LA CROSSE BOILING WATER REACTOR

LICENSE TERMINATION PLAN

CHAPTER 5

FINAL RADIATION SURVEY 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
BIL	Basement Inventory Limit
CAQ	Conditions Adverse to Quality
CsI	Cesium Iodide
CoC	Chain of Custody
Dairyland	Dairyland Power Cooperative
DCGL	Derived Concentration Guideline Levels
DQA	Data Quality Assessment
DQO	Data Quality Objectives
DF	Dose Factors
EMC	Elevated Measurement Comparison
ETD	Easy to Detect
FOV	Field of View
FRS	Final Radiation Survey
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
MARSAME	Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual
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
STS	Source Term Survey
TEDE	Total Effective Dose Equivalent
TSD	Technical Support Document
UBGR	Upper Bound of the Gray Region
WGTV	Waste Gas Tank Vault
WTB	Waste Treatment Building

5. Final Radiation Survey Plan

The purpose of the Final Radiation Survey (FRS) Plan is to describe the methods to be used in planning, designing, conducting, and evaluating the FRS at the La Crosse Boiling Water Reactor (LACBWR). The FRS 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 FRS 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 FRS 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 criterion. The dose modeling methods and source term assumptions affect the FRS process for the various media with residual radioactivity at LACBWR. The source term for the Basement Fill Model (BFM) is the total inventory summed over the remaining below ground and backfilled walls and floors in each basement/structure. The BFM is a mixing model that is independent of the range and distribution of residual radioactivity and therefore the typical concentration-based Derived Concentration Guideline Levels (DCGL) are not applicable. For soils and buried pipe, site-specific, concentration-based DCGLs were calculated.

Because of differences in the conceptual models and required source terms, the FRS will include two somewhat different approaches depending on the media; 1) Final Status Survey (FSS) will be conducted in accordance with MARSSIM on soil and buried piping to demonstrate that residual radionuclide concentrations are equal to or below site-specific DCGLs, and 2) a "Source Term Survey" (STS) will be conducted in accordance with section 5.5 of this Chapter to demonstrate that the inventory of residual radioactivity in building basements is below a source term inventory commensurate with the dose criterion in 10 CFR 20.1402.

The term "FRS" is used in this chapter to represent both the FSS and STS surveys. When the discussion applies to only one of the survey methods, either FSS or STS, they are specifically called out.

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

The FRS Plan includes the radiological assessment of all impacted backfilled structures, buried piping and open land areas that will remain following decommissioning. DCGLs and BFM Dose Factors (DF) have been established for soil/buried piping and backfilled structures respectively, to determine compliance with the 25 mrem/yr dose criterion. After successful implementation of this FRS Plan, LaCrosse*Solutions* (Solutions) intends to release for unrestricted use the impacted open land areas, remaining backfilled structures and buried piping 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 FRS Plan does not address non-impacted areas as identified in Chapter 2.

As indicated in Chapters 3 and 4 of this License Termination Plan (LTP), all LACBWR buildings, structures and components, other than the LACBWR Administration building and Crib House, will be demolished and removed to a depth of at least 3 feet below grade. 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, including the G-3 Crib House, will remain intact and functional for G-3 power operations. The LACBWR Administrative Building and Crib House will also remain intact. Other minor structures such as the Turbine Building Switchyard Fire Service Water System will also remain. These structures are not expected to contain residual radioactivity. They will be free released in accordance with the guidance in NUREG-1575, Supplement 1, *Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual* (MARSAME) (7). 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 STS include the basements of the Reactor Building, Waste Treatment Building (WTB) and Waste Gas Tank Vault (WGTV). The portions of the Piping and Ventilation Tunnel, Reactor/Generator Plant, a one foot thick portion of the Chimney Foundation, the Turbine sump and the Turbine pit below 636 foot elevation will also remain and are grouped together for the purposes of the BFM dose assessment and FRS. The group of structures is designated as the "Remaining Structures" survey unit. All systems, components 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 WTB, the only portion of the structure that will remain is the 630 and 635 foot floor, sump and concrete footers below the 636 foot elevation. In the

WGTV and the Remaining Basements, the remaining structure will consist of the floors and foundation walls as well as concrete piling cap and piles below the 636 foot elevation.

The backfilled structural surfaces 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 structures to be plausibly occupied. The dose model that will be used to calculate and quantify the dose to the AMCG is referred to as the BFM. The approach and methodologies that will be used to calculate the remaining total inventory of residual activity remaining in the subsurface steel-reinforced concrete structures and the corresponding dose are described in this plan. Rather than using an adjusted gross DCGL to demonstrate compliance with the release criteria, the steel-reinforced concrete walls and floors of the backfilled structures/basements will be remediated to levels that are equal to or below the 25 mrem/yr dose criterion.

5.1. Radionuclides of Concern and Mixture Fractions

Energy*Solutions* Technical Support Document (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. The survey data for basements is based on core samples obtained at biased locations with elevated contact dose rates, contamination levels, and/or evidence of leaks/spills. Surface and subsurface soil samples were taken in each impacted open land survey units 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 and Remaining Structures and refines the initial suite of potential ROC by evaluating the dose significance of each radionuclide. No core samples were taken in the WGTV structure during initial characterization. Based on process knowledge, the radionuclide mixture found in the Reactor Building, WTB and Remaining Structures was applied to the WGTV as it is expected to be the most representative mixture..

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 and Sr-90 contribute greater than 91.5% of the total dose. In most scenarios the dose percentage was much greater. The remaining radionuclides were designated as insignificant dose contributors and are eliminated from further detailed evaluation. Therefore, the final ROCs for LACBWR soil and basement concrete are Cs-137, Co-60 and Sr-90.

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
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

LTP Chapter 6, section 6.5.2 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 5-2 Dose Significant Radionuclides and Mixture

Radionuclide(2)	% of Total Activity (normalized)(1)
Co-60	1.40%
Sr-90	1.00%
Cs-137	97.60%

(1) Dose significant ROC in accordance with TSD RS-TD-313196-001.
(2) Normalized radionuclide mixture for dose significant ROC from Table 20 of TSD RS-TD-313196-001.

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 FRS 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 criterion. In addition, the FRS 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 potential uncertainty in dose.

Sufficient characterization samples have been taken of the Reactor Building, WTB and Remaining Structures concrete to derive the radionuclide mixture and assess the dose impact of HTD radionuclides. The only remaining structure that has not been fully characterized to date is the concrete in the WGTV. When the basement concrete in the WGTV becomes accessible, the concrete will be characterized. If a characterization sample and/or measurement is taken on a concrete surface in the WGTV, and the result indicates a SOF in excess of 0.5 based on gamma spectroscopy results (and inferring Sr-90), then a sample will be collected at the location of the highest accessible individual measurement and analyzed for the initial suite of radionuclides. In this unlikely situation, if the analysis indicates HTD radionuclides (other than Sr-90) at concentrations exceeding the Minimum Detectable Concentration (MDC), then additional investigation/sampling will be performed. If the insignificant contributor dose is less than the insignificant contributor dose assigned to the given area for planning purposes (see LTP Chapter 6, Section 6.13.1), then the current adjustment to the Basement Dose Factors will be retained. If the insignificant contributor dose from the additional characterization of the WGTV is greater than the insignificant contributor dose assigned for planning purposes, then the Basement Dose Factors or soil DCGLs for the WGTV will be re-adjusted to account for the increased dose. It is possible, but not likely, that the data could indicate different ROC for the WGTV. If so, a specific ROC list will be applied to the WGTV. The characterization data will also be reviewed to determine if the ratio of Sr-90/Cs-137 is significantly different from the ratio currently assigned. If so, the ratios from the characterization data will be applied to the FRS surrogate calculations for the WGTV.

