SAFETY EVALUATION REPORT (SER)

BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO AMENDMENT NOS. 191 AND 178

TO FACILITY OPERATING LICENSE NO. DPR-39 AND DPR-48

ZION*SOLUTIONS*, LLC

ZION NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NO. 50-295 AND 50-304

1.0 INTRODUCTION

On December 19, 2014, Zion*Solutions*, LLC (ZS, the licensee)[1] submitted a license termination plan (LTP) and accompanying license amendment request (LAR), "License Amendment Request for the License Termination Plan," for Zion Nuclear Power Station, Units 1 and 2 (ZNPS or the facility) (Agencywide Documents Access and Management System Package Accession number ML15005A336). The LTP was submitted as a supplement to the ZNPS defueled safety analysis report (DSAR), and was accompanied by a proposed license amendment (in the form of a license condition) that establishes the criteria for when changes to the LTP require prior U.S. Nuclear Regulatory Commission (NRC) approval. On February 26, 2015 (ML15061A281), ZS submitted additional information, including site-specific decommissioning cost information. On July 20, 2017 (ADAMS Package ML17215A095), the licensee submitted Revision 1 to the LTP with corresponding changes to the LAR that reflect, among other things, changes made in response to requests for additional information (RAI). Subsequently, on February 7, 2018 (ADAMS Package ML18052A851), the licensee submitted Revision 2 to the LTP. Finally, on August 28, 2018 (ML18242A082), the licensee submitted a change to the license conditions associated with the LTP. The supplements dated February 26, 2015, March 8, 2016, July 20, 2016, July 20, 2017, February 7, 2018, April 10, 2018, and August 28, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 14, 2015, (80 FR 20020).

2.0 REGULATORY EVALUATION

The ZNPS is located near the city of Zion in northeast Illinois on the west shore of Lake Michigan. The site is approximately 40 miles north of Chicago, Illinois and 42 miles south of Milwaukee, Wisconsin. In September 1996, ZNPS Unit 2 was permanently shut-down after approximately 23 years of operation. In February 1997, ZNPS Unit 1 was permanently shut-down after approximately 24 years of operation. On February 13, 1998, in accordance with 10 CFR 50.82(a)(1)(i) and (ii), Commonwealth Edison Company (now Exelon Generating

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[1] The terms "ZS" and "licensee" are used interchangeably throughout this document.

Company, LLC. (Exelon)) notified the NRC of the permanent cessation of operations at the ZNPS and the permanent removal of all spent fuel assemblies from the reactor vessels to the spent fuel pool (ADAMS Legacy Accession Nos. 9902200407 and 9803110251).

On September 1, 2010, the NRC transferred Facility Operating License Numbers DPR-39 and DPR-48 from Exelon to ZS (ADAMS Accession No. ML102290437). The ZNPS was acquired by ZS to conduct the decommissioning of the facility and then return the decommissioned site back to Exelon. The spent fuel has been moved from the spent fuel pool to the on-site Zion Independent Spent Fuel Storage Installation. The decontamination and dismantlement of the ZNPS is actively underway.

In accordance with the requirements of section 50.82(a)(9) of Title 10, U.S. Code of Federal Regulations, "All power reactor licensees must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval." The licensee has not submitted an application for termination of the licenses at this time.

Per 10 CFR 50.82(a)(9)(i), the LTP must be a supplement to the updated final safety analysis report (FSAR), or equivalent. Per 10 CFR 50.82(a)(9)(ii), the LTP must include:

(A) A site characterization;
(B) Identification of remaining dismantlement activities;
(C) Plans for site remediation;
(D) Detailed plans for the final radiation survey;
(E) A description of the end use of the site, if restricted;
(F) An updated site-specific estimate of remaining decommissioning costs;
(G) A supplement to the environmental report, pursuant to § 51.53, describing any new information or significant environmental change associated with the licensee's proposed termination activities; and
(H) Identification of parts, if any, of the facility or site that were released for use before approval of the license termination plan.

The approval criteria of the LTP are given in 10 CFR 50.82(a)(10), which states:

> If the license termination plan demonstrates that the remainder of decommissioning activities will be performed in accordance with the regulations in this chapter, will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment and after notice to interested persons, the Commission shall approve the plan, by license amendment, subject to such conditions and limitations as it deems appropriate and necessary and authorize implementation of the license termination plan.

The licensee also requested a new license condition to allow the licensee to make certain changes to this approved LTP without prior NRC review or approval. Each new License Condition 2.C.17 would read as follows:

2. C. (17) License Termination Plan (LTP)

Zion*Solutions* shall implement and maintain in effect all provisions of the approved License Termination Plan as approved in License Amendment No. 191 subject to and as amended by the following stipulations:

Zion*Solutions* may make changes to the LTP without prior approval provided the proposed changes do not meet any of the following criteria:

(A) Require Commission approval pursuant to 10 CFR 50.59.

(B) Result in significant environmental impacts not previously reviewed.

(C) Detract or negate the reasonable assurance that adequate funds will be available for decommissioning.

(D) Decrease a survey unit area classification (i.e., impacted to not impacted; Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3) without providing the NRC a minimum 14 day notification prior to implementing the change in classification.

(E) Increase the derived concentration guideline levels (DCGL) and related minimum detectable concentrations (for both scan and fixed measurement methods).

(F) Increase the radioactivity level, relative to the applicable DCGL, at which an investigation occurs.

(G) Change the statistical test applied other than the Sign test.

(H) Increase the approved Type I decision error above the level stated in the LTP.

(I) Change the approach used to demonstrate compliance with the dose criteria (e.g., change from demonstrating compliance using derived concentration levels to demonstrating compliance using a dose assessment that is based on final concentration data).

(J) Change parameter values or pathway dose conversion used to calculate the dose such that the resultant dose is lower than in the approved LTP and if a dose assessment is being used to demonstrate compliance with the dose criteria.

(K) Reuse concrete from demolished structures, other than from the list of areas specified in Section 2.1.1 of TSD 17-010, “Final Report - Unconditional Release Surveys at the Zion Station Restoration Project, Revision 1”, as backfill.

(L) Assign a dose for reuse concrete other than the dose values provided along with the LTP (as shown in Table 6-53 (Revision 2) of the LTP) and documented in Section 8 and Table 33 of TSD 14-010, “RESRAD Dose Modeling for Basement Fill Model and Soil DCGL and Calculation of Basement Fill Model Dose Factors and DCGLs, Revision 6.”

(M) Use area-specific surrogate ratios that are less than the maximum surrogate ratios (H-3/Cs-137, Ni-63/Co-60, Sr-90/Cs-137) presented in Table 5-15 (Revision 2) of the LTP.

As stated in 10 CFR 50.92(a), "[i]n determining whether an amendment to a license ... will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses ... to the extent applicable and appropriate." The considerations that govern issuance of initial operating licenses include those stated in 10 CFR 50.40 "Common Standards" and 10 CFR 50.57 "Issuance of operating license."

3.0 TECHNICAL EVALUATION

3.1 Method of Review

To be approved under 10 CFR 50.82(a)(10), the LTP must demonstrate that "the remainder of decommissioning activities will be performed in accordance with the [Commission's] regulations ..., will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment." To perform its review of the LTP, the staff used the guidance in NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," Rev. 1, April 2003 (ML031270391) and Rev. 2, April 2018 (ML18116A124).

The licensee also requested an amendment to its licenses which would add to each license a new license condition (LC) giving a process for changing the LTP after approval. In reviewing the LC, the staff used the guidance and model license condition in Appendix 2 of NUREG 1700, Rev. 1 and/or Appendix B of NUREG-1700, Rev. 2.

The LTP describes ZS's approach for demonstrating compliance with radiological criteria for unrestricted use. Those criteria are set forth in 10 CFR 20.1402, which states:

> A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal.

3.2 Site Characterization

3.2.1 Facility Radiological Status

Licensees conduct site characterization surveys to determine the nature and extent of radioactive contamination in buildings, plant systems and components, site grounds, and surface and groundwater. The major objectives of characterization activities are to: permit the planning and conduct of radiological remediation activities; confirm the effectiveness of radiological remediation methods; provide information to develop specifications for final status surveys (FSS); define site and building areas as survey units and assign survey unit classifications; and provide information for dose modeling. The licensee conducted radiological site characterization activities that included a historical site assessment[2] (HSA) (ML15342A281), scoping surveys, and a characterization survey. The HSA included a review of records maintained to satisfy the requirements of 10 CFR 50.75(g)(1)(requiring the licensee to keep "[r]ecords of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site"), as well as environmental reports, Radiological Environmental Monitoring Reports, Radioactive Effluent Release Reports, Licensee Event Reports, Plant Operating Reports, Plant Safety Analyses, Radiological Surveys, and Plant Operating Logs. Additionally, personal interviews of current and former ZNPS site personnel and reviewed collections of photographs were performed for the HSA.

LTP Section 2.1.3 provides a description of the operational history of the ZNPS beginning with the authorization of construction on December 26, 1968, to the units' shutdown in 1998, and subsequent decommissioning activities to the present. LTP Table 2-1 summarizes the operational and post-operational history. Spent fuel remained onsite in the spent fuel pool (SFP) until January 2015, when the licensee transferred the fuel to the independent spent fuel storage installation (ISFSI), that is generally licensed by the NRC under 10 CFR 72.210 (issuing a general license for the storage of spent fuel in an ISFSI at power reactor sites to persons authorized to possess or operate nuclear power reactors under 10 CFR part 50 or 10 CFR part 52). LTP Section 2.1.4 provides a summary of site incidents based on a review of plant records. Events included radiological spills, chemical spills, loss of radioactive material control, and system cross-contaminations. As part of the HSA process, the ZNPS facilities and grounds were divided into preliminary survey areas and assigned initial area classifications based on the operational history and the incidents and processes documented for that survey unit. Survey units included Class 1, 2 and 3 structures; class 1, 2 and 3 open land areas; and non-impacted areas.

To make the best use of resources for decommissioning, MARSSIM places greater survey efforts on areas that have, or had, the highest potential for contamination. Areas that have no reasonable potential for residual contamination are classified as non-impacted areas. Class 1 Areas have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the DCGL for the average residual radioactivity in a survey unit (DCGLw). Class 2 Areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGLw. Class 3 areas are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGLw, based on site operating history and previous radiation surveys.

The licensee designed a characterization survey plan consistent with the data quality objectives (DQO) outlined in MARSSIM, Appendix D. The survey plan specified the number of static measurements and/or samples, determination of static measurement or sample locations, scan coverage, types of measurements or samples, concrete core samples, and material sampling consistent with the DQOs. The selection and use of instrumentation, laboratory requirements, and quality assurance were also specified.

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[2] In this context, the term "historical site assessment" concerns the assessment of radiological contamination on the site (essentially, a history of the radioactivity on the site as a result of the ZNPS operation). The historical site assessment does not concern historic properties or cultural resources.

ZS TSD 11-001, "Potential Radionuclides of Concern during the Decommissioning of Zion Station" (ML15344A389) was prepared to establish the basis for an initial suite of potential Radionuclides of Concern (ROCs) 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 potential neutron activation products, noble gases and radionuclides with a half-life less than 2 years, an initial suite of potential ROC for the decommissioning of the ZNPS was prepared. The list of potential radionuclides was provided in Table 2-12 of the LTP.

The licensee conducted radiological site characterizations beginning in November 2011. Several background studies were performed on the ZNPS site to assess background for soils and concrete. The first study was performed in February of 2012 as part of the work scope pertaining to the characterization of the ISFSI construction area. During this study, soil, concrete and asphalt was assessed though surface scanning and volumetric sampling and analysis. In March and April of 2012, ZS conducted a comprehensive background study of non-contaminated concrete. In July of 2012, an additional study was performed to evaluate background for soils. ZS performed surveys and sampling of the various structures, open land areas, as well as surface and groundwater and determined the ROCs for each. The results of those characterizations are included in Section 2.3 of the LTP.

The LTP states that the licensee will continue to characterize the site as decommissioning progresses making additional areas accessible, collecting additional sampling data as needed, and that the licensee will continue to evaluate data as collected to determine the impact on the radioisotopes present, nuclide fractions, and the classification of structures and environmental media. The LTP concludes that the characterization data collected and analyzed to date are of sufficient quantity and quality to provide the basis for the initial classification of survey areas, decommissioning and decontamination (D&D) activities, estimating radioactive waste types and volumes, and for the development of the DCGLs.

3.2.2 Site Characterization

The NRC staff has reviewed the information in the LTP for the ZNPS site according to Section B.2 "Site Characterization" of the standard review plan (SRP), NUREG-1700 (NRC, 2003). As described therein, the purposes of the staff's review are (1) to ensure that the site characterization presented in the LTP is complete; and (2) to verify that the licensee obtained the data using sufficiently sensitive instruments and proper quality assurance procedures to obtain reliable data that are relevant to determining whether the site will meet the decommissioning limits if characterization data is used as final survey data. The acceptance criteria of Section B.2 of the SRP, states that the LTP should (1) identify all locations where activities (including spills) could have resulted in contamination; (2) summarize the status of the site; (3) be sufficiently detailed to allow a reader to determine the contamination levels; (4) identify survey instruments and practices; (5) identify background radiation levels; and (6) describe areas and equipment that need further remediation.

The LTP summarizes the original shutdown and current radiological status of the site. The LTP identifies all locations, where spills, disposals, operational activities, or other accidents and or incidents occurred that could have resulted in contamination within and outside the facility. The LTP describes the areas and equipment that need additional remediation, identifies background activity concentrations and radiation exposure readings used during scoping and characterization surveys, and identifies the survey instruments and supporting QA practices used in the site characterization program. The licensee has sufficiently detailed the status of the ZNPS to allow the staff to determine the extent and range of radiological contamination of the ZNPS facility structures and site. Therefore, the LTP meets the acceptance criteria as delineated in SRP Section B.2. Based on this review, the staff has determined that the licensee has met the objectives of providing an adequate site characterization as required by 10 CFR 50.82(a)(9)(ii)(A)(requiring the LTP to include "[a] site characterization").

3.3 Remaining Site Dismantlement Activities

In accordance with 10 CFR 50.82 (a)(9)(ii)(B)(requiring the LTP to include "[i]dentification of remaining dismantlement activities"), and following the guidance of NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," the licensee provided a description of the major remaining dismantlement activities. Those activities were to be undertaken pursuant to the current 10 CFR 50 license consistent with the Zion Nuclear Station "Post Shutdown Decommissioning Activity Report" (PSDAR) (ML080840398).

The staff has reviewed the LTP against the acceptance criteria in SRP Section B.3 "Identification of Remaining Site Dismantlement Activities." The acceptance criteria in Section B.3 are:

- The LTP discusses the remaining tasks associated with decontamination and dismantlement, estimates the quantity of radioactive material to be shipped for disposal or processing, describes the proposed control mechanisms to ensure that areas are not recontaminated, and contains occupational exposure estimates and radioactive waste characterization.
- The LTP describes the remaining dismantlement activities in sufficient detail for the NRC staff to identify any associated inspection or technical resources that will be needed.
- The LTP is sufficiently detailed to provide data for use in planning further decommissioning activities. As such, the LTP includes decontamination techniques, projected schedules, costs, waste volumes, dose assessments (including groundwater assessments), and health and safety considerations.
- The LTP lists the remaining activities that do not require any additional licensing action.

The information included those areas and equipment that need further radiological remediation and an estimate of radiological conditions that the licensee may encounter. The licensee provided an overview and describes the major remaining components of radiologically contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations. The LTP also provided information related to the remaining D&D tasks. This information included an estimate of the quantity of radioactive material to be released in accordance with the licensed material disposal requirement of 10 CFR 20.2001(a)(1), a description of proposed control mechanisms to ensure areas are not re-contaminated, estimates of occupational exposures, and characterization of radiological conditions to be encountered and the types and quantities of radioactive waste.

The licensee has sufficiently detailed data for use in planning further D&D activities and lists the remaining activities that do not require any additional licensing action. Based on this review, the staff has determined that the licensee has identified, in sufficient detail, the remaining dismantlement activities necessary to complete decommissioning of the facility, as required by 10 CFR 50.82(a)(9)(ii)(B)(requiring the LTP to include "[i]dentification of remaining dismantlement activities").

