determined from Table 5-12 using linear or exponential interpolation as necessary.

Any identified elevated areas are each compared to the specific $DCGL_{EMC}$ value calculated for the size of the affected area. If the individual elevated areas pass, then they are combined and evaluated under the unity rule. This will be performed by determining the fraction of dose contributed by the average radioactivity across the survey unit and by adding the additional dose contribution from each individual elevated area following the guidance as provided in section 8.5.1 and section 8.5.2 of MARSSIM.

The average activity of each identified elevated areas is determined as well as the average activity value for the survey unit. The survey unit average activity value is divided by the Base Case DCGL, the survey unit average value is then subtracted from the average activity value for the elevated area and the result is divided by the appropriate $DCGL_{EMC}$. The net average activity for each identified elevated area is evaluated against its applicable $DCGL_{EMC}$. The fractions are summed and the result must be less than unity for the survey unit to pass. This is summarized in the equation as follows;

Equation 5-14

$$\frac{\delta}{DCGL_W} + \frac{\tau_1 - \delta}{DCGL_{EMC_1}} + \frac{\tau_2 - \delta}{DCGL_{EMC_2}} + \cdots \frac{\tau_n - \delta}{DCGL_{EMC_n}} < 1$$

where:

δ = the survey unit average activity;

$DCGL_W$ = the survey unit Base Case DCGL concentration,

τ_n = the average activity value of hot spot *n*, and

$DCGL_{EMCn}$ = the $DCGL_{EMC}$ concentration of hot spot *n*.

5.10.5. Data Conclusions

The results of the statistical testing, including the application of the EMC, allow for one of two conclusions to be made. The first conclusion is that the survey unit meets the site release criterion through the rejection of the null hypothesis. The data provide statistically significant evidence that the level of residual radioactivity within the survey unit does not exceed the release criteria. The decision to release the survey unit will then be made with sufficient confidence and without any further analyses.

The second conclusion that can be made is that the survey unit fails to meet the release criteria. The data may not be conclusive in showing that the residual radioactivity is less than the release criteria. As a result, the data will be analyzed further to determine the reason for failure. Potential reasons may include:

- The average residual radioactivity exceeds the Operational DCGL;
- The average residual radioactivity in soils is less than the Base Case DCGL; however, the survey unit fails the EMC test;
- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release, (i.e., an adequate number of measurements was not performed); or,
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

"Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. The power of the statistical test is a function of the number of measurements made and the standard deviation of the measurement data. Quantitatively, the power is $1 - \beta$, where β is the Type II error rate (the probability of accepting the null hypothesis when it is actually false). A retrospective power analysis can be used in the event that a survey unit is found not to meet the release criterion to determine if this is indeed due to excess residual radioactivity or if it is due to an inadequate sample size. In the case of such a failure, a retrospective power analysis will be performed using the methods as described in section I.9 and section I.10 of MARSSIM.

If the retrospective power analysis indicates insufficient power, then an assessment will be performed to determine whether the observed median concentration and/or observed standard deviation are significantly different from the estimated values used during the DQO process. The assessment will identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions can include failing the unit and starting the DQO process over, remediating some or all of the survey unit and starting the DQO process over and adjusting the LBGR to increase sample size. For example, the assessment determines that the median residual concentration in the survey unit exceeds the Operational DCGL or is higher than was estimated and planned for during the DQO process. A likely course of action might be to fail the unit or remediate and resurvey using a new sample design.

There may be cases where the decision was made during the DQO process by the planning team to accept lower power. For instance, during the DQO process the calculated relative shift was found to be less than one. The planning team adjusts the LBGR, evaluates the impact on power and accepts the lower power. In this case, the DQA process would require the planning team to compare the prospective power analysis with the retrospective power analysis and determine whether the lower power is still justified and the DQOs satisfied.

5.11. Final Status Survey Reporting

Documentation of the FSS will be contained in two types of reports and will be consistent with section 8.6 of MARSSIM. An FSS Release Record will be prepared to provide a complete record of the as-left radiological status of an individual survey unit, relative to the specified release criteria. Survey Unit Release Records will be made available to the NRC for review as appendices to the appropriate FSS Final Report. An FSS Final Report, which is a written report that is provided to the NRC for its review, will be prepared to provide a summary of the survey results and the overall conclusions which demonstrate that the site, or portions of the site, meets the radiological criteria for unrestricted use including ALARA.