If a characterization sample and/or measurement is taken in soil or buried pipe, and the result indicates a SOF in excess of 0.5 based on gamma spectroscopy results (and inferring Sr-90), then a sample will be collected at the location of the highest accessible individual measurement and analyzed for HTD radionuclides. In this unlikely situation, if the analysis indicates HTD radionuclides (other than Sr-90) at concentrations exceeding the MDC, then additional investigation/sampling will be performed. Based on the results of the analysis, LACBWR commits to calculate the insignificant contributor dose. If the insignificant contributor dose is less than the insignificant contributor dose assigned for planning purposes (see LTP Chapter 6, Section 6.13.1), then the current adjustment to the Basement Dose Factors will be retained. If the insignificant contributor dose from the additional characterization of the soil or buried pipe is greater than the insignificant contributor dose assigned for planning purposes, then the soil or buried pipe DCGLs will be re-adjusted to account for the increased dose. It is possible, but not likely, that the data could indicate different ROC for the area where the additional characterization data was taken. If so, a specific ROC list will be applied to the area. The characterization data will also be reviewed to determine if the ratio of Sr-90/Cs-137 is significantly different from the ratio currently assigned. If so, the ratios from the characterization data will be applied to the FRS surrogate calculations. Based upon the analysis of radionuclide fractions and dose contribution in TSD RS-TD-313196-001, the dose contribution from HTD fractions is expected to be very low with even the most extreme HTD ratios.

5.2. Release Criteria

Before the FSS and STS process can proceed, the DCGLs and BFM DFs that are used to demonstrate compliance with the 25 mrem/yr unrestricted release criterion must be established. The DCGLs and BFM DFs are calculated by analysis of various pathways (direct radiation, inhalation, ingestion, etc.), media (concrete, soils, and groundwater) and scenarios through which exposures could occur. They are used in the survey design process to establish the minimum sensitivities required for the survey instruments and techniques and, in some cases, the spacing of measurements or samples to be made within a survey unit. Chapter 6 of this LTP describes in detail the approach, modeling parameters and assumptions used to develop the unrestricted release criterion.

For soil and buried pipe, concentrations are calculated that correspond to the dose criterion and are referred to as DCGLs. Each radionuclide-specific 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. When applied to soil, the DCGLs are expressed in units of activity per unit of mass (pCi/g). For 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 DCGL that is established for the average residual radioactivity in a survey unit is called a $DCGL_W$. Values of the $DCGL_W$ may then be increased through use of Area Factors (AF) to obtain a 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 $DCGL_{EMC}$ is only applicable to Class 1 survey units.

For basements/structures below 636 foot elevation, building-specific BFM DFs are calculated in units of mrem/yr per mCi. The total inventory remaining for each ROC will be multiplied by the BFM DFs for each radionuclide to calculate the actual dose from the remaining basement source term. These BFM DFs are radionuclide-specific values that are determined for each ROC. The dose contribution from each ROC is accounted for using the SOF to ensure that the total dose from all ROC does not exceed the dose criterion.

5.2.1. Basement Dose Factors for Structures

The BFM applies to the steel-reinforced concrete walls and floors of the backfilled Reactor Building, WTB, WGTV and the Remaining Structures below the 636 foot elevation.

BFM DFs were calculated in LTP Chapter 6, sections 6.6.6 and 6.6.8 for three scenarios resulting in BFM Groundwater DFs, BFM Drilling Spoils DFs and BFM Excavation DFs. The insignificant dose contributor percentages for the most limiting basement scenario was used to adjust each BFM DF to account for the dose from the eliminated insignificant contributor radionuclides. The BFM DFs from LTP Chapter 6, section 6.6.9 are reproduced in Table 5-3.

Table 5-3 Basement Dose Factors (mrem/yr per mCi)

	Reactor Building			Waste Treatment Building		
	Groundwater Scenario	Drilling Spoils Scenario	Excavation Scenario	Groundwater Scenario	Drilling Spoils Scenario	Excavation Scenario
	mrem/yr per mCi			mrem/yr per mCi		
Co-60	4.87E-01	1.19E-01	1.58E+00	6.66E-02	1.17E+00	2.10E+01
Sr-90	5.05E+00	2.11E-04	3.08E-03	2.93E+00	2.08E-03	4.08E-02
Cs-137	1.76E-01	2.91E-02	3.50E-01	4.16E-02	2.86E-01	4.63E+00

	Waste Gas Tank Vault			Remaining Basements		
	Groundwater Scenario	Drilling Spoils Scenario	Excavation Scenario	Groundwater Scenario	Drilling Spoils Scenario	Excavation Scenario
	mrem/yr per mCi			mrem/yr per mCi		
Co-60	1.20E-01	6.08E-01	7.00E+00	1.40E-02	1.34E-01	3.52E+00
Sr-90	1.10E+00	1.07E-03	1.37E-02	1.19E+00	2.36E-04	6.85E-03
Cs-137	4.79E-02	1.48E-01	1.54E+00	5.28E-03	3.26E-02	7.75E-01

5.2.2. 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, *LACBWR Soil DCGL and Concrete BFM Dose Factors* (9) 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 6.13.1 are reproduced in Table 5-4. 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.

Table 5-4 DCGLs for Soils (pCi/g)

Radionuclide	Soil DCGL (pCi/g)
Co-60	1.25E+01
Sr-90	6.40E+03
Cs-137	5.65E+01

5.2.3. 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 5-5 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 5-5 DCGLs for Buried Piping (dpm/100cm^2)

Radionuclide	Buried Pipe Group	Circulating Water Discharge Pipe
Co-60	4.09E+04	4.32E+04
Sr-90	5.56E+05	6.58E+05
Cs-137	1.57E+05	1.66E+05

5.2.4. Surrogate Radionuclides

The instrumentation and methods used for FRS 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. Surrogates may also be developed between beta-gamma emitting radionuclides if gross gamma counting instrumentation is used.

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 derived using pre-remediation site characterization data collected prior to the FRS. 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 and listed in Table 5-2 are the scaling factors that will be used to infer a surrogate relationship. Cs-137 is expected to be the principle surrogate radionuclide for the LACBWR site. The only HTD ROC that will be inferred at LACBWR is Sr-90. 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.

5.2.5. Sum-of-Fractions

The SOF or "unity rule" will be applied to the data used for the survey planning, and data evaluation and statistical tests for soil sample analyses since multiple radionuclide-specific measurements may 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) or the BFM DF becomes one (1). The use and application of the unity rule will be performed in accordance with section 4.3.3 of MARSSIM.

5.2.6. Dose from Groundwater

Based upon the results of groundwater monitoring performed on the LACBWR site since 1987, when the reactor was permanently shutdown 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.