As part of the discussion of remaining dismantlement activities, the licensee discussed the decommissioning activities that have already been completed, including the large components that have been shipped offsite for disposal in radiological waste repositories, at landfills, or for recycling. The submission of this information meets the submittal requirements of 10 CFR 50.82(a)(9)(ii)(H).

3.4 Plans for Radiological Site Remediation

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(C)(requiring the LTP to include "[p]lans for site remediation"), the licensee provided its plans for completing radiological remediation of the site. The licensee plans to remediate the site, including structures and systems that remain on the site, to the criteria of 0.25 mSv/yr (25 mrem/yr) for all pathways, which is the unrestricted use criteria specified in 10 CFR Part 20, Subpart E. The licensee also stated that the remaining residual radioactivity must also satisfy the ALARA [As Low As Reasonably Achievable] criterion in 10 CFR 20.1402 (giving the radiological criteria for unrestricted use), which requires an evaluation as to whether it is feasible to further reduce residual radioactivity to levels below those necessary to meet the dose criterion (i.e., to levels that are ALARA).

Since the licensee will be remediating the site to the unrestricted release criteria, no submission regarding a restricted end use of the site is required. Therefore, the licensee has complied with the requirement of 10 CFR 50.82(a)(9)(i)(E).

The staff compared the information in the LTP against the acceptance criteria in SRP Section B.4 "Remediation Plans." Those criteria are:

- The LTP addresses any changes in the radiological controls to be implemented to control radiological contamination associated with the remaining decommissioning and remediation activities.
- The LTP discusses in detail how facility and site areas will be remediated to meet the proposed residual radioactivity levels (DCGLs) for license termination.
- The LTP includes a schedule that demonstrates how and in what time frames the licensee will complete the interrelated decommissioning activities.

Under the asset sale agreement with Exelon, ZS must demolish and remove all on-site buildings, structures, and components to a depth of at least three feet below grade. Remediation techniques that may be used for the structural surfaces below 588 foot elevation include washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting.

The licensee will remove and dispose of soil contamination above the site specific $DCGL_{EMC}$ as radioactive waste. Soil remediation equipment will include, but not be limited to, shovels, back hoe and track hoe excavators. Other equipment including soil dredges and vacuum trucks may also be used. As practical, when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique may be used to minimize the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. The licensee commits to the use of excavation safety and environmental control procedures to remediate radiologically contaminated soils. The licensee will augment the excavation safety and environmental control procedures with procedural requirements to ensure the licensee maintains adequate erosion, sediment, and air emission controls during soil remediation.

Section 4.3 of the LTP states that the Radiation Protection Program approved for decommissioning is similar to the regulatory approved program that was implemented during commercial power operation and the subsequent SAFSTOR period. During power operations, contaminated structures, systems and components were decontaminated in order to perform maintenance or repair actions. These techniques are the same or similar to the radiological controls implemented at Zion Station Restoration Project (ZSRP) for the decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas. The licensee provided its ALARA analysis process in LTP Section 4.4. The licensee's formulas for calculating the remediation levels conform to the guidance provided in Appendix N of NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance -Characterization, Survey, and Determination of Radiological Criteria."

The LTP discusses in detail how the licensee intends to remediate ZNPS to meet the proposed residual radioactivity levels (DCGLs) for license termination. The LTP includes a schedule that demonstrates how and in what time frames the licensee intends to complete the interrelated decommissioning activities.

Based on this review, the staff determined that the licensee has met the requirements of 10 CFR 50.82(a)(9)(ii)(C)(requiring the LTP to include "[p]lans for site remediation") by providing a detailed discussion of its radiological remediation site plans for the remaining decommissioning activities.

3.5 Final Status Survey Plan

3.5.1 Final Status Survey Plan Overview

In Section 5 of the License Termination Plan (LTP) for the Zion Nuclear Power Station (ZNPS) facility the licensee describes the final status survey (FSS) process to demonstrate that the ZNPS facility and site comply with the radiological criteria for unrestricted use, as specified in 10 CFR 20.1402. A plan for the FSS is also one of the requirements to terminate a power reactor license, as specified in 10 CFR 50.82(a)(9)(ii)(D)(requiring the LTP to include "[d]etailed plans for the final radiation survey").

The staff compared the information in the LTP against the acceptance criteria in SRP Section B.5 "Final Radiation Survey Plan." Those criteria are:

- The LTP includes the "Information To Be Submitted," as described in section 4 "Facility Radiation Surveys" of NUREG-1757 Vol 2.
- The LTP includes the following information: identification of the major radiological contaminants; methods used for addressing hard-to-

detect radionuclides; access control procedures to control recontamination of clean areas; description of the Quality Assurance (QA) Program; and methods for surveying embedded piping.
- The final survey plan meets the evaluation criteria defined in Section 4 of NUREG-1757, Vol. 2.

The licensee notes in Section 5 of the LTP that the following guidance was used to develop the FSS plan:

- NUREG-1575, Revision 1, “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM);” August 2000;
- NUREG-1505, Revision 1,, “A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys;” June 1998;
- NUREG-1507, Revision 0, “Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions;” June 1998;
- NUREG-1700, Revision 1,, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans;” April 2003;
- NUREG-1757, Volume 2, Revision 1, “Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report;” September 2006; and
- Regulatory Guide 1.179, Revision 1, “Standard Format and Content of License Termination Plans for Nuclear Power Reactors” June 2011.

The licensee generated site-specific Derived Concentration Guideline Levels (DCGLs) in order to assess compliance with the 25 mrem/yr dose criterion for unrestricted use, as specified in 10 CFR 20.1402. These DCGLs were developed for soils, buried pipe, basement surfaces, basement penetrations, and basement embedded pipe. The dose from basement surfaces, penetrations, and embedded pipe will be summed to determine the total dose for a given basement, and the resultant basement dose will be summed with calculated soil, groundwater and buried pipe doses in order to demonstrate compliance.

The scope of the FSS plan includes the radiological assessment of all impacted sub-grade structures, buried piping, and open land areas that will remain following decommissioning. The licensee indicates in the LTP that it is their 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 ISFSI.

Section 5 of the LTP indicates that the following major structures will remain at license termination and will require FSS: 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 (including the Spent Fuel Pool), 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). The licensee intends to remove all structures above the 588 foot elevation during the decommissioning process and dispose of them as waste, with the exception of several minor structures such as the Switchyard, the ISFSI warehouse, the microwave tower, and the Sewage Lift Station, as well as all roadways and rail lines.

The licensee discusses the planned demolition of several structures and their anticipated end state in the LTP as follows. All concrete will be removed from the interior side of the steel liner in both containment buildings (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. All interior walls and floors will be removed from the Auxiliary Building, leaving only the exterior walls and basement floor. Only the reinforced concrete floors and exterior foundation walls and the sub-grade portions of the pedestals below the 588 foot elevation will remain in the Turbine Building basement. The only portion of the Fuel Handling Building (FHB) 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. The LTP additionally notes that the following below ground structures will remain after decommissioning: the lower concrete portions of the Waste Water Treatment Facility (WWTF), Main Steam Tunnels, and Circulating Water Inlet Piping and Discharge Tunnels. The licensee also indicates in Section 5 of the LTP that they intend to reuse concrete from building demolition as fill material, and notes that “concrete that meets the non-radiological definition of Clean Concrete Demolition Debris and where radiological surveys demonstrate that the concrete meets the criteria for unconditional release will be used.”

NRC staff notes there are several points to consider with regard to how reused materials are assessed in terms of 10 CFR 20, Subpart E. These considerations are further discussed in Sections 3.5.7 and 3.5.12.2 of this SER.

3.5.2 Radionuclides of Concern and Release Criteria

Section 5.1 of the LTP discusses the licensee’s anticipated radionuclides of concern (ROC) and mixture fractures to be encountered during decommissioning. The licensee prepared Technical Support Document (TSD) 11-001, “Potential Radionuclides of Concern during the Decommissioning of Zion Station” in November 2011 to establish the basis for the initial suite of potential ROCs. Industry guidance and analytical results from the sampling of various media during past plant operations were utilized by the licensee to develop the ROCs. Additionally, some of the theoretical neutron activation products, noble gases and radionuclides with a half-life less than two years were eliminated. The initial list of potential ROCs was provided in Table 5-1 of the LTP. A final suite of ROCs was developed using the results of concrete core analyses from the Containment and Auxiliary Building, as was documented in TSD 14-019, “Radionuclides of Concern for Soil and Basement Fill Model Source Terms.” The licensee determined several insignificant dose contributors based upon the guidance contained in Section 3.3 of NUREG-1757. The suite of dose significant ROCs for use in decommissioning was provided in Table 5-2 of the LTP.

An evaluation of the process used to derive the final suite of dose significant ROC and to eliminate insignificant dose contributors from the ROC is also addressed in Section 3.6 of this SER.

In Section 5.2 of the LTP, the licensee discusses the FSS criteria that will be used to demonstrate compliance with the radiological criteria for unrestricted use, as specified in 10 CFR 20.1402. As previously noted, the release criteria for the FSS program will be concentration based DCGLs (e.g., pCi/g for land areas, pCi/m2 for basement surfaces, and dpm/100 cm2 for buried piping internal surfaces), with Base Case DCGLs calculated to demonstrate compliance with the 25 mrem/yr dose limit.

In order to ensure that the summation of dose from each source term is 25 mrem/yr or less after FSS is completed, the licensee reduced the Base Case DCGLs and developed “Operational”

DCGLs to be used for FSS planning and implementation. These Operational DCGLs were set to an expected fraction of the compliance dose based on the results of site characterization, process knowledge and the extent of planned remediation. Additional details on the development of the Operational DCGLs are provided in Chapter 6 of the LTP and in TSD 17-004, “Operational Derived Concentration Guideline Levels for Final Status Survey.”

Compliance will ultimately be demonstrated through the summation of dose from four distinct source terms for the end-state (basements, soils, buried pipe and groundwater), as shown in Equation 5-3 of the LTP. Additionally, basement dose used in this calculation will be comprised of four structural source terms (surfaces, embedded pipe, penetrations and fill). Operational DCGLs and Base Case DCGLs are provided in Section 5.2 of the LTP for basement surfaces, soils, buried piping, embedded pipe, and penetrations. Section 5.2.2 of the LTP discusses the development of surface and subsurface DCGLs. The licensee has defined surface soil at the site as the first 0.15 m layer of soil and subsurface soil as the layer of soil beginning at the surface and extending to 1 m. The licensee calculated site-specific DCGLs for both the 0.15 m and 1m thicknesses.

As the licensee plans to survey structures prior to surveying soils, there is an indication in the LTP that once the FSS of structures is complete, the Operational DCGLs for soils and 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.

Since multiple ROCs exist at the site, a sum of fractions (unity rule) approach will be utilized for FSS. The LTP also indicates that the Base Case DCGL, which is established for the average residual radioactivity in a survey unit, is equivalent to a $DCGL_W$ (as typically described in MARSSIM guidance). In that sense, the $DCGL_W$ can be multiplied times an Area Factor to obtain a Base Case DCGL that represents a dose of 25 mrem/yr to an individual as a result of a smaller area of contamination within a survey unit. This LTP defines this concept 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. The usage of Area Factors and for soil is described in Sections 5.2.15 and 6.11 of the LTP.

The licensee discusses surrogate radionuclides in Section 5.2.11 of the LTP. This process is used to develop a ratio between an easy to detect radionuclide and a hard to detect (HTD) for use in compliance measurements. Section 5.2.11 notes that “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,” and “this is accomplished through the application of a surrogate relationship.” For FSS surveys, a modified DCGL is required as described in LTP Equation 5-1 (per Section 4.3.2 of MARSSIM). A modified sum-of-fractions (unity rule) approach is also required when multiple ROCs exist. The licensee notes in Section 5.2.12 of the LTP that the use and application of the unity rule will be in accordance with Section 4.3.3 of MARSSIM. The licensee also notes in Section 5.2.11 of the LTP that the final ROCs for decommissioning 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. The licensee intends to infer HTD concentrations for Ni-63, Sr-90, and H-3 during FSS using the surrogate approach. Cs-137 will be the principle surrogate radionuclide for H-3 and Sr-90 and Co-60 will be the principle surrogate radionuclide for Ni-63. The licensee established a mean, maximum and 95% Upper Confidence Level (UCL) of the surrogate ratios for concrete core samples taken in the Containment and Auxiliary

Building basements during characterization, as described in the licensee's TSD 14-019 document (and presented in Table 5-15 of the LTP). Section 5.2.11 of the LTP indicates that the licensee will used the maximum ratios for FSS surrogate calculations unless area specific ratios are determined by continuing characterization. In Section 5.1 of the LTP, the licensee indicates that "survey unit-specific surrogate ratios, in lieu of the maximum ratios from section 5.2.11 Table 5-15, may be used for compliance if sufficient radiological data exists to demonstrate that a different ratio is representative for the given survey unit," and that "in these cases, the survey unit-specific radiological data and the derived surrogate ratios will be submitted to the NRC for approval."

NRC staff evaluated the licensee's proposed radionuclides of concern and release criteria in accordance with acceptance criteria from NUREG-1757, Vol. 2, Rev. 1, Section 4.1 and NUREG-1700, Rev. 1, Section 5. The staff finds that the radionuclides of concern are reasonable and appropriate for the site and that the release criteria are adequate to demonstrate compliance with the "Radiological Criteria for Unrestricted Use," per 10 CFR 20.1402. Further discussion on the derivation of the licensee's release criteria is provided in Section 3.6 of this SER.

3.5.3 Summary of Characterization Survey Methods and Results

Characterization survey results are summarized in Section 5.3 of the LTP and a discussion of the characterization results as they relate to the FSS is provided. Section 5.3.1 of the LTP discusses surveys of impacted media and notes that characterization of the impacted and non-impacted open land survey units (as designated by the Zion Historical Site Assessment) as well as the structural building basements that would remain and be subjected to FSS was completed in October of 2013. This characterization effort included the following land area surveys: scanning of 145,730 m^2 of surface soil, the analysis of 1,037 surface soil samples and 699 subsurface samples, and taking 282 static measurements on surface soils using a Canberra In Situ Object Counting System (ISOCS). Structural surveys were completed as follows: 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.

Chapter 2 of the LTP provides a more extensive discussion of the previous characterization programs along with a discussion of the Historical Site Assessment and contamination history. The characterization survey design is discussed in Section 2.2.2 of the LTP, and the licensee notes that a graded approach using Data Quality Objectives (DQOs) was established for the design. The licensee used the MARSSIM DQO process to design each survey unit, and notes that "for example, an open land survey unit was designated as Class 1 because it may contain levels of radiological contamination greater than the unrestricted release criteria," and that "characterization surveys that were performed in a Class 1 survey unit focused on bounding the contamination where contamination was potentially present." The survey design for areas classified as non-impacted, Class 2 or Class 3 included a combination of systematic and biased survey measurement locations and scan areas. Biased surveys were designed using known information to select locations for static measurements and/or samples, and systematic surveys were designed to select static measurement and/or sample locations at random or by using a systematic sampling design with a random start. The licensee used the DQO process to establish the appropriate design for each survey unit. The licensee notes that "a biased approach was warranted when the characterization effort was designed to delineate the extent of an area that requires remediation," and "a systematic approach was warranted if the characterization effort was designed to verify the basis for the classification of a survey unit."

Since site-specific DCGLs were not established at the time the characterization survey was performed, alternate action levels were selected. The licensee used screening DCGLs as presented in NUREG-1757 and the concentration values found in NUREG/CR-5512 Volume 3, "Residual Radioactive Contamination from Decommissioning Parameter Analysis", Table 6.91 (Pcrit = 0.10) for soils. For structures, a 7,100 dpm/100 cm^2 action level (which is the nuclide-specific screening value for total gross beta-gamma surface activity of Co-60 from NUREG-1757, Appendix H) was used. As DCGLs were not developed at the time, the usage of this screening level was considered conservative since there is an assumption of 100% Co-60 (Cs-137 is anticipated as a surface contaminant, but also has a higher screening level listed in NUREG-1757).

