ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 5, REVISION 1
FINAL STATUS 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
CAQ	Conditions Adverse to Quality
CCDD	Clean Concrete Demolition Debris
CsI	Cesium Iodide
CoC	Chain of Custody
CVS	Contamination Verification Survey
DCGL	Derived Concentration Guideline Levels
DQA	Data Quality Assessment
DQO	Data Quality Objectives
EMC	Elevated Measurement Comparison
ETD	Easy to Detect
FOV	Field of View
FSS	Final Status Survey
GPS	Global Positioning System
HPGe	High-Purity Germanium
HSA	Historical Site Assessment
HTD	Hard to Detect
IC	Insignificant Contributor
ISFSI	Independent Spent Fuel Storage Installation
ISOCS	*In Situ* Object Counting System
LBGR	Lower Bound of the Gray Region
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
NAD	North American Datum
NaI	Sodium Iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
ODCM	Off Site Dose Calculation Manual
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control

RA	Radiological Assessment
RASS	Remedial Action Support Survey
ROC	Radionuclides of Concern
SFP	Spent Fuel Pool
SOF	Sum of Fractions
SOP	Standard Operating Procedures
TEDE	Total Effective Dose Equivalent
TSD	Technical Support Document
UCL	Upper Confidence Level
URS	Unrestricted Release Survey
VCC	Vertical Concrete Cask
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

5. FINAL STATUS SURVEY PLAN

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

The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 are; 1) the residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an Average Member of the Critical Group (AMCG) that does not exceed 25 millirem/year (mrem/yr), including that from groundwater sources of drinking water, and 2) the residual radioactivity has been reduced to levels that are As Low As Reasonably Achievable (ALARA).

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

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

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

Dose modeling as discussed in Chapter 6 was performed to develop the residual radioactivity levels that correspond to the 25 mrem/yr dose criterion. Site-specific, concentration-based Derived Concentration Guideline Levels (DCGL) were calculated for soils, buried pipe, basement surfaces, basement penetrations and basement embedded pipe. The dose from basement surfaces, penetrations and embedded pipe are summed to determine the total dose for a given basement.

It is Zion*Solutions* expectation that the NRC will choose to conduct confirmatory measurements during the implementation of FSS. Zion*Solutions* acknowledges that the purpose of the confirmatory measurements will be to assist the NRC in making a determination that the FSS was performed in accordance with this Plan and that they verify that the site is suitable for unrestricted use in accordance with the dose criterion in 10 CFR 20.1402.

The FSS Plan includes the radiological assessment of all impacted sub-grade structures (including embedded piping and penetrations), buried piping and open land areas that will remain following decommissioning. It is Zion*Solutions* intention to release for unrestricted use the impacted open land areas and remaining below grade structures and piping from the 10 CFR 50 license, with the exception of the immediate area surrounding the Independent Spent Fuel Storage Installation (ISFSI), through the successful implementation of this FSS Plan. The ISFSI was established under the general license provisions of 10 CFR 72.210. This FSS Plan does not address non-impacted areas as identified in Chapter 2.

Section 8.5 of Exhibit C, Lease Agreement, titled "Removal of Improvements; Site Restoration" integral to the *"Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement"* (Reference 5-7) requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least three feet below grade. The ISFSI Monitoring Building and the ISFSI Warehouse will remain, however they are not part of the scope and will be included as part of the ISFSI license. The FSS of other minor solid items, such as but not limited to the switchyard structures, the microwave tower, telephone poles, fencing, culverts, duct banks and electrical conduit will be included in the open land FSS unit in which they reside.

The major structures that will remain at license termination and be subjected to FSS, are the basements of the Unit 1 Containment Building, Unit 2 Containment Building, Auxiliary Building, Turbine Building, Waste Water Treatment Facility (WWTF), the lower portion of the Fuel Handling Building (FHB), including the Spent Fuel Pool (SFP) and the Fuel Transfer Canal, Crib House and Forebay, Unit 1 and Unit 2 Steam Tunnels and the Circulating Water Intake and Discharge Tunnels below the 588 foot elevation (3 feet below grade). All systems, components as well as all structures above the 588 foot elevation (with the exception of the minor structures previously noted) will be removed during the decommissioning process and disposed of as a waste stream.

In both Containment basements, all concrete will be removed from the interior side of the steel liner above the 565 foot elevation, leaving only the remaining exposed liner below the 588 foot elevation (to the 565 foot elevation), the concrete in the In-core Instrument Shaft leading to and including the area under vessel (or Under-Vessel area), and the structural concrete outside of the liner. In the Auxiliary Building, all interior walls and floors will be removed, leaving only the exterior walls and basement floor. In the Turbine Building basement, the remaining structures will consist of reinforced concrete floors and exterior foundation walls and the sub-grade portions of the pedestals below the 588 foot elevation. For the FHB, the only portion of the structure that will remain is the lower 12 feet of the SFP below 588 foot elevation and the concrete structure of the Fuel Transfer Canal once the steel liner has been removed. Other below ground structures that will remain are the lower concrete portions of the WWTF, Main Steam Tunnels, and Circulating Water Inlet Piping and Discharge Tunnels.

The current decommissioning approach for ZSRP also calls for the beneficial reuse of concrete from building demolition as clean fill. Concrete that meets the non-radiological definition of Clean Concrete Demolition Debris (CCDD) and where radiological surveys demonstrate that the concrete meets the criteria for unconditional release will be used. See section 2.4 of this License Termination Plan (LTP) for additional discussion.

The structural surfaces that will remain at ZNPS following the termination of the license are constructed of solid steel and 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 structural surfaces, if excavated, to be realistically occupied.

The End State will also include a range of buried piping, embedded piping and penetrations. Buried piping is defined as pipe that runs through soil. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of buried piping, penetrations and embedded piping to remain is provided in Zion*Solutions* Technical Support Document (TSD) 14-016, "*Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State*" (Reference 5-8).

5.1. Radionuclides of Concern and Mixture Fractions

Zion*Solutions* TSD 11-001, *"Potential Radionuclides of Concern during the Decommissioning of Zion Station"* (Reference 5-9) was prepared and approved in November 2011. The purpose of this document was to establish the basis for an initial suite of potential radionuclides of concern (ROC) for the 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 the ZNPS was prepared. The initial suite of potential ROC is provided in Table 5-1.

Table 5-1 Initial Suite of Radionuclides

Radionuclide	Half Life (years)	Radionuclide	Half Life (years)	Radionuclide	Half Life (years)
H-3	1.24 E 01	Tc-99	2.13 E 05	Pu-238	8.77 E 01
C-14	5.73 E 03	Sb-125	2.77 E 00	Pu-239/240	2.41 E 04
Fe-55	2.70 E 00	Cs-134	2.06 E 00	Pu-241	1.44 E 01
Ni-59	7.50 E 04	Cs-137	3.00 E 01	Np-237	2.14 E 06
Co-60	5.27 E 00	Pm-147	2.62 E 00	Am-241	4.32 E 02
Ni-63	9.60 E 01	Sm-146	1.03 E 08	Am-243	7.38 E 03
Sr-90	2.91 E 01	Sm-151	9.00 E 01	Cm-244	1.81 E 01
Mo-93	3.50 E 03	Eu-152	1.33 E 01		
Nb-94	2.03 E 04	Eu-154	8.80 E 00		

LTP Chapter 2 provides detailed characterization data that describes current contamination levels in the basements and the results of surveys taken of soils. The survey data for basements is based on core samples obtained at biased locations with elevated contact dose rates 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. Zion*Solutions* TSD 14-019, *"Radionuclides of Concern for Soil and Basement Fill Model Source Terms"* (Reference 5-10) evaluates the results of the concrete core analysis data from the Containments and Auxiliary Building and refines the initial suite of radionuclides potential ROC by evaluating the dose significance of each radionuclide.

LTP Rev 1 Chapter 6, section 6.5.2 discusses the process used to derive the ROC for the decommissioning of ZNPS, including the elimination of insignificant dose contributors from the initial suite consistent with the guidance in Section 3.3 of NUREG-1757. Based upon the analysis of the mixture in TSD 14-019, Table 19, it was determined that Co-60, Ni-63, Sr-90, Cs-134 and Cs-137 accounted for 99.5% of all dose in the contaminated concrete mixes. For activated concrete, H-3, Eu-152, and Eu-154, in addition to the five aforementioned nuclides, accounted for 99% of the dose.

Table 5-2 presents the ROC for the decommissioning of ZNPS and the normalized mixture fractions based on the radionuclide mixture presented for the Auxiliary Building and Containment in TSD 14-019, Table 19.

Table 5-2 Dose Significant Radionuclides and Mixture

Radionuclide	Containment % of Total Activity (normalized)(1)	Auxiliary Building(2) % of Total Activity (normalized)(1)
H-3	0.08%	NA
Co-60	4.72%	0.92%
Ni-63	26.50%	23.71%
Sr-90	0.03%	0.05%
Cs-134	0.01%	0.01%
Cs-137	68.17%	75.32%
Eu-152	0.44%	NA
Eu-154	0.06%	NA

(1) Based on maximum percent of total activity from Table 20 of TSD 14-019, normalized to one for the dose significant radionuclides.
(2) Does not include dose significant radionuclides for activated concrete (H-3, Eu-152, Eu-154).

The results of surface and subsurface soil characterization in the impacted area surrounding ZNPS indicate that there is minimal residual radioactivity in soil. Based on the characterization survey results to date, ZSRP does not anticipate the presence of significant soil contamination in any remaining subsurface soil that has not yet been characterized. In addition, based on process knowledge, minimal contamination is expected in any of the buried piping that ZSRP plans to abandon in place. Consequently, due to the absence of any significant source term in soil or in buried piping, the suite of ROC and radionuclide mixture derived for the Auxiliary Building concrete was considered as a reasonably conservative mixture to apply to soils and buried piping for FSS planning and implementation.

The characterization surveys of several inaccessible or not readily accessible subsurface soils or structural surfaces have been deferred until safe access is available. These areas are specified in section 5.3.4.4 of this Chapter, as well as section 2.5 of LTP Chapter 2 and will be characterized during the continuing characterization process. In order to verify that the IC dose does not change prior to implementing the FSS, and to verify the HTD to surrogate radionuclide ratios used for the surrogate calculation (see LTP Rev 1 Chapter 6, section 6.5.2) are still valid, ZSRP will obtain and analyze concrete core and soil samples during continuing characterization (including radiological assessments) and FSS as described below.

For continuing characterization, 10% of all media samples collected in a survey unit, with a minimum of one sample, will be analyzed for the full initial suite of radionuclides. The samples selected will be biased to those areas exhibiting the highest gamma activity. The IC dose will be calculated for each individual sample result. If the IC dose calculated from the sample result is greater than the IC dose assigned for DCGL adjustment (see LTP Chapter 6, Section 6.5.2.3), then an investigation will be performed to determine if the DCGL requires readjustment. It is possible, but not likely, that the continuing characterization could indicate different ROC for a given survey unit. If so, the different ROC will be applied to the affected survey unit. For sample(s) with positive results for both a HTD ROC and the corresponding surrogate radionuclide (Cs-137 or Co-60), the HTD to surrogate ratio will be derived. The maximum ratio (see section 5.2.11) will be used unless specific survey information from continuing characterization supports the use of a surrogate ratio that is specific to the area. In these cases, the area-specific ratios as determined by actual survey data will be used in lieu of the maximum ratios. The area-specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit.

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

Soil samples and concrete cores will be collected during FSS to confirm the HTD to surrogate radionuclide ratios used for the surrogate calculation. Only HTD radionuclides included as ROC (H-3, Ni-63, Sr-90, for Containment and Ni-63 and Sr-90 for all other structures and soils) will be analyzed in the FSS confirmatory samples. Concrete cores will be collected from the Auxiliary Building basement, SFP/Transfer Canal, and the Under-Vessel areas in Containment where concrete will remain. The number of cores collected and analyzed for ROC HTD will be ten percent (10%) of the FSS ISOCS measurements. The concrete core locations will be selected from the floor and lower walls in the survey unit to alleviate safety concerns from working at heights and to focus on the areas expected to contain the majority of residual radioactivity. For soil, ten percent (10%) of the FSS samples collected from open land survey units will also be analyzed for ROC HTD radionuclides. Additionally, if levels of residual radioactivity in an individual soil sample exceed a SOF of 0.1, then the sample(s) will be analyzed for ROC HTD radionuclides. For soil samples or concrete cores with positive results for both a HTD ROC and the corresponding surrogate radionuclide (Cs-137 or Co-60), the HTD to surrogate ratio will be derived. The maximum ratio (see section 5.2.11) will be used unless specific survey information from continuing characterization supports the use of a surrogate ratio that is specific to the area. In these cases, the area-specific ratios as determined by actual survey data will be used in lieu of the maximum ratios. The area-specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit.

5.2. Release Criteria

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

Each radionuclide-specific Base Case DCGL is equivalent to the level of residual radioactivity (above background levels) that could, when considered independently, result in a TEDE of 25 mrem per year to an AMCG. To ensure that the summation of dose from each source term is 25 mrem/yr or less after all FSS is completed, the Base Case DCGLs are reduced based on an expected, or *a priori*, fraction of the 25 mrem/yr dose limit from each source term. The reduced DCGLs, or "Operational" DCGLs can be related to the Base Case DCGLs as an expected fraction of dose based on an *a priori* assessment of what the expected dose should be based on the results of site characterization, process knowledge and the extent of planned remediation. The Operational DCGL is then used as the DCGL for the FSS design of the survey unit (calculation of surrogate DCGLs, investigations levels, etc.). Details of the Operational DCGLs derived for each dose component and the basis for the applied *a priori* dose fractions are provided in TSD 17-004, "*Operational Derived Concentration Guideline Levels for Final Status Survey"* (Reference 5-11).

At ZNPS, compliance is demonstrated through the summation of dose from four distinct source terms for the end-state (basements, soils, buried pipe and groundwater). Basements are comprised of the summation of four structural source terms (surfaces, embedded pipe, penetrations and fill). When applied to backfilled basement surfaces below 588 foot elevation, embedded pipe and penetrations, the DCGLs are expressed in units of activity per unit of area (pCi/m^2). When applied to soil, the DCGLs are expressed in units of activity per unit of mass (pCi/g). For buried piping, DCGLs are calculated and expressed in units of activity per surface area (dpm/100 cm^2).

Multiple ROC are known to be present at ZSRP. 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.

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

5.2.1. Base Case Derived Concentration Guideline Levels for Basement Surfaces

The Basement Fill Model (BFM) applies to the steel and concrete walls and floor surfaces, below the 588 foot elevation, of the Unit 1 Containment Building, Unit 2 Containment Building, Auxiliary Building, Turbine Building, WWTF, the lower portion of the SFP, Fuel Transfer Canals, Crib House and Forebay, Unit 1 and Unit 2 Steam Tunnels and the Circulating Water Intake Piping and Circulating

Water Discharge Tunnels. The BFM source term also includes the end-state embedded piping and penetrations as specified in TSD 14-016. The DCGLs for embedded pipe and penetrations are provided in sections 5.2.7 and 5.2.9. Basement Surface DCGLs are referred to as "Base Case" DCGLs in this LTP Chapter to ensure a clear distinction from Operational DCGLs. Base Case DCGLs for basement surfaces are in units of pCi/m^2. Additional information pertaining to the calculation of DCGLs for basement surfaces is provided in LTP Chapter 6, section 6.6.8.