5.2.7. Demonstrating Compliance with Dose Criterion

The BFM DFs, soil DCGLs and buried piping DCGLs for each ROC are presented in Tables 5-3, 5-4, and 5-5, 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 the BFM, the total inventory remaining for each ROC will be multiplied by the BFM DFs for each radionuclide to calculate the actual dose from the remaining basement source term. For soils, the dose from the residual radioactivity from each ROC (radionuclide *i*) can be expressed as shown in the following equation:

Equation 5-2

$$\text{Dose}_{\text{Media}} = \frac{\text{Conc (or Inventory)}_{\text{Radionuclide } i}}{\text{DCGL}_{\text{Radionuclide } i}}$$

The "End-State" condition of LACBWR will include dose contributions from residual radioactivity in soil, existing groundwater, buried pipes and backfilled basements. The backfilled basements or BFM dose is comprised of three different components that pertain to the maximum dose from each pathway scenario (future groundwater, drilling spoils and excavation). The final demonstration of compliance with the dose criterion will be made through the summation of dose from exposure scenarios that are not mutually exclusive and could potentially occur in parallel.

After compliance is demonstrated independently through the FRS of each basement, open land area and buried pipe, the mean concentrations or inventory values from the FRS will be used to calculate dose and summed as shown in Equation 5-3. Note that the BFM Groundwater dose from the basement assigned the Maximum BFM Excavation dose in Equation 5-3 will not be included in the "BFM Groundwater Summation" term in Equation 5-3. The two scenarios are mutually exclusive; the contamination can either be retained in the concrete and excavated or released to the fill and contribute to dose from well water, not both.

Equation 5-3

Compliance Dose = Max BFM $InSitu_{DS}$ + Max BFM_{EX} + Sum BFM_{GW} and Buried Pipe Insitu + Max Open Land + Max Buried Pipe + Max Existing Groundwater

where:

Compliance Dose = Dose to Industrial Worker AMCG (mrem/yr),

Max BFM $Insitu_{DS}$ = maximum BFM Drilling Spoils dose from basements (mrem/yr),

Max BFM_{EX} = maximum BFM Excavation dose from basements (mrem/yr),

Sum BFM_{GW} and Buried Pipe Insitu = summation of BFM Groundwater dose from all basements (mrem/yr) and Buried Pipe Insitu dose (mrem/yr) from all pipe,

Max Open Land = maximum dose from open land survey units (mrem/yr),

Max Buried Pipe = maximum dose from buried piping (mrem/yr),

Max Existing Groundwater = maximum dose from existing groundwater.

5.2.8. Area Factors

For the decommissioning of LACBWR, AF apply to soil DCGLs only. 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 5-6.

Table 5-6 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. Summary of Characterization Survey Results

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 FRS planning.

5.3.1. Survey of Impacted and Non-Impacted Media

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 would remain and be subjected to FRS before backfill 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.

5.3.2. 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 a 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.

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.

5.3.3. 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.

5.3.4. Summary of Survey Results

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

5.3.4.1. 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. 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 two Class 2 survey units. 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.

5.3.4.2. 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 charcaterization results.

The non-impacted open land area is approximately 352,360 square meters 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.

5.3.4.3. 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. Currently accessible basements that will be subjected to STS 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 STS 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 STS. See section 5.5 for the STS design criteria for survey unit boundaries and the approach to determining survey area coverage.

5.3.4.4. Inaccessible or Not Readily Accessible Areas

The survey of inaccessible or not readily accessible subsurface soils or surfaces has been deferred. Examples of areas where surveys are deferred include soils under structures, soils under concrete or asphalt coverings, currently inaccessible concrete basement surfaces and the interiors of buried pipe that may remain. 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. Soils around and under structures may be accessed using GeoProbe® technology and analyzed for the presence of residual radioactivity. 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. 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 characterization database. This additional characterization data will be used in validating the initial classification and in planning for the FRS.

5.4. Remedial Action Support (In-Process) Surveys

Remedial Action Support Surveys (RASS) are conducted to: 1) guide remediation activities; 2) determine when an area or survey unit has been adequately prepared for the FRS; and, 3) provide updated estimates of the FRS design parameters (such as standard deviation) to be used for planning the FRS.

RASS of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-12. In addition to direct and scan radiation measurements, the RASS will include the collection of samples of soil, sediment and surface residue for laboratory analysis as appropriate.

RASS of structural surfaces and systems that will be remediated, or where there is a potential for residual surface contamination, 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.

5.4.1. Description of Field Screening Methods and Instrumentation

Table 5-12 shows typical field instruments that will be used for performing FRS. The same or similar instruments will be used during the performance of the RASS. The typical MDCs for field instruments used for scanning are provided in Table 5-13 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 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) (14).

5.4.2. 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 a MDC of 50% of the applicable DCGL 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 applicable DCGL. The information obtained during the 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 DCGL and require further excavation;
- contain radioactivity concentrations that are less than the applicable DCGL, 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 applicable DCGL, 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 applicable DCGL, 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 applicable DCGL, the area will be considered suitable for FSS.

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

5.4.3. Field Screening Methods for RASS of Below-Grade Structural Surfaces

All remaining structural surfaces will be remediated to the criteria specified in Energy*Solutions* TSD RS-TD-313196-005, *La Crosse Open Air Demolition Limits* (15). These conditions or indicators are used to characterize the acceptable removable contamination and contact exposure rate levels that are allowable for open air demolition. This approach calculates the acceptable levels of fixed and removable contamination based upon re-suspension factors and ground level release and dispersion models.

Based upon the calculations, comparisons, and conclusions documented in TSD RS-TD-313196-005, the following open air demolition limits will be implemented:

- Less than 2 mR/hr beta-gamma total surface contamination on contact with structural surface.
- Less than 1,000 dpm/100cm^2 beta-gamma loose surface contamination.
- Less than 300 dpm/100cm^2 beta-gamma average contamination.

All structural surfaces will be remediated to the open air demolition limits prior to demolition. Confirmatory radiological surveys will be performed using approved procedures following remediation and prior to demolition to ensure that contamination levels are acceptable. The radiological surveys will include extensive surveys on the structural surfaces (walls and floors) located below the 636 foot elevation that will remain. These surveys will be performed using conventional gamma instruments in typical scanning and measurement modes. Scanning coverage for pre-remediation surveys on structures prior to open air demolition could include up to 100% of the accessible surface area depending on the contamination potential. Consequently, the pre-remediation surveys performed to prepare building surfaces for open air demolition will provide confidence that structural surfaces that have significant elevated activity will be removed. Surveys performed following remediation will ensure that the remediation was successful, that the remediation was sufficient to meet the unrestricted release criterion and that STS may commence. Once remediation is complete, structural surfaces located above the 636 foot elevation will be demolished, reduced in size, packaged and shipped off-site to a licensed disposal facility.

5.5. Source Term Survey (STS)

As described in section 5.4.3, all remaining floor and wall concrete surfaces will be remediated to levels that will allow demolition of above ground structures in open air with minimum contamination controls.

This criterion, presented in TSD RS-TD-313196-005, provides an upper bound on the residual radioactivity concentration that could remain in concrete at license termination. This is an important consideration in the evaluation of the potential risk from non-uniform distributions to affect the mean inventory value calculated from STS measurements. The open air demolition criteria do not include any additional remediation that may be required to meet the 25 mrem/yr license termination criteria in accordance with 10 CFR 20 Subpart E.

When a basement structure has been successfully remediated to the open air demolition criterion, a STS will be conducted to demonstrate that the inventory of residual radioactivity in building basements corresponds to a dose below the 25 mrem/yr dose criterion. The dose is calculated by multiplying the remaining inventory by the BFM DFs. The mean inventory determined by performing the STS is used in the dose calculation.

Several aspects of the BFM conceptual model and source term requirements are different from the assumptions intrinsic to a typical FSS survey design. These differences are:

1) The BFM source term is the total inventory (mCi) in all walls and floor of each building as opposed to a concentration based $DCGL_w$ (pCi/g or dpm/100cm^2) that can be represented by a single measurement.