Section 2.2.2.1 of the LTP discusses the licensee's strategies for determining the number of characterization samples as follows:

- For the characterization of structural survey units that would not remain at license termination and not be subjected to FSS, the numbers of static measurements and/or samples taken were a sufficient quantity to determine the general radiological condition of the survey unit, including average and maximum concentration of loose surface contamination and total surface contamination if possible.
- For the characterization of impacted Class 1 open land areas and Class 1 basement structures that will remain and be subjected to FSS, the sample size was based upon the necessary number of samples needed to assess the lateral and vertical extent of the contamination.
- For the characterization of impacted Class 2 open land areas and impacted Class 2 or Class 3 impacted basement structures that will be subjected to FSS, the minimum number of static measurements and/or samples that were taken in each survey unit was commensurate with the probability of the presence of residual radioactive contamination in the survey unit.

For non-impacted and Class 3 open land survey units, the primary characterization DQO was to validate the basis of the classification. Section 2.2.2.3 of the LTP discusses scan coverage and notes that survey units were scanned to the extent practical in accordance with their classification, and that the area scanned was contingent upon the accessibility of the surface areas in the survey unit and the recommended scan coverage guidelines which were presented in Table 2-5 of the LTP (which is re-created below as Table 3.5-1 of this SER).

Table 3.5-1: Scan Coverage Guidelines for Characterization

Area Classification	**Recommended Characterization Scan Coverage**
Class 1	No scanning required unless compelled by a specific survey objective.
Class 2	50% to 100%, concentrating on areas with an increased probability of exhibiting elevated activity (such as Class 1 boundaries, vehicle transit routes, etc.).
Class 3	10% to 50%, with emphasis on areas that were used for plant activities during operation and areas downwind or downstream of known effluent release points.

Non-Impacted	1% to 5%, with emphasis on areas adjacent to impacted areas.

The types of characterization surveys for land areas and building surfaces are discussed in Section 2.2.2.4 of the LTP. These consisted of a combination of surface scans (beta and gamma), static beta measurements, and material samples/smears for building surfaces. For any concrete and/or asphalt-paved open land areas that will remain and be subjected to FSS combination of surface scans (beta and gamma), static beta measurements, and volumetric samples were performed. Surveys of open land areas consisted of gamma scans and the sampling of surface and subsurface soil, sediment and surface water for isotopic analysis. Details on static measurements, beta surface scans, gamma surface scans, removable surface contamination, concrete core sampling, and material sampling were addressed in subsections of Section 2.2.2.4 of the LTP, as summarized below.

The licensee performed static measurements to detect direct contamination levels on structural surfaces of the buildings or on concrete or asphalt paved areas, and these were primarily performed using ~126 cm^2 scintillation or gas-flow proportional detectors. Static measurements were conducted by placing the detector on or very near the surface to be counted and acquiring data over a pre-determined count time. Instrument count times were adjusted as appropriate to achieve an acceptable minimum detectable concentration (MDC) for static measurements.

The licensee performed scanning to locate areas of residual activity above the 7,100 dpm/100 cm^2 action level. Beta scans were performed over accessible structural surfaces including: floors, walls, ceilings, roofs, asphalt and concrete paved areas. Floor monitors with large area gas-flow proportional detectors (typically 584 cm2) were used for floor and other larger accessible horizontal surfaces. Hand-held beta scintillation and/or gas-flow proportional detectors (typically 126 cm2) were used for surfaces not accessible by a floor monitor. The licensee performed beta scanning with the detector position maintained within 1.27 cm (0.5 inch) of the surface and with a scanning speed of one detector active window per second. If scanning at the specified standoff distance was not possible because of surface conditions, the detection sensitivity for an alternate distance was determined, and the scanning technique adjusted accordingly. Scanning speed was calculated prior to the surveys to ensure that the MDC for scanning was appropriate for the stated objective of the survey. Technicians monitored the audible response of instruments when not affected by ambient noise and flagged areas of elevated contamination for additional investigation or decontamination.

The licensee performed gamma scans over open land surfaces, typically using 2” x 2” sodium iodide (NaI) gamma scintillation detectors. The response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 when used for scanning surface soils was described in ZionSolutions TSD 11-004, “Ludlum Model 44-10 Detector Sensitivity.” The licensee performed gamma scans by moving the detector in a serpentine pattern, while advancing at a rate not to exceed 0.5 m (20 in) per second. The distance between the detector and the surface was maintained within 15 cm (6 in) of the surface if possible. Audible signals were monitored, and locations of elevated direct levels were flagged for further investigation

The licensee performed removable beta contamination or smear surveys where applicable to verify that loose surface contamination is less than an action level of 1,000 dpm/100cm2. To accomplish this, the licensee sampled a 100 cm2 surface area using a circular cloth or paper filter, under moderate pressure. These smears were analyzed for the presence of gross beta activity using a proportional counting system or equivalent.

The licensee utilized concrete core boring and sampling to assess contamination in concrete walls and floors that will remain and be subjected to FSS. A diamond bit core drill was used for concrete sampling, and the sample was typically sliced into ½-inch wide "pucks" to assess contamination at depth. Static measurements were performed on the top and bottom of the pucks to determine contaminant intrusion depth and/or the activation of the concrete matrix. Concrete pucks were also pulverized and analyzed for isotopic content.
The licensee obtained samples of soil, sediment, and sludge from judgmental and systematic sample locations per survey designs, as well as other biased locations of elevated activity identified by scanning. Surface soil was collected using a split spoon sampling system or, by using hand trowels, bucket augers, or other suitable sampling tools, while subsurface soil was sampled by direct push sampling systems (e.g. GeoProbe®) or by the excavation of test pits.

Field instrumentation and sensitivities are discussed in Section 2.2.3 and summarized in Section 5.3.2 of the LTP. Section 5.3.2 of the LTP reiterates the characterization action levels and notes that "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." Instrumentation and nominal MDC values that were employed during characterization are listed in Table 2-6 of the LTP (which is re-created below as Table 3.5-2 of this SER).

Table 3.5-2: Instrument Types and Nominal MDC

Detector Model [b]	**Meter Model**	**Application**	**Typical Detection Sensitivity**	
			MDC_{scan} (dpm/100 cm^2)	**MDC_{static}** [a] **(dpm/100 cm^2)**
Ludlum 44-9	Ludlum 2350-1	β static & scan	2900	985
Ludlum 43-5	Ludlum 2350-1	α static & scan	150	75
Ludlum 43-68 β mode	Ludlum 2350-1	β static & scan	1050	330
Ludlum 43-68 α mode	Ludlum 2350-1	α static & scan	170	70
Ludlum 44-116	Ludlum 2350-1	β static & scan	1300	415
Ludlum 43-90	Ludlum 2350-1	α static & scan	130	55
Ludlum 44-10	Ludlum 2350-1	γ scan	3.5 pCi/g ^{60}Co 6.5 pCi/g ^{137}Cs	N/A
Ludlum 43-37	Ludlum 2350-1	β scan	1000	N/A
Tennelec LB5100 proportional counting system	N/A	α and/or β smear	N/A	18
HPGe Gamma Spectroscopy System [c]	N/A	γ Analysis	N/A	~0.15 pCi/g for Co-60 and Cs-137

a Based on 1-minute count time; and default values for surface efficiencies (ε_s) as specified in International Standard, ISO 7503-1 (1998)
b Functional equivalent instrumentation may be used
c MDC Requirements per Regulatory Guide 4.8

Laboratory instrument methods and sensitivities are summarized in Section 5.3.3 and Section 2.2.4 of the LTP. Table 2-7 of the LTP lists typical analytical methods employed and the laboratory MDC achieved by the off-site vendor laboratories used during characterization, and is re-created in Table 3.5-3 of this SER.

Table 3.5-3: Typical Vendor Laboratory Standard MDC Values

Test	Technique	Method	Solid (pCi/g)	Water (pCi/L)
Gamma radionuclides	Gamma Spectroscopy	LANL EM-9	0.1	10
Alpha	Gas Flow Proportional	EPA 900.0	4.0	5.0
Beta	Gas Flow Proportional	EPA 900.0	10.0	5.0
H-3	Liquid Scintillation	EPA 906.0 Mod	6.0	700
C-14	Liquid Scintillation	EPA EERF C	2.0	50.0
Fe-55	Liquid Scintillation	DOE RESL Fe-1	5.0	100.0
Ni-59	Low Energy Gamma Spectroscopy	DOE RESL Ni-1	10.0	20.0
Ni-63	Liquid Scintillation	DOE RESL Ni-1	4.0	50.0
Sr-90	Gas Flow Proportional	EPA905.0 Mod	2.0	2.0
Tc-99m	Liquid Scintillation	DOE EML HASL 300	5.0	50.0
Pm-147	Liquid Scintillation	EPA EERF PM-1-1	10	10
Np-237	Alpha Spectroscopy	DOE EML HASL	0.5	1.0
Pu-238-240	Alpha Spectroscopy	DOE EML HASL 300	0.5	1.0
Pu-241	Liquid Scintillation	DOE EML HASL 300	15.0	15.0
Am-241 & 243	Alpha Spectroscopy	DOE EML HASL 300	0.5	1.0
Pu-242	Alpha Spectroscopy	DOE EML HASL 300	0.5	1.0
Cm-242-246	Alpha Spectroscopy	DOE EML HASL 300	0.5	1.0

The need for additional ongoing characterization has also been identified, and it is noted in Section 5.1 of the LTP that the characterization surveys of several inaccessible or not readily accessible subsurface soils or structural surfaces have been deferred until safe access is available. Additional details on those areas were provided in Sections 2.5 and 5.3.4.4 of the LTP. The licensee has also committed to obtain and analyze concrete core and soil samples during continuing characterization (including radiological assessments) and FSS in order to verify that the dose from Insignificant Contributors (ICs) does not change prior to FSS and to verify that surrogate ratios used for HTD radionuclide calculations are still valid. Section 5.1 of the LTP describes the process that will be utilized to sample for HTDs during continuing characterization. This includes analyzing for HTDs in at least 10% of all media samples collected in a survey unit during continuing characterization, and additionally a minimum of one sample beyond the 10% will be selected at random to also be analyzed for HTDs. The HTD analyses performed during continuing characterization will be for the full suite of radionuclides, as shown in Table 5-1 of the LTP. The IC contribution will also be assessed for each individual sample result using the DCGLs from TSD 14-019 Table 27 for structures and Table 28 for soils. The LTP notes that “if the IC dose calculated is less than the IC dose assigned for DCGL adjustment (1.25 mrem/yr for all basement structures other than the Containments and 2.5 mrem/yr for the Containments and soils), then no further action will be taken,” and that “if the actual IC dose calculated from the sample result is greater than the IC dose assigned for DCGL adjustment, then a minimum of five (5) additional investigation samples will be taken around the original sample location.” The actual calculated maximum IC dose from an individual investigation sample would then be used to readjust the DCGLs in the respective survey unit, and if the maximum IC dose exceeds 10%, the additional radionuclides that were the cause of the IC dose exceeding 10% will be added as ROCs for that survey unit.

The licensee has additionally indicated in Section 5.1 of the LTP that 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). Section 5.4.1 of the LTP indicates that radiological assessments of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation, as described in Table 5-27 of the LTP, and samples will also be collected from potentially impacted soil, sediment and/or surface residues for laboratory analysis. In that case, the licensee commits in Section 5.1 of the LTP to analyze 10% of any soil samples collected during the radiological assessment of a survey area (with a minimum of one sample) for the full initial suite of radionuclides. It is additionally noted in Section 5.1 of the LTP that 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.

Section 5.3.4.4 of the LTP further describes the continuing characterization processes and notes that "surveys will be performed in accordance with ZionSolutions ZS-LT-02, 'Characterization Survey Plan' (Reference 5-18) using the same processes, quality, instruments, plans and procedures as described in section 2.2 of Chapter 2," and that "continuing characterization surveys will be designed to gather the appropriate data using the DQO process as outlined in MARSSIM, Appendix D." The LTP also notes that, at a minimum, continuing characterization will takes place in the following areas:

- The underlying concrete of the SFP/Transfer Canal below the 588 foot elevation after the steel liner has been removed;
- The concrete walls and floor of the Under-Vessel areas in Unit 1 and Unit 2 Containments;
- The floors and walls of the Hold-Up Tank (HUT) cubicle;
- The floor of the Auxiliary Building 542 foot elevation Pipe Tunnel floors;
- The floor and lower walls of the 542 foot elevation of the Auxiliary Building;
- The subsurface soils in the "keyways" between the Containment Buildings and the Turbine Building;
- The soils under the basement concrete of the Containment Buildings, the Auxiliary Building and the SFP/Transfer Canal;
- When the interior surfaces become accessible, several potentially contaminated embedded and buried pipe systems that will be abandoned in place will be characterized;
- The Containment basements, after concrete removal.

NRC staff evaluated the licensee's characterization survey methods and results in accordance with acceptance criteria from NUREG-1757, Vol. 2, Rev. 1, Section 4.2 and NUREG-1700, Rev. 1, Section 2. The staff evaluated completed characterization at the time of the LTP submittal. The staff also understands that certain areas of the site were inaccessible or not readily able to be characterized prior to remediation or at the time of the LTP submittal. In these cases, the licensee has proposed plans for continuing characterization, which were evaluated by the staff.

The staff finds that the completed radiological characterization of the site and the proposed continuing characterization plans are adequate to permit planning for a remediation that will be effective and will not endanger the remediation workers, to demonstrate that it is unlikely that significant quantities of residual radioactivity has not gone undetected, and to provide information that will be used to design the final status survey. As such, the staff finds that the licensee's site characterization methods are adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(A) and with 10 CFR 20.1501(a) and (b).

3.5.4 Decommissioning Support Surveys

Section 5.4 of the LTP discusses decommissioning support surveys at Zion, including Radiological Assessment surveys, Remedial Action Support (In-Process) Surveys, Contamination Verification Surveys (CVS), and Post-Demolition Surveys.
Radiological Assessment surveys are addressed previously in Section 3.5.4 of this SER. According to Section 5.4 of the LTP, Remedial Action Support Surveys will be conducted to guide remediation activities, determine when an area or survey unit has been adequately prepared for the FSS, and 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. In a similar fashion to "Radiological Assessment" surveys, Remedial Action Support Surveys will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation (as described in Table 5-27 of the LTP) and samples will also be collected from potentially impacted soil, sediment and/or surface residues for laboratory analysis. Instrumentation and field screening methods for Remedial Action Support Surveys are also provided in Section 5.4.3 and 5.4.4 of the LTP.

Section 5.4.5 of the LTP describes Contamination Verification Surveys of basement structural surfaces, which will be performed to identify areas requiring remediation to meet the open air demolition limits presented in ZionSolutions TSD 10-002, "Technical Basis for Radiological Limits for Structure/Building Open Air Demolition." These surveys will be performed using hand-held beta-gamma instrumentation as presented in Table 5-27 of the LTP. The licensee will determine scan coverage based on the contamination potential of the structural surface being surveyed, and Class 1 survey units will require 100% scan coverage of all exposed concrete surface areas. The LTP indicates that any areas identified in excess of the open air demolition limits will be earmarked for remediation. For basement surfaces below the 588 foot elevation (which will remain and be subjected to FSS), the licensee also commits to remediate areas to ensure that any individual In Situ Object Counting System (ISOCS) measurement will not exceed the Operational $DCGL_B$ from LTP Table 5-4 during FSS. The LTP additionally indicates that any area of elevated activity that could potentially approach the Operational $DCGL_B$ will be identified as a location for a judgmental ISOCS measurement during FSS.

Post-demolition surveys are described in Section 5.4 of the LTP as additional scan surveys that will be performed (following demolition) to ensure that any individual ISOCS measurement will not exceed the Operational $DCGL_B$ from LTP Table 5-4 during FSS. These will be performed using hand-held beta-gamma instrumentation as presented in Table 5-27 of the LTP.

NRC staff evaluated the licensee's decommissioning support survey methods in accordance with acceptance criteria from NUREG-1757, Vol. 2, Rev. 1, Section 4.3 (Remedial Action Support Surveys). The staff finds that the plans for decommissioning support surveys are consistent with the NRC's guidance on remedial action support surveys, and the plans are adequate to assist the licensee in determining when remedial actions have been successful and that the FSS may commence. As such, the staff finds that the licensee's decommissioning support survey methods area adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D) and with 10 CFR 20.1501(a) and (b).