It is anticipated that the FSS Final Report will be provided to the NRC in phases as remediation and FSS are completed with related portions of the site. The phased approach for submittal is intended to provide NRC with detailed insight regarding the remediation and FSS early in the process, to provide opportunities for improvement based on feedback, and to support a logical and efficient approach for technical review and independent verification.

5.11.1. FSS Release Records

An FSS Release Record will be prepared upon completion of the FSS for a specific survey unit. Sufficient data and information will be provided in the release record to enable an independent re-creation and evaluation at some future time. The FSS Release Record will contain the following information:

- Survey unit description, including unit size, descriptive maps, plots or photographs and reference coordinates;
- Classification basis, including significant HSA and characterization data used to establish the final classification;
- DQOs stating the primary objective of the survey;
- Survey design describing the design process, including methods used to determine the number of samples or measurements required based on statistical design, the number of biased or judgmental samples or measurements selected and the basis, method of sample or measurement locating, and a table providing a synopsis of the survey design;
- Survey implementation describing survey methods and instrumentation used, accessibility restrictions to sample or measurement location, number of actual samples or measurements taken, documentation activities, QC requirements and scan coverage;
- Survey results including types of analyses performed, types of statistical tests performed, statement of pass or failure of the statistical test(s);
- QC results to include discussion of split samples and/or QC replicate measurements;
- Results of any investigations;
- Any remediation activities, both historic and resulting from the performance of the FSS;
- Any changes from the FSS survey design including field changes;

- DQA conclusions;
- Any anomalies encountered during performance of the survey or in the sample results; and,
- Conclusion as to whether or not the survey unit satisfied the release criteria and whether or not sufficient power was achieved.

5.11.2. FSS Final Reports

The ultimate product of FSS is an FSS Final Report which will be, to the extent practical, a stand-alone document. To facilitate the data management process, as well as overall project management, FSS Final Reports will usually incorporate multiple FSS Release Records. To minimize the incorporation of redundant historical assessment and other FSS program information, and to facilitate potential partial site releases from the current license, FSS Final Reports will be prepared and submitted in a phased approach. FSS Final Reports will contain the following information:

- A brief overview discussion of the FSS Program including descriptions regarding survey planning, survey design, survey implementation, survey data assessment, and QA and QC measures;
- A description of the site, the applicable survey area(s) and survey unit(s), a summary of the applicable HSA information, conditions at the time of survey, identification of potential contaminants, and radiological release criteria;
- A discussion regarding the DQOs, survey unit designation and classification, background determination, FSS plans, survey design input values and method for determining sample size, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), survey methodology, QC surveys, and a discussion of any deviations during the performance of the FSS from what was described in this LTP;
- A description of the survey findings including a description of surface conditions, data conversion, survey data verification and validation, evaluation of number of sample/measurement locations, a map or drawing showing the reference system and random start systematic sample locations, and comparison of findings with the appropriate Operational DCGL or Action Level including statistical evaluations;
- Description of any judgmental and miscellaneous sample data collected in addition to those required for performing the statistical evaluation;
- Description of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the Operational DCGL;
- If survey unit fails the statistical test, a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure and the impact that the failure has on the conclusion that the facility is ready for final radiological surveys, and a discussion of the impact of the failure on survey design and result for other survey units;
- Description of how good housekeeping and ALARA practices were employed to achieve final activity levels.

As appendices to the Final Report, the applicable FSS Unit Release Record(s), all applicable implementing procedures and all applicable TSDs will be attached. If during a phased submittal, procedures and TSDs are submitted with the initial report, all subsequent submittals will only contain any revisions or additions to the applicable implementing procedures and/or TSDs.