2) The BFM is a mixing model with a source term based on total inventory that is independent of concentration levels and areal distribution of residual radioactivity. Therefore, the typical scan coverage guidance as presented in MARSSIM, which relies on concentration based $DCGL_W$ and $DCGL_{EMC}$ values, is not directly applicable.

3) The standard approach in MARSSIM for calculating AFs in conjunction with $DCGL_W$ values to determine the acceptability of elevated areas of activity does not apply to the BFM inventory source term which is independent of concentration levels and areal distribution.

4) The application of statistical testing to determine compliance on a survey unit basis does not directly apply to the BFM inventory source term if more than one survey unit is contained in a given building. In this case, the inventories of the survey units are additive, which requires a slight adjustment to the approach specified in MARSSIM to survey design and assessment.

The STS design incorporates the concept of a graded survey approach based upon the contamination potential, and retains the conceptual processes for survey design and data assessment from MARSSIM.

5.5.1. Instruments Selected for Performing STS

The Canberra ISOCS has been selected as the primary instrument that will be used to perform STS. A TSD will be 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 inventory 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 STS 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 inventory with depth in concrete.
- One of the most significant advantages of the ISOCS system in the STS application is the analytical capability to perform comprehensive uncertainty analysis of various potential source term geometries (depth and areal distribution). This can be used to ensure and document that the reported inventory data are reasonably conservative.

During the survey design for STS, the DQO process will be used to determine the most limiting geometry for the ISOCS measurements based on the physical conditions of the remediated surface and the depth and distribution of activity in the concrete surface as identified by existing characterization data and surveys taken during structure remediation. The analysis of concrete core samples taken during characterization indicate that the variability in the geometry of residual radioactivity detected at depth is not excessive. Consequently, it is not anticipated that additional concrete core sampling and/or scan surveys will be necessary to confirm the areal and depth distribution of activity in concrete in support of ISOCS geometry assumptions and sensitivity analysis. However, the results of any continuing characterization data will be taken into account during survey design to ascertain if the ISOCS geometry is sufficiently conservative to capture the variability of residual radioactivity at depth and if additional concrete core samples are necessary to confirm the depth profile.

5.5.2. STS Survey Units

STS will be designed and documented in the same manner as a traditional FSS and performed in accordance with approved procedures and in compliance with FSS quality requirements in the QAPP. The BFM is not sensitive to the concentration or areal distribution of residual radioactivity. Therefore, there is no intrinsic survey unit size limitation for the basement structures analogous to MARSSIM recommended survey unit sizes, which are based on a building occupancy scenario.

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 STS survey units will be comprised of the combined wall and floor surfaces of each remaining building basement. Contamination potential is the prime consideration for grouping STS survey units. Contiguous surface areas with the same contamination potential will minimize uncertainty in the estimate of the mean inventory 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), WTB (630 and 635 foot elevations), WGTV (618 and 621 foot elevations) and the Remaining Structures (including the Piping and Ventilation Tunnel, Reactor/Generator Plant, the one foot thick portion of the Chimney Foundation, the Turbine sump and the Turbine pit) were deemed to be unique STS units. Characterization data, radiological surveys performed to support commodity removal and surveys performed to support structural remediation for open air demolition will provide additional information to validate survey unit boundaries or possibly lead to survey design changes.

5.5.2.1. STS Areal Coverage

For the purpose of STS design, the BFM DFs for the three scenarios listed for each basement in Table 5-3 were summed resulting in one BFM DF for each of the four survey units. Compliance for each survey unit will be based on the summed BFM DF. This ensures that not only will the dose from each individual scenario be less than 25 mrem/yr but that there is also margin to ensure that the compliance dose summation across all basements and media, as indicated in Equation 5-3, will be less than 25 mrem/yr.

The single, summed BFM DFs were used to calculate a hypothetical maximum inventory level for the ROC in each basement as listed in Table 5-7. These calculated values, which are designated as Basement Inventory Levels (BIL), do not represent the inventory levels expected to remain at license termination. The BILs are calculated by dividing 25 mrem/yr by the summed BFM DFs (in units of mrem per mCi). The actual mean inventory determined by performing the STS will be used as the source term in the BFM to determine the dose to the AMCG. As described below, the actual inventory expected to remain in the basements is a small fraction of the BIL. The BIL is used during STS survey design to determine a reasonable areal coverage based upon the theoretical potential of exceeding the inventory level based upon characterization survey data.

Table 5-7 Basement Inventory Levels (BIL) Used for STS Design (mCi Total Inventory)

	Reactor Building (mCi)	WTB (mCi)	WGTV (mCi)	Remaining Basements (mCi)
Co-60	1.14E+01	1.13E+00	3.23E+00	6.82E+00
Sr-90	4.95E+00	8.41E+00	2.24E+01	2.08E+01
Cs-137	4.51E+01	5.05E+00	1.44E+01	3.08E+01

The primary consideration for determining STS areal coverage is the potential for the presence of a residual radioactivity inventory in a STS survey unit that could exceed the BIL. Survey areal coverage can be a low percentage of the total wall and floor surface area in survey units that have a low potential for exceeding the BIL. Conversely, in areas with a high potential for approaching the BIL, a higher percentage of coverage is justified.

A secondary consideration is the potential for the presence of small areas of elevated radioactivity in a STS survey unit that could exceed the BIL. The areal coverage of the STS should be commensurate with the probability that a small area of elevated radioactivity could exist within a STS survey unit with a total activity exceeding the BIL and the likelihood that such an area would be detected by the STS ISOCS measurements. As discussed in section 5.4.3, extensive surface scan surveys, in some cases 100% of the surface area, will be performed to identify areas exceeding the open air demolition criteria prior to remediation. The scanning performed during the RASS or radiological survey during remediation is integral to the STS survey planning process and will provide a high degree of confidence that areas with contamination exceeding the open air demolition criteria will be identified and remediated.

It is highly unlikely that an area containing the hypothetical maximum BIL activity inventory could remain after remediation to the 2 mR/hr open air demolition criteria. Note that the 630 and 635 foot elevations of the WTB are the only areas where contamination above the 2 mR/hr criteria is expected to be present. All contaminated concrete will be removed from the Reactor Building basement. Once the concrete and the liner are removed, it is anticipated that the residual radioactivity source term in the remaining Reactor Building basement structure will be minimal.

It is also highly unlikely that the ISOCS, with a nominal Field-of-View (FOV) of 28 m^2 would not detect and account for elevated areas approaching the BIL inventory limits if they were present assuming a reasonable areal coverage. In addition, given the information on contamination potential provided by characterization surveys performed to date, coupled with the information that will be provided by radiological surveys performed to support commodity removal, radiological surveys performed to support remediation of structural surfaces to the open air demolition criteria specified in TSD RS-TD-313196-005 and any RASS performed to demonstrate the adequacy of any remediation that was performed, there is a high degree of confidence that any area with elevated activity that could possibly exceed the BIL will be identified prior to STS. However, to provide a higher degree of confidence that all potential areas with elevated activity will be accounted for in the total inventory estimate, 100% areal coverage will be required in all STS survey units that will require remediation to meet the open air demolition criteria specified in TSD RS-TD-313196-005. As stated above, this is expected to apply only to the 630 and 635 foot elevations of the WTB. The WTB STS survey unit is classified analogous to a Class 1 as defined in MARSSIM, section 2.2 and the areal coverage from MARSSIM, Table 5.9 that corresponds to that classification.