3.5.5 Final Status Survey

3.5.5.1 Final Status Survey of Basement Structures

Final Status Surveys for basement structures are described in Section 5.5 of the LTP, where basement structures are defined as basement surfaces (concrete and steel liner), embedded pipe, and penetrations. After the licensee has remediated basement structures to ensure that the remaining floor and wall concrete surfaces are below the Operational $DCGL_B$ as measured by ISOCS an FSS will be completed to demonstrate compliance with the 25 mrem/yr dose criterion for unrestricted release (per 10 CFR 20.1402).

Section 5.5 of the LTP indicates that the Canberra ISOCS will primarily be used to perform FSS of basement surfaces, as "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." However, the licensee has committed to performing additional core samples during additional characterization and during FSS, as discussed in Sections 5.1 of the LTP. With regard to FSS, it is noted in Section 5.1 of the LTP that "the number of cores collected and analyzed for ROC HTD will be ten percent (10%) of the FSS ISOCS measurements."

According to Section 5.5.2 of the LTP, the basement surface FSS units will be comprised of the combined wall and floor surfaces of each remaining building basement, which includes the Auxiliary Building, Unit 1 Containment, Unit 2 Containment, the Turbine Building, the Crib House/Forebay, the WWTF and remnants of the SFP/Fuel Transfer Canal. The Containment Buildings will be further divided into two separate surface survey units, one for the walls and floors of the exposed steel liner and one for the Under-Vessel area where concrete will remain.

Section 5.5.2 of the LTP indicates that that activity in the Circulating Water Intake Pipes, Circulating Water Discharge Tunnels, Circulating Water Discharge Pipes, and Buttress Pits/Tendon Tunnels will be included with the Turbine Building survey unit through the associated DCGL calculation. The activity in the Circulating Water Intake Pipe will also be included with the Crib House/Forebay survey unit through that associated DCGL calculation. The Circulating Water Discharge Tunnels will be addressed as a separate survey unit within the Turbine Building. Section 6.6.8 of the LTP provides additional details on these DCGL calculations.

FSS classification for basement surfaces is discussed in Section 5.5.2.1 of the LTP, where it is noted that the "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," which will be "evaluated by the potential for an individual ISOCS measurement to exceed the Operational $DCGL_B$." The licensee also notes in Section 5.5.2.1 of the LTP that multiple sources of radiological status information will be used to validate survey unit classification. Specifically, the LTP indicates that "extensive surface scan surveys, in some cases 100% of the surface area, will be performed during Contamination Verification Survey (CVS)," and that "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."

ISOCS areal coverage determination for basement surfaces is discussed in Section 5.5.2.1 of the LTP, where it is noted that the graded approach in selecting scan surveys from MARSSIM Section 2.2 will be utilized in determining ISOCS areal coverage for survey units. The licensee also refers to MARSSIM Table 5.9, which discusses recommended survey coverage for structures and land areas, in determining the ISOCS area coverage. As such, the ISOCS areal surveys will be used by the licensee in place of scanning surveys recommended by MARSSIM.

The determination of FSS sample sizes is discussed in Section 5.5.2.2 of the LTP. The licensee calculates minimum sample sizes two ways; first, based upon the simple quotient of the required areal coverage surface area divided by the ISOCS Field of View (FOV), and second, by using methods described in MARSSIM Section 5.5. The larger of the two values is used for FSS planning purposes. Additionally, the number of ISOCS samples is further adjusted for Class 1 areas to ensure that 100% coverage was achieved, as the ISOCS FOV is circular and must overlap to achieve the required coverage. Table 5-19 of the LTP presents the survey unit classifications and the “Adjusted Minimum Number of ISOCS Measurements per FSS Unit,” as re-created in this SER as Table 3.5-4. It is worth noting that the survey unit sizes for Class 1 structures, as shown in Table 5-19 of the LTP, exceed the MARSSIM recommended survey unit sizes. However, the sampling density is such that 100% ISOCS areal coverage is achieved in these basement survey units. While there are currently no Class 2 areas proposed for basement structures, the licensee notes that “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 [LTP] Chapter,” and “the number of measurements will also be increased in survey units that exceed 1,000 m2 to correspond with the MARSSIM recommended survey size density for a Class 2 structure (measurements/1,000 m2).”

Table 3.5-4: 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(1)	6,503	100%
Unit 1 CTMT above 565 foot elevation	Class 1	2,465	155(1)	2,465	100%
Unit 1 CTMT Under-Vessel Area	Class 1	294	19(1)	294	100%
Unit 2 CTMT above 565 foot elevation	Class 1	2,465	155(1)	2,465	100%
Unit 2 CTMT Under-Vessel Area	Class 1	294	19(1)	294	100%
SFP/Transfer Canal	Class 1	723	45(1)	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 1	1,124	71(1)	1,124	100%

(1) 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).

A discussion on data assessment for basement surface FSS results is provided in Section 5.5.4 of the LTP. The licensee plans to utilize a sum of fractions (SOF) approach 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 SOF for gamma-emitting ROCs will be based on the measured activity (along with any other gamma emitting radionuclides positively detected by the ISOCS), and the SOF for HTD ROCs will be inferred utilizing the surrogate approach from Section 5.2.11 of the LTP. All individual ROC fractions will be summed to provide a total SOF value for each measurement.

At the time of this SER, the licensee identified some challenges with using surrogates in areas where remediation has occurred, particularly in remediated under-vessel areas of the containment buildings. In these cases, the gamma emitting ROCs were remediated while HTDs remained as the dominant dose contributors. As the surrogate ratio is no longer valid, it becomes necessary for the licensee to perform discrete sampling for the HTDs. Since there is no way to perform scanning for the HTDs, considerations for sampling density must be taken into account. It is additionally important that continuing characterization sampling ensure that the full extent of HTD contamination is understood. These situations will be subject to ongoing review and inspection to ensure consistency with NRC survey guidance and to ensure that adequate surveys are performed to demonstrate compliance.

Section 5.5.4 of the LTP notes that the Sign Test will be used to evaluate the residual radioactivity against the compliance dose criterion of 25 mrem/yr, and the SOF for each measurement will be used as the sum value for the Sign Test. The licensee further notes that "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," and that "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."

Methods for evaluating elevated areas of activity in basements are also described in Section 5.5.4 of the LTP. For building surfaces, the licensee defines areas of elevated activity as "any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL," and indicates that "any area that exceeds the Base Case DCGL will be remediated." Section 5.5.4 of the LTP notes that the "SOF (based on the Operational DCGL) for a systematic or a judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign Test and, the mean SOF (based on the Operational DCGL) for the survey unit does not exceed one." If the Sign Test (using Operational DCGLs) is passed, the following equation (Equation 5-5 of the LTP) will be used (with Base Case DCGLs) to assess the SOF for basement structural surface survey units:

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

where:

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

The SOF_B will then be multiplied by 25 mrem/yr in order to assign the dose from residual radioactivity to the FSS unit.

Surveys of embedded piping and penetrations are described in Section 5.5.5 of the LTP, and the associated Operational DCGLs are provided in Tables 5-13 and 5-14 of the LTP, respectively. The LTP notes that “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,” while “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.” Timed measurements will be made at specified distances within the pipe/sleeve in accordance with survey areal coverage requirements.

According to the LTP, gamma surface activities for piping and penetration measurements will be converted to gamma measurement results (in units of pCi/m2) for each gamma ROC based on the mixture applicable to the pipe/sleeve surveyed, and HTD ROCs will be inferred for each measurement. Dose fractions for piping and penetrations, relative to the applicable Operational DCGL, will be developed, and individual ROC dose fractions will be summed to produce an SOF for the measurement.

Section 5.5.5 of the LTP addresses elevated areas of activity for embedded piping/penetrations and indicates that for “embedded pipe and penetrations, areas of elevated activity will be defined as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL,” and “any area that exceeds the Base Case DCGL will be remediated.”

Section 5.5.5 of the LTP further notes that the “SOF (based on the Operational DCGL) for a systematic or a judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign Test and, the mean SOF (based on the Operational DCGL) for the survey unit does not exceed one.” If the Sign Test (using Operational DCGLs) is passed, the following equation (Equation 5-6 of the LTP) will be used (with Base Case DCGLs) to assess the SOF for basement embedded pipe or penetration FSS units.

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

where:

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

The SOF will then be multiplied by 25 mrem/yr in order to assign the dose from residual radioactivity to the FSS unit.

Because of the relatively small surface area within embedded piping and penetrations, the associated Operational DCGLs for piping/penetrations are higher than the DCGLs for the floors and walls of the basements. In order to address the potential for the release of contamination from piping at activity levels above the surrounding basement, the licensee proposes several remediation and grouting strategies in Section 5.5.5 of the LTP as follows:

- If maximum activity exceeds the Base Case DCGLEP 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 DCGLEP 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 DCGLEP, 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.

The licensee provides similar remediation and grouting strategies for penetrations, as follows:

- If maximum activity exceeds the Base Case DCGLPN 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 DCGLPN 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 DCGLPN, 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.

A description of the summation of dose for basement structures is provided in Section 5.5.6 of the LTP, and this topic is discussed in greater detail in Chapter 3.6 of this SER. Equation 5-7 of the LTP sums basement doses as follows:

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

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

3.5.5.2 Final Status Survey Design

Final Status Survey (FSS) design is presented in Section 5.6 of the LTP, where it is noted that the design discussion “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.”

Survey planning is discussed in Section 5.6.1 of the LTP, and the licensee indicates that 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 LTP notes that the FSS process will include the following four principal elements: planning, design, implementation, and data assessment. The usage of the Data Quality Objectives (DQO) and Data Quality Assessment (DQA) processes is additionally described in the LTP. The licensee commits to utilizing the DQO process from MARSSIM Appendix D, which includes the following actions: state the problem, identify the decision, identify inputs to the decision, define the study boundaries, develop a decision rule, specify limits on decision errors, and optimize the design for obtaining data.

The licensee discusses the FSS design process in Section 5.6.4 of the LTP, and indicates that the concepts on sampling size determination and scanning coverage will be implemented as described in MARSSIM and NUREG-1757. Decision errors are addressed, and the licensee commits to setting Type I and II decision errors as follows:

- 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.

The licensee notes in Section 5.6.4.1.6 of the LTP that the relative shift calculations will utilize Operational DCGLs in planning the number of required FSS measurements.

Section 5.6.4.2 of the LTP discusses statistical tests to evaluate FSS results, and the licensee indicates that "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."

Small areas of elevated activity are discussed in Section 5.6.4.3 of the LTP and investigation levels are presented as a comparison to the DCGLEMC (a DCGL modified by an area factor to account for small areas of elevated activity). The licensee notes 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 DCGLw," and that "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." Section 5.6.4.3 of the LTP proceeds to discuss the fact that sampling sizes may need to be adjusted to ensure that FSS scan surveys are able to adequately detect the DCGLEMC, and the licensee commits to using methods from Section 5.5.2.4 of MARSSIM for that purpose.

FSS scan coverage is discussed in Section 5.6.4.4 of the LTP, and the licensee indicates that Table 5-9 of MARSSIM was utilized to determine the recommended survey coverage. This coverage is provided in Table 5-24 of the LTP and is shown in Table 3.5-5 of this SER.

Table 3.5-5: 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

The licensee indicates in Section 5.6.4.5 of the LTP that reference grids and sampling locations will be developed in accordance with MARSSIM Sections 4.8.5 and 5.5.2.5 and provides the methods which will be used to determine sampling locations during FSS.

The licensee’s FSS investigation process is described in Section 5.6.4.6 of the LTP where it is noted that any areas of concern will be investigated during FSS, and “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.” FSS investigation levels are presented in Table 5-25 of the LTP, and are shown in Table 3.5-6 of this SER.

Table 3.5-6: Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	>Operational 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 $DCGL_W$

Depending upon FSS results and the results of any investigations, there may be a need for remediation, reclassification, and resurveys, and these concepts are described in Section 5.6.4.6.1 of the LTP. Remediation, reclassification, and resurvey actions are also described in Table 5-26 of the LTP.

3.5.5.3 Final Status Survey Implementation

Section 5.7 of the LTP describes FSS implementation and discusses the various FSS survey methods, including scanning, volumetric sampling, and fixed measurements. Specific sampling strategies for surface and subsurface soils are provided in Sections 5.7.1.5 and 5.7.1.6, respectively. The licensee defines surface soil as “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),” and defines subsurface soil as “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.” Surface soil samples will be taken for FSS of land areas at designated systematic locations and at areas of elevated activity identified by gamma scans. Subsurface samples will be sampled during FSS in Class 1 open land areas at 10% of the systematic soil sampling locations, with the location(s) selected at random. Section 5.7.1.6.2 of the LTP additionally notes that “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.”

Sampling of subsurface soils below basement structure foundations is discussed in Section 5.7.1.6.3 of the LTP. The licenses notes that "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.” Strategies to sample subsurface soils below basement structures include soil borings or Geoprobe® sampling and will be biased to locations having a high potential for the accumulation and migration of radioactive contamination. The licensee also commits in Section 5.7.1.6.3 of the LTP that “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." Additionally, the licensee indicates that survey designs will also consider the possibility of coring through the basement concrete floor slabs to facilitate the collection of soil samples. In order to address any potential for groundwater intrusion through core borings while assessing the potential for migration of contamination from buildings to sub-foundation soils, the licensee indicates that "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." Those core samples 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. The licensee also notes that "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," but "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." Any detection of residual radioactivity in subsurface soils adjacent to or under a basement surface will result in an investigation to assess the potential contamination of the exterior of the structure.
Reuse of excavated soils is addressed in Section 5.7.1.7 of the LTP where the licensee indicates that soil will not be excavated or stockpiled for reuse as backfill in basements, but overburden soils will be created to expose building components such as concrete pads, buried pipe, buried conduit, etc. The licensee presents survey strategies for overburden soils in Section 5.7.1.7, as follows:

- 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 [of the LTP]. A soil sample will be acquired at any scan location that indicates activity in excess of 50% of the soil Operational DCGL. Any soil confirmed as containing residual radioactivity at concentrations exceeding 50% of the soil Operational DCGL will not be used to backfill the excavation and will be disposed of as waste.

During Radiological Assessment surveys of excavated soils, the licensee would be expected to follow all commitments for Radiological Assessments, as described in Section 5.1 of the LTP (and previously discussed in Section 5.4 of this SER), including HTD analyses.

Additional survey strategies are discussed within Section 5.7 of the LTP, including those for pavement covered areas, buried piping, groundwater, sediments and surface water, and these are discussed briefly in this SER as follows.

Pavement covered areas will be incorporated into larger open land survey units and will be surveyed according to the classification of the survey units in which they are located. Surface soil DCGLs will be used for these surveys, and sample media will be pulverized and analyzed by gamma spectroscopy for comparison to the Operational DCGL. The licensee indicates in Section 5.7.1.8 of the LTP that if pavement exhibits residual radioactivity above the surface soil Base Case DCGLs then pavement will be disposed of as radiological waste, and an investigation of underlying soil will be performed.

Section 5.7.1.9 of the LTP discusses buried piping surveys and notes that designated sections of buried piping will be remediated in place and undergo FSS. The licensee provided an inventory of buried piping located below the 588 foot grade that will remain and be subjected to FSS in TSD 14-016. The licensee also indicates in Section 5.7.1.9 of the LTP that " compliance with the Operational DCGL values, as presented in Table 5-10 [of the LTP], will be demonstrated by measurements of total surface contamination and/or the collection of sediment samples," and "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 [of the LTP]."

Groundwater assessment is described in Section 5.7.1.10 of the LTP, where it is noted that "assessments of any residual radioactivity in groundwater at the site will be via groundwater monitoring wells installed at ZNPS," and "ongoing monitoring of surface water and groundwater at ZNPS include REMP, Radiological Groundwater Protection Program (RGPP) and NPDES Monitoring."

Sediments and surface water are addressed in Section 5.7.1.11 of the LTP, and the licensee notes that these samples will be evaluated against the site-specific soil Operational DCGLs for each of the potential ROCs, as shown in Table 5-7 of the LTP. Section 5.7.1.11 of the LTP indicates that "the assessment of residual radioactivity levels in surface water drainage systems will be made through the sampling of sediments, total surface contamination measurements, or both, at traps and other access points where it is expected that radioactivity levels will be representative or bounding of the residual radioactivity on the interior surfaces."