5.12. Surveillance Following FSS

Isolation and control measures will be implemented in accordance with approved site procedures as described in section 5.6.3. Isolation and control measures will remain in force throughout FSS activities and until there is no risk of recontamination from decommissioning or the survey area has been released from the license. In the event that isolation and control measures established for a given survey unit are compromised, evaluations will be performed and documented to confirm that no radioactive material was introduced into the area that would affect the results of the FSS.

To provide additional assurance that open land survey units that have successfully undergone FSS remain unchanged until final site release, documented routine surveillances of the completed survey units will be performed. The surveillances will be performed in areas following FSS completion to monitor for indications of recontamination and verification of postings and access control measures. These routine surveillances will consist of;

- Review of access control entries since the performance of FSS or the last surveillance,
- A walk-down of the areas to check for proper postings,
- Check for materials introduced into the area or any disturbance that could change the FSS including the potential for contamination from adjacent decommissioning activities,
- Perform and document a biased scan of the survey area, focusing on access and egress points and any areas of disturbance and/or concern.

A routine surveillance will be performed in each completed FSS unit on a semi-annual basis. In addition, a surveillance will be performed at any time when an activity occurs that may have radiologically impacted the survey unit (e.g., transiting a radioactive material package though an FSS area, etc.). These surveillances will be controlled and documented in accordance with the QAPP and approved procedures. If a routine surveillance identifies physical observations and/or radiological scan measurements that require further investigation, then FSS may be repeated in the affected survey unit.

5.13. References

1. U.S. Nuclear Regulatory Commission, NUREG-1575, Revision 1, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), August 2000.
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3. U.S. Nuclear Regulatory Commission NUREG-1507, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions – June 1998.
4. U.S. Nuclear Regulatory Commission NUREG-1700, Revision 1, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans – April 2003.

5. U.S. Nuclear Regulatory Commission, NUREG-1757, Volume 2, Revision 1, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report - September 2006.

6. U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, Standard Format and Content of License Termination Plans for Nuclear Power Reactors - June 2011.

7. Energy*Solutions* Technical Support Document RS-TD-313196-004, Revision 4, LACBWR Soil DCGL and Concrete BFM Dose Factors.

8. Energy*Solutions* Technical Support Document RS-TD-313196-001, Revision 5, Radionuclides of Concern during LACBWR Decommissioning.

9. Energy*Solutions* Technical Support Document LC-FS-TSD-002, Revision 2, Operational Derived Concentration Guideline Levels for Final Status Survey.

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11. Sandia National Laboratories, NUREG/CR-5512, Volume 3, Residual Radioactive Contamination from Decommissioning Parameter Analysis – October 1999.

12. Energy*Solutions* Technical Support Document RS-TD-313196-006, Revision 0, Ludlum Model 44-10 Detector Sensitivity.

13. Energy*Solutions* GP-EO-313196-QA-PL-001, Quality Assurance Project Plan LACBWR Site Characterization Project (Characterization QAPP).

14. Energy*Solutions* Procedure LC-FS-PN-002, Revision 0, Characterization Survey Plan.

15. Energy*Solutions* Procedure LC-FS-PR-003, Revision 0, Radiological Assessments and Remedial Action Support Surveys.

16. Energy*Solutions* Technical Support Document LC-RS-PN-164017-001, Revision 0, 2015 Characterization Survey Report.

17. Energy*Solutions* LC-FS-PR-002, Revision 0, Final Status Survey Package Development

18. Energy*Solutions* LC QA-LTP-PL-001, Revision 0, Quality Assurance Project Plan LACBWR License Termination Plan (LTP) Development, Site Characterization and Final Radiation Survey Projects (QAPP).

19. Energy*Solutions* Technical Support Document RS-TD-313196-005, Revision 0, La Crosse Open Air Demolition Limits.

20. Energy*Solutions* Technical Support Document LC-FS-TSD-001, Revision 0, Use of ISOCS for FSS of End State Sub Structures at LACBWR.

21. International Standard ISO 7503-1, Part 1, Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters – August 1998.

22. U.S. Nuclear Regulatory Commission Inspection Procedure No. 84750 Radioactive Waste Treatment, and Effluent and Environmental Monitoring – March 1994.

Figure 5-1 Characterization/LTP/FSS Organization Chart