The Base Case $DCGL_B$ is directly analogous to the $DCGL_W$ as defined in MARSSIM and is the DCGL used during FSS to demonstrate compliance. The IC dose percentage of 5% for the Auxiliary Basement and 10% for the Containment Basement was used to adjust the Base Case $DCGL_B$ to account for the dose from the eliminated IC radionuclides. The Base Case $DCGL_B$ values, equivalent to the Basement Surfaces DCGLs from LTP Chapter 6, section 6.6.8.1 are reproduced in Table 5-3.

Table 5-3 Base Case DCGLs ($DCGL_B$) for Basements (pCi/m^2)

Nuclide	Auxiliary Building	Containment	SFP/Transfer Canal	Turbine Building	Crib House /Forebay	WWTF
H-3	5.30E+08	2.38E+08	2.38E+08	1.29E+08	1.93E+08	1.71E+07
Co-60	3.04E+08	1.57E+08	1.57E+08	7.03E+07	5.52E+07	2.83E+07
Ni-63	1.15E+10	4.02E+09	4.02E+09	2.18E+09	3.25E+09	2.89E+08
Sr-90	9.98E+06	1.43E+06	1.43E+06	7.74E+05	1.16E+06	1.03E+05
Cs-134	2.11E+08	3.01E+07	3.01E+07	1.59E+07	2.13E+07	2.31E+06
Cs-137	1.11E+08	3.94E+07	3.94E+07	2.11E+07	2.96E+07	2.93E+06
Eu-152	6.47E+08	3.66E+08	3.66E+08	1.62E+08	1.23E+08	7.55E+07
Eu-154	5.83E+08	3.19E+08	3.19E+08	1.43E+08	1.12E+08	5.74E+07

Note 1: The Base Case DCGL for the SFP/Transfer Canal set equal to the lower of either the Auxiliary Building or Containment Base Case DCGL. The Containment Base Case DCGLs were lower for all ROC, therefore the SFP/Transfer Canal Base Case DCGLs were set equal to Containment Base case DCGLs.

5.2.2. Operational Derived Concentration Guideline Levels for Basement Surfaces

The operational DCGLs for FSS of basement structural surfaces are shown in Table 5-4. Additional information pertaining to Operational DCGLs is provided in TSD 17-004.

Table 5-4 Operational DCGLs ($OpDCGL_B$) for Basements (pCi/m^2)

ROC	Auxiliary Building	Unit 1 & Unit 2 Containment		SFP/ Transfer Canal	Turbine Building		Crib House/ Forebay	WWTF
		(above 565 ft)	Under-vessel		(Floors & Walls) (1)	(Circ Water Discharge Tunnel)		
H-3	1.71E+08	3.25E+07	2.37E+08	4.98E+07	1.10E+07	5.39E+07	7.43E+07	3.28E+06
Co-60	9.81E+07	2.15E+07	1.56E+08	3.28E+07	5.98E+06	2.94E+07	2.13E+07	5.43E+06
Ni-63	3.71E+09	5.50E+08	4.00E+09	8.41E+08	1.85E+08	9.11E+08	1.25E+09	5.55E+07
Sr-90	3.22E+06	1.96E+05	1.42E+06	2.99E+05	6.58E+04	3.24E+05	4.47E+05	1.98E+04
Cs-134	6.81E+07	4.12E+06	2.99E+07	6.30E+06	1.35E+06	6.65E+06	8.20E+06	4.44E+05
Cs-137	3.58E+07	5.39E+06	3.92E+07	8.24E+06	1.79E+06	8.82E+06	1.14E+07	5.63E+05
Eu-152	2.09E+08	5.00E+07	3.64E+08	7.66E+07	1.38E+07	6.77E+07	4.74E+07	1.45E+07
Eu-154	1.88E+08	4.36E+07	3.17E+08	6.67E+07	1.22E+07	5.98E+07	4.31E+07	1.10E+07

(1) The Operational DCGLs for Floors & Walls will be applied to the surfaces in the Circulating Water Intake Pipe and Circulating Water Discharge Pipe

5.2.3. Base Case Derived Concentration Guideline Levels for Soil

The results of surface and subsurface soil characterization in the impacted area surrounding ZNPS show that there is minimal residual radioactivity in soil. At this time, based on the characterization survey results to date, ZSRP does not anticipate the presence of significant concentrations of soil contamination.

Surface soil is defined as soil residing in the first 0.15 m layer of soil. A subsurface soil category, which is defined as a layer of soil beginning at the surface but extending to a depth of 1 m is also assessed to allow for flexibility in compliance demonstration if contamination deeper than 0.15 m is encountered. Site-specific DCGLs for soil were calculated for both the 0.15 m and 1 m thicknesses. 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. Zion*Solutions* TSD 14-011, *"Soil Area Factors"* (Reference 5-12) and LTP Chapter 6, section 6.8 provides the exposure scenarios and modeling parameters that were used to calculate the site-specific DCGLs for soils (referred to a Base Case Soil DCGLs in this Chapter). The surface and subsurface soil Base Case DCGLs for the unrestricted release of open land survey units are provided in Tables 5-5 and 5-6, respectively. The IC dose percentage of 10% was used to adjust the DCGLs in Tables 5-5 and 5-6 to account for the dose from the eliminated IC radionuclides.

Table 5-5 Base Case DCGLs for Surface Soils ($DCGL_{SS}$)

Radionuclide	Surface Soil DCGL (pCi/g)
Co-60	4.26
Cs-134	6.77
Cs-137	14.18
Ni-63	3572.10
Sr-90	12.09

Table 5-6 Base Case DCGLs for Subsurface Soils ($DCGL_{SB}$)

Radionuclide	Subsurface Soil DCGL (pCi/g)
Co-60	3.44
Cs-134	4.44
Cs-137	7.75
Ni-63	763.02
Sr-90	1.66

5.2.4. Operational Derived Concentration Guideline Levels for Soil

The operational DCGLs for FSS of surface and subsurface soils are presented in Tables 5-7 and 5-8, respectively. Once the FSS of structures is complete, the Operational DCGLs for soils may be revised by incorporating the difference between the *a priori* fraction of dose for the maximum basement and the actual fraction of dose for the maximum basement as measured by FSS results. Additional information pertaining to Operational DCGLs is provided in TSD 17-004.

Table 5-7 Operational DCGLs for Surface Soils ($OpDCGL_{SS}$)

Radionuclide	Surface Soil (pCi/g)
Co-60	1.091
Cs-134	1.733
Cs-137	3.630
Ni-63	914.458
Sr-90	3.095

Table 5-8 Operational DCGLs for Subsurface Soils ($OpDCGL_{SB}$)

Radionuclide	Subsurface Soil (pCi/g)
Co-60	0.881
Cs-134	1.137
Cs-137	1.984
Ni-63	195.333
Sr-90	0.425

5.2.5. Base Case Derived Concentration Guideline Levels for Buried Piping

The residual radioactivity in buried piping located below the 588 foot grade that will remain and be subjected to FSS is discussed in LTP Chapter 2, section 2.3.3.7 and TSD 14-016. The dose assessment methods and resulting DCGLs for buried piping are described in detail in TSD 14-015, *"Buried Pipe Dose Modeling & DCGLs"* (Reference 5-13) and LTP Chapter 6, section 6.12. Table 5-9 presents the DCGLs for buried piping from LTP Chapter 6, section 6.12 (referred to as Base Case DCGLs for buried piping in this Chapter).

Table 5-9 Base Case DCGLs for Buried Piping ($DCGL_{BP}$)

Radionuclide	Buried Piping DCGL (dpm/100 cm^2)
Co-60	2.64E+04
Cs-134	4.54E+04
Cs-137	1.01E+05
Ni-63	4.89E+07
Sr-90	4.50E+04

5.2.6. Operational Derived Concentration Guideline Levels for Buried Piping

The operational DCGLs for the FSS of buried piping are presented in Table 5-10. Once the FSS of structures is complete, the Operational DCGLs for buried piping may be revised by incorporating the difference between the *a priori* fraction of dose for the maximum basement and the actual fraction of dose for the maximum basement as measured by FSS results. Additional information pertaining to Operational DCGLs is provided in TSD 17-004.

Table 5-10 Operational DCGLs for Buried Piping ($OpDCGL_{BP}$)

Radionuclide	Buried Piping (dpm/100 cm^2)
Co-60	6.76E+03
Cs-134	1.16E+04
Cs-137	2.59E+04
Ni-63	1.25E+07
Sr-90	1.15E+04

5.2.7. Base Case Derived Concentration Guideline Levels for Embedded Pipe

The BFM groundwater source term transport and dose assessment pathways applicable to embedded pipe are the same as those assumed for concrete, i.e., the activity in the pipe is assumed to be released and mixed with the water in the interstitial spaces of the fill material with the water then used for drinking and irrigation. Note that the DCGLs calculated for embedded pipe are based on an assumption of instant release of all activity into the basement fill.

A FSS will be conducted on the interior surfaces of embedded piping to demonstrate that the concentrations of residual radioactivity are equal to or below DCGLs corresponding to the dose criterion in 10 CFR 20.1402 ($DCGL_{EP}$). $DCGL_{EP}$ were calculated for each of the embedded pipe survey units. The $DCGL_{EP}$ values from LTP Chapter 6, section 6.13 are reproduced in Table 5-11 (referred to as Base Case DCGLs for embedded piping in this Chapter). The IC dose percentages of 10% for Containment and 5% for all other survey units was used to adjust the $DCGL_{EP}$ values in Table 5-11 to account for the dose from the eliminated IC radionuclides.

Table 5-11 Base Case DCGLs for Embedded Pipe ($DCGL_{EP}$)

Radionuclide	Auxiliary Bldg. Basement Embedded Floor Drains (pCi/m^2)	Turbine Bldg. Basement Embedded Floor Drains (pCi/m^2)	Unit 1 & Unit 2 Containment In-Core Sump Embedded Drain Pipe (pCi/m^2)	Unit 1 & Unit 2 Steam Tunnel Embedded Floor Drains (pCi/m^2)	Unit 1 & Unit 2 Tendon Tunnel Embedded Floor Drains (pCi/m^2)
H-3	N/A	N/A	8.28E+09	N/A	1.61E+10
Co-60	7.33E+09	6.31E+09	5.47E+09	4.07E+10	1.06E+10
Ni-63	2.78E+11	1.96E+11	1.40E+11	1.26E+12	2.72E+11
Sr-90	2.41E+08	6.94E+07	4.98E+07	4.48E+08	9.70E+07
Cs-134	5.10E+09	1.43E+09	1.05E+09	9.22E+09	2.04E+09
Cs-137	2.68E+09	1.89E+09	1.37E+09	1.22E+10	2.67E+09
Eu-152	N/A	N/A	1.28E+10	N/A	2.48E+10
Eu-154	N/A	N/A	1.11E+10	N/A	2.16E+10

5.2.8. Operational Derived Concentration Guideline Levels for Embedded Pipe

The operational DCGLs for the FSS of buried piping are presented in Table 5-12. Additional information pertaining to Operational DCGLs is provided in TSD 17-004.

Table 5-12 Operational DCGLs for Embedded Pipe ($OpDCGL_{EP}$)

Radionuclide	Auxiliary Bldg. Basement Embedded Floor Drains (pCi/m^2)	Turbine Bldg. Basement Embedded Floor Drains (pCi/m^2)	Unit 1 Containment In-Core Sump Embedded Drain Pipe (pCi/m^2)	Unit 2 Containment In-Core Sump Embedded Drain Pipe (pCi/m^2)	Unit 1 & Unit 2 Steam Tunnel Embedded Floor Drains (pCi/m^2)	Unit 1 Tendon Tunnel Embedded Floor Drains (pCi/m^2)	Unit 2 Tendon Tunnel Embedded Floor Drains (pCi/m^2)
H-3	N/A	N/A	6.62E+08	6.62E+08	N/A	3.22E+08	3.22E+08
Co-60	7.33E+09	2.52E+08	4.38E+08	4.38E+08	1.63E+09	2.12E+08	2.12E+08
Ni-63	2.78E+11	7.84E+09	1.12E+10	1.12E+10	5.04E+10	5.44E+09	5.44E+09
Sr-90	2.41E+08	2.78E+06	3.98E+06	3.98E+06	1.79E+07	1.94E+06	1.94E+06
Cs-134	5.10E+09	5.72E+07	8.40E+07	8.40E+07	3.69E+08	4.08E+07	4.08E+07
Cs-137	2.68E+09	7.56E+07	1.10E+08	1.10E+08	4.88E+08	5.34E+07	5.34E+07
Eu-152	N/A	N/A	1.02E+09	1.02E+09	N/A	4.96E+08	4.96E+08
Eu-154	N/A	N/A	8.88E+08	8.88E+08	N/A	4.32E+08	4.32E+08

5.2.9. Base Case Derived Concentration Guideline Levels for Penetrations

A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end.

A penetration survey unit is defined for each basement. The direction that the residual radioactivity may migrate, i.e., into which basement, cannot be predicted with certainty. Therefore, a given penetration that begins in one basement and ends in another will be included in the survey units for both basements. The residual radioactivity in the penetration is assumed to release to both basements simultaneously.

The BFM groundwater source term transport and dose assessment pathways applicable to penetrations are the same as those assumed for concrete, i.e., the activity in the penetration is released and mixed with the water in the interstitial spaces of the fill material with the water then used for drinking and irrigation. Note that the DCGLs calculated for penetrations are based on an assumption of instant release of all activity into the basement fill.

A FSS will be conducted on the interior surfaces of penetrations to demonstrate that the concentrations of residual radioactivity are equal to or below DCGLs corresponding to the dose criterion in 10 CFR 20.1402 ($DCGL_{PN}$). By definition a given penetration interfaces two basements. The lesser $DCGL_{PN}$ of the two basements will be used for remediation and grouting action levels. $DCGL_{PN}$ were calculated for each of the embedded pipe survey units. The $DCGL_{PN}$ values from LTP Chapter 6, section 6.14 are reproduced in Table 5-13 (referred to as Base Case DCGLs for penetrations in this Chapter). The IC dose percentages of 10% for Containment and 5% for all other survey units was used to adjust the $DCGL_{PN}$ values in Table 5-13 to account for the dose from the eliminated IC radionuclides.