The licensee discusses considerations for buildings, structures, and equipment in Section 5.7.1.12 of the LTP. The LTP indicates that all above grade buildings will be removed, and the survey approach provided in Section 5.5 of the LTP will be used to assess the residual radioactivity on below-grade basement surfaces. Section 5.7.1.12 of the LTP also notes that the FSS of minor solid structures, such as but not limited to the Switchyard, the microwave tower, the Sewage Lift Station, telephone poles, fencing, culverts, duct banks and electrical conduit will be included in the open land FSS units in which they reside.

In Section 5.7.1.12 of the LTP the licensee discusses the intentions to reuse concrete from several demolished structures onsite and notes that prior to demolition the materials will be surveyed using the site unconditional release program with surveys that meet the statistical rigor and quality of MARSSIM. Once a satisfactory survey is completed, the licensee plans to remove all metal and crush concrete pieces to 10 inches in diameter or less. Section 5.7.1.12 of the LTP further notes that "if the unconditional release surveys positively detect plant-derived radionuclides in any concentration, then the concrete will not be used as clean fill," and "in this case, it will be segregated, packaged and disposed of as low level radioactive waste." The licensee also provided two reports with the LTP regarding reuse of concrete on site: TSD 17-007 - Evaluation of Static Measurements Performed for Unconditional Release Surveys of Building Materials Used for Backfill at the Zion Decommissioning Project, Revision 1, and TSD 17-010 - Final Report - Unconditional Release Surveys at the Zion Station Restoration Project, Revision 1. According to TSD-17-010 the following structures were surveyed and the concrete, after the structure's demolition, was utilized as backfill:

- Chlorination / Dechlorination Buildings (North and South)
- Contractor Break Building
- Crib House

- East/West Service Building
- ENC
- Fire Training Buildings
- Interim Radwaste Storage Facility (IRSF)
- Mechanical Maintenance Training Center (MMTC)
- NGET
- Old Gatehouse
- Old Sewage Lift Station
- South Gatehouse
- Turbine Building
- Unit 1 Containment Building (Exterior)
- Unit 2 Containment Building (Exterior)
- Vertical Concrete Cask (VCC) Pad

3.5.5.4 Final Status Survey Instrumentation

Section 5.8 of the LTP discusses instrument selection and indicates that the DQO process will be utilized in selecting instruments. With regard to instrument detection capability, the licensee notes that "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," and "instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the Base Case DCGL." Section 5.8.1 of the LTP additionally notes that the "target MDC for measurements obtained using laboratory instruments will be 10 percent of the applicable Operational DCGL."

The licensee's proposed FSS instrumentation is listed in Table 5-27 of the LTP, and instrument MDCs are discussed in Section 5.8.4 of the LTP, with nominal MDC values presented in Table 5-28 of the LTP. The licensee also indicates in Section 5.8.1 of the LTP that "other measurement instruments or techniques may be utilized," and "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." The licensee commits to developing technical basis evaluations for alternate final status survey instruments or techniques that will be provided for NRC review 30 days prior to use. That 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.

The calibration of instrumentation used for FSS is discussed in Section 5.8.2 of the LTP, where it is stated that "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." Instrument response checks and measurement sensitivity are described in Section 5.8.3. and 5.8.4 of the LTP, respectively. Section 5.8.4 of the LTP also describes the scan and static MDC calculations for FSS, and utilizes guidance from NUREG-1507 and ISO 7503-1 (1988). Scan MDC calculations for gamma scans of land areas are described in Section 5.8.4.4 of the LTP along with a basic description of the scanning procedures that will be utilized for gamma walkover surveys of land areas.

Sections 5.8.4.5 and 5.8.4.6 of the LTP address HPGe spectrometer analysis and pipe survey instrumentation, respectively. With regard to HPGe analysis, the licensee indicates that “on-site laboratory counting systems are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil.” With regard to pipe survey instrumentation, the licensee provides detection sensitivities of approximately 350 dpm/100 cm2 to 5,200 dpm/100 cm2.

3.5.5.5 Quality Assurance

The licensee’s quality assurance program is described in Section 5.9 of the LTP which states that “ZionSolutions 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.” Project management and the decommissioning organizational structure is described in Section 5.9.1 of the LTP, and the licensee notes that further details on key positions are described in the project Quality Assurance Project Plan (QAPP). The basic elements of the QAPP are described in Section 5.9.2.1 of the LTP, and include: written procedures; training and qualifications; measurement and data acquisitions; instrument selection, calibration and operation; chain of custody; control of consumables; control of vendor-supplied services; database control; and data management.

Measurement and data acquisition is described in Section 5.9.3 of the LTP, and the licensee lists the following quality control measures for use during decommissioning: replicated measurements and surveys; duplicate and split samples; field blanks and spiked samples; and QC investigations. The licensee’s quality assurance assessment and oversight strategies are described in Section 5.9.4 of the LTP, which includes focused self-assessments; independent review of survey results; and the corrective action process.

Data validation is described in Section 5.9.5 of the LTP, where it is noted that survey data will be reviewed for completeness and for outliers. Comparisons to investigation levels will be performed and procedurally verified data will be subjected to the Sign Test and unity rule.

3.5.5.6 Final Status Survey Data Assessment

FSS data assessment is described in Section 5.10 of the LTP and includes the DQA process to complete the following:

- 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.

The licensee indicates in Section 5.10 of the LTP that simple assessment methods such as comparing the survey data mean result to the appropriate Operational DCGL will be performed first on FSS results. An SOF will be calculated as several radioisotopes are measured and a non-parametric statistical test (i.e., the Sign Test) will be applied to the final data set. In Class 1 soil areas an EMC test may be performed if elevated activity is encountered.

Data validation is discussed in Section 5.10.2.1 of the LTP, and the licensee notes that, at a minimum, the following actions should occur:

- 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.

Graphical data analyses are discussed in Section 5.10.2.2 of the LTP and will include, at a minimum, posting plots and frequency plots, while it is noted that additional data review methodologies are found in MARSSIM Section 8.2.2.

The FSS statistical tests are provided in Section 5.10.3 of the LTP. The licensee notes that the unity rule, or sum of fractions, will be applied to data in accordance with guidance in NUREG-1757, Volume 2, Section 2.7. The sum of fractions calculation will be based on the Operational DCGL, and if a surrogate DCGL is used, the “unity rule equivalents” will be calculated using the surrogate adjusted Operational DCGL as shown in Equation 5-16 of the LTP, and as follows (shown for Cs-137 as a surrogate for Sr-90):

$$\text{SOF} \leq 1 = \frac{\text{Conc}_{\text{Cs-137}}}{\text{DCGL}_{\text{Cs-137}_s}} + \frac{\text{Conc}_{\text{Co-60}}}{\text{DCGL}_{\text{Co-60}}} + \dots \frac{\text{Conc}_n}{\text{DCGL}_n}$$

where:

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

The process for an elevated measurements comparison (EMC) evaluation is shown in Section 5.10.4 of the LTP. As previously noted, the application of the EMC will only be to Class 1 land survey units. The DCGLEMC, which is a DCGL modified by an area factor to account for small areas of elevated activity, will be used in accordance with Sections 8.5.1 and 8.5.2 of

MARSSIM. This analysis will use the Base Case DCGLs as presented in Equation 5-17 of the LTP, and as follows:

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

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*.

The fractions for all of the terms (including all elevated areas within a survey unit) will be summed, and must be less than unity to pass the evaluation.

3.5.5.7 Final Status Survey Reporting

Section 5.11 of the LTP discusses final radiation survey reporting, and Sections 5.11.1 and 5.11.2 of the LTP discuss survey unit release records and final FSS reports, respectively.

In Section 5.11.11 of the LTP, the licensee indicates that FSS survey unit release records will include the following:

- 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.

Section 5.11.2 of the LTP notes that FSS final reports will be written, to the extent practical, as stand-alone documents that will usually incorporate multiple survey unit release records. The licensee commits in Section 5.11.2 of the LTP to include the following information in these final reports:

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

3.5.5.8 Surveillance Following Final Status Survey

The licensee discusses surveillance following FSS in Section 5.12 of the LTP and notes that "Isolation and control measures will be implemented in accordance with ZionSolutions site procedures as described in section 5.6.3 [of the LTP]." Section 5.12 of the LTP also notes that documented surveillances of open land survey units will be performed per the following:

- 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 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.

3.5.5.9 NRC Staff Evaluations and Conclusions

NRC staff evaluated Chapter 5 (Final Status Survey Plan) of the licensee's LTP to ensure that the licensee's proposed decommissioning strategies will be consistent with, or comparable to, the NRC's applicable decommissioning guidance, including: NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM); NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions;" NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans;" and the NUREG-1757, Volumes 1 and 2, "Consolidated Decommissioning Guidance," in order to ultimately ensure that the licensee complies with the "radiological criteria for unrestricted use" as specified in 10 CFR 20.1402. NRC staff provides evaluations with regard to the FSS plan and FSS strategies as follows.

3.5.5.9.1 Evaluation of Final Status Survey Approaches for Basement Surfaces

With regard to FSS methodologies for basement surfaces, the licensee utilizes MARSSIM in the development of the FSS plans for basement structures, but there are deviations from the MARSSIM process that warrant elaboration in this SER. A typical MARSSIM process utilizes gamma walkover scanning in conjunction with random/systematic sampling (or discrete measurements). Random/systematic sampling provides a non-biased and representative approach to determine the radiological status of a survey unit, while scanning is utilized for the purpose of locating and delineating any elevated areas of activity. The licensee proposes in the LTP to use in-situ gamma surveys (i.e., ISOCS) to accomplish both scanning and discrete sampling tasks for the basement surface surveys, and the licensee notes in Section 5.5.1 of the LTP that "the surface area covered by a single ISOCS measurement is large (a nominal range of 10-30 m2) which essentially eliminates the need for scan surveys." NRC staff finds the licensee's overall approach is adequate.

The NRC notes that ISOCS should not arbitrarily be considered as a replacement for gamma scanning to locate elevated areas of activity. This is because ISOCS results represent an average of all detectable radioactivity within the instrument's field of view, which can present challenges in delineating small areas of elevated activity within a relatively large field of view.

The licensee discusses the detection of elevated areas in "TSD 14-022 - Use of In-Situ Gamma Spectroscopy for Final Status Survey of End State Structures, Revision 1," which was provided with the LTP. In that document, the licensee notes that an over- or under-estimate of activity can occur depending on the location of an elevated area relative to the center of the field of view. The licensee concludes in the TSD that "the distribution and location of non-uniform elevated areas are expected to be randomly located with equal probability of being at any distance from the detector centerline," and makes no adjustment for potential non-uniform areal distribution. The licensee further notes in TSD 14-022, Revision 1, that "the location(s) of small isolated spots, if any, in contaminated areas such as the Auxiliary basement should be well known through the results of the scan surveys that will be performed to identify areas exceeding the 2 mR/hr open air demolition criteria," and that "after remediation of these areas additional scan surveys will be conducted to ensure that the remediation was successful providing additional information in regards to the potential for elevated areas at the time of FSS." The TSD also indicates that "scan information will further inform the FSS survey design process to ensure that there is no obvious source term geometry present that would result in underestimating total activity in a given basement considering the sample plan and locations for ISOCS FSS measurements." The licensee commits to using additional scan surveys and investigations to inform the FSS, and has noted in Section 5.5.2 of the LTP that “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.” As such, NRC staff considers the additional scan surveys as a necessary part of the overall FSS process.

Additional factors considered in the NRC staff’s evaluation of basement surface surveys include their context within the Basement Fill Model and the currently proposed survey unit classification and investigation processes. The Basement Fill Model is unique with regard to how overall residual radioactivity is assessed, which is discussed in greater detail in Section 6 of this SER. With regard to current survey unit classification, a majority of the survey units are Class 1, which ensures greater overall survey coverage throughout the entire decommissioning process. With regard to investigation levels for basement surfaces, the ISOCS investigation levels (as shown in Table 5-25 of the LTP and in Table 3.5-6 of this SER) are based on an Operational DCGL which is below the Base Case DCGL. This ensures that investigations will be triggered at lower levels (relative to the Base Case), but NRC staff notes that if ISOCS surveys were performed for the sole purpose of locating small areas of elevated activity (as is the case under a typical MARSSIM scanning model) that investigations may need to be designed differently. As a point of reference, NRC staff notes that an NRC sponsored study was completed in 2006 titled “Spatially-Dependent Measurements of Surface and Near-Surface Radioactive Material Using In situ Gamma Ray Spectrometry (ISGRS) For Final Status Surveys,” (ML17284A121). This study addressed various survey and investigation considerations when a discrete particle is located within a larger in-situ field of view and states the following with respect to investigation levels:

- It is important to understand that scanning is performed to identify or detect the presence of areas of elevated contamination, which may be discrete particles. The purpose of scanning is not to quantify the activity in the elevated area. The difference is one of detectability versus measurability. According to MARSSIM, "Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action" (MARSSIM 2000). Therefore, it is necessary to define ISGRS investigation levels during scanning, and to specify the nature of the further investigation once the investigation level is triggered.

The 2006 ISGRS study goes on to note that if the investigation level is exceeded then conventional scan surveys might be conducted to confirm and/ or identify the location of the discrete particle. The licensee’s LTP does not specify that conventional scan surveys will be performed in the event that an ISOCS investigation level is exceeded, but rather seems to rely on preliminary demolition and characterization surveys to locate areas of elevated activity.

Upon the NRC staff’s evaluation of the entire survey process for basement surfaces, the staff finds the overall survey approach adequate, but reiterates previous points of discussion within this SER that the licensee’s commitment to perform pre-FSS scanning surveys is necessary to the overall decommissioning process. The ISOCS instrumentation and detection capabilities are adequate to quantify the average residual radioactivity within its field of view. The licensee’s process for developing a sum of fractions result from basement surfaces is acceptable.

3.5.5.9.2 Evaluation of Concrete Reuse Surveys and Implementation

NRC staff evaluated the licensee’s proposal to reuse concrete from multiple onsite locations as backfill at the site, as previously discussed in Sections 3.5.1 and 3.5.8 of this SER. The licensee discussed concrete reuse in Section 5.7.1.12 of the LTP and also provided two reports with the LTP regarding reuse of concrete on site: TSD 17-007 Evaluation of Static Measurements Performed for Unconditional Release Surveys of Building Materials Used for Backfill at the Zion Decommissioning Project, Revision 1, and TSD 17-010 - Final Report - Unconditional Release Surveys at the Zion Station Restoration Project, Revision 1. The TSD reports were reviewed and NRC staff recognizes that a non-MARSSIM-based graded approach was utilized to perform scanning and direct measurements of structural surfaces along with alpha/beta smear analyses. The approach utilized a survey action level at the minimum detectable count rate plus background. Scan alarms were set at that action level, and alarms triggered investigations. Investigations generally resulted in an evaluation of the area using a gamma sensitive multichannel analyzer, remediation of an area, or analysis of materials via gamma spectroscopy.

The licensee's overall survey approach is one that was developed for "Unconditional Release Surveys (URS)," and the licensee notes in TSD 17-010 that the URS process was developed to be consistent with NRC IE Circular No. 81-07 - "Control of Radioactively Contaminated Materials;" HPPOS-071, "Control of Radioactively Contaminated Material;" and HPPOS-073, "Surveys of Wastes from Nuclear Reactor Facilities Before Disposal." The licensee's original intent, as discussed in multiple RAI responses was to unconditionally release these materials in accordance with the detection capability guidelines provided within IE Circular No. 81-07, HPPOS-071, and HPPOS-073 and to leave the materials onsite as backfill. NRC staff notes that the referenced NRC communication documents essentially inform reactor licensee's "how hard" they must look to assess radioactivity prior to release of materials and equipment offsite, but do not pertain to onsite disposition for license termination purposes. It is also worth noting that there are no release "limits" specified in these NRC communication documents, as no detectable radioactivity should be released on materials and equipment from a reactor site. This point was further clarified in NRC Information Notice No. 85-92: "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities," which states that:

- In 1981, IE Circular 81-07 was issued by the NRC. That circular provided guidance on the control of radioactively contaminated material and identified the extent to which licensees should survey for contamination. It did not establish release limits. The criteria in the circular that addressed surface contamination levels were based on the best information available at the time and were related to the detection capability of portable survey instruments equipped with thin-window "pancake" Geiger-Mueller (G.M.) probes, which respond primarily to beta radiation.