For the remaining STS survey units (the Reactor Building, WGTV and Remaining Structures), the criteria for selecting reasonable and risk-informed areal coverage will be based on a graded approach similar to the MARSSIM, Table 5.9 scan survey guidance for Class 2 and Class 3 structures. Because elevated areas will be well known and identified, the primary basis for determining the areal coverage for the STS ISOCS measurements is the potential for the BIL to be exceeded in the survey unit. The criteria for selecting reasonable and risk-informed areal coverage are based on a graded approach similar to the guidance for scan surveys for FSS in MARSSIM section 2.2. However, instead of basing the coverage on the expected fraction of the DCGL, the coverage will be based on the fraction of the BIL expected to remain in the survey unit.

5.5.2.1.1. STS Survey Units for the Reactor Building, WGTV and Remaining Structures

TSD RS-TD-313196-001 examines the source term based upon concrete cores collected from the Reactor Building, Piping and Ventilation Tunnels and WTB. Concrete core samples were collected at locations exhibiting the highest contact dose rate in each of these areas. No samples were taken in the WGTV. However, process knowledge would indicate minimal source term in the concrete. This assumption will be confirmed by continuing characterization.

In the Reactor Building basement, all concrete will be removed to expose the steel liner. In addition, the steel liner will be removed and disposed of as radioactive waste. As a consequence, the entire source term in the Reactor Building basement will be removed as well. Note that because all of the concrete will be removed, the typical approach of classifying an area based on pre-remediation levels is not applicable. The guidance in MARSSIM would apply if the concrete were scabbled or shaved, leaving the majority of the concrete intact. This is not the case for Reactor Building basement, where all of the media will be removed, not simply the contaminated portion. The only residual radioactivity that could remain in the end-state is contaminated concrete dust remaining from the demolition process. Additionally, it is anticipated that prior to turning over the Reactor Building basement for STS, the building basements will be decontaminated to remove loose surface contamination, minimizing the potential for remaining contaminated concrete dust. Once remediation and demolition is complete, the fraction of the BIL in the Reactor Building basement from any remaining source term is expected to be small. As per the definition of a Class 2 area from MARSSIM, section 2.2, it becomes an area that had, prior to remediation, a potential for radioactive/contamination or known contamination, but is not expected to exceed the release criteria.

Concrete core samples taken in the Piping and Ventilation Tunnels indicate minimal source term in concrete in these areas, with maximum concentrations of Co-60, Sr-90 and Cs-137 of 1.39 pCi/g, 0.24 pCi/g and 19.8 pCi/g respectively in the first ½ inch of concrete surface. Consequently, the fraction of the BIL in these three STS survey units is also expected to be very small.

The STS survey units for the Reactor Building basement, WGTV and Remaining Structures are classified analogous to Class 2 as defined in MARSSIM, section 2.2. While the definition of a Class 2 area refers to the potential for exceeding the release criteria, the potential for the residual radioactivity inventory in the Reactor Building basement, WGTV and Remaining Structures exceeding the BILs as presented in Table 5-7 is highly unlikely. As a conservative measure, these STS survey units will be subjected to an areal coverage commensurate with the guidance pertaining to Class 2 scan coverage as presented in MARSSIM, Table 5.9. Sufficient ISOCS measurements will be taken to ensure that at least 10% of the surface area in each survey unit is subjected to STS survey. In addition to the prescribed areal coverage, additional judgmental measurements may be collected at locations with higher potential for containing elevated concentrations of residual radioactivity based on professional judgment.

5.5.2.2. STS Sample Size Determination

Based on the contamination potential of each STS survey unit that was determined in the previous section, along with the corresponding areal coverage, the number of ISOCS measurements required in each STS survey unit can be calculated as the quotient of the ISOCS FOV divided into the surface area required for areal coverage. Table 5-8 presents the STS survey 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 5-8 Number of ISOCS Measurements per STS Survey Unit based on Areal Coverage

STS Survey Unit	Classification	Area (m^2)	Minimum Areal Coverage (% of Area)	# of ISOCS Measurements (based on 28 m^2 FOV)
Reactor Building basement	Class 2	511.54	10%	2
WTB basement	Class 1	101.90	100%	4
WGTV basement	Class 2	460.04	10%	2
Remaining Structures	Class 2	667.26	10%	3

To ensure that the number of ISOCS measurements based on the necessary areal coverage in a STS survey unit was sufficient to satisfy a statistically based sample design, a simplified calculation was performed to determine sample size using the guidance in MARSSIM. This calculation was applied to the Class 2 STS survey units. The process was not applied to the Class 1 STS survey unit as sufficient

ISOCS measurements will be taken in the survey units to provide 100% areal coverage. For the Class 2 STS survey units, the statistical determination of sample size was accomplished by replacing DCGL with the BIL. If the sample size based on the statistical design required more ISOCS measurements than the number of ISOCS measurement required by the areal coverage, then the number of ISOCS measurements was adjusted to meet the larger sample size.

Following the guidance in MARSSIM, the Type I decision error that was used for this calculation was set at 0.05 and the Type II decision error was set at 0.05. The upper boundary of the gray region was set at the BIL. The Lower Bound of the Gray Region (LBGR) was set at 50% of the BIL in each Class 2 STS survey unit. The standard deviation of the concrete core samples taken in the Piping and Ventilation Tunnels, converted to units of total inventory (mCi), was used for sigma (σ) in the STS survey units for the Remaining Basements. For the Reactor Building basement, the entire concrete source term will be removed. Consequently, the results of any concrete core samples taken in the Reactor Building basement would not be representative of the conditions at the time of STS. For the WGTV, no concrete core data was available from initial characterization. As reasonable value for sigma (σ) cannot be determined for the Reactor Basement and WGTV STS survey units based on existing survey data, a coefficient of variation of 30% was used in accordance with the guidance in MARSSIM, section 5.5.2.2.

Using the values above, the relative shift (Δ/σ) was calculated as discussed in section 5.6.4.1.6 of this Chapter. In all cases, the relative shift (Δ/σ) was greater than three (3). Consequently, a value of three (3) was used as the adjusted relative shift (Δ/σ). Using Table 5-5 of MARSSIM, the required number of measurements (N) for use with the Sign Test, using a value of 0.05 for the Type I and Type II decision errors, is 14 measurements. Consequently, the number of ISOCS measurements in the three Class 2 STS survey units was adjusted to meet the larger sample size. Table 5-9 presents the STS survey units and the adjusted minimum number of ISOCS measurements that will be taken in each for STS.

The MARSSIM statistical approach for planning does not apply to the Class 1 areas because 100% of the area is measured which encompasses the entire statistical population. However, as noted in section 5.5.4, the Sign Test will still be performed on the Class 1 results to demonstrate with 95% confidence that the survey unit is in compliance.