Table 5-13 Base Case DCGLs for Penetrations ($DCGL_{PN}$)

Radionuclide	Auxiliary Bldg. (pCi/m^2)	U1/U2 Containment (pCi/m^2)	SFP/ Transfer Canal (pCi/m^2)	Turbine Bldg. (pCi/m^2)	Crib House/ Forebay (1) (pCi/m^2)	WWTF[1] (pCi/m^2)
H-3	3.99E+09	3.42E+09	4.84E+16	3.23E+09	N/A	N/A
Co-60	8.82E+07	2.26E+09	4.45E+08	1.76E+09	N/A	N/A
Ni-63	6.79E+10	5.78E+10	1.86E+14	5.48E+10	N/A	N/A
Sr-90	2.41E+07	2.06E+07	9.26E+10	1.94E+07	N/A	N/A
Cs-134	3.28E+08	4.32E+08	7.48E+08	4.00E+08	N/A	N/A
Cs-137	6.17E+08	5.66E+08	1.46E+09	5.29E+08	N/A	N/A
Eu-152	3.29E+08	5.26E+09	9.44E+08	4.06E+09	N/A	N/A
Eu-154	2.33E+08	4.58E+09	8.53E+08	3.58E+09	N/A	N/A

(1) The Base Case $DCGL_{PN}$ for the Crib House/Forebay and WWTF are listed a not applicable due the very small surface area of the penetrations present. These penetrations are included with the Crib House/Forebay and WWTF surface survey units and the surface $DCGL_B$ will apply.

5.2.10. Operational Derived Concentration Guideline Levels for Penetrations

The operational DCGLs for the FSS of penetrations are presented in Table 5-14. Because a given penetration interfaces two basements, the lesser $OpDCGL_{PN}$ of the two basements will be used for FSS design and implementation. Additional information pertaining to Operational DCGLs is provided in TSD 17-004.

Table 5-14 Operational DCGLs for Penetrations ($OpDCGL_{PN}$)

Radionuclide	Auxiliary Bldg. (pCi/m^2)	Unit 1/Unit 2 Containment (pCi/m^2)	SFP/ Transfer Canal (pCi/m^2)	Turbine Bldg. (pCi/m^2)	Crib House/ Forebay (pCi/m^2)	WWTF (pCi/m^2)
H-3	3.14E+08	2.33E+08	1.13E+16	2.58E+08	N/A	N/A
Co-60	6.95E+06	1.54E+08	1.04E+08	1.41E+08	N/A	N/A
Ni-63	5.35E+09	3.93E+09	4.33E+13	4.38E+09	N/A	N/A
Sr-90	1.90E+06	1.40E+06	2.16E+10	1.55E+06	N/A	N/A
Cs-134	2.58E+07	2.94E+07	1.74E+08	3.20E+07	N/A	N/A
Cs-137	4.86E+07	3.85E+07	3.40E+08	4.23E+07	N/A	N/A
Eu-152	2.59E+07	3.58E+08	2.20E+08	3.25E+08	N/A	N/A
Eu-154	1.84E+07	3.11E+08	1.99E+08	2.86E+08	N/A	N/A

5.2.11. Surrogate Radionuclides

The instrumentation and methods used for FSS will be based on the measurement of beta-gamma emitting radionuclides by either gamma spectroscopy or gross counting. The option is available to use gross beta measurements for survey of piping but this approach is not currently planned. Assuming gamma measurements are used for the survey, the concentrations of the HTD radionuclide(s) will be based on known ratio(s) of the HTD radionuclide(s) to beta-gamma radionuclide(s) when demonstrating compliance with the release criteria. This is accomplished through the application of a surrogate relationship. 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 in most cases derived using pre-remediation site characterization data collected prior to the FSS. A surrogate ratio DCGL allows the DCGLs specific to HTD radionuclides in a mixture to be expressed in terms of a single radionuclide that is more readily measured or easy-to-detect (ETD). The ETD or measured radionuclide is called the surrogate radionuclide.

The final ROC for the decommissioning of Zion are Co-60, Cs-134 and Cs-137 (as well as Eu-152 and Eu-154 for Containment), which are gamma emitters and Ni-63, Sr-90 and H-3 (applicable only to Containment), which are HTD radionuclides. During FSS, HTD concentrations will be inferred using a surrogate approach. Cs-137 is the principle surrogate radionuclide for H-3 and Sr-90 and Co-60 is the principle surrogate radionuclide for Ni-63. The mean, maximum and 95% Upper Confidence Level (UCL) of the surrogate ratios for concrete core samples taken in the Containment and Auxiliary Building basements were calculated in TSD 14-019 and are presented in Table 5-15. The maximum ratios will be used in the surrogate calculations during FSS unless area specific ratios are determined by continuing characterization. Note that the 95% UCL is conservatively based in the standard deviation of the individual values as opposed to the standard deviation of the mean.

Table 5-15 Surrogate Ratios

Ratios	Containment			Auxiliary Building		
	Mean	Max	95%UCL	Mean	Max	95%UCL
H-3/Cs-137	0.208	1.760	0.961	N/A	N/A	N/A
Ni-63/Co-60	30.623	442	193.910	44.143	180.450	154.632
Sr-90/Cs-137	0.002	0.021	0.010	0.001	0.002	0.002

Any future continuing characterization or FSS data that contains positive results for H-3, Ni-63 and Sr-90 will be reviewed. In these cases, the area specific ratios as determined by actual survey data will be used in lieu of the maximum ratios presented in Table 5-15. The area-specific ratios used and the survey data serving as the basis for the ratios will be documented in the release record for the survey unit.

Using the appropriate scaling factors, the DCGL of the measured radionuclide is modified to account for the represented radionuclide(s) 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 represented radionuclide, and
$Conc_{ETD}$ = Ratio of the ETD or surrogate radionuclide.

5.2.12. Sum-of-Fractions

The SOF or "unity rule" is applied to the data used for the survey planning, and data evaluation and statistical tests for soil sample analyses since multiple radionuclide-specific measurements 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) becomes one (1). The basement structure DCGLs ($DCGL_B$), embedded pipe DCGLs ($DCGL_{EP}$) and penetration DCGLs ($DCGL_{PN}$) are directly analogous to the $DCGL_W$ as defined in MARSSIM. The use and application of the unity rule will be performed in accordance with section 4.3.3 of MARSSIM.

5.2.13. Dose from Groundwater

Based upon the results of groundwater monitoring performed on the Zion site since June 1998, when both Zion units were placed in a SAFSTOR condition through the current period of active decommissioning, the dose from existing residual radioactivity in groundwater is expected to be diminutive. However, if groundwater contamination is identified during decommissioning, the dose will be calculated using the Groundwater Exposure Factors presented in Chapter 6.

5.2.14. Demonstrating Compliance with Dose Criterion

The Base Case DCGLs for backfilled basements, surface soil, subsurface soil, buried piping, embedded piping and penetrations for each ROC are presented in Tables 5-3, 5-5, 5-6, 5-9, 5-11 and 5-13, 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 an AMCG. For all media, the dose from the residual radioactivity from each ROC (radionuclide *i*) can be expressed as shown in the following equation:

Equation 5-2

$$\mathrm{Dose}_{\mathrm{Media}} = \frac{\mathrm{Conc}_{\ \mathrm{Radionuclide}\ i}}{\mathrm{DCGL}_{\mathrm{Radionuclide}\ i}} \text{ x } 25 \text{ mrem/yr}$$

The final compliance dose will be calculated using Equation 5-3 after FSS has been demonstrated independently through FSS in all survey units. The results of the FSS performed for each FSS unit will be reviewed to determine the maximum dose from each of the four source terms (e.g., basement, soil, buried pipe and existing groundwater if applicable) using the Base Case DCGLs to derive the mean SOF of FSS systematic results plus the dose from any identified elevated areas. For all media except soils, areas of elevated activity are defined in this context as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. The SOF (when using the Operational DCGL) for a systematic or judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign Test and, the mean SOF (when using the Operational DCGL) for the survey unit does not exceed one. For all media except soils, if the SOF for a sample/measurement(s) exceeds one when using Base Case DCGLs, then remediation is required. For soils, the EMC as described in section 5.10.4 of this Chapter will apply. Detailed information pertaining to the calculation of the compliance dose is provided in TSD 17-004 (see LTP Chapter 6, section 6.17 for additional discussion).

Equation 5-3

Compliance Dose = (Max $\mathrm{SOF}_{\mathrm{BASEMENT}}$ + Max $\mathrm{SOF}_{\mathrm{SOIL}}$ + Max $\mathrm{SOF}_{\mathrm{BURIED\ PIPE}}$ + Max $\mathrm{SOF}_{\mathrm{GROUNDWATER}}$) x 25 mrem/yr

where:

Compliance Dose	=	must be less than or equal to 25 mrem/yr,
Max $\mathrm{SOF}_{\mathrm{BASEMENT}}$	=	Maximum SOF (mean of FSS systematic results plus the dose from any identified elevated areas) for backfilled Basements (including surface, embedded pipe, penetrations and fill [if required]),
Max $\mathrm{SOF}_{\mathrm{SOIL}}$	=	Maximum SOF (mean of FSS systematic results plus the dose from any identified elevated areas) for open land survey units,
Max $\mathrm{SOF}_{\mathrm{BURIED\ PIPE}}$	=	Maximum SOF (mean of FSS systematic results plus the dose from any identified elevated areas) from buried piping survey units,
Max $\mathrm{SOF}_{\mathrm{GROUNDWATER}}$	=	Maximum SOF from existing groundwater

The dose summation described in the equation above is conservative because the various source terms may not in fact be contiguous. For example, the maximum soil survey unit dose may be from an area that is not within the footprint of the Basement with the maximum dose. Another example is the buried pipe that delivers the greatest dose may not be under or contiguous with the soil survey unit with the maximum dose.

5.2.15. Soil Area Factors

Section 2.5.1.1 and section 5.5.2.4 of MARSSIM address the concern of small areas of elevated radioactivity in a survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas.

The investigation level for this comparison in soils is the $DCGL_{EMC}$, which is the $DCGL_w$ modified by an AF to account for the small area of the elevated radioactivity. At the ZSRP, $DCGL_{EMC}$ only applies to soils as all other media (structural surfaces, embedded pipe, buried pipe and penetrations) will be remediated at their applicable Base Case DCGL. 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 following equation defines the calculation of a $DCGL_{EMC}$.

Equation 5-4

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

AFs are calculated using RESRAD for each ROC and for source area sizes ranging from 1 m^2 up to the full source area of 64,500 m^2. The AFs for surface and subsurface soils were calculated in TSD 14-011 and are provided in Tables 5-16 and 5-17.

Table 5-16 Area Factors for Surface Soils

Area (m^2)	Area Factors for Radionuclides of Concern				
	Cs-137	Co-60	Cs-134	Ni-63	Sr-90
0.01	1.50E+03	1.23E+03	1.33E+03	3.31E+05	8.40E+04
0.03	4.98E+02	4.09E+02	4.42E+02	1.76E+05	3.03E+04
0.1	1.50E+02	1.23E+02	1.33E+02	6.92E+04	8.52E+03
0.3	4.98E+01	4.09E+01	4.42E+01	2.57E+04	2.88E+03
1	1.50E+01	1.23E+01	1.33E+01	8.06E+03	8.90E+02
3	6.46E+00	5.24E+00	5.73E+00	2.73E+03	3.13E+02
10	3.06E+00	2.47E+00	2.72E+00	8.23E+02	1.03E+02
30	2.10E+00	1.68E+00	1.86E+00	2.75E+02	4.02E+01
100	1.62E+00	1.29E+00	1.44E+00	8.26E+01	1.64E+01
300	1.46E+00	1.16E+00	1.30E+00	2.75E+01	6.14E+00
1,000	1.33E+00	1.08E+00	1.20E+00	8.26E+00	1.88E+00
3,000	1.26E+00	1.05E+00	1.16E+00	4.68E+00	1.73E+00
10,000	1.13E+00	1.02E+00	1.08E+00	1.86E+00	1.33E+00
64,500	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

Table 5-17 Area Factors for Subsurface Soils

Area (m^2)	Area Factors for Radionuclides of Concern				
	Cs-137	Co-60	Cs-134	Ni-63	Sr-90
0.01	2.04E+03	1.10E+03	1.52E+03	5.16E+05	1.45E+05
0.03	6.80E+02	3.65E+02	5.08E+02	1.98E+05	4.95E+04
0.1	2.04E+02	1.10E+02	1.52E+02	6.30E+04	1.50E+04
0.3	6.80E+01	3.65E+01	5.08E+01	2.14E+04	5.01E+03
1	2.04E+01	1.10E+01	1.52E+01	6.49E+03	1.50E+03
3	9.26E+00	4.91E+00	6.92E+00	2.17E+03	5.23E+02
10	4.48E+00	2.36E+00	3.35E+00	6.51E+02	1.64E+02
30	3.23E+00	1.70E+00	2.42E+00	2.18E+02	5.72E+01
100	2.59E+00	1.37E+00	1.95E+00	6.51E+01	1.76E+01
300	2.29E+00	1.26E+00	1.77E+00	2.17E+01	5.92E+00
1,000	1.90E+00	1.16E+00	1.56E+00	6.53E+00	1.78E+00
3,000	1.72E+00	1.13E+00	1.46E+00	4.12E+00	1.65E+00
10,000	1.32E+00	1.07E+00	1.22E+00	1.81E+00	1.30E+00
64,500	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00

In Class 1 open land FSS units, any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated. The $DCGL_{EMC}$ calculation for soils will use Base Case DCGLs ($DCGL_{SS}$ from Table 5-5 and/or $DCGL_{SB}$ from Table 5-6). Note that the soil FSS unit must pass the Sign Test using Operational DCGLs.

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 FSS planning.

5.3.1. Survey of Impacted Media

The characterization of the site commenced in November of 2011 with the characterization of the section of impacted land designed for construction of the future ISFSI facility, the "non-impacted" location where the Vertical Concrete Cask (VCC) Construction Area was to be located and the pathway for the new rail tracks. Characterization of the impacted and non-impacted open land survey units, as designated by the Zion *"Historical Site Assessment"* (HSA) (Reference 5-14), as well as the structural building basements that would remain and be subjected to FSS was accomplished in the following 23 months with the initial site characterization campaign concluding in October of 2013. During this period, 145,730 m^2 of surface soil was scanned, 1,037 surface soil samples were acquired and analyzed, 699 subsurface samples were acquired and analyzed, 282 static measurements were taken on surface soils using a Canberra *In Situ* Object Counting System (ISOCS), direct scans were performed over approximately 17,700 m^2 of basement surfaces below the 588 foot elevation, 109 concrete core samples were acquired from subsurface basement surfaces, and samples and measurements were taken inside building drain systems.

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 ZSRP at levels sufficiently below the established action levels. For characterization of soils, the interim screening DCGLs presented in NUREG-1757, Appendix H, Table H.1 and NUREG/CR-5512 Volume 3, *"Residual Radioactive Contamination from Decommissioning Parameter Analysis"*, (Reference 5-15), 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 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 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 anticipated 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. Floor monitors using large area gas-flow proportional detectors (typically with 584 cm^2) were used to scan the basement floor in the Turbine Building.

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. Zion*Solutions* TSD 11-004, "*Ludlum Model 44-10 Detector Sensitivity*" (Reference 5-16) examines the response and scan Minimum Detectable Concentration (MDC) of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. ISOCS measurements were taken in several open land survey units in lieu of scanning and soil sampling.