NRC staff communicated to the licensee during the RAI process that actual detection capabilities should be considered as current technology may be capable of detecting residual radioactivity at levels below the levels specified in IE Circular 81-07, and for license termination purposes any detectable residual radioactivity above background must be accounted for in the dose assessment. In order to release a site for unrestricted use per 10 CFR 20.1402, the licensee must demonstrate that "the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA)." As such, licensees would typically survey structures or soil to remain onsite against a Derived Concentration Guideline Level (DCGL), or another approved level that is tied to a dose assessment, for comparison to the license termination criteria. Any residual radioactivity that is distinguishable from background must be considered in that dose assessment, even if it is below the DCGL (i.e., anything less than the DCGL, but distinguishable from background, is not considered zero). NRC staff also noted during the RAI process that the

Commission did not specifically address the reuse of concrete as backfill for the sake of the License Termination Rule (LTR), though consideration for the assessment of approved onsite disposals per 10 CFR 20.2002 was addressed (which has been accepted to represent a few millirem per year, similar to dose assessments for released materials as described in IE Circular 81-07). In particular, the Federal Register Notice associated with the LTR (62 FR 39091) notes in Section C.3.2 that “with regard to burials, as discussed in the preamble to the proposed rule, the determination of whether the licensee meets the radiological criteria of the final rule includes consideration of all residual radioactivity at the site, including burials made in conformance with 10 CFR part 20 (both existing § 20.2002 and formerly used §§ 20.302 and 20.304).”

The concept of “rubblization” of concrete for usage as backfill has previously been addressed by NRC staff, as noted in SECY-00-0041 (Use of Rubblized Concrete Dismantlement to Address 10 CFR Part 20, Subpart E, Radiological Criteria for License Termination) which responded to a proposal from Maine Yankee to leave demolished concrete onsite. The NRC staff noted in that paper that “this procedure, referred to as ‘rubblization,’ appears compatible with the radiological performance criteria for license termination,” but that “it was not specifically considered in the ‘Statement of Consideration’ to the final rule, and is somewhat controversial.” The paper concludes that “at this time, the staff believes that it is technically possible to approve a LTP that includes rubblization,” and that “the staff's belief is premised on a licensee's demonstration that rubblization meets the requirements of Part 20, Subpart E, considering the scenarios of intrusion, excavation, and reuse of buried material and recognizes that, in some cases, mixing/diluting of contaminated material may occur.” Several additional points that should be considered when evaluating applications proposing to use rubblization to demonstrate compliance with 10 CFR Part 20, Subpart E were also provided as Attachment 8 to the SECY paper. Many of these considerations deal with the conceptual dose modeling, and that paper notes in particular that:

- Current dose assessment guidance does not address the proposed conceptual models for rubblization including the acceptability of mixing/diluting contaminated material, nor does the current guidance define the scenarios that should be addressed, and the staff will need to develop supporting guidance for both these areas. Until dose modeling guidance is developed, NRC staff will have to review each applicant’s dose modeling proposal on a case-by-case basis, and this will increase the review time of the application. The acceptability of mixing/diluting contaminated material must be addressed.

As a path forward, the May 31, 2016 NRC comments on the licensee’s RAIs (ML16138A280) indicated that “the licensee should justify the usage of IE Circular 81-07 and the ODCM criteria to meet the LTR,” and that “in doing so, the licensee must also account for potential doses from all impacted materials remaining onsite (including applicable hard-to-detect or insignificant radionuclides).”

As a result of the RAI process, the licensee performed a dose assessment of reuse concrete by assuming that the entire surface of the concrete is contaminated with Cs-137 at the IE Circular 81-07 detection levels, and by using surrogate ratios to estimate the levels of any HTD radionuclides. The licensee indicates in their July 2016 RAI response that:

- The "detection limit" used for the dose calculation is conservatively assumed to be the 5,000 dpm/100 cm2 value in I&E Circular 81-07. Actual detection limits in the unconditional release program are lower than this value. Note that if the use of the 5,000 dpm/100 cm2 maximum non-detect limit is deemed to be too conservative, this dose calculation will be revised based on actual survey detection limits as opposed to the conservative 5,000 dpm/100 cm2 upper value.

This dose assessment is described and evaluated in more detail in Chapter 3.6 of this SER.

NRC staff evaluated the approach and considered the locations where these surveys of reuse concrete were performed and finds the surveys adequate to detect surface contamination. As the original surveys were not radionuclide-specific and did not specifically address the potential for HTD radionuclides, the licensee's dose assessment was evaluated and was found adequate. However, the licensee indicates in Section 6.16 of the LTP that the dose will be adjusted based on the actual maximum scan MDC after all URS surveys are completed. This final dose assessment will be subject to NRC evaluation.

3.5.5.9.3 Evaluation of Final Status Survey Approaches for Embedded Piping/Penetrations, and Buried Piping

NRC staff evaluated the licensee's plans for FSS of embedded piping/penetrations and buried piping, as described in Sections 5.5 and 5.7 of the LTP, respectively, and as previously discussed in this SER. The survey methods and instrumentation proposed for embedded piping/penetration surveys, including the usage of hand held portable detectors and push-pull methods, are adequate, and the proposed strategies to maintain adequate survey coverage within piping are acceptable. The methods to address grouting or remediation of piping (as discussed in Section 5.5.5 of the LTP are adequate. The licensee's process for developing a sum of fractions result for embedded piping or penetrations is acceptable. As noted in Section 5.7.1.9 of the LTP, "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." As such, buried piping survey methodologies are considered adequate.

3.5.5.9.4 Evaluation of Final Status Survey Approaches for Open Land Survey Areas

NRC staff evaluated the licensee's final status survey approaches for open land survey units, as provided in Sections 5.6 and 5.7 of the LTP, and as previously discussed in this SER. The NRC staff finds that the FSS design and usage of the DQO and DQA processes is consistent with MARSSIM and the NUREG-1757 "Consolidated Decommissioning Guidance." The proposed survey instrumentation and detection sensitivities are adequate. The scan coverage and systematic sampling strategies for open land areas are consistent with MARSSIM and the NUREG-1757 "Consolidated Decommissioning Guidance," and are considered adequate. The sum of fractions and elevated measurements comparison methods, as presented in Equations 5-16 and 5-17 of the LTP, respectively, are acceptable and consistent with MARSSIM and the NUREG-1757 "Consolidated Decommissioning Guidance."

3.5.5.9.5 NRC Staff Conclusions on the Licensee's Final Status Survey Plan

NRC staff evaluated the licensee's Final Status Survey plans in accordance with acceptance criteria from NUREG-1757, Vol. 2, Rev. 1, Section 4.4 and NUREG-1700, Rev. 1, Section 5. The staff finds that the plans provide reasonable assurance that the licensee will be able to perform adequate surveys to demonstrate compliance with the radiological criteria for unrestricted use, as specified in 10 CFR 20.1402. Additionally, the staff finds that the licensee's Final Status Survey plans are adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D) and with 10 CFR 20.1501(a) and (b).

3.6 Compliance with the Radiological Criteria for License Termination

Section 3.6 of this SER covers the review of Chapter 6 of the LTP, in which the licensee describes their site-specific DCGLs which they commit to use in order to demonstrate compliance with the radiological criteria for license termination during final status surveys. Staff reviewed this information using Section 2.6 of NUREG 1700 "Compliance with the Radiological Criteria for License Termination" which refers to NUREG-1757, Vol. 2, Section 5.2 and Appendix I.

3.6.1 Radiological Criteria for License Termination

Chapter 6 of the ZSRP LTP, Rev. 2 (Package ML18052A851) included a dose scenario and calculations for site-specific DCGLs for Radionuclides of Concern (ROCs) to be applied to the Zion Nuclear Power Station site to meet the U. S. Nuclear Regulatory Commission (NRC) criteria for release for unrestricted use. Subpart E to 10 CFR Part 20, "Radiological Criteria for License Termination," establishes criteria for the release of sites for unrestricted use. Specifically, the residual radioactivity that is distinguishable from background level must result in a Total Effective Dose Equivalent (TEDE) to the average member of the critical group that does not exceed 0.25 mSv/yr (25 mrem/yr), and the residual radioactivity must also be reduced to levels that are as low as reasonably achievable (ALARA).

ZSRP has chosen to develop DCGLs to demonstrate compliance with the dose based criteria for soil, buried piping, and basements to remain. Base Case DCGLs are levels of each ROC that would result in a dose equivalent to the dose limit for each source. When more than one radionuclide is present, the sum of fractions rule is applied to ensure that the total dose for a particular survey unit remains within the limit. The sum of the fractions methodology takes the radionuclide concentration or activity for each radionuclide present and divides it by the DCGL of the same radionuclide for all of the ROCs and sums the ratios. The sum of the ratios of all the ROCs must be less than or equal to one.

In addition, the licensee must demonstrate that the dose from all sources at the site meets the radiological criteria for license termination. The four source terms for the end-state at Zion are as follows: backfilled basements, soil, buried piping, and groundwater. Backfilled basements are further delineated into basement surfaces, embedded pipe, penetrations, and fill.

The licensee will apply Equation 1 to determine compliance (reproduced from Equation 6-11 in the LTP).

Equation 1. Compliance Dose Equation

$$\text{Compliance Dose} = (\text{Max } SOF_{BASEMENT} + \text{Max } SOF_{SOIL} + \text{Max } SOF_{BURIED\ PIPE} + \text{Max } SOF_{GROUNDWATER}) \times 25 \text{ mrem/yr}$$

where:

Compliance Dose = must be less than or equal to 25 mrem/yr,
Max $SOF_{BASEMENT}$ = Maximum SOF for backfilled Basement (mean of FSS systematic results plus the dose from any identified elevated areas) for backfilled Basement FSS unit

(including surface, embedded pipe, penetrations and fill [if required]),

Max SOF_{SOIL} = Maximum SOF (mean of FSS systematic results plus the dose from any identified elevated areas) for open land survey units,

Max $SOF_{BURIED\ PIPE}$ = Maximum SOF (mean of FSS systematic results plus the dose from any identified elevated areas) from buried piping survey units,

Max $SOF_{GROUNDWATER}$ = Maximum SOF from existing groundwater (based on the analysis of water samples taken from eleven active sample wells established at and around Zion which are monitored on a routine frequency. These wells will remain active and will be monitored through license termination. The two years of monitoring prior to Final Report submittal will be used to establish the Max $SOF_{GROUNDWATER}$.)

The details of how the terms in this equation are determined from sample measurement data are described in TSD 17-004, Rev. 3 (ML18052A539). The technical support document also defines Operational DCGLs which will be applied during FSS to help ensure that the dose from all source terms at the site remains below the dose criterion. The Operational DCGLs are a fraction of the Base Case DCGLs, equivalent to the fraction which has been assigned to each source term. While compliance is not dependent the specific Operational DCGL values as long as the overall dose is less than or equal to 25 mrem/yr, the concept of applying operational DCGLs is integral to meeting compliance with Equation 1. Furthermore, as specified in the HP Chapter of this document, the design of the FSS and investigation levels are based on Operational DCGLs.

This section of the SER describes the staff's review of the development of the DCGLs for Zion Nuclear Power Station. The areas to be surveyed and sampled for unrestricted release within each basement are the wall, floor, and ceiling (if applicable) surfaces, and the penetrations and embedded piping associated with each remaining building basement, (i.e., Auxiliary Building, Unit 1 Containment, Unit 2 Containment, Turbine Building, Crib House/Forebay, Wastewater Treatment Facility (WWTF), Spent Fuel Pool/Fuel Transfer Canal, the Circulating Water Intake Tunnel and Circulating Water Discharge Tunnel). The soil and remaining buried piping will also be surveyed and sampled. As per the LTP, buried pipe is 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 NRC staff has reviewed the information provided in the LTP pertaining to the licensee's assessment of the potential doses resulting from exposure to residual radioactivity remaining at the end of the decommissioning process according to NUREG-1757, Volume 2, Section 5.2. The findings and conclusions of the review under this SER evaluate the licensee's ability to comply with 10 CFR 20.1402 using the LTP.

3.6.2 Identification of Potential Radionuclides of Concern

3.6.2.1 Potential Radionuclides of Concern and Insignificant Radionuclide Contribution

To generate an initial suite of ROCs, the licensee reviewed relevant guidance as well as 19 analytical samples from ZPNS. The licensee reviewed potential radionuclides due to neutron activation of reactor components generated from tables within NUREG/CR-3474. The licensee also reviewed data provided in NUREG/CR-4289 of both fission and activation products in actual samples from seven nuclear power plants. The theoretical listing of radionuclides expected to be found in both PWRs and BWRs, including both fission and activation products, found in WINCO-1191 was also reviewed by the licensee to generate the initial suite of radionuclides. The licensee reviewed the analytical results of 19 samples including resins, smears, and sludge collected from within various waste/process streams at the Zion Site following shut down. From the 19 samples, a list of radionuclides with relative fractions equal to, or greater than, 0.0001 (0.01%) was generated. The licensee then eliminated noble gases since they are not expected to be present at the time of final status survey. The licensee also eliminated any radionuclide with a half-life less than two years, and theoretical neutron activation products that have calculated activity concentrations less than 0.0001 (0.01%) of both the calculated activity concentrations of Co-60 and Ni-63. Based on the review of calculated activity concentrations the following radionuclides were deleted from the list; Cl-36, Ca-41, Mn-53, Se-79, Nb-92m, Zr-93, Mo-93, Ag-108m, Sn-12lm, I-129, Ba-133, Pm-145, Sm-146, Sm-151, Eu-155, Tb-158, Ho-166m, Hf-187m, Pb-205, and U-233. The set of radionuclides generated from the above analysis is listed in Table 5-2 of TSD-11-001, Rev. 1. The licensee refined the list from Table 5-2 of TSD-11-001, Rev. 1 with data from further site characterization. The licensee added radionuclides Ag-108m and Eu-155 to the initial list since they were positively identified in the characterization samples. Furthermore, Pm-147 was eliminated due to false positive results due to interference from very high Cs-137 concentrations (ML15344A389 and ML17208A117).

The licensee determined the relative percent mixture of the initial suite of radionuclides for containment and auxiliary concrete based on the results from the core on-site and off-site analyses decay corrected to July 1, 2018. The estimates for containment were based upon 39 concrete cores (20 in Unit 1, 19 in Unit 2) collected from the 568 foot elevation concrete floors inside the liner (9 Unit 1 and 8 Unit 2 cores were analyzed for HTDs) and from 541 foot elevation under-vessel floors (4 floor cores were analyzed for HTDs). The auxiliary building mixtures are based on based upon a total of 20 cores collected (17 floor and 3 wall) from the 542 foot elevation of the auxiliary building (five floor cores and one wall core were analyzed for the HTDs). The top ½ inch slice of a total of 27 concrete cores were sent for off-site laboratory analysis that included 39 radionuclides (as opposed to only those identified in TSD 11-001 Rev. 1 (ML15344A389)). Cores were collected from locations that exhibited the highest contact dose rates. Several radionuclides in the initial suite were not positively identified in any of the core sample analyses. The mixture percentages for those radionuclides were determined using the reported MDC values. The scaling factors applied were specific to each area (containment, or auxiliary building) based upon the ratio of radionuclides in the concrete cores for that area. Ratios were based on the average concentration of the cores analyzed for the HTD radionuclides within each area (ML17208A117).