Table 5-9 Adjusted Minimum Number of ISOCS Measurements per STS Survey Unit

STS Survey Unit	Classification	Required Areal Coverage (m^2)	Adjusted # of ISOCS Measurements (FOV-28 m^2)	Adjusted Areal Coverage (m^2)	Adjusted Areal Coverage (% of Area)
Reactor Building basement	Class 2	51.15	14	392	77%
WTB basement	Class 1	101.90	4	101.90	100%
WGTV basement	Class 2	46.00	14	392	85%
Remaining Structures	Class 2	66.73	14	392	59%

As a conservative measure, the areal coverage represented by the adjusted increase in sample size when using a FOV of 28 m^2 will be surveyed. However, in the case where the physical configuration or measurement geometry would make the acquisition of a 28 m^2 FOV difficult or prohibitive, then the FOV for the ISOCS measurement may be reduced provided that the adjusted number of samples remains constant and the minimum areal coverage represented by the STS survey unit classification (100% areal coverage for a Class 1 STS survey unit or 10% areal coverage for a Class 2 STS survey unit) is achieved. In addition to the prescribed areal coverage, additional judgmental measurements may be collected at locations with higher potential for containing elevated concentrations of residual radioactivity based on professional judgment.

In STS survey units where less than 100% ISOCS coverage is required, the location of the center of each ISOCS measurement FOV will be determined at a distance equal to the radius of the ISOCS FOV from the boundaries of the STS survey unit and the FOV radius of other measurement locations. If possible, the FOV for individual measurements should not overlap. If FOV overlap cannot be avoided, then adjustments shall be made, including taking additional measurements if necessary to ensure that the required areal coverage is achieved. If a selected location is found to be either inaccessible or unsuitable, then the location will be adjusted to the closest adjacent suitable location. In these cases, a notation will be made in the field log and the coordinates of the new location documented.

5.5.3. STS Survey Approach

STS 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 STS survey unit. The same area preparation, area turnover and control measures specified in section 5.6.3 will also apply to STS survey units. The Quality Assurance (QA) requirements specified in section 5.9 will also apply to the acquisition of STS measurements.

As previously stated, the ISOCS was selected as the instrument of choice to perform STS. In summary, the ISOCS detector will be oriented perpendicular to the STS 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.

The detector to source distance may be reduced to accommodate physical constraints of a particular survey unit. In this case, the FOV will be reduced and the number of measurements increased to ensure the required STS coverage as presented in Table 5-9 is achieved.

If during the course of performing a STS, 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. STS Data Assessment and Application of Results to BFM

After a sufficient number of ISOCS measurements are taken in a STS unit in accordance with the areal coverage requirements specified in Table 5-9, 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 5.2.4. 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 applicable BIL for each ROC, after converting the BIL to the same units as the ISOCS measurement (pCi/m^2). The individual ROC fractions will then be summed to provide a total SOF value for the measurement.

As described in section 5.10.3.2, the Sign Test will be used to evaluate the remaining residual radioactivity in each survey unit against the dose criterion. The SOF for each measurement will be used as the weighted sum for the Sign Test. If the Sign Test demonstrates that the mean activity of the survey unit is less than the BIL 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 (designated as the "Mean Inventory Fraction") is calculated.

If the Sign Test fails, or if the Mean Inventory Fraction in a basement exceeds one, then the survey unit will fail STS. If a survey unit fails STS, then the STS survey unit may be reclassified, additional remediation will be performed and the STS performed again.

Once the survey data set passes the Sign test, the mean radionuclide activity (mCi) for each ROC will be used to perform a conservative calculation of future dose from residual radioactivity for each STS survey unit in accordance with the following equation;

Equation 5-5

$$Dose_{STS} = ((DF_{GW}+DF_{DS}+DF_{EX}) \times \text{Mean Activity}_{ROC_1}) + ((DF_{GW}+DF_{DS}+DF_{EX}) \times \text{Mean Activity}_{ROC_2}) + ((DF_{GW}+DF_{DS}+DF_{EX}) \times \text{Mean Activity}_{ROC_i})$$

where:

$Dose_{STS}$ = Reported Dose from residual radioactivity in STS survey unit (mrem/yr)

DF_{GW} = Dose Factor for groundwater scenario for ROC*i* (mrem/yr per mCi)

DF_{DS} = Dose Factor for drilling spoils scenario for ROC*i* (mrem/yr per mCi)

DF_{EX} = Dose Factor for excavation scenario for ROC*i* (mrem/yr per mCi)

$\text{Mean Activity}_{ROC_i}$ = Mean activity of radionuclide *i* in STS survey unit (mCi)

5.6. Final Status Survey (FSS) Design

FSS design is the process used to generate FSS packages and sample plans that when implemented, are designed to demonstrate compliance with the dose-based unrestricted release criteria at LACBWR. This process will pertain primarily to open land survey units and buried pipe at LACBWR. The future dose associated with remaining backfilled concrete structures will be quantified by the STS. Any above-grade structures that will remain on the site (e. g. LACBWR Administration building and Crib House, etc.) will be released for unrestricted use using the graded survey approach from MARSAME.

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 open land survey units and survey unit classifications that will be used for the FSS of LACBWR are presented in LTP Chapter 2, section 2.1.6 and Table 2-1 and shown on Figure 2-1. A FSS Package will be prepared for each applicable survey unit. This survey package is a collection of documentation detailing FSS Sample Plan survey design, survey implementation and data evaluation. A FSS Package may 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 may include no action, investigation, resurvey, remediation and 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 applicable $DCGL_W$ 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 applicable $DCGL_W$. 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 are discussed in more detail in section 5.6.4.

Sample uncertainty is controlled by collecting a small frequency of additional samples from each survey unit. 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 may be used for the FSS of LACBWR and their respective MDC values are presented in section 5.8 and Tables 5-12 and 5-13.

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

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 as appropriate.

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. Sample Size Determination

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. By establishing the decision errors as specified above based on acceptable risk, the number of sampling and measurement locations may be controlled within reason.

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 $DCGL_W$.

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 $DCGL_W$ 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 may initially be calculated from preliminary 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 may 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 open land 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 to open land survey units 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 may 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 Elevated Measurement Comparison (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 the consideration of small areas of elevated radioactivity typically applies only to Class 1 survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_w$.

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 5.2.8.

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 5-6 for soils corresponding to the bounded area (A) calculated. If the calculated bounded area (A) falls between two area categories on Tables 5-6, 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

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

Next, the adjusted AF is used to look up a new adjusted area (A') from Table 5-6. 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 5-10, which is reproduced from the portion of Table 5.9 from MARSSIM pertaining to the FSS of open land survey units. 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 survey units are:

- For Class 1 survey units, 100 percent of the accessible soil surface will be scanned;
- For Class 2 survey units, between 10 percent and 100 percent of the accessible soil 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 may include surface drainage areas and collection points.

Table 5-10 Recommended Survey Coverage for Open Land Areas

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

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.

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 may 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 may 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 may 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 may not be identified. 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, may 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, 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 5-11.

Table 5-11 FSS Investigation Levels

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

5.6.4.6.1. Remediation and Reclassification

Any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated to reduce the residual radioactivity to acceptable levels.

If an individual survey measurement (direct measurement or sample analysis) in a Class 2 survey unit exceeds the $DCGL_w$, the survey unit, or portion of the survey unit, will be investigated. If small areas of elevated activity are confirmed by this investigation or, suggests that there may be a reasonable potential that contamination is present in excess of the $DCGL_w$, then all or part of the survey unit will be reclassified as Class 1 and the survey strategy for that survey unit redesigned accordingly. If an individual survey measurement in a Class 3 survey unit exceeds 50 percent of the $DCGL_w$, 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 $DCGL_w$, then the survey unit will be reclassified to a Class 1 or a Class 2 survey unit and the survey re-designed and re-performed accordingly.

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.

5.6.4.6.2. Resurvey

If a survey unit is re-classified (in whole or in part), or if remediation is performed within a unit, then the affected areas are subject to re-survey. Any re-surveys will be designed and performed as specified in this plan based on the appropriate classification of the survey unit. That is, if a survey unit is re-classified or a new survey unit is created, the survey design will be based on the new classification.