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. ZSRP ensured that 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 Zion*Solutions* ZS-LT-01, "*Quality Assurance Project Plan (for Characterization and FSS*" (QAPP) (Reference 5-17). 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 ZNPS is presented in Chapter 2.

5.3.4.1. Impacted and Non-Impacted Areas

The size of the entire ZNPS site is approximately 331 acres. Structures and open land classified as "impacted" by the operation of the facility are defined by a surrounding single-security fence line that has been designated as the "Radiologically Restricted Area". The area defined by the double-security fence line designated as the "Security Restricted Area" contains all structures and open land areas initially classified as Class 1.

In addition to the area within the "Radiologically Restricted Area", several additional areas have been deemed as "impacted". These include the site parking lot, the open land area directly north of the site and the area along Shiloh Boulevard designated as the West Training Area. The parking lot and the north field were designated as impacted as they represent the major path for material egress on and off of the site. The West Training Area was once the location of the Training Building which housed a Westinghouse Nuclear Training Reactor. The training reactor was decommissioned and the license terminated by the NRC in 1988 and the structure was demolished in 2003.

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, and operational radiological surveys as documented in the HSA was conducted. Based on the review, the open land areas included in the "owner-controlled" property outside of the footprint of the 87 acre, fence-enclosed "Radiologically Restricted Area", minus the additional impacted areas cited in the previous section, were deemed not impacted by licensed activities or materials.

From June to September 2013, sufficient survey coverage and an adequate number of samples were obtained in the areas designated as non-impacted to serve as the basis for this classification. Cs-137 was the only radionuclide positively identified that could potentially be classified as plant-derived. However, the concentrations observed are well within the range of activity defined as background due to global fallout. The summary of these survey results is presented in LTP Chapter 2, section 2.3.4.

5.3.4.3. Adequacy of the Characterization

The site characterization of ZNPS 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 water, 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, ZSRP considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected.

The initial soil (i.e., open land) survey units and survey unit classifications that will be used for the FSS of open land at ZNPS are presented in LTP Chapter 2, section 2.1.6 and Table 2-4. These classifications may be changed to a more restrictive classification as decommissioning progresses.

Currently accessible structures that will remain and be subjected to FSS 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 588 foot elevation that were developed for characterization and decommissioning planning purposes are presented in LTP Chapter 2, section 2.1.6 and Table 2-3. However, the FSS that will be applied to structures below 588 foot uses a different design criterion that is not directly driven by the preliminary classifications selected for characterization. Therefore, the preliminary survey unit boundaries and classifications will not apply to the FSS of structures (basements) below 588 foot elevation. See section 5.5 for the FSS design criteria for basement surface survey unit boundaries and the approach to determine survey area coverage.

5.3.4.4. Inaccessible or Not Readily Accessible Areas

ZSRP has characterized end-state concrete structures to assess the current residual radioactivity concentration, radionuclide mixture and to ensure the correct classification of each FSS unit. ZSRP has also characterized surface and subsurface soils surrounding ZNPS.

The following are areas at Zion where additional or "continuing" characterization will occur. These areas are;

- The underlying concrete of the SFP/Transfer Canal below the 588 foot elevation after the steel liner has been removed. Continuing characterization will consist of scanning of the exposed concrete surfaces and the acquisition of concrete core sample(s) at the location of highest activity. The number and location of the additional concrete core sample(s) will be determined by DQO during survey design.
- The concrete walls and floor of the Under-Vessel areas in Unit 1 and Unit 2 Containments. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The floors and walls of the Hold-Up Tank (HUT) cubicle. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The floor of the Auxiliary Building 542 foot elevation Pipe Tunnel floors. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The floor and lower walls of the 542 foot elevation of the Auxiliary Building to augment the existing characterization data. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The subsurface soils in the "keyways" between the Containment Buildings and the Turbine Building once subsurface utilities have been removed and the removal of subsurface structures in this area create access (e.g., Waste Annex Building). Continuing characterization will consist of the scanning of soils exposed by the demolition and building removal, the acquisition of soil sample(s) of the exposed soil and the acquisition of additional subsurface soil samples using test pits or soil

borings. The number and location of the additional subsurface soil samples will be determined by DQO during survey design.

- The soils under the basement concrete of the Containment Buildings, the Auxiliary Building and the SFP/Transfer Canal once commodity removal and building demolition have progressed to a point where access can be achieved. Continuing characterization will consist of soil borings at the nearest locations along the foundation walls that can be feasibly accessed, angled soil bores to access the soils under the concrete, and deep cores from building floors, but not entirely through the foundation, at bias locations to assess migration potential from building interiors to soils under basement concrete. The number and location of the additional subsurface soil samples will be determined by DQO during survey design. Additional investigations and sampling will be performed in accordance with a sample plan if activity is positively identified.
- When the interior surfaces become accessible, several potentially contaminated embedded and buried pipe systems that will be abandoned in place. Continuing characterization will consist of direct measurements on pipe openings and the acquisition of sediment and/or debris samples (if available) for analysis. The number and location of the additional direct measurements and sediment samples will be determined by DQO during survey design.
- The Containment basements after concrete removal. Continuing characterization of the steel liner will consist of beta gamma scans and swipe samples. The number and location of the beta scans and swipe samples will be determined by DQO during survey design.

All exposed surface soil at ZNPS has been adequately characterized and additional characterization of surface soil is not anticipated during continuing characterization. Radiological Assessment (RA) surveys will be performed in currently inaccessible soil areas that are exposed after removal of asphalt or concrete roadways and parking lots, rail lines, or building foundation pads (slab on grade).

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

5.4. Decommissioning Support Surveys

5.4.1. Radiological Assessment (RA)

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

The RA of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-26. In addition to direct and scan radiation measurements, the RA will include the collection of samples of soil, sediment and surface residue for laboratory analysis, as appropriate.

5.4.2. Remedial Action Support (In-Process) Surveys

Remedial Action Support Surveys (RASS) are performed while remediation is being conducted, and guides cleanup in a real-time mode. RASS are conducted to: 1) guide remediation activities; 2) determine when an area or survey unit has been adequately prepared for the FSS; and, 3) provide updated estimates of the parameters (e.g., variability, and in some instances, a verification of the radionuclide mixture) to be used for planning the FSS.

RASS of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-26. 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 undergo remediation will be performed using surface contamination monitors, augmented with sampling for removable surface contamination. RASS surveys may also be performed using the ISOCS, especially where personnel safety is of concern. Examples include: overhead ceilings, upper walls and cavity locations where the use of scaffolding and areal lifts is impractical.

5.4.3. Instrumentation for RA and RASS

Table 5-26 shows typical field instruments that will be used for performing FSS. The same or similar instruments will be used during the performance of the RA and RASS. The typical MDCs for field instruments used for scanning are provided in Table 5-27 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. ZSRP will ensure that the quality programs of any contracted off-site vendor laboratory that is used for the receipt, preparation and analysis of RA and RASS samples will provide the same level of quality as the on-site laboratory under the QAPP.

5.4.4. Field Screening Methods for RA and RASS

A gamma walk-over survey will be performed over the surface area, 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 Operational DCGLs for surface soil ($OpDCGL_{SS}$ from Table 5-7) is achieved. Locations of elevated count rate will be identified for additional scanning and/or the collection of biased soil samples to determine if the elevated count rate indicates the presence of soil concentration in excess of the $OpDCGL_{SS}$. The information obtained during the RA and RASS (scan results and the analytical data from any associated soil samples) will be used to determine if the remaining exposed soils:

- contain radioactivity concentrations above the $OpDCGL_{SS}$ and may require further excavation;

- contain radioactivity concentrations that are less than the $OpDCGL_{SS}$, but require removal in order to access additional soil/debris that potentially contains radioactivity concentrations above the applicable DCGL; or,
- contain radioactivity concentrations that are less than the $OpDCGL_{SS}$, and not requiring removal.

Zion*Solutions* TSD 11-004 examines the response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. If the survey instrument scan MDC is less than the $OpDCGL_{SS}$, then scanning will be the primary method for determining if the area is suitable for FSS. Once the scan surveys and the laboratory data obtained from any biased soil samples that may have been collected indicate residual concentrations are less than the $OpDCGL_{SS}$, then the area will be considered suitable for FSS.

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

The Canberra ISOCS system may be used in lieu of the walk over survey using a NaI detector provided the scan sensitivity meets the field screening requirements.

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

All remaining structural surfaces will be surveyed to meet the criteria for open air demolition specified in Zion*Solutions* TSD 10-002, *"Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"* (Reference 5-18). These criterions are the acceptable removable contamination and contact exposure rate levels that are allowable for open air demolition. A contamination verification survey (CVS) will be performed to identify areas requiring remediation to meet the open air demolition limits. A CVS will be performed within any structure that contains, or previously contained, radiological controlled areas. The CVS will be performed using hand-held beta-gamma instrumentation as presented in Table 5-26 in typical scanning and measurement modes.

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

For structural surfaces below the 588 foot elevation that will remain and be subject to a FSS (primarily any basement floor and outer walls), additional remediation will be performed to ensure that any individual ISOCS measurement will not exceed the Operational $DCGL_B$ from Table 5-4 during FSS. Any areas identified that have the potential to exceed the Operational $DCGL_B$ by ISOCS measurement during the performance of CVS in these areas will be earmarked for remediation. Any areas of elevated activity that could potentially approach the Operational $DCGL_B$ will be identified as a location for a judgmental ISOCS measurement during FSS.

5.4.6. Post-Demolition Survey

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

$DCGL_B$ from Table 5-4 during FSS. The survey will be performed using hand-held beta-gamma instrumentation as presented in Table 5-21 in typical scanning and measurement modes.

5.5. Final Status Survey of Basement Structures

Basement structures are defined as basement surfaces (concrete and steel liner), embedded pipe, and penetrations. As described in section 5.4.5, all remaining floor and wall concrete surfaces will be remediated to levels below the Operational $DCGL_B$ as measured by ISOCS during FSS. After remediation, a FSS will be conducted to demonstrate that the residual radioactivity in building basements corresponds to a dose below the 25 mrem/yr criteria.

5.5.1. Instruments Selected for Performing FSS of Basement Surfaces

The Canberra ISOCS has been selected as the primary instrument that will be used to perform FSS of basement surfaces. Direct beta measurements taken on the concrete surface will not provide the data necessary to determine the residual radioactivity at depth in concrete and therefore, would have to be augmented with core sampling. The ISOCS was selected as the instrument of choice to perform FSS of basement surfaces for the following reasons:

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

Additional concrete core sampling will be taken as continuing characterization in the SFP/Transfer Canal, HUT cubicles and Auxiliary Building 542 foot elevation Pipe Tunnels to confirm the depth distribution of activity in concrete in support of ISOCS geometry assumptions and sensitivity analysis. The additional concrete cores samples will be evaluated to ensure that the ISOCS geometry used for efficiency calculations is sufficiently conservative.

5.5.2. Basement Surface FSS Units

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

The survey units designated for structures below 588 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 FSS units will be comprised of the combined wall and floor surfaces of each remaining building basement, i.e., Auxiliary Building, Unit 1 Containment, Unit 2 Containment, Turbine Building, Crib House/Forebay, WWTF and remnants of the SFP/Fuel Transfer Canal. The Containment Buildings will contain two surface survey units, the walls and floors of the exposed steel liner and the Under-Vessel area where concrete will remain (see section 5.5.2.1).

The activity in the Circulating Water Intake Pipes, Circulating Water Discharge Tunnels, Circulating Water Discharge Pipes, and Buttress Pits/Tendon Tunnels is included with Turbine Building through the DCGL calculation. The activity in the Circulating Water Intake Pipe is also included with the Crib House/Forebay through the DCGL calculation. See LTP Rev 1 Chapter 6, section 6.6.8 for discussion of the DCGL calculations. The Circulating Water Discharge Tunnels will be addressed as a separate survey unit within the Turbine Building. Access to the Circulating Water Intake Pipes, Discharge Pipes, and Buttress Pits/Tendon Tunnels are very limited and therefore, these areas will be surveyed as biased areas using judgmental samples. The area-weighted mean of the judgmental sample population in these survey units will be added to the systematic mean of the Turbine Building and Crib House/Forebay FSS unit results. The entire surface areas will be included in the area-weighted average calculation (see section 5.5.6.1).

Contamination potential was the prime consideration for grouping FSS units. Contiguous surface areas with the same contamination potential will minimize uncertainty in the estimate of the mean concentration and ensure the appropriate level of areal coverage. Characterization data, radiological surveys performed to support commodity removal and surveys performed to support structural remediation for open air demolition have and will continue to be used to verify that the contamination potential within each FSS unit is reasonably uniform throughout all walls and floor surfaces. The FSS of Class 1 survey units include ISOCS measurements over 100% of wall and floors surfaces, eliminating uncertainty in the assumptions regarding uniformity of the underlying population.

5.5.2.1. Classification and Areal Coverage for FSS of Basement Surfaces

The primary consideration for determining FSS classification and areal coverage in basement surfaces is the potential for an individual measurement in a FSS unit to exceed the dose criterion. This is evaluated by the potential for an individual ISOCS measurement to exceed the Operational $DCGL_B$.

As discussed in section 5.4.5, extensive surface scan surveys, in some cases 100% of the surface area, will be performed during CVS. In addition to the CVS, information on contamination potential is also provided by characterization surveys performed to date and radiological surveys to be performed to support commodity removal. All of this information has been and will continue to be used to validate survey unit classification.

Above the 565 foot elevation, all concrete will be removed from each of the Containment basements to expose the steel liner. As a consequence, the entire source term above the 565 foot elevation will be removed as well. After all of the concrete above the 565 foot elevation is removed, it is anticipated that the residual radioactivity remaining in the Containment basement surfaces (comprised of steel liner only) above the 565 foot elevation will correspond to a small fraction of the dose criterion.

The FSS units for the Auxiliary Building 542 foot elevation floor and walls, the concrete (and any exposed liner) in both Containment Under-Vessel areas and the remaining SFP/Transfer Canal structural surfaces are designated as Class 1 and the FSS areal coverage will be 100% which is consistent with MARSSIM, Table 5.9. For the remaining basement surface FSS units (the Containment basements including the 565 foot elevation liner and above, the Turbine Building basement, the Crib House/Forebay, Circulating Water Discharge Tunnels and the WWTF), the criteria for selecting reasonable and risk-informed ISOCS areal coverage will be based on the MARSSIM, Table 5.9 scan survey guidance for Class 2 and Class 3 structures. The criteria for selecting areal coverage is based on a graded approach consistent with the guidance for scan surveys for FSS in MARSSIM section 2.2.

5.5.2.1.1. FSS Units for the Containment Basements

In both Containment basements, all concrete above the 565 foot elevation will be removed to expose the steel liner. As a consequence, the entire source term in the Containments above the 565 foot elevation will be removed as well. Below the 565 foot elevation, the concrete in the Under-Vessel areas of both Containments will remain, primarily due to safety concerns associated with removing the concrete. Previous characterization surveys indicate that the concrete in the Under-Vessel areas are volumetrically contaminated due to activation and, that the concrete surfaces are contaminated at concentrations exceeding the Operational $DCGL_B$ values in Table 5-4. Consequently, the concrete in both Under-Vessel areas have been designated as Class 1. As discussed in section 5.3.4.4, additional samples will be taken in the Under-Vessel areas as part of continuing characterization.