The concentrations and source term for the gamma-emitting radionuclides were determined for containment and the auxiliary Building. The calculated surface area of the walls and floors of the end state structure was multiplied by ½ in (1.27 cm) to calculate the volume of concrete equal to a ½ inch thickness for the location sampled. A density of 2.40 g/cm^3 was used to calculate the mass equal to the ½ in thick volume. The estimated masses at each ½ inch depth were multiplied times the average in-house gamma spectroscopy concentration to determine the total estimated source term in each area based upon the on-site gamma spectroscopy results. Since pucks from only 27 of the 69 concrete cores were sent for off-site analysis, the positively identified HTD radionuclide source terms were estimated by multiplying the scaling factor times the on-site gamma spectroscopy source term of the scaling nuclide. The activity mixture percentages shown in Table 1 are based on the inventories calculated in this manner decay corrected to July 1, 2018 since this is the earliest possible date for license termination (ML17208A117)

Table 1. Initial Suite of Potential Radionuclides for ZNPS and Radionuclide Mixture Based on Auxiliary and Containment Concrete [Table 6-2 of LTP ML18052A851]

Radionuclide	% Activity Containment	% Activity Auxiliary
H-3	0.074%	0.174%
C-14	0.008%	0.044%
Fe-55	0.174%	0.106%
Ni-59	0.156%	0.498%
Co-60	4.675%	0.908%
Ni-63	26.275%	23.480%
Sr-90	0.027%	0.051%
Nb-94	0.178%	0.013%
Tc-99	0.008%	0.016%
Ag-108m	0.282%	0.017%
Sb-125	0.025%	0.017%
Cs-134	0.008%	0.010%
Cs-137	67.582%	74.597%
Eu-152	0.436%	0.017%
Eu-154	0.058%	0.009%
Eu-155	0.018%	0.008%
Np-237	0.000%	0.0004%
Pu-238	0.001%	0.001%
Pu-239	0.000%	0.0005%
Pu-240	0.000%	0.001%
Pu-241	0.007%	0.028%
Am-241	0.007%	0.001%
Am-243	0.000%	0.001%
Cm-243	0.001%	0.0003%
Cm-244	0.001%	0.0003%
Total	100.00%	100.00%

Assuming the activity mixture percentages listed in Table 1 (Table 6-2 of the LTP), the licensee calculated the dose attributable to each radionuclide, and identified the insignificant dose contributors in order to select the final ROCs (ML17208A117). The licensee refers to this calculation as the Best Estimate for either containment or auxiliary buildings. The Best Estimate dose contribution for ROCs in structures is calculated by multiplying the percent radionuclide activity (for either containment or auxiliary basements in Table 6-2 of the LTP) by the BFM dose factors determined in TSD-14-010, Rev. 6 (ML17215A110). A similar approach is applied for determining the dose contribution from each ROC for soil. The LTP describes 10 soil samples that were analyzed for HTDs (Section 2.3.5.1 and Section 2.3.5.2). Section 6.8.2 states, "there were very few positive soil sample results identified during characterization and the levels were insufficient to provide a meaningful evaluation of HTD radionuclides. Therefore, the radionuclide mixture for the auxiliary basement cores was applied to soil for planning purposes." The percent mixture for the auxiliary basement was multiplied by the dose to source ratios for soil in TSD-14-010, Rev. 6 to calculate dose contribution from each ROC for soil (ML17215A110).

The licensee also performed several alternative analyses to assess the dose contribution from the insignificant contributors in response to RAIs as further described in Section 3.6.2.2 of this SER. The LTP summarizes the dose contribution from insignificant ROCs from buildings (Table 6-3 and Table 6-4 of the LTP), and soil (Tables 6-31 and 6-32 of the LTP). As a result of the RAI responses, the licensee increased the assumed percent for insignificant radionuclides to be 10% for soil and 5% or 10% for concrete in auxiliary and containment respectively.

The majority of the dose is from Cs-137 at around 96%, with Ni-63 and Sr-90 as the next highest dose contributors. The licensee retained the gamma emitters Co-60 and Cs-134 as ROCs. Sr-90 and Ni-63 are also retained as ROCs even though they are relatively low dose contributors since the licensee stated there is potential for these radionuclides to be present at levels above the MDC at the time of license termination. The licensee includes Eu-152, Eu-154 and H-3 in the containment ROC list because of their potential for being present in activated concrete, not due to their dose contribution, which is less than 0.1% total.

Table 2. Radionuclides of Concern and Percent Dose Contribution for Concrete and Soil [LTP Table 6-5, and Table 6-31 ML18052A851]

	Containment		Auxiliary		Soil
	Percent Activity	Percent Annual Dose	Percent Activity	Percent Annual Dose	Percent Annual Dose
H-3	0.074	0.017	N/A	N/A	N/A
Co-60	4.675	1.699	0.908	0.783	3.878
Ni-63	26.275	0.366	23.480	0.270	0.119
Sr-90	0.027	1.072	0.051	0.742	0.072
Cs-134	0.008	0.015	0.010	0.039	0.028
Cs-137	67.582	96.269	74.597	96.959	95.733
Eu-152	0.436	0.067	N/A	N/A	N/A
Eu-154	0.058	0.010	N/A	N/A	N/A
"Best Estimate" Percent Insignificant Radionuclides Calculated	0.864	0.512	0.954	1.313	0.171

Percent Insignificant Radionuclides Assigned for Final Radiation Survey Planning	N/A	10	N/A	5	10

3.6.2.2 NRC Evaluation of Radionuclides of Concern and Insignificant Radionuclide Contribution

NUREG 1757, Vol 2, Section 5.2 outlines the criteria for evaluating the description of the source term including radionuclides of concern, configuration of the source, areal variability of the source, and chemical form of the source. NUREG 1757, Vol 2, Section 3.3 provides guidance on conditions under which radionuclides or exposure pathways may be considered insignificant and may be eliminated from further consideration. Specifically, NRC staff has determined it is reasonable that radionuclides or pathways that are insignificant contributors to dose (<10% of the dose criteria) may be eliminated from further detailed consideration.

In reviewing the source term information, the NRC staff note that the initial suite of radionuclides in Table 5-2 of TSD-11-001, Rev. 1 does not exactly match the initial suite of radionuclides listed in the LTP (LTP Table 5-1 or Table 6-2 in the LTP). Table 5-1 in the LTP includes Mo-93, Sm-146, Sm-151, and Eu-155. TSD 11-001, Rev. 1 states that Mo-93, Sm-146, and Sm-151 were eliminated based on low activity fractions. Table 6-2 in the LTP does not include Pm-147 and does include Ag-108m and Eu-155. NRC staff noted the table discrepancies to the licensee as part of the RAI process and the licensee responded with additional explanation in the revised LTP, the public meetings held with the licensee, and the RAI responses. Ag-108m and Eu-155 are included in the dose assessment list since they were positively identified during characterization (Chapter 6 of the LTP). Pm-147 was not included in the dose assessment list (Table 6-2 of the LTP) because it was considered a false positive due to interference from very high Cs-137 concentrations. (ML17208A117). Given this explanation of the discrepancies in the various tables, the NRC staff finds the initial suite of potential radionuclides listed in Table 6-2 of the LTP acceptable.

The NRC staff considers radionuclides and exposure pathways that contribute no greater than 10% of the dose criteria (2.5 mrem/yr) to be insignificant contributors. NUREG 1757, Vol. 2, Appendix O, Question 2, states:

> When developing derived concentration guideline levels (DCGLs) for the FSS, which radionuclides can be deselected from further consideration?
>
> It is incumbent on the licensee to have adequate characterization data to support and document the determination that some radionuclides may be deselected from further detailed consideration in planning the Final Status Survey (FSS). Radionuclides that are undetected may also be considered insignificant, as long as the MDCs are sufficient to conclude that the dose contribution is less than 10% of the dose criterion (i.e., with the assumption that the radionuclides are present at the MDCs).

Concrete cores taken in structures other than the auxiliary building or containment buildings (e.g., turbine building, steam tunnels, and crib house) indicated minimal activity. In addition, samples of surface and subsurface soils taken in and around Class 1 survey units have not contained significant activity levels. Due to the absence of a significant source term in soil or in other end-state structures, ZSRP applies the radionuclide mixture derived for the Auxiliary Basement concrete to all basement structures, soils and buried piping for FRS planning and implementation. Therefore, the licensee relies on the auxiliary building core samples (5 floor cores and 1 wall core analyzed for HTDs) for all basements besides containment. Using the data from the auxiliary building cores, the licensee calculated 0.171 % of the dose for the insignificant radionuclides for soil and 1.31% for all basements except containment. Based on the results from containment characterization (21 cores analyzed for HTDs), the licensee calculated 0.51% to insignificant radionuclides for containment basements.

With regard to soil, the NRC staff did not agree that the characterization data was adequate to support the assumption in the LTP Rev. 0 that the insignificant dose contribution for soil is 0.171%. The NRC staff also did not agree that the auxiliary building activity ratios are necessarily appropriate to apply to the soil given the differences in material. As a result of the RAI process, the licensee revised the assumed insignificant dose contribution percentage to be 10% for soil in Rev. 2 of the LTP Section 5.2.3. The staff finds the licensee's revised assumption in Rev. 2 of the LTP of a 10% contribution for soil acceptable (Package No. ML18052A851). Assigning a 10% value is conservative given that insignificant radionuclides were not detected in the 10 soil samples which were analyzed for HTDs; only Cs-137 and Co-60 were identified in these soil samples. Furthermore, in soils in higher risk areas that have not yet been characterized (e.g., under containment and SFP foundation slabs), the licensee has committed to analyzing for the initial suite of radionuclides as described below.

With regard to basements, the NRC staff did not agree that the characterization data provided was adequate to support the assumption in the LTP Rev. 0 that the insignificant dose contribution for concrete for uncharacterized areas was 1.313%. Specifically, the NRC staff did not agree that the auxiliary building activity ratios, based on 6 cores analyzed for HTDs, were necessarily appropriate to apply to uncharacterized areas such as the Spent Fuel Pool/Transfer Canal, Circulating Water Tunnels, and embedded piping. Therefore, the NRC requested the licensee perform additional analyses to support insignificant contribution assumptions. In response, the licensee assessed the dose contribution from the individual cores (normalized to a total of 25 mrem/yr when appropriate). As a result of this analysis and the RAI process the licensee increased the assumed contribution for containment concrete to 10% and all other basement concrete to 5% as detailed in Section 5.2.1 of the LTP Rev. 2. This increase in assumed percentage of insignificant radionuclide dose contribution allows the licensee additional margin in meeting the dose criterion, and the licensee has committed to analyzing for the full suite of radionuclides during continuing characterization for areas that have not yet been characterized as outlined in Section 5.1 of the LTP, Rev. 2 (Package No.ML18052A851).

In the response to the NRC Request for Additional Information dated July 20, 2016, the licensee proposed to eliminate the further analysis of Np-237 (ML16211A200). The licensee stated that Np-237 would be excluded from any further analysis (including during the continuing characterization of places that have not yet been characterized (the SFP/transfer canal, Circulating Discharge Water Tunnels, soils between the turbine building and containment, soils along foundation walls, and embedded pipes). The licensee's reason was that Np-237 has not been found in the 19 samples (Resin or Smear from auxiliary building) sent for full Part 61 analysis, 10 soil samples sent for HTD analysis, 6 core samples from the auxiliary building or 21 core samples from the containment building which were all analyzed for HTDs. Also, the licensee stated that "Np-237 relative concentration in NUREG/CR-4289, Table 4.4 is orders of magnitude below the 0.01 % threshold applied in TSD 11-001 to exclude radionuclides from the initial suite and therefore should not have been included in the initial suite." NUREG/CR-4289 aggregates data from seven NPPs in US. Np-237 was analyzed in three plants (Indian Point (PWR), Turkey Point (PWR), and Dresden (BWR)).

The NRC staff noted during the review process that the dose factor for soils for Np-237 is relatively high. For example, the dose from Np-237 (provided in response to RAIs) assuming the average MDC of the 10 soil samples of 3.79×10^{-2} pCi/g was 1.18 mrem/yr or 5% of the 25 mrem [ML16081A010]. Given the potential risk of Np-237, in response to the staff's RAIs, the licensee has committed to analyze for Np-237 in the samples that are analyzed for the full initial suite during continuing characterization in order to verify that the dose contribution from Np-237 is not significant. Initial suite analysis will be performed in 10% of all media samples, with a minimum of one sample, taken during continuing characterization in the areas specified in LTP Rev 2, Chapter 2 (ML17208A113).

The licensee stated that H-3, Eu-152, and Eu-154 are activation products and are therefore applicable to containment building only. However, H-3 is formed not only from activation of concrete (Li-6) but more commonly from neutron capture by boron (B-10) in PWRs (boric acid added to PWR reactor coolant system). Tritium can potentially build up in the SFP due to mixing with reactor coolant. Fuel cladding defects could also allow tritium transfer from fuel. H-3 was detected in a groundwater monitoring well at the site in 2006 (ML15188A113). Although H-3 has not been detected in any groundwater monitoring well above the MDC since 2006, the fact that it was detected above MDC is evidence of a past release. The licensee stated that the well in which H-3 was detected is located up gradient from the groundwater flow direction and should not be impacted by the decommissioning activities, but the fact that the well is up gradient does not explain how H-3 transported to that location. H-3 was also positively detected above MDC in the auxiliary basement cores, in addition to the containment cores. The licensee has committed to performing analyses of the full initial suite of radionuclides (including H-3) during continuing characterization as described below.

In the response to RAI PAB 1a (January 2018 RAI Response), the licensee defines all areas where continuing characterization will occur and commits to provide continuing characterization sample plans to the NRC for review and results to NRC for evaluation.

The following areas had not yet been characterization at the time of the submission of the final revision of the LTP:

- underlying concrete of the SFP/transfer canal below 588 ft (179 m) elevation
- concrete walls and floor of the Under-Vessel areas in Unit 1 and Unit 2 containments
- floors and walls of the Hold-Up Tank (HUT) cubicle
- floor of the auxiliary building 542 foot elevation Pipe Tunnel floors
- floor and lower walls of the 542 foot elevation of the auxiliary building
- 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
- 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
- several potentially contaminated embedded and buried pipe systems that will be abandoned in place

- the exposed steel liner in the containment basements after concrete removal

In Section 5.3.4.4 of the LTP, as well as Section 2.5 of LTP, Rev 2, the licensee commits to analyze selected samples taken from these areas for the initial suite of radionuclides and commits to calculate the insignificant contributor dose as described in Section 5.1 of the LTP. If the insignificant contributor dose from each individual sample is less than the insignificant contributor dose assigned for planning purposes, then no further adjustments for insignificant contributor dose will be made. If the insignificant contributor dose from the characterization sample is greater than the insignificant contributor dose assigned for planning purposes, then an investigation will be conducted and the insignificant contributor dose will be revised as described in Section 5.1 of the LTP. Furthermore, if the data indicate different ROCs for the given area, a specific ROC list will be applied to the area. The licensee will also to determine if the ratios of the HTD ROCs (Sr-90, Ni-63, and H-3 are different from the ratios currently assigned. The evaluation of the use of surrogate ratios is discussed in Section 3.5.14 of this SER.

The NRC staff finds that the approach used by Zion*Solutions* to identify the potential radionuclides of concern in its LTP is acceptable because of the licensee's commitments for additional characterization and verification of the mixture activity percentages and dose contribution discussed in this section. The determination of insignificant radionuclides performed by Zion*Solutions* is consistent with the NRC's guidance in NUREG-1757 Vol 2, Section 5.2. This guidance states that radionuclides may be considered to be insignificant as long as the total dose from the radionuclides that were identified as insignificant is less than 10% of the dose criteria. The guidance further states that the dose from the insignificant radionuclides must be accounted for when demonstrating compliance.

3.6.3 Backfilled Basements, Embedded Piping, and Penetrations DCGLs

This section of the SER discusses the licensee's approach and proposed DCGLs for backfilled basements, embedded piping and penetrations. The staff evaluation and analysis of the licensee's DCGLs is in Section 3.6.3.4, NRC Evaluation and Independent Analysis of DCGLs for Backfilled Basements, Embedded Piping, and Penetrations. The staff is assessing the licensee's compliance with the requirements set forth in 10 CFR Part 50.82(a)(9)(ii)(D),"Detailed plans for the final radiation survey" because the DCGLs are used in the final radiation surveys.