For example, a Class 3 area that is subdivided due to the unexpected presence of radioactivity will be divided into at least two areas. One of these may remain as a Class 3 area while the other may be a Class 2 area. In order to maintain the survey design Type I and Type II decision error rates in the Class 3 area, additional measurements may be required to be performed at randomly selected locations until the required total number of measurements is met. The new sub-divided Class 2 survey area will then be surveyed using a new survey design. The Type I and II decision error rates used will be documented in the FSS report.

A Class 2 area that is subdivided due to the levels of radioactivity identified will be divided into at least two areas as well. In this case if the original survey design criteria has been satisfied, no additional action is required, otherwise the remaining Class 2 survey unit will be redesigned. The new sub-divided survey unit will be surveyed against a new survey design.

If remediation is required in only a small area of a Class 1 survey unit (defined as if the Elevated Radioactivity Fraction (f_{EMC}) exceeds unity in 5% or less of the survey unit area), then additional measurements will be taken to determine the effectiveness of the remediation and FSS will be repeated in the remediated area and replace the measurement using the same survey design. If remediation is required in a larger area of a Class 1 survey unit (defined as if the f_{EMC} unity in greater than 5% of the survey unit area), then the FSS will be performed again under 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. At LACBWR, FSS implementation only pertains to the assessment of surface and subsurface soils as well as any buried piping that will remain at license termination. The survey approach that will be used to quantify the remaining residual radioactivity in backfilled structures that will remain at license termination is presented in section 5.5 of this FRS Plan.

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 may be based on the $DCGL_w$, a fraction of the $DCGL_w$, or the $DCGL_{EMC}$, depending upon the detection capability (instrument and surveyor) to identify radioactivity.

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 $DCGL_w$. 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.

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 may be sampled include soil, sediments, concrete and groundwater for open land areas. Bulk material samples will be analyzed via gamma spectroscopy, alpha spectroscopy 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 FRS 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 may be collected at random locations in a survey unit or may be collected at systematic locations and supplement scanning surveys for the identification of small areas of elevated activity. Fixed measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify 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 5.2.4.

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 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.

5.7.1.5. Subsurface Soils

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.

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 $DCGL_w$ will be investigated and likely remediated. The remediation process will include performing RASS of the open excavations in accordance with section 5.4.2 of this FRS 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 $DCGL_w$ may be utilized 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 $DCGL_w$, 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 subsurface $DCGL_w$, 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 $DCGL_w$, 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 may 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. No HTD analysis will perfomed during FSS. However, if a characterization sample and/or measurement is taken in subsurface soil, and the result indicates a SOF in excess of 0.5 based on gamma spectroscopy results (and inferring Sr-90), then a sample will be collected at the location of the highest accessible individual measurement and analyzed for HTD radionuclides. See section 5.1 for additional detail.

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, WTB, WGTV, Remaining Basements 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 ($DCGL_w$) as presented in Table 5-4. However, prior to license termination, 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.

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. This area will be addressed during continued characterization after the Turbine Building floor is removed. 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. No HTD analysis will be perfomed during FSS. However, if a characterization sample and/or measurement is taken in subsurface soil, and the result indicates a SOF in excess of 0.5 based on gamma spectroscopy results (and inferring Sr-90), then a sample will be collected at the location of the highest accessible individual measurement and analyzed for HTD radionuclides. See section 5.1 for additional detail.

5.7.1.6. Stored Excavated Soils

In several areas, clean overburden soils may be removed and stockpiled on site for use as backfill materials. Prior to reuse, excavated soil will be surveyed to determine its suitability. The scope of the survey will be designed and documented and will be comparable to the rigor of a Final Status Survey. Soils satisfying the criteria for unconditional release may be stockpiled for use as onsite backfill material. Soils with detectable plant-derived radioactivity at concentrations greater than background will not be used as backfill for building basements. These soils may be used to backfill excavation voids outside of the building basement footprints.

Stockpiled soils will be controlled using the methods described in section 5.6.3. Scanning requirements and soil sample frequency after stockpiled soil is used as backfill shall also be determined in accordance with the classification of the area where the soil had originated. Controls will be instituted to prevent mixing of soils from more restrictive survey area classifications (e.g., Class 2 material could be used in either Class 1 or 2 areas and Class 1 material could only be used in Class 1 areas).

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 site-specific $DCGL_w$ for soil for each of the potential ROC. If pavement exhibits residual radioactivity in excess of the site-specific $DCGL_w$ 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. Compliance with the DCGL values, as presented in Table 5-5, will be primarily demonstrated by measurements of total surface contamination and by the collection of sediment samples when available. The acquisition of direct measurements using "pipe-crawling" technology, primarily equipped with adequately sized NaI detectors. In some instances, primarily with larger sized pipe, in-situ gamma-spectroscopy may be utilized provided adequate instrument efficiencies and detection limits can be achieved. The gamma-emissions detected by the NaI detector will be scaled to infer Sr-90 in accordance with section 5.2.4. Based on the FOV of the detector used as well as the classification of the pipe, a sufficient number of measurements will be taken in each pipe to provide sufficient areal coverage of the pipe interior that is commensurate with the scan coverage requirements presented in Table 5-10. 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 DCGLs for each of the potential ROC as presented in Table 5-4. Assessment of residual radioactivity levels in surface water drainage systems will be via sampling of sediments, total surface contamination measurements, or both, as appropriate, making measurements at traps and other appropriate access points where radioactivity levels should be representative or bound those 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. At LACBWR, the BFM will be used to calculate the remaining total activity inventory of residual radioactivity remaining in the subsurface concrete structures and the corresponding dose. Consequently, these structures will not be subjected to FSS as defined by MARSSIM. The survey approach that will be used to radiologically assess the residual radioactivity in these structures is presented in section 5.5 of this FRS Plan.

5.8. Final Radiation Survey Instrumentation

Radiation detection and measurement instrumentation for performing FRS 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 FRS. The specific DQOs for instruments are established early in the planning phase for FRS 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.

Radiation detection and measurement instrumentation will be selected based on the type and quantity of radiation to be measured. The target MDC for measurements obtained using field instruments will be 50 percent of the applicable $DCGL_w$. The target MDC for field instruments is the maximum acceptable value. The actual MDCs expected to be used during FSS will be much lower. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the $DCGL_{EMC}$. The target MDC for measurements obtained using laboratory instruments will be 10 percent of the applicable $DCGL_w$. 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 FRS will be justified in a technical basis evaluation document prior to use. Technical basis evaluations for alternate final status survey 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 FRS is listed in Table 5-12. Instrument MDCs are discussed in section 5.8.4 and nominal MDC values for the proposed instrumentation are presented in Table 5-13.

Table 5-12 Typical FRS 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 5-13 Typical FRS 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.

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 FRS 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 FRS. MDC is defined as the smallest amount or concentration of radioactive material that will yield a net positive count with a 5% probability of falsely interpreting background responses as true activity from contamination and a 5% probability of interpreting a result at the MDC level as being 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* (16).