Due to the fact that all of the contaminated concrete above the 565 foot elevation steel liner will be completely removed in each Containment, the typical approach of classifying the Containment(s) above the 565 foot elevation 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 the Containments above the 565 foot elevation, where all of the media will be removed, not simply the contaminated portion. The only residual radioactivity that could remain in the end state Containments above the 565 foot elevation is contaminated concrete dust remaining on the steel liner from the demolition process. Additionally, prior to turning over the Containment buildings for FSS, the building basements will be decontaminated to remove loose surface contamination, minimizing the potential for remaining contaminated concrete dust. Once remediation is complete, the fraction of the Operational $DCGL_B$ on the Containment basement(s) liner walls and floor at and above the 565 foot elevation, from any remaining source term is expected to be small. After all concrete is removed, continuing characterization will be performed on the steel liner floor and walls above 565 foot elevation to confirm the assumption that minimal residual radioactivity will remain. All continuing characterization sample plans will be provided to the NRC for information and results will be provided to NRC for evaluation.

The FSS units for the 565 foot elevation steel liner floor and walls above the 565 foot elevation floor in the Containment basements are designated as 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 concentrations of residual radioactivity in the Containment basement(s) above the 565 foot elevation to exceed the Operational $DCGL_B$ as presented in Table 5-4 is very low due to complete removal of the contaminated media (i.e., concrete). As a conservative measure, these FSS units will be subjected to an areal coverage commensurate with the Class 2 scan coverage as presented in Table 5-24. Sufficient ISOCS measurements will be taken to ensure that at least 10% of the surface area in each survey unit is subjected to FSS. 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 characterization, the results of CVS or professional judgment.

5.5.2.1.2. FSS Units for Turbine Building Basement, Crib House/Forebay, WWTF and Circulating Water Discharge Tunnels

Extensive characterization has been performed in the 560 foot and 570 foot elevations of the Turbine Building and in the Crib House. A series of concrete core samples were taken in all three locations. In addition, the entire floor of the Turbine Building 560 foot elevation was scanned using a Ludlum Model 43-37 floor monitor. The maximum radiological concentration observed in the analysis of all the concrete core samples taken in the Turbine Building basement or Crib House was 46.70 pCi/g. The scan of the Turbine Building 560 foot elevation resulted in a maximum observed count rate of 3,922 cpm/100 cm^2.

At the time of LTP submittal, the Forebay and the Circulating Water Intake Piping and Discharge Tunnels were completely underwater and not accessible. In addition, the portion of the WWTF located below the 588 foot elevation that will remain was also not accessible. Process knowledge and the results of environmental monitoring of radiological conditions at effluent outfalls in the past indicates that the probability of residual radioactivity in these FSS units exceeding 50% of the Operational $DCGL_B$ as presented in Table 5-4 is very low.

The FSS units for the basements of the Turbine Building, the Crib House/Forebay, WWTF and the Circulating Water Discharge Tunnels are designated as Class 3 as defined in MARSSIM, section 2.2 in that the FSS units are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGLs, based on site operating history and previous radiation surveys. These four FSS units will be subjected to an areal coverage commensurate with the guidance pertaining to Class 3 scan coverage as presented in MARSSIM, Table 5.9, which states that the scan coverage guidance is "judgmental". In this context, judgmental areal coverage will be defined as sufficient ISOCS measurements to ensure that at least 1% of the surface area in each survey unit is subjected to FSS.

5.5.2.2. Sample Size Determination for FSS of Basement Surfaces

Based on the contamination potential of each FSS unit that was determined in the previous section, along with the corresponding areal coverage, the number of ISOCS measurements required in each FSS unit can be calculated as the quotient of the ISOCS FOV divided into the surface area required for areal coverage. Table 5-18 presents the FSS units, the classification based on contamination potential, the surface area to be surveyed and the minimum number of ISOCS measurements that will be required based on a measurement FOV of 28 m^2.

To ensure that the number of ISOCS measurements based on the necessary areal coverage in a basement surface FSS unit was sufficient to satisfy a statistically based sample design, a calculation was performed to determine sample size using the process described in section 5.6.4.1. This calculation was applied to the Class 1, Class 2 and Class 3 basement surface FSS units.Class 1Class 2 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. For Class 1 FSS units where 100% areal coverage by ISOCS will be performed, the number of measurements are expected to exceed that required by the statistical test but the process to determine sample size is followed to confirm that this is the case.

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 Operational $DCGL_B$. The Lower Bound of the Gray Region (LBGR) was set at the expected fraction of the Operational $DCGL_B$. The expected fraction of the Operational $DCGL_B$ in the Class 1 and Class 2 FSS units was set at 50% and the expected fraction of the Operational $DCGL_B$ in the Class 3 FSS units was set at 1%. The standard deviation of the concrete core samples taken in the Turbine Building was used for sigma (σ) in the FSS units for the Turbine Building, Crib House/Forebay and WWTF. For the Class 2 FSS units in the Containment basements, the entire concrete source term above the 565 foot elevation will be removed. Consequently, the results of the concrete core samples taken in the Containments are also not representative of the conditions at the time of FSS. As reasonable value for sigma (σ) cannot be determined 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. For the Class 1 survey unit in the Auxiliary Building, the standard deviation of the concrete cores analyzed for the full initial suite were used for sigma (σ). For the concrete in the Containment Under-Vessel area the standard deviation of the cores collected during characterization were used for sigma (σ).

The relative shift (Δ/σ) was calculated as discussed in section 5.6.4.1.6 of this Chapter. The Δ/σ calculations will be confirmed and documented as part of the FSS design process including evaluation of any additional data from continuing characterization. With the exception of the Under-Vessel area, the relative shift (Δ/σ) was greater than three in all cases. The relative shift (Δ/σ) for the Under-Vessel area was two. Consequently, a value of three was used as the adjusted relative shift (Δ/σ) for all FSS units other than the Under-Vessel area where a value of two was used. From 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 for a Δ/σ value of three and 15 for a Δ/σ value of 2. Consequently, the number of ISOCS measurements in several basement surface FSS units was adjusted to meet the larger sample size. Table 5-19 presents the basement surface FSS units and the adjusted number of ISOCS measurements that will be taken in each for FSS.

Table 5-18 Number of ISOCS Measurements per FSS Unit based on Areal Coverage

FSS Unit	Classification	Area (m^2)	Minimum Areal Coverage (% of Area)	Minimum # of ISOCS Measurements (FOV-28 m^2)
Aux Bldg. 542 foot Floor and Walls	Class 1	6,503	100%	233
Unit 1 Containment Basement above 565 foot elevation	Class 2	2,465	10%	9
Unit 1 CTMT Under-Vessel Area	Class 1	294	100%	11
Unit 1 Containment Basement above 565 foot elevation	Class 2	2,465	10%	9
Unit 2 CTMT Under-Vessel Area	Class 1	294	100%	11
SFP/Transfer Canal	Class 1	723	100%	26
Turbine Building Basement	Class 3	14,864	1%	6
Circulating Water Discharge Tunnels	Class 3	4,868	1%	2
Crib House/Forebay	Class 3	13,843	1%	5
WWTF	Class 3	1,124	1%	1

As previously noted, the required areal coverage for a Class 1 basement survey unit is 100%. Sufficient measurements will be taken in the Class 1 FSS unit to ensure that 100% of the surface area is surveyed (ISOCS FOV will be overlapped to ensure that there are no un-surveyed corners and gaps). 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 FSS unit classification (100% areal coverage for a Class 1 FSS unit or 10% areal coverage for a Class 2 FSS unit) is achieved.

Table 5-19 Adjusted Minimum Number of ISOCS Measurements per FSS Unit

FSS 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)
Aux Bldg. 542 foot Floor & Walls	Class 1	6,503	407(2)	6,503	100%
Unit 1 CTMT above 565 foot elevation	Class 2	247	35(1)	980	40%
Unit 1 CTMT Under-Vessel Area	Class 1	294	19(2)	294	100%
Unit 2 CTMT above 565 foot elevation	Class 2	247	35(1)	980	40%
Unit 2 CTMT Under-Vessel Area	Class 1	294	19(2)	294	100%
SFP/Transfer Canal	Class 1	723	45(2)	723	100%
Turbine Building Basement	Class 3	149	14	392	3%
Circulating Water Discharge Tunnels	Class 3	49	14	392	8%
Crib House/Forebay	Class 3	138	14	392	3%
WWTF	Class 3	11	14	392	35%

(1) Number of ISOCS adjusted to meet MARSSIM recommended survey size for a Class 2 structure (1,000 m^2).
(2) Adjusted to ensure number of measurements that will be taken in Class 1 FSS units will ensure 100% areal coverage, including overlap to ensure that there are no un-surveyed corners and gaps (FOV based on a 4m x 4m grid system).

In the Class 2 basement surface FSS units (where less than 100% ISOCS coverage is required), measurement spacing will be determined in accordance with section 5.6.4.5.2 of this Chapter. The number of measurements will also be increased in survey units that exceed 1,000 m^2 to correspond with the MARSSIM recommended survey size density for a Class 2 structure (measurements/1,000 m^2). If the grid spacing allows, 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 FSS 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. 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 the Class 3 basement surface FSS units, each measurement location will be randomly selected using a random number generator. 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. 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.

5.5.3. Survey Approach for FSS of Basement Surfaces

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

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

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 FSS coverage as presented in Table 5-18 is achieved. In most cases, the measurement will be acquired using the ISOCS with a geometry that evaluates residual activity over the activity depth.

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

5.5.4. Basement Surface FSS Data Assessment

After a sufficient number of ISOCS measurements are taken in a FSS unit in accordance with the areal coverage requirements specified in Table 5-19, 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 that is positively detected by ISOCS) 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 HTD ROC using the surrogate approach specified in section 5.2.11. The surrogate ratios that will be used are presented in Table 5-15. A sum of fractions (SOF) calculation will be performed for each measurement by dividing the reported concentration of each ROC by the Operational $DCGL_B$ for each ROC to calculate an individual ROC fraction. 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 against the dose criterion. The SOF for each measurement will be used as the sum value for the Sign Test. If the Sign Test demonstrates that the mean activity for each ROC is less than the Operational $DCGL_B$ at a Type 1 decision error of 0.05, then the mean of all the total SOFs for each measurement in a given survey unit is calculated. If the Sign Test fails, or if the mean of the total SOFs in a basement exceeds one (using Operational DCGLs), then the survey unit will fail FSS. If a survey unit fails FSS, then the survey unit may be reclassified, additional remediation will be performed and the FSS performed again.

Once the survey data set passes the Sign Test (using Operational DCGLs), the mean radionuclide activity (pCi/m^2) for each ROC from systematic measurements along with any identified elevated areas will be used with the Base Case DCGLs to perform a SOF calculation for each surface FSS unit in a basement in accordance with the following equation. The dose from residual radioactivity assigned to the FSS unit is the SOF_B multiplied by 25 mrem/yr.

Equation 5-5

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

where:

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

5.5.5. FSS of Embedded Piping and Penetrations

The end state will include embedded piping and penetrations. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of penetrations and embedded piping to remain is provided in TSD 14-016. Embedded pipe and penetrations have separate Operational DCGLs as listed in Tables 5-12 and 5-14. However, the survey methods are the same for both. The survey units for embedded pipe and penetrations are presented in Table 5-20.

Table 5-20 Embedded Pipe and Penetration Survey Units

Basement FSS Unit	Embedded Pipe	Penetrations
Auxiliary Building Basement	• Basement Floor Drains (542 ft. elevation)	• Auxiliary Building Penetrations
Containment Basement	• Unit 1 and Unit 2 In-Core Sump Drains (541 ft. elevation) • Unit 1 and Unit 2 Tendon Tunnel Drains	• Containment Penetrations
SFP/Transfer Canal	N/A	• SFP/Transfer Canal Penetrations
Turbine Building Basement	• Unit 1 and Unit 2 Basement Floor Drains (560 ft. elevation) • Unit 1 and Unit 2 Steam Tunnel Floor Drains (570 ft. elevation) • Unit 1 and Unit 2 Tendon Tunnel Drains (1)	• Turbine Penetrations

(1) Buttress Pits/Tendon Tunnels hydraulically connected to Steam Tunnel/Turbine Building so include with Turbine Building as well as Containment

The residual radioactivity remaining in each section of embedded piping/penetration applicable to each FSS unit will be assessed and quantified by direct survey. Shallow penetrations or short lengths of embedded pipe that are directly accessible will be surveyed using hand-held portable detectors, such as a gas-flow proportional or scintillation detector. Lengths of embedded pipe or penetrations that cannot be directly accessed by hand-held portable detectors will be surveyed using applicable sized NaI or Cesium Iodide (CsI) detectors that will be inserted and transported through the pipe using flexible fiber-composite rods or attached to a flexible video camera/fiber-optics cable. The specific types of instruments that may be used for both types of scenarios are presented in section 5.8 and Table 5-26.

The interior of embedded pipe or penetration sections that cannot be accessed directly will be inspected prior to survey using a miniature video camera designed to assess the physical condition of the pipe/sleeve interior surfaces. The miniature camera with supporting lighting components as well as the subsequent detectors that will be used to survey the pipe/sleeve interior surfaces will be maneuvered through the pipe/sleeve by the manipulation of fiber-composite rods which will be manually pushed or pulled to provide locomotion. The detectors will be deployed into the actual pipe/sleeve and a timed measurement acquired at a specified distance traversed into the pipe. This distance will be determined as a DQO based on the contamination potential in the pipe/sleeve. As an example, based upon a conservative “area of detection” for the detectors used, a measurement interval of one measurement for each foot of pipe will conservatively provide 100% areal coverage of all accessible pipe/sleeve interior surfaces.

The detector output will represent the gamma activity in gross cpm. This gamma measurement value in cpm will then be converted to dpm using an efficiency factor based on the calibration source. The total activity in dpm will adjusted for the assumed total effective surface area commensurate with the pipe/penetration diameter, resulting in measurement results in units of dpm/100 cm^2. This measurement result will then represent a commensurate and conservative gamma surface activity.

The gamma surface activity for each FSS measurement is then converted to a gamma measurement result (in units of pCi/m^2) for each gamma ROC based on the mixture applicable to the pipe/sleeve surveyed. HTD ROC are inferred to the applicable gamma radionuclide concentration to derive a concentration for each ROC for each measurement taken. The measurement concentration for each ROC is then divided by the applicable Operational DCGL to produce a dose fraction for each ROC. The individual ROC dose fractions are then summed to produce a SOF for the measurement. There is no EMC applicable to embedded pipe or penetrations. Consequently, a measurement SOF that exceeds one would require investigation. For embedded pipe and penetrations, areas of elevated activity are defined as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. The SOF (based on the Operational DCGL) for a systematic measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign Test (using the Operational DCGL) and, the mean SOF (using the Operational DCGL) for the survey unit does not exceed one. If the SOF for a sample/measurement(s) exceeds one when using Base Case DCGLs, then remediation is required.

Once the survey data set passes the Sign Test (using the Operational DCGL), the mean radionuclide activity (pCi/m^2) for each ROC from systematic measurements along with any identified elevated areas will be used with the Base Case DCGLs to perform a SOF calculation for the embedded pipe or penetration FSS unit in the basement accordance with the following equation. The dose from residual radioactivity assigned to the FSS unit is the SOF multiplied by 25 mrem/yr.

Equation 5-6

$$SOF_{EP/PN} = \sum_{i=1}^{n} \frac{Mean\ Conc_{EP/PN_{ROC_i}}}{BcDCGL_{EP/PN_{ROC_i}}} + \frac{\left(Elev\ Conc_{EP/PN_{ROC_i}} - Mean\ Conc_{EP/PN_{ROC_i}}\right)}{\left[BcDCGL_{EP/PN_{ROC_i}} \times \left(\frac{SA_{SU}}{SA_{Elev}}\right)\right]}$$

where:

$SOF_{EP/PN}$	=	SOF for embedded pipe or penetration survey unit within a Basement using Base Case DCGLs
$Mean\ Conc_{EP/PN\ ROC_i}$	=	Mean concentration for the systematic measurements taken during the FSS of embedded pipe or penetrations in survey unit for each ROC_i
$BcDCGL_{EP/PN\ ROC_i}$	=	Base Case DCGL for structural surfaces ($DCGL_B$) for each ROC_i
$Elev\ Conc_{EP/PN\ ROC_i}$	=	Concentration for ROC_i in any identified elevated area
SA_{Elev}	=	surface area of the elevated area
SA_{SU}	=	surface area of FSS unit

The total embedded pipe located within a basement will be treated as a separate FSS unit. FSS units for penetrations are grouped by basement. In situations where there are multiple survey units in a basement (e.g., surface, embedded pipe, penetration), the sum of the dose from all survey units must be less than 25 mrem/yr. As such, the FSS results for embedded pipe and penetration survey units will be part of the summation of the compliance dose calculated for each building basement in which they are located (see LTP Chapter 6, section 6.17).

Embedded pipe survey units have a relatively small surface area leading to Operational DCGLs that are higher than the wall/floor Operational DCGL. This is due to the total internal surface area of the embedded pipe survey unit in a given basement being less than the total wall/floor surface area of the basement containing them. To eliminate the potential for activity levels in embedded pipe that could lead to releases greater than surrounding walls and floors, the following remediation and grouting action levels will be applied to measurements of surface activity in embedded pipe.

- If maximum activity exceeds the Base Case $DCGL_{EP}$ from Table 5-11 (SOF >1), then remediation will be performed.
- If the maximum activity in an embedded pipe exceeds the surface Operational $DCGL_{B}$ from Table 5-4 (SOF>1) in the building that contains it, but is below the Base Case $DCGL_{EP}$ from Table 5-12, then the embedded pipe will be remediated or grouted.
- If an embedded pipe is remediated and the maximum activity continues to exceed the surface Operational $DCGL_{B}$ from Table 5-4 (SOF>1), but is less than the Operational $DCGL_{EP}$, then the embedded pipe will be grouted.
- If the maximum activity is below the surface Operational $DCGL_{B}$ from Table 5-4, then grouting of the pipe will not be required.

As with embedded pipe, penetration survey units also have total surface areas that are less than the area of the wall/floor surface survey unit that the penetrations interface. To eliminate the potential for activity levels in penetrations that could lead to releases greater than the adjacent basement walls and floors, the following remediation and grouting action levels will be applied to measurements of surface activity in penetrations.

- If maximum activity exceeds the Base Case $DCGL_{PN}$ from Table 5-13 (SOF >1), then remediation will be performed.
- If the maximum activity in a penetration exceeds the most limiting Operational $DCGL_{B}$ from Table 5-4 of the two basements where a penetrations interface (SOF>1), but is below the Base Case $DCGL_{PN}$ from Table 5-13, then the penetration will be remediated or grouted.
- If a penetration is remediated and the maximum activity continues to exceed the most limiting Operational $DCGL_{B}$ from Table 5-4 of the two basements where a penetrations interface (SOF>1), but is less than the Operational $DCGL_{PN}$, then the penetration will be grouted.
- If the maximum activity is below the surface Operational $DCGL_{B}$ from Table 5-4, then grouting of the penetration will not be required.

An alternate drilling spoils scenario was evaluated to determine the maximum hypothetical dose from drilling into penetrations or embedded pipe assuming activity is present at the $DCGL_{PN}$ concentrations. The alternate scenario is categorized as “less likely but plausible” in accordance with NUREG-1757, Table 5.1 which states that assessment of a “less likely but plausible scenario” is “not analyzed for compliance, but is used to risk-inform the decision”. The alternate scenario drilling spoils dose was less than 25 mrem/yr for all penetrations and embedded pipe with the exception of the Steam Tunnel Floor Drains, which resulted in a dose of 71.16 mrem/yr (see LTP Chapter 6 section 6.7). Although the alternate scenario is not a compliance scenario, as a conservative measure the remediation levels for the

Steam Tunnel Floor Drains will be reduced by a factor of 2.89, i.e., from the $DCGL_{EP}$ to $DCGL_{EP} \div 2.89$.

5.5.6. Summation of Dose for Basement Structures

The BFM source term for a given basement structure includes the contributions from basement surfaces, (concrete and steel liner for Containment), embedded pipe and penetrations that are contained in, or interface with, the basement. Each dose component (surface, embedded pipe, penetrations) has a unique DCGL. Concrete fill is another dose component applicable to any basement where clean concrete debris is used as fill. This is discussed further in LTP Chapter 6, section 6.16. The total dose attributed to the use of concrete fill for each basement, including all ROC, is presented in Table 5-21, which is reproduced from Table 6-51 from LTP Chapter 6, section 6.16. The dose values in Table 5-21 will be added to any basement where concrete fill is used regardless of the volume of concrete fill used.

Table 5-21 Dose Assigned to Clean Concrete Fill

Basement Structure	Dose (mrem/yr)
Auxiliary Building	0.99
Containment	1.77
SFP/Transfer Canal	0.15
Turbine Building	1.58
Crib House/Forebay	1.57
WWTF	6.40

The *a priori* dose from clean concrete fill in Table 5-21 is currently based on a maximum allowable MDC of 5,000 dpm/100cm^2, which is a conservative assumption. This is solely a bounding value and not indicative of the actual MDC values experienced when URS were performed on the concrete, which were significantly lower. After all Unrestricted Release Surveys (URS) have been completed on the remainder of the concrete that will be reused as clean fill, the dose from fill in Table 5-21 will be recalculated based on the actual maximum MDC observed during the performance of the URS.

After the FSS of all dose components in a given basement is complete and all dose component survey units pass the Sign Test, the SOF for each dose component is calculated using Equations 5-5 or 5-6 as applicable. The SOF for concrete fill is calculated by dividing the basement-specific assigned dose in Table 5-21 by 25 mrem/yr. The total dose for the Basement is then calculated by summing the SOF from all dose components using Equation 5-7 and multiplying by 25 mrem/yr.

Equation 5-7:

$$SOF_{BASEMENT} = SOF_B + SOF_{EP} + SOF_{PN} + SOF_{CF}$$

where:

$SOF_{BASEMENT}$	=	SOF (mean of FSS systematic results plus the dose from any identified elevated areas) for backfilled Basements
SOF_B	=	SOF for structural survey unit(s) within the Basement (mean of FSS systematic results plus the dose from any identified elevated areas)

SOF_{EP}	=	SOF for embedded pipe survey unit(s) within the Basement (mean of FSS systematic results plus the dose from any identified elevated areas)
SOF_{PN}	=	SOF for penetration survey unit(s) within the Basement (mean of FSS systematic results plus the dose from any identified elevated areas)
SOF_{CF}	=	SOF for clean concrete fill (if applicable) based on maximum MDC during URS

5.5.6.1. Basement Surface Dose Calculation Including Multiple Survey Units

The calculation of dose from specific building surfaces (Auxiliary Building, Containments, Turbine Building and Crib House/Forebay) is the sum of the contributions from two or more surface survey units within, or connected to, the given basement. In addition, the source term from biased judgmental FSS results from the surface of the Circulating Water Intake Pipe are added to the Turbine Building and the Crib/House Forebay. The process for calculating the combined basement surface dose from each of the survey units and the calculation method is provided below. Table 5-22 lists the surface survey units that contribute to each basement.

Table 5-22 Surface Survey Units Contributing to Each Basement

Basement	Surface Survey Unit 1	Surface Survey Unit 2	Surface Survey Unit 3	Surface Survey Unit 4	Surface Survey Unit 5
Auxiliary	All walls and floors	SFP/Transfer Canal	N/A	N/A	N/A
Containment	565'elevation steel liner floor and walls above 565' elevation	Under-Vessel Area	SFP/Transfer Canal	N/A	N/A
SFP/ Transfer Canal	All walls and floors	N/A	N/A	N/A	N/A
Turbine	All walls and floors	Circulating Water Discharge Tunnel	Circulating Water Intake Pipe (1)	Buttress Pits/ Tendon Tunnels(1)	Circulating Water Discharge Pipe (1)
Crib House/Forebay	All walls and floors	Circulating Water Intake Pipe (1)	N/A	N/A	N/A
WWTF	All walls and floors	N/A	N/A	N/A	N/A

(1) Judgmental samples only – Circulating Water Intake Pipe, CW Discharge Pipe and Buttress Pits/Tendon Tunnels are not survey units.

After passing the Sign test, the mean dose contribution for multiple surface survey units in a given basement (and the mean of the judgmental samples in Circulating Water Intake Pipe, Circulating Water Discharge Pipe and the Buttress Pits/Tendon Tunnels) is determined on an area-weighted basis. The total basement area used in the weighted average calculation is the adjusted surface area used to calculate the DCGLs in section 6.6.8. Residual radioactivity at the DCGL will result in 25 mrem/yr only if residual radioactivity is uniformly distributed over 100% of the adjusted surface area. The adjusted areas used for the DCGL calculations, and applied in the weighted average calculation of total basement surface dose are provide in Table 5-23, which is reproduced from Chapter 6, section 6.6.8.1, Table 6-23.

Table 5-23 Adjusted Basement Surface Areas for Area-Weighted SOF Calculation

Basement	Structures Included in Area-Weighted SOF Calculation(1)	Adjusted SA m^2
Containment	Containment + SFP/Transfer Canal	3,482
Auxiliary Building	Auxiliary + SFP/Transfer Canal	7,226
Turbine Building	Turbine + Circulating Water Discharge Tunnel + Circulating Water Intake Pipe + Circulating Water Discharge Pipe + Buttress Pits/Tendon Tunnels	27,135
Crib House/Forebay	Crib House/Forebay + Circulating Water Intake Pipe	18,254
SFP/Transfer Canal	SFP/Transfer Canal	723
WWTF	WWTF	1,124

(1) Surface areas of individual structures listed are provided in LTP Chapter 6, Tables 6-22 and 6-23.

The area-weighted SOF for Basements that have dose contributions from multiple surface survey units is calculated in accordance with Equation 5-8. For the areas specified in Footnote 1 of Table 5-22, the $SOF_{Bi,B}$ to be used in Equation 5-8 is based on the mean of the judgmental samples.

Equation 5-8

$$SOF_{B,B} = \sum_{i=1}^{n} \frac{SA_{SUi,B}}{SA_{Adjust,B}} * SOF_{Bi,B}$$

where:

$SOF_{B,B}$ = total surface SOF including all surface survey units in basement (B)
$SA_{SUi,B}$ = surface area of survey unit (i) in basement (B)
$SA_{Adjust,B}$ = adjusted surface area for DCGL calculation (Table 1) for basement (B)
$SOF_{Bi,B}$ = SOF_B for survey unit (i) in basement (B)

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 ZSRP. Survey design in this section specifically pertains to open land survey units and buried pipe; however the application of survey planning, survey package development, DQOs, data quality, investigations and data assessment as specified in this section is applicable to all FSS, including basement surfaces as described section 5.5. Buried piping is defined as pipe that runs through soil and is addressed in section 5.7.1.9. The FSS design for basement surfaces, embedded pipe and penetrations is described in section 5.5.

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 Zion are presented in LTP Chapter 2, section 2.1.6 and Table 2-4 and shown on Figures 2-4, 2-5, 2-6, and 2-7. 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 Zion*Solutions* 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 ZNPS, 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 Lower Bound of the Gray Region (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 is 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 Zion and their respective MDC values are presented in section 5.8 and Tables 5-26 and 5-27.

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, RA, and/or RASS that indicate residual radioactivity is unlikely to exceed the applicable Operational 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. <u>Sample Size Determination</u>

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

5.6.4.1.1. Decision Errors

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

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

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

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

Appendix E of NUREG-1757 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the guidance in NUREG-1757, the decision error(s) that will be used for the FSS at ZSRP 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 5.2.12, 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

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

5.6.4.1.5. Lower Bound of the Gray Region

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

5.6.4.1.6. Relative Shift

The relative shift (Δ/σ) for the survey unit data set is defined as shift (Δ), which is the upper boundary of the gray region, or Operational DCGL, 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 units where no significant concentrations of residual radioactivity are identified or anticipated, then survey design for FSS will 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 Tests

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

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

The number of sampling and measurement locations (N) that will be collected 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 attain the desired power level with the statistical tests and allow for possible lost or unusable data. Equation 5-2 of MARSSIM may alternatively be used to calculate the number of sampling and measurement locations. The result will be increased 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 EMC. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_w$ modified by an AF to account for the small area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_w$. Note that at ZSRP, the consideration of small areas of elevated radioactivity will only be applied to Class 1 open land (soil) survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_w$. For all other media (structural surfaces, embedded pipe, penetrations and/or buried pipe), any residual radioactivity identified by a FSS measurement at concentrations in excess of the respective Base Case DCGL will be remediated.

The statistical tests that determine if the residual radioactivity exceeds the $DCGL_w$ are not adequate for providing assurance that small areas of elevated radioactivity are successfully detected, as discussed in section 5.5.2.4 of MARSSIM. Systematic sampling and measurement locations in conjunction with surface scanning are used to obtain adequate assurance that small elevated areas comply with the $DCGL_{EMC}$; however, the number of statistical systematic sampling and measurement locations must be compared to the scan sensitivity to determine the adequacy of the sampling density. The calculation of the $DCGL_{EMC}$ is detailed in section 5.2.15.

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 5-9

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

where:

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

The AF is selected from Tables 5-16 and 5-17 for soils corresponding to the bounded area (A) calculated. If the calculated bounded area (A) falls between two area categories on Tables 5-16 and 5-17, 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 5-10

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

Next, the adjusted AF is used to look up a new adjusted area (A') from Tables 5-16 and 5-17. Finally, using the adjusted area (A'), an adjusted number of statistical systematic sampling and measurement locations (n_{EMC}) is calculated as follows;

Equation 5-11

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

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 5-24, which is reproduced from the portion of Table 5.9 from MARSSIM. As intended by the guidance, the emphasis will be placed on a higher frequency of scans in areas of higher risk. The scan coverage requirements that will be applied for scans performed in support of the FSS 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 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.