5.8.4.2. Static Minimum Detectable Concentration

For static (direct) surface measurements with conventional detectors, such as those listed in Table 5-12, the MDC is calculated using the following equation:

Equation 5-10

$$MDC_{static} = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{\varepsilon_t\left(\frac{A}{100cm^2}\right)}$$

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 one detector window width per second however; other scan speeds may be used. 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) may be adjusted 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

Designated sections of buried piping will be remediated in place, if necessary, and undergo FSS. The remaining buried pipe will be surveyed to ensure that the remaining residual radioactivity is less than the DCGLs derived for the unrestricted release of buried pipe as presented in Table 5-5. 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 DCGLs derived for the unrestricted release of buried pipe as presented in Table 5-5.

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 QAPP which adequately encompasses all risk-significant decommissioning activities. The QAPP is currently an Energy*Solutions* document, which will be converted to a LACBWR*Solutions* document after NRC agrees to the license transfer. 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 FRS 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 FRS 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 FRS. Compliance with the QAPP will serve to ensure that FRS 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 FRS operations will be verified through 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 FRS.

Characterization and FRS encompasses all survey and sampling activities related to the LTP. This includes site characterization surveys, RASS, RA, STS 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 as appropriate. An organizational chart is provided as Figure 5-1.

5.9.2. Quality Objectives and Measurement Criteria

The QA objectives for FRS 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 FRS design, analysis and evaluation. Compliance with the QAPP ensures that the following items are accomplished:

- The elements of the FRS 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 FRS results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for FRS measurements.

5.9.2.2. Training and Qualifications

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

- Procedures governing the conduct of the FRS,
- Operation of field and laboratory instrumentation used in the FRS, and
- Collection of FSS and STS 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 FRS records will be designated as quality documents and will be governed by site quality programs and procedures. Generation, handling and storage of the original FRS design and data packages will be controlled by site procedures. Each FRS 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 FRS 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 FRS 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 FRS 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 FRS 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 or STS survey package in locations chosen at random. QC replicate surveys, conducted during the FSS or STS, 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

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* (17) 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 typically involve 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. When deemed appropriate, 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 FRS 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 and STS 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 and STS. 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 as appropriate.

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 STS, 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 and STS 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 FRS to ensure the validity of FRS results and demonstrate achievement of the survey plan objectives. In this context, STS is considered as a FSS. 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 $DCGL_w$ will be performed first. The SOF will be calculated for soil data 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 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 $DCGL_w$. 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 $DCGL_w$ are observed. In these cases, statistical tests must be performed to determine whether the survey unit meets the release criterion. The statistical tests that may be required to make decisions regarding the residual radioactivity levels in a survey unit relative to the applicable $DCGL_w$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

For LACBWR, the Sign Test is expected to be the most appropriate test for FRS. Characterization surveys indicate that Cs-137 found in background due to global fallout constitutes a small fraction of the $DCGL_w$. Consequently, the Sign Test will be applied to open land 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, mCi) and compared to investigation levels to determine if the action levels for investigation have been exceeded. Measurements exceeding investigation action levels will be investigated. If confirmed within a Class 1 survey unit, the location of elevated concentration may be evaluated using the EMC, or the location may 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 FRS 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 $DCGL_w$ for evidence of small areas of elevated radioactivity or results that are statistical outliers relative to the rest of the measurements. For soils and buried pipe, interpreting the results from a survey is most straightforward when all measurements are higher or lower than the $DCGL_w$. For structures, the preliminary data review will involve the calculation of a SOF for each measurement and the derivation of a Mean Inventory Fraction. 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 $DCGL_w$ (or an SOF of one).

5.10.2.1. Data Validation

The initial step in the preliminary review of the FRS 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 below the $DCGL_w$ or if not, it was below the $DCGL_{EMC}$ for Class 1, below the $DCGL_w$ for Class 2 and below 0.5 $DCGL_w$ for Class 3 survey units (The target MDC for field instruments is the maximum acceptable value. The actual MDCs expected to be used during FSS will be much lower),
- Ensure that the instrument calibration was current and traceable to NIST standards,
- Ensure that the field instruments used for FRS 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 should consist of a posting plot and a frequency plot or histogram. Additional data review methodologies may be used and are detailed in section 8.2.2 of MARSSIM.

5.10.2.2.1. Posting Plot

Posting plots may 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 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 may 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 $DCGL_W$. There must be sufficient survey data at or below the $DCGL_W$ to statistically reject the null hypothesis and conclude the survey unit meets the site release criteria. These statistical analyses may 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 DCGL for each sample or measurement by dividing the reported concentration by the $DCGL_W$. If a sample has multiple ROC, then the fraction of the DCGL for each ROC will be summed to provide a SOF for the sample.

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

Equation 5-13

$$SOF \leq 1 = \frac{Conc_{Cs\text{-}137}}{DCGL_{Cs\text{-}137_s}} + \frac{Conc_{Co\text{-}60}}{DCGL_{Co\text{-}60}} + \frac{Conc_{Sr\text{-}90}}{DCGL_{Sr\text{-}90}}$$

where:

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

$DCGL_{Cs\text{-}137s}$ = Surrogate $DCGL_w$ for Cs-137,

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

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

$Conc_{Sr\text{-}90}$ = inferred mean concentration for Sr-90,

$DCGL_{Sr\text{-}90}$ = $DCGL_w$ for Sr-90.

The unity rule equivalent results will be used to demonstrate compliance assuming the 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 $DCGL_w$ or BIL. 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 FRS 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 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 survey units when an elevated area is identified by surface scans and/or biased and systematic samples or measurements.

The investigation level for the EMC is the $DCGL_{EMC}$, which is the $DCGL_w$ modified by an AF. Locations identified by surface scans or sample analyses which exceed the $DCGL_w$ 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 5-6 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 $DCGL_W$, 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 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 $DCGL_W$;
- The average residual radioactivity is less than the $DCGL_W$; 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 - β, 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. A retrospective power analysis may 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 may identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions may 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 $DCGL_W$ or is higher than was estimated and planned for during the DQO process. A likely cause 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 Radiation Survey Reporting

Documentation of the FRS will be contained in two types of reports and will be consistent with section 8.6 of MARSSIM. An FSS or STS Survey Unit 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 or STS Final Report. An FSS or STS 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.

It is anticipated that the FSS or STS Final Report will be provided to the NRC in phases as remediation and FSS or STS 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/STS 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 and STS Unit Release Records

An FSS or STS Unit Release Record will be prepared upon completion of the FSS or STS 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 or STS Unit 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 or STS;
- Any changes from the FSS or STS 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 and STS Final Reports

The ultimate product of FRS is an FSS or STS 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 or STS Final Reports will usually incorporate multiple FSS or STS Unit Release Records. To minimize the incorporation of redundant historical assessment and other FRS program information, and to facilitate potential partial site releases from the current license, FSS or STS Final Reports will be prepared and submitted in a phased approach. FSS or STS Final Reports will contain the following information:

- A brief overview discussion of the FRS 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 or STS 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 or STS from what was described in this LTP;
- A description of the survey findings including 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 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 $DCGL_W$;
- 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 ALARA practices were employed to achieve final activity levels.

As appendices to the Final Report, the applicable FSS or STS 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 and STS

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 and STS 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 or STS.

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 may 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

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4. U.S. Nuclear Regulatory Commission NUREG-1700, Revision 1, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans – April 2003.
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8. EnergySolutions Technical Support Document RS-TD-313196-001, Radionuclides of Concern during LACBWR Decommissioning.

9. EnergySolutions Technical Support Document RS-TD-313196-004, LACBWR Soil DCGL and Concrete BFM Dose Factors.

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

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

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16. 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.

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Figure 5-1 Characterization/LTP/FSS Organization Chart