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.

Table 5-24 Recommended Survey Coverage

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

5.6.4.5.1. Reference Grid

A reference grid will be used for reference purposes and to locate the sampling and measurement locations. The reference grid 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 5-12

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

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

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 0.866*L* 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, any areas identified during post-processing and reviewing of scan survey data, and any results of soil or bulk material analyses that exceed the Operational 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 5-25.

Table 5-25 Investigation Levels

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

5.6.4.6.1. Remediation and Reclassification

The DQO process will be used as appropriate to evaluate the reclassification/resurvey action if an investigation level is exceeded. In Class 1 open land survey units, any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated to reduce the residual radioactivity to acceptable levels. In Class 1 survey units for media other than soil (structural surfaces, embedded pipe, buried pipe and/or penetrations), any areas of elevated residual radioactivity above the Base Case DCGL will be remediated. If an area is remediated, then a RASS will be performed to ensure that the remediation was sufficient.

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

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

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.

If the survey unit fails to demonstrate compliance with the release criterion, the data that led to the decision will be reviewed. Upon completion of the review, the DQO process will be used to identify and evaluate potential solutions. The level of residual radioactivity in the survey unit will be determined to help define the problem. Once the problem has been defined, the decision concerning the survey unit will be developed into a decision rule. Additional data, if any, will be acquired to document that the survey unit demonstrates compliance with the release criterion. Alternatives to resolving the decision statement will be developed for each survey unit that fails the tests. These alternatives will be evaluated against the DQOs, and a survey design that meets the objectives will be selected.

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, then the survey design will be based on the new classification. Additional guidance, including examples regarding the failure and re-survey of a survey unit is provided in section 8.5.3 of MARSSIM.

5.7. <u>Class 3Final Status Survey Implementation</u>

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

5.7.1. Survey Methods

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

5.7.1.1. Scanning

Scanning is performed in order to locate small, elevated areas of residual activity above the investigation level. It is the process by which a surveyor passes a portable radiation detector within close proximity of a surface with the intent of identifying residual radioactivity. Scan surveys that identify locations where the magnitude of the detector response exceeds an investigation level indicate that further investigation is warranted to determine the amount of residual radioactivity. The investigation levels may be based on the Operational DCGL, a fraction of the Operational DCGL, or the $DCGL_{EMC}$ for Class 1 soils, 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 Operational DCGL. If the scanning indicates that the survey unit or a portion of the survey unit has been improperly classified, then the survey design process must be evaluated to either assess the effect of reclassification on the survey unit as a whole (if the whole unit requires reclassification) or a new design must be established for the new unit(s) (in the case of sub-division). A new survey design will require a re-evaluation of the survey strategy to decide if it can meet the requirements of the revised survey design. If not, the survey strategy must be revised based on the available instrumentation and methods.

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

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

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

5.7.1.2. Beta-Gamma Scanning

Scanning is performed in order to locate small, elevated areas of residual activity above the investigation level. Miscellaneous materials may be scanned for beta-gamma radiation with appropriate instruments such as those listed in Table 5-26. The measurements will typically be performed at a distance of 1 cm or less from the surface and at a scan speed of 5 cm/sec for hand-held instruments. Adjustments to scan speed and distance may be made in accordance with approved procedures.

5.7.1.3. 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. ZSRP will utilize a chain-of-custody (COC) process to ensure sample integrity.

Quality assurance (QA) requirements for FSS activities that apply to sample collection (e.g., split samples, duplicates, etc.) and onsite and offsite laboratories employed to analyze samples as a part of the FSS process will be controlled by approved procedures, in conformance with the QAPP and is further described in section 5.9. Performance of laboratories will be verified periodically in accordance with the QAPP and ZS-QA-10, *"Quality Assurance Project Plan - Zion Station Restoration Project"* (Reference 5-19).

5.7.1.4. 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.

5.7.1.5. Surface Soils

In this context, surface soil refers to outdoor areas where the soil is, for purposes of dose modeling, considered to be uniformly contaminated from the surface down to a depth of 15 cm (6 in). These areas will be surveyed through combinations of sampling, scanning, and in-situ measurements, as appropriate.

5.7.1.5.1. 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. Zion*Solutions* TSD #11-004 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. Where appropriate, gamma emitters such as Co-60 and Cs-137 will be used as surrogates to infer the amount of any HTD radionuclide(s) that may be present in the distribution in accordance with section 5.2.12.

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).

5.7.1.5.2. 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.

All surface soil samples taken during continuing characterization and FSS will be analyzed by gamma spectrometry.

5.7.1.6. Subsurface Soils

Subsurface soil refers to soil that resides at a depth greater than 15 cm 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 decommissioning of Zion, any subsurface soil contamination that is identified by continuing characterization or operational radiological surveys that is in excess of the site specific Base Case DCGLs for each of the potential ROC as presented in Table 5-2 will be remediated. The remediation process will include performing RASS of the open excavations in accordance with section 5.4.2 of this FSS Plan. The RASS will include scan surveys and the collection of soil samples during excavation to gauge the effectiveness of remediation, and to identify locations requiring additional excavation. The scan surveys and the collection of and subsequent laboratory analysis of soil samples will be performed in a manner that is intended to meet the DQOs of FSS. The data obtained during the RASS is expected to provide a high degree of confidence that the excavation, or portion of the excavation, meets the criterion for the unrestricted release of open land survey units. Soil samples will be collected to depths at which there is high confidence that deeper samples will not result in higher concentrations. Alternatively, a NaI detector or intrinsic germanium detector of sufficient sensitivity to detect residual radioactivity at the Operational DCGL 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 Operational DCGL, additional confirmatory investigation and analyses of soil samples of the suspect areas will be performed.

5.7.1.6.1. 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.

5.7.1.6.2. 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, 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 Zion facility have shown that there is minimal residual radioactivity in subsurface soil. Consequently, Zion proposes to perform minimal subsurface sampling 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 Operational DCGL, then additional biased subsurface soil sample(s) will be taken within the area of concern as part of the investigation.

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

GeoProbe®, split spoon sampling or other methods 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 one-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 taken during continuing characterization and FSS will be analyzed by gamma spectrometry.

5.7.1.6.3. Sampling of Subsurface Soils below Structure Basement Foundations

The foundation walls and basement floors below the 588 foot elevation of the Unit 1 Containment, Unit 2 Containment, Auxiliary Building, Turbine Building, Crib House/Forebay, WWTF and remnants of the SFP will remain at the time of license termination. Based on the results of subsurface soil sampling performed during site characterization, it is not likely that the residual radioactivity concentrations in soil beneath these building foundations exceed the site-specific Base Case DCGLs as presented in Table 5-6. However, prior to license termination, it will be necessary to ascertain the radiological conditions of these sub-slab soils to demonstrate suitability for unrestricted release.

As stated in section 5.3.4.4, the soils under the basement concrete of the Containment Buildings, the Auxiliary Building and the SFP/Transfer Canal have been designated as "continuing characterization" areas once commodity removal and building demolition have progressed to a point where access can be achieved. Continuing characterization will consist of soil borings or use of GeoProbe® technology at the nearest locations along the foundation walls that can be feasibly accessed. The under-basement soil activity will be determined by interpreting results from borings collected at the nearest locations. 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. Angled soil borings will also be performed to directly access the sub-slab soils. The exact number and location of the soil borings will be determined by DQO during the survey design. All samples taken from sub-slab soils will be analyzed by gamma spectrometry. Ten percent (10%) of any sub slab soil samples taken will be analyzed for the initial suite of HTD radionuclides as well as any individual sample where analysis indicates gamma activity in excess of a SOF of 0.1.

If possible, survey design will also consider the possibility of coring through the basement concrete floor slabs to facilitate the collection of soil samples. However, to date, this has been not possible due to the intrusion of groundwater into the basements through the bore hole. This is especially true for the Containment basements, as sampling through the foundation would require compromising the integrity of the internal steel liner. To address the issue of groundwater intrusion and still investigate the potential for migration of contamination from building interiors to the sub-foundation soils, any continuing characterization performed in the Auxiliary Building basement, Under-Vessel area of the Containments and the SFP/Transfer Canal will include cores into the concrete floor, but not fully through the foundation or liner. The cores will be biased to areas with higher potential of providing a pathway for migration of contamination to sub-foundation soil including stress cracks, floor and wall interfaces, and penetrations through walls and floors for piping. If the analysis of the deepest 0.5 inch "puck" from the core in the foundation does not contain detectable activity, then it will be assumed that the location was not a source of sub-foundation soil contamination. If activity is positively detected at the deepest point in the core, continuing the core to the soil under the foundation will be considered depending on the levels of activity identified and the potential for groundwater intrusion.

If residual radioactivity is detected in subsurface soils adjacent to or under a basement surface, then the investigation will also include an assessment of the potential contamination of the exterior of the structure. A sample plan for the investigation will be created as specified by procedure and the plan and investigation results will be provided to NRC for evaluation. Based on the results of the investigation, ZSRP will assess the dose consequences of the subsurface soil contamination or will remediate as necessary.

5.7.1.7. Reuse of Excavated Soils

ZSRP will not stockpile and store excavated soil for reuse as backfill in basements. However, overburden soils may be excavated to expose buried components (e.g. concrete pads, buried pipe, buried conduit, etc…) that will be removed and disposed of as waste or, to install a new buried system. In these cases, the overburden soil will be removed, the component will be removed or installed, and the overburden soil will be replaced back into the excavation. In these cases, a RA will be performed. The footprint of the excavation, and areas adjacent to the excavation where the soil will be staged, will be scanned prior to the excavation. In addition, periodic scans will be performed on the soil as it is excavated and the exposed surfaces of the excavated soil will be scanned after it is piled next to the excavation for reuse. Scanning will be performed in accordance with section 5.7.1.5.1. A soil sample will be acquired at any scan location that indicates activity in excess of 50% of the soil Operational DCGL. Any soil confirmed as containing residual radioactivity . All samples taken from sub-slab soils will be analyzed by gamma spectrometry. at concentrations exceeding 50% of the soil Operational DCGL will not be used to backfill the excavation and will be disposed of as waste.

A RA is performed prior to introducing off-site material to ZSRP for use as backfill in a basement, or for any other use. The RA will be performed at the borrow pit, landfill, or other location from where the material originated and will consist of gamma scans and material sampling. Gamma scans are performed in situ, or by package (using a hand-held instrument or through the use of a truck monitor). Soil samples of overburden soils will be analyzed by gamma spectroscopy.

5.7.1.8. Pavement Covered Areas

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

5.7.1.9. Buried Piping

Designated sections of buried piping will be remediated in place and undergo FSS. The inventory of buried piping located below the 588 foot grade that will remain and be subjected to FSS is provided in TSD 14-016. Compliance with the Operational DCGL values, as presented in Table 5-10, will be demonstrated by measurements of total surface contamination and/or the collection of sediment samples.

The survey of buried pipe will be achieved in the same manner as described for the survey of embedded pipe as discussed in section 5.5.5. The radiological survey of pipe system interiors involves the insertion of appropriately sized detectors into the pipe interior by a simple “push-pull” methodology, whereby the position of the detector in the piping system can be easily determined in a reproducible manner.

The detectors are configured in a fixed geometry relative to the surveyed surface, thus creating a situation where a defensible efficiency can be calculated. The detectors are then deployed into the actual pipe and timed measurements are acquired at intervals commensurate with the contamination potential of the pipe. A conservative "area of detection" is assumed for each pipe size. It is also conservatively assumed that any activity is uniformly distributed in the area of detection.

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

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

5.7.1.10. Groundwater

Assessments of any residual radioactivity in groundwater at the site will be via groundwater monitoring wells installed at ZNPS. Ongoing monitoring of surface water and groundwater at ZNPS include REMP, Radiological Groundwater Protection Program (RGPP) and NPDES Monitoring. This is further described in Chapter 2, section 2.3.6.4.

5.7.1.11. Sediments and Surface Water

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

Sediment samples will be evaluated against the site-specific soil Operational DCGLs for each of the potential ROC as presented in Table 5-7. 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.

5.7.1.12. Survey Considerations for Buildings, Structures and Equipment

All above grade buildings will be removed in the end-state for ZSRP. The survey approach that will be used to radiologically assess the residual radioactivity in below-grade basement surfaces is presented in section 5.5 of this FSS Plan. The FSS of minor solid structures, such as but not limited to the Switchyard, the microwave tower, and the Sewage Lift Station, telephone poles, fencing, culverts, duct banks and electrical conduit will be included in the open land FSS unit in which they reside. These items will be scanned in accordance with recommended survey coverage in Table 5-24.

Prior to demolition, the standing concrete surface(s) (designated by process knowledge and previous characterization results as a suitable candidate(s) for potential use as clean fill) will be surveyed using the site program for unconditional release of material offsite. The unconditional release surveys will meet the statistical rigor and quality of a MARSSIM FSS. Once the concrete has been determined to be suitable for unconditional release, the structure will be demolished, all metal removed and the concrete crushed to pieces that are 10 inches in diameter or less. The material will then be stockpiled and controlled as "non-radioactive clean fill" (as per section 5.6.3) until such time that it is placed in the basement void. If the unconditional release surveys positively detect plant-derived radionuclides in any concentration, then the concrete will not be used as clean fill. In this case, it will be segregated, packaged and disposed of as low level radioactive waste. The results of static measurements taken during the unconditional release surveys will be provided to NRC in a Final Report.

5.8. Final Status Survey Instrumentation

Radiation detection and measurement instrumentation for performing FSS is selected to provide both reliable operation and adequate sensitivity to detect the ROC identified at the site at levels sufficiently below the Operational DCGL. 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 Operational DCGL. Detector selection will be based upon detection sensitivity, operating characteristics, and expected performance in the field. The instrumentation will, to the extent practicable, use data logging to automatically record measurements to minimize transcription errors. Commercially available portable and laboratory instruments and detectors are typically used to perform the three basic survey measurements: 1) surface scanning; 2) static measurements; and 3) radionuclide specific analysis of media samples such as soil and other bulk materials.

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

5.8.1. Instrument Selection

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

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

Other measurement instruments or techniques may be utilized. The acceptability of additional or alternate instruments or technologies for use in the FSS will be justified in a technical basis evaluation document prior to use. Technical basis evaluations for alternate 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 FSS is listed in Table 5-26. Instrument MDCs are discussed in section 5.8.4 and nominal MDC values for the proposed instrumentation are presented in Table 5-27.

5.8.2. Calibration and Maintenance

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

Instrumentation used for FSS will be calibrated and maintained in accordance with approved Zion*Solutions* 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.

Table 5-26 Typical FSS Survey Instrumentation

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

Table 5-27 Typical FSS Instrument Detection Sensitivities

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

[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] 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.

5.8.3. 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 Zion*Solutions* 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 is discarded, replacement data will be collected at the original locations.

5.8.4. Measurement Sensitivity

The measurement sensitivity or MDC will be determined *a priori* for the instruments and techniques that will be used for FSS. The MDC is defined as the *a priori* activity level that a specific instrument and technique can be expected to detect 95% of the time. When stating the detection capability of an instrument, this value should be used. The MDC is the detection limit, (*LD*), multiplied by an appropriate conversion factor to give units of activity. The critical level, (*LC*), is the lower bound on the 95% detection interval defined for *LD* and is the level at which there is a 5% chance of calling a background value "greater than background. This is the value used when actually counting samples or making direct radiation measurements. Any response above this level should be considered as above background (i.e., a net positive result). This will ensure 95% detection capability for *LD*. 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) for surface measurements 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, *"Evaluation of surface contamination - Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters"* (Reference 5-20).

5.8.4.2. Static Minimum Detectable Concentration

For static (direct) surface measurements with conventional detectors, such as those listed in Table 5-26, the MDC is calculated using the following equation:

Equation 5-13

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

where:

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

5.8.4.3. Beta-Gamma Scan Measurement Minimum Detectable Concentration

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

Equation 5-14

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

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 "observe" 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 Zion*Solutions* TSD 11-004, which examines the gamma sensitivity for 5.08 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 11-004 derives the MDC for the radionuclide mixtures at various detector distances and scan speeds. The model in TSD 11-004 uses essentially the same geometry configuration as the model used in MARSSIM. TSD 11-004 provides MDC values for the expected ZSRP 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 Zion*Solutions* laboratory 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. On-site laboratory counting systems are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil; this is calculated in accordance with the following equation:

Equation 5-15

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

where:

B	=	number of background counts during the count interval t;
K	=	proportionality constant that relates the detector response to the activity level in a sample for a given set of measurement conditions,
V	=	mass of sample (g), and
t	=	count time (minutes).

5.8.4.6. Pipe Survey Instrumentation

Pipe survey instruments proposed for use with pipe having diameters between 0.75 and 18 inches have been shown to have efficiencies ranging from approximately 0.02 to 0.5. This equates to detection sensitivities of approximately 350 dpm/100 cm^2 to 5,200 dpm/100 cm^2. This level of sensitivity is adequate to detect residual radioactivity below the Operational DCGLs derived for the unrestricted release of buried pipe as presented in Table 5-10.

5.9. Quality Assurance

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

Zion*Solutions* has developed and is implementing a comprehensive QA Program to assure conformance with established regulatory requirements. The quality requirements and quality concepts are presented in ZS-QA-10 which adequately encompasses all risk-significant decommissioning activities. The participants in the Zion*Solutions* 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 Zion*Solutions* QA Program complies with the requirements set forth in Appendix B of 10 CFR 50, Appendix H of 10 CFR 71, Appendix G of 10 CFR 72. References to specific industry standards for QA and QC measures governing FSS activities are reflected in the QAPP as well as all applicable supporting procedures, plans, and instructions. Effective implementation of QA and QC measures will be verified through audit activities, with corrective actions being prescribed, implemented and verified in the event any deficiencies are identified. These measures will also apply to the any FSS related services provided by off-site vendors, in addition to on-site sub-contractors.

The QAPP has been prepared to ensure the adequacy of data being developed and used during FSS. It supplements the quality requirements and quality concepts presented in ZS-QA-10. Compliance with the QAPP will serve to ensure that FSS are performed by trained individuals using approved written procedures and properly calibrated instruments that are sensitive to the suspected ROC. In addition, QC measures will be taken to obtain quantitative information to demonstrate that measurement results have the required precision and are sufficiently free of errors to accurately represent the area being investigated. QC checks will be performed as prescribed by the QAPP for both field measurements and laboratory analysis. Effective implementation of FSS operations will be verified through 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.

5.9.1. Project Management and Organization

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

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

Characterization and FSS encompasses all survey and sampling activities related to the LTP. This includes site characterization surveys, RASS, RA, and FSS. The duties and responsibilities of key Zion*Solutions* 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 FSS is to ensure the survey data collected are of the type and quality needed to demonstrate, with sufficient confidence, that the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. Compliance with the QAPP ensures that the following items are accomplished:

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

The following describe the basic elements of the QAPP.

5.9.2.1. Written Procedures

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

5.9.2.2. Training and Qualifications

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

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

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

5.9.2.3. Measurement and Data Acquisitions

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

5.9.2.4. Instrument Selection, Calibration and Operation

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

5.9.2.5. Chain of Custody

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

5.9.2.6. Control of Consumables

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

5.9.2.7. Control of Vendor-Supplied Services

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

5.9.2.8. Database Control

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

5.9.2.9. Data Management

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

5.9.3. Measurement/Data Acquisition

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

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

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

5.9.3.2. Duplicate and Split Samples

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

The NRC Inspection Procedure No. 84750 *"Radioactive Waste Treatment, and Effluent and Environmental Monitoring"* (Reference 5-21) 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 Zion*Solutions* procedure ZS-AD-08, *"Corrective Action Program"* (Reference 5-22). 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 FSS activities will be performed in accordance with applicable guidance. The findings will be tracked and trended.

5.9.4.2. Independent Review of Survey Results

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

5.9.4.3. Corrective Action Process

The corrective action process, already established as part of the site QA Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. The process will be conducted in accordance with ZS-AD-08, which describes the methods used to identify potential conditions adverse to quality (CAQ), condition reporting, self-assessment resolution and corrective action issues related to FSS. The CAQ evaluation effort is commensurate with the classification of the CAQ and could include root cause determination, extent of condition reviews, and preventive and remedial actions.

5.9.4.4. Corrective Action Process

Reports of audits and trend data will be reported to management in accordance with the QAPP and approved procedures.

5.9.5. Data Validation

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

5.9.6. NRC Confirmatory Measurements

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

5.10. Final Status Survey Data Assessment

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

- verification that the measurements were obtained using approved methods;
- verification that the quality requirements were met;
- verification that the appropriate corrections were made to any gross measurements and that the data is expressed in the correct reporting units;

- verification that the measurements required by the survey design, and any measurements required to support investigation(s) have been included;
- verification that the classification and associated survey unit design remain appropriate based on a preliminary review of the data;
- subjecting the measurement results to the appropriate statistical tests;
- determining if the residual radioactivity levels in the survey unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

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

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

For ZSRP, the Sign Test is the most appropriate test for FSS. Characterization surveys indicate that Cs-137 found in background due to global fallout constitutes a small fraction of the DCGL. Consequently, the Sign Test will be applied to open land, basements surfaces (to include steel liner) embedded pipe, penetrations and buried piping when demonstrating compliance with the unrestricted release criteria without subtracting background.

Survey results will be converted to appropriate units of measure (e.g., dpm/100 cm^2, pCi/g, pCi/m^2) and compared to investigation levels to determine if the action levels for investigation have been exceeded. Measurements exceeding investigation action levels will be investigated. If confirmed within a Class 1 soil 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, in most cases, the entire survey unit will be reclassified and a re-survey performed consistent with the change in classification.

5.10.1. Review of DQOs and Survey Plan Design

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

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

5.10.2. Preliminary Data Review

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

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

5.10.2.1. Data Validation

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

- Ensure that the instrumentation MDC for direct measurements and sample analyses was less than 10% of the Operational DCGL, which is preferable. MDCs up to 50% of the Operational DCGL are acceptable,
- Ensure that the instrument calibration was current and traceable to NIST standards,
- Ensure that the field instruments used for FSS were source checked with satisfactory results before and after use each day that data were collected,
- Ensure that the MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey,
- Ensure that the survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed,
- Ensure that the sample was controlled from the point of sample collection to the point of obtaining results,

- Ensure that the data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility, and
- Ensure that the data have been properly recorded.

If the data review criteria are not met, the discrepancy(s) will be evaluated and the decision to accept or reject the data will be documented in accordance with approved procedures. A condition report generated in accordance with ZS-AD-08 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 Operational DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Incongruities in the background data may be the result of residual, undetected activity, or they may just reflect background variability.

5.10.2.2.2. Frequency Plot

Frequency plots 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 Operational DCGL. There must be sufficient survey data at or below the Operational DCGL to statistically reject the null hypothesis and conclude the survey unit meets the site release criteria. These statistical analyses 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 Operational DCGL for each sample or measurement by dividing the reported concentration by the Operational

DCGL. If a sample has multiple ROC, then the fraction of the Operational DCGL for each ROC will be summed to provide a SOF for the sample.

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

Equation 5-16

$$SOF \leq 1 = \frac{Conc_{Cs\text{-}137}}{DCGL_{Cs\text{-}137_s}} + \frac{Conc_{Co\text{-}60}}{DCGL_{Co\text{-}60}} + \ldots \frac{Conc_n}{DCGL_n}$$

where:

$Conc_{Cs\text{-}137}$ = measured mean concentration for Cs-137,
$DCGL_{Cs\text{-}137s}$ = Surrogate Operational DCGL for Cs-137,
$Conc_{Co\text{-}60}$ = measured mean concentration for Co-60,
$DCGL_{Co\text{-}60}$ = Operational DCGL for Co-60,
$Conc_n$ = measured mean concentration for radionuclide n,
$DCGL_n$ = Operational DCGL for radionuclide n.

The unity rule equivalent results will be used to perform the Sign Test.

5.10.3.2. Sign Test

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

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

5.10.4. Elevated Measurement Comparison Evaluation

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

The investigation level for the EMC is the $DCGL_{EMC}$, which is the Base Case DCGL modified by an AF. Locations identified by surface scans or sample analyses which exceed the Base Case DCGL are subject to additional surveys to determine compliance with the elevated measurement criteria. Based upon the size of the elevated measurement area, the corresponding AF will be determined from Tables 5-16 and 5-17 using linear or exponential interpolation as necessary.

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

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

Equation 5-17

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

where:

δ = the survey unit average activity;
$DCGL_W$ = the survey unit Base Case DCGL concentration,
τ_n = the average activity value of hot spot *n*, and
$DCGL_{EMCn}$ = the $DCGL_{EMC}$ concentration of hot spot *n*.

5.10.5. Data Conclusions

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

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

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

"Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. The power of the statistical test is a function of the number of measurements made and the standard deviation of the measurement data. Quantitatively, the power is 1 - 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 Operational DCGL 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 FSS will be contained in two types of reports and will be consistent with section 8.6 of MARSSIM. An FSS 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 Final Report. An FSS Final Report, which is a written report that is provided to the NRC for its review, will be prepared to provide a summary of the survey results and the overall conclusions which demonstrate that the site, or portions of the site, meets the radiological criteria for unrestricted use, including ALARA.

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

5.11.1. FSS Unit Release Records

An FSS Unit Release Record will be prepared upon completion of the FSS for a specific survey unit. Sufficient data and information will be provided in the release record to enable an independent re-creation and evaluation at some future time. The FSS 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, surrogate ratios, statement of pass or failure of the statistical test(s);
- QC results to include discussion of split samples and/or QC replicate measurements;
- Results of any investigations;
- Any remediation activities, both historic and resulting from the performance of the FSS;
- Any changes from the FSS survey design including field changes;
- DQA conclusions;
- Any anomalies encountered during performance of the survey or in the sample results; and,
- Conclusion as to whether or not the survey unit satisfied the release criteria and whether or not sufficient power was achieved.

5.11.2. FSS Final Reports

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

- A brief overview discussion of the FSS Program including descriptions regarding survey planning, survey design, survey implementation, survey data assessment, and QA and QC measures;
- A description of the site, the applicable survey area(s) and survey unit(s), a summary of the applicable HSA information, conditions at the time of survey, identification of potential contaminants, and radiological release criteria;
- A discussion regarding the DQOs, survey unit designation and classification, background determination, FSS plans, survey design input values and method for determining sample size, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), ISOCS Efficiency Calibration geometry, survey methodology, QC surveys,

and a discussion of any deviations during the performance of the FSS from what was described in this LTP;

- A description of the survey findings including data conversion, survey data verification and validation, evaluation of number of sample/measurement locations, a map or drawing showing the reference system and random start systematic sample locations, and comparison of findings with the appropriate Operational DCGL or Action Level including statistical evaluations.
- Description of any judgmental and miscellaneous sample data collected in addition to those required for performing the statistical evaluation.
- Description of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the Operational DCGL.
- If survey unit fails the statistical test, a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure and the impact that the failure has on the conclusion that the facility is ready for final radiological surveys, and a discussion of the impact of the failure on survey design and result for other survey units.
- Description of how ALARA practices were employed to achieve final activity levels.

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

5.12. Surveillance Following FSS

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

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

- Review of access control entries since the performance of FSS or the last surveillance,
- A walk-down of the areas to check for proper postings,
- Check for materials introduced into the area or any disturbance that could change the FSS including the potential for contamination from adjacent decommissioning activities,

- If evidence is found of materials that may have been introduced into the survey unit or any disturbance that could change the FSS, then perform and document a biased scan of the survey unit, 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

5-1 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" – August 2000

5-2 U.S. Nuclear Regulatory Commission NUREG-1505, Revision 1, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys" – June 1998 draft

5-3 U.S. Nuclear Regulatory Commission NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions" – June 1998

5-4 U.S. Nuclear Regulatory Commission NUREG-1700, Revision 1, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" – April 2003

5-5 U.S. Nuclear Regulatory Commission NUREG-1757, Volume 2, Revision 1, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report" – September 2003

5-6 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" – January 1999

5-7 "Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement" – December 2007

5-8 Zion*Solutions* Technical Support Document 14-016, "Description of Embedded Piping, Penetrations and Buried Piping to Remain in Zion End State"

5-9 Zion*Solutions* Technical Support Document 11-001, "Potential Radionuclides of Concern during the Decommissioning of Zion Station"

5-10 Zion*Solutions* Technical Support Document 14-019, "Radionuclides of Concern for Soil and Basement Fill Model Source Terms"

5-11 Zion*Solutions* Technical Support Document 17-004, "Operational Derived Concentration Guideline Levels for FSS"

5-12 Zion*Solutions* Technical Support Document 14-011, "Soil Area Factors"

5-13 Zion*Solutions* Technical Support Document 14-015, "Buried Pipe Dose Modeling & DCGLs"

5-14 "Zion Station Historical Site Assessment" (HSA) – September 2006

5-15 Sandia National Laboratories, NUREG/CR-5512, Volume 1, Final Report, "Residual Radioactive Contamination from Decommissioning Parameter Analysis" – October 1992

5-16 Zion*Solutions* Technical Support Document 11-004, "Ludlum Model 44-10 Detector Sensitivity"

5-17 Zion*Solutions* ZS-LT-01, "Quality Assurance Project Plan (for Characterization and FSS)" (QAPP)

5-18 Zion*Solutions* Technical Support Document 10-002, "Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"

5-19 Zion*Solutions* ZS-QA-10, "Quality Assurance Project Plan - Zion Station Restoration Project"

5-20 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

5-21 U.S. Nuclear Regulatory Commission Inspection Procedure No. 84750 "Radioactive Waste Treatment, and Effluent and Environmental Monitoring" – March 1994

5-22 ZionSolutions ZS-AD-08, "Corrective Action Program"

Figure 5-1 Characterization/LTP/FSS Organization Chart