3.6.3.1 Scenarios, Parameters, and Uncertainty Analysis for Backfilled Basements, Embedded Piping, and Penetrations

On site buildings will be demolished and removed to a depth of at least three feet below grade (i.e., an elevation of 588 ft [179 m]) above mean sea level). The LTP states that all contaminated systems, components, piping, buildings, and structures above the 588 ft (179 m) elevation will be removed and disposed of as waste. The LTP calls for the rubblization and reuse of onsite concrete that meets the licensee's definition of "Clean Concrete Demolition Debris". The basements and below ground structures included in the ZNPS end state include: Unit 1 containment Building, Unit 2 containment Building, auxiliary building, turbine building, Crib House and Forebay, WWTF, Spent Fuel Pool, Main Steam Tunnels for Unit 1 and Unit 2, Circulating Water Intake Piping, and Circulating Water Discharge Tunnels (Table 3 and Table 4).

Table 3 Basements and Below Ground Structures included in the ZNPS End State (LTP Table 6-1)

Basement/Structure	Material remaining	Lowest Internal Elevation feet (m) above mean sea level
Unit 1 Containment Building	Steel Liner over Concrete	568 (173.1)
Unit 2 Containment Building	Steel Liner over Concrete	568 (173.1)
Auxiliary Building	Concrete	542 (165.2)
Turbine Building	Concrete	560 (170.7)
Crib House and Forebay	Concrete	552 (168.2)
WWTF	Concrete	577 (175.9)
Spent Fuel Pool	Concrete	576 (175.6)
Main Steam Tunnels (Unit 1 and Unit 2)	Concrete	570 (173.7)
Circulating Water Intake Piping	Steel Pipe	Site 552 (168.2), Lake 543 (165.5)
Circulating Water Discharge Tunnels	Concrete	Site 552 (168.2), Lake 543 (165.5)

Table 4 Adjusted Basement Surface Areas for Base Case DCGL Calculation [LTP Table 6-23]

Basement	**Structures Included in Calculation**	**Total Adjusted m²**
Containment	• Containment Above 565 ft Elevation • Containment Under-Vessel Area • SFP/Transfer Canal	3,482
Auxiliary	• Auxiliary • SFP/Transfer Canal	7,226
Turbine	• Turbine • Circulating Water Discharge Tunnel • Circulating Water Intake Pipe • Circulating Water Discharge Pipes • 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

Embedded piping and penetrations are expected to remain in the basements. The licensee defined embedded pipe as pipe that runs vertically through a concrete wall or horizontally through a concrete floor and defined a penetration as a pipe (or pipe sleeve or concrete if the pipe has been removed) that traverses a wall and is cut on both sides of the wall. The licensee provided a bounding list of end-state embedded pipe and penetrations in Attachment F to TSD 14-016, Rev. 0. The licensee committed that no pipe that is not listed in Attachment F will remain, but pipes can be removed from the list and disposed of as waste. The basements will be backfilled and will covered by at least three feet (0.91 m) of clean soil. In the response to RAI PAB 4b (January 2018 RAI responses), the licensee indicated that there are penetrations present that provide hydraulic connections between the containment, auxiliary, turbine, and SFP/transfer canal basements.

Table 5 Embedded Pipe and Penetration Survey Units [LTP 5-20]

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 (1)	• Containment Penetrations
SFP/Transfer Canal	N/A	N/A
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 are hydraulically connected to Steam Tunnel/turbine building so include with turbine building as well as containment

The licensee used a model they called the Basement Fill Model (BFM) to evaluate the dose from residual radioactivity remaining in the basements and associated structures at the time of license termination. In this model, it is assumed that the basements will fill with water and that the residual radioactivity associated with the walls and floors will be released into this water either instantly or by diffusion over time, depending on whether the activity is surficial or volumetric. The residual radioactivity in the fill water is assumed to be instantly mixed over the fill water in the basement and is allowed to partition back onto the fill. A well is assumed to be drilled into the basement and the fill water is assumed to support a farm (i.e., well water is used for drinking, garden irrigation, pasture/crop irrigation, and livestock water supply).

The Disposal Unit Source Term – Multiple Species (DUST-MS) model was used to calculate the maximum water concentrations in the fill material of each basement for a given total inventory of each radionuclide in the basement. The specific composition of the backfill has not been determined yet, but the licensee expects it to be a combination of sand and debris from building demolition. Because the composition of the backfill has not been finalized, the bulk density, porosity, and Kd values are not known. Zion*Solutions* therefore selected values for these parameters that they believed to be conservative.

The RESRAD Model (Version 7.0) was then used to determine the dose to a resident farmer receptor as a function of the water concentration in the particular basement. The exposure pathways included for the resident farmer scenario include: direct exposure to external radiation, inhalation dose from airborne radioactivity, ingestion dose from plants grown with irrigation water from onsite well, ingestion dose from meat and milk from livestock consuming fodder from fields irrigated with onsite well water and consuming onsite well water, ingestion dose from onsite well water, and soil ingestion. The parameters used in the RESRAD calculation for the BFM are provided in Attachment 2 to Chapter 6 of the LTP. The results of the DUST-MS and RESRAD models were then multiplied to get a dose rate of each radionuclide per unit inventory of the radionuclide in the basement.

The projected dose from drilling spoils brought to the surface during installation of the hypothetical well was also evaluated as part of the BFM. The inventory in the drilling spoils considered both the hypothetical equilibrium residual radioactivity on the fill calculated using DUST-MS and residual radioactivity remaining in the concrete. The hypothetical well was assumed to be drilled to the concrete floor of the basement and to have a borehole diameter of 8 inches (20.32 cm). The material brought to the surface was assumed to be homogeneously mixed, but it was assumed that no additional dilution of the spoils would occur once it was brought to the surface. The spoils were assumed to be spread out over an area of 0.92 to 3.56 m^2, depending on the size of the basement. The exposure pathways included for exposure to the drilling spoils included direct exposure, inhalation dose, and ingestion dose. The dose of each radionuclide from the drilling spoils per activity of the radionuclide in the basement was calculated using the surface soil DCGLs and area factors.

The licensee did not calculate the dose to a well driller for the SFP because the elevation of the SFP floor is only 3 ft (0.914 m) below the water table and the licensee did not think that this was a sufficient amount of water to support a well. However, the dose from drilling spoils from a hypothetical well was evaluated on the basis that a well driller could inadvertently drill into the SFP and then reject the location due to insufficient yield. In addition, because the SFP/transfer canal will be hydraulically connected to the containment and auxiliary basements, the inventory in the SFP/transfer canal and its associated area were added to the containment and auxiliary basements in the calculation of the dose to the resident farmer from contaminated well water.

The licensee calculated $DCGL_{BS,i}$ value for each radionuclide corresponding to a dose of 25 mrem/yr per unit activity from either the groundwater (fill water) or drilling spoils scenario using Equation 2 and the dose factors described above..

$$\boldsymbol{DCGL_{BS,i}} = \frac{\mathbf{25}\,\frac{\boldsymbol{mrem}}{\boldsymbol{yr}}}{\boldsymbol{BFM\ Scenario\ DF_i}} * \frac{\mathbf{1}}{\boldsymbol{SA_{b(adjusted)}}} * \mathbf{1}\boldsymbol{E} + \mathbf{09}\ \frac{\boldsymbol{pCi}}{\boldsymbol{mCi}} * \boldsymbol{IC\ Dose\ Adjustment} \qquad \textbf{Equation 2}$$

Where:

$DCGL_{BS,i}$	= Groundwater or Drilling Spoils DCGL for radionuclide (i) (pCi/m^2)
BFM Scenario DF_i	= BFM Dose Factor for radionuclide (i) (mrem/yr per mCi)
$Sa_{b(adjusted)}$	= Adjusted Surface Area of Basement (b) (m^2)
IC Dose Adjustment	= Insignificant Contributor Dose Adjustment Factor (0.9 for Containment and 0.95 for all other basements)

The licensee derived Basement Derived Concentration Levels ($DCGL_B$) that correspond to a dose of 25 mrem/yr per radionuclide considering the dose from both the use of fill water and from exposure to drilling spoils using Equation 3 (Table 6).

$$DCGL_{B,i} = \frac{1}{\left(\frac{1}{GW\ DCGL_{BS,i}} + \frac{1}{DS\ DCGL_{BS,i}}\right)} \qquad \textbf{Equation 3}$$

Where:
$DCGL_{B,i}$ = Basement Surface DCGL for radionuclide (i) (pCi/m^2)
GW $DCGL_{BS,i}$ = Groundwater (fill water) DCGL for radionuclide (i) (pCi/m^2)
DS $DCGL_{BS,i}$ = Drilling Spoils DCGL for radionuclide (i) (pCi/m^2)

Table 6. Base Case Basement Surface DCGLs ($DCGL_B$) – (from LTP Chapter 5, Table 5-3)

Nuclide	Auxiliary Building	Containment	SFP/Transfer Canal	Turbine Building	Crib House /Forebay	WWTF
	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)
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 DCGL for the SFP/transfer canal was set equal to the lower of either the auxiliary building or containment DCGLs. The containment DCGLs were lower for all ROC; therefore, the SFP/transfer canal DCGLs were set equal to containment DCGLs.
Note 2: To convert pCi/m^2 to Bq/ m^2 multiply by 0.037

The licensee developed DCGL values for embedded piping ($DCGL_{EP}$) in terms of activity of the given radionuclide per area of pipe (Table 7). The residual radioactivity was modeled as being released into the basement where the piping is contained. The $DCGL_{EP}$ values were calculated from the BFM dose factors described above using Equation 4.

$$DCGL_{EP}(b,i) = \frac{25\,\frac{mrem}{yr}}{BFM\ DF_{b,i}} * \frac{1}{EP\ SU\ Area} * 1E+09\ \frac{pCi}{mCi} * IC\ Dose\ Factor \qquad \textbf{Equation 4}$$

Where:
$DCGL_{EP}$ (b,i) = Embedded Pipe DCGL for radionuclide (i) in basement (b) (pCi/m^2)
BFM DF(b,i) = Sum of BFM Dose Factors for Groundwater and Drilling Spoils for radionuclide (i) in basement (b) (mrem/yr per mCi)
EP SU Area = Total internal surface area of embedded pipe in the survey unit (m^2)
IC Dose Adjustment = Insignificant Contributor Dose Adjustment Factor (0.9 for Tendon Tunnel and 0.95 for all other embedded piping)

Table 7. Base Case DCGLs for Embedded Pipe ($DCGL_{EP}$) [LTP Table 5-11]

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

* To convert pCi/m^3 to Bq/m^3 multiply by 0.037

DCGL values were also generated for penetrations ($DCGL_{PN}$) that run through a wall and/or floor between two buildings and are open at the wall or floor of each building (Table 8). In the development of these DCGL values, it was assumed that the residual radioactivity is released to both of the buildings that the penetration interfaces. The $DCGL_{PN}$ values were calculated in the same way as the $DCGL_{EP}$.

Table 8. Base Case DCGLs for Penetrations ($DCGL_{PN}$) [LTP Table 5-13]

Nuclide	**Auxiliary Bldg.**	**Containment**	**SFP/ Transfer Canal**	**Turbine Bldg.**	**Crib House/ Forebay**	**WWTF**
	(pCi/m^2)	**(pCi/m^2)**	**(pCi/m^2)**	**(pCi/m^2)**	**(pCi/m^2)**	**(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

* To convert pCi/m^3 to Bq/m^3 multiply by 0.037

Because the above Base Case DCGL values correspond to a dose of 25 mrem/yr for each radionuclide for each component of the basement (i.e., basement surfaces, embedded piping, and penetrations), the licensee developed operational DCGL values to ensure that the combined dose from the basement and other source terms (soil, buried piping, and groundwater) is less than 25 mrem/yr (TSD 17-004, Rev. 3). As described in Section 3.5 the licensee used the operational DCGL values as the DCGL for the purpose of survey design. The operational DCGL values were obtained by reducing the Base Case DCGL values by the expected fraction of the 25 mrem/yr dose limit for each source term. The operational DCGL values developed in TSD 17-004, Rev. 3 and provided in Chapter 5 of the LTP are based on the assumption that the maximum dose fraction for a basement will be 0.448.

The licensee assigned a dose to the onsite concrete used to backfill the basements (Table 9). This dose accounts for the potential dose from the presence of residual radioactivity below the level of detection. The onsite concrete used in backfill is surveyed through the Unconditional Release Survey (URS) program according to procedure ZS-LT-400-001-001, "Unconditional Release of Materials, Equipment and Secondary Structures". The licensee stated that materials unconditionally released through this program have been verified to contain no detectable plant-derived radioactivity. A detection limit of 5,000 dpm/100 cm^2 was used to generate the potential dose. The licensee stated that actual detection limits are lower than this value. The potential dose was evaluated using the ROC mixtures ratios for both the Containment and Auxiliary buildings. Because the Containment building ROC mixture was slightly higher, it was used for all concrete when calculating the assigned dose. The licensee stated in the LTP that they plan to adjust the dose values in Table 9 based on the actual maximum scan MDC after all the URS surveys are completed

Table 9.Dose Assigned to Concrete Backfill [LTP Table 6-53]

	Auxiliary Bldg.	**Containment**	**SFP/ Transfer Canal**	**Turbine Bldg.**	**Crib House/ Forebay**	**WWTF**
	(mrem/yr)	**(mrem/yr)**	**(mrem/yr)**	**(mrem/yr)**	**(mrem/yr)**	**(mrem/yr)**

Dose	9.94E-01	1.77E+00	1.52E-01	1.58E+00	1.57E+00	6.40E+00

* To convert mrem/yr to mSv/yr multiply by 0.01

3.6.3.2 Backfilled Basements, Embedded Piping, and Penetrations Elevated Area Consideration

For structural surfaces, embedded pipe, and penetrations, sample measurements that are above the Operational DCGLs listed in TSD 17-004, Rev 1 but are less than the Base Case DCGL values are considered elevated areas. In Chapter 5 of the LTP, the licensee committed to investigating any area identified by measurement (systematic or judgmental) that exceeds the Operational DCGL. Any residual radioactivity on structural surfaces, embedded pipe, or penetrations greater than the Base Case DCGLs will be remediated. Furthermore, any residual radioactivity in Class 1 embedded pipe and penetration greater than the Base Case DCGLs for the structural surfaces where the pipe or penetration interfaces will require further remediation and/or grouting of the pipe. The dose contribution from elevated areas on the basement surfaces, embedded pipes, and penetrations will be incorporated in the calculation of the SOF in the basement.

In addition to the BFM, the licensee also evaluated alternate scenarios in TSD 14-010, Rev 6. In one of these scenarios, the potential dose from drilling a well for water that goes through contaminated penetrations, embedded pipes, or basement surfaces was evaluated. This activity was assumed to be brought to the surface in the drilling spoils. The exposure of both a worker and a resident farmer to the drill spoils was evaluated. The potential dose to a driller at the maximum concentrations based on the DCGLs was found to be less than 25 mrem/yr.

The licensee also considered the construction of a basement for a house for a resident farmer within the fill material. In this scenario, the basement was assumed to be 3 m deep, which is not deep enough to reach the expected depth of residual contamination on site at the time of license termination. This scenario therefore only considered the direct radiation dose to the resident farmer, which was found to be negligible. A third alternate scenario analyzed by the licensee was a large-scale excavation of the backfilled structures.

3.6.3.3 Total Basement Dose

The SOF for the basement surface survey units will be calculated by the licensee according to Equation 5. Similarly, the SOF for embedded pipes and penetrations will be calculated according to Equation 6.

$$\boldsymbol{SOF_B} = \sum_{i=1}^{n} \frac{\boldsymbol{Mean\ Conc_{B,ROC_i}}}{\boldsymbol{Base\ Case\ DCGL_{B,ROC_i}}} + \frac{(\boldsymbol{Elev\ Conc_{B,ROC_i}} - \boldsymbol{Mean\ Conc_{B,ROC_i}})}{\left[\boldsymbol{Base\ Case\ DCGL_{B,ROC_i}} \times (\frac{SA_{SU}}{SA_{Elev}})\right]} \qquad \textbf{Equation 5}$$

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 $DCGL_{B,\ ROCi}$

Elev $Conc_{B,\ ROCi}$ = Concentration for ROC_i in any identified elevated area (systematic of judgmental)

SA_{Elev} = surface area of the elevated area

SA_{SU} = adjusted surface area of FSS unit for DCGL calculation

$$SOF_{EP/PN} = \sum_{i=1}^{n} \frac{Mean\ Conc_{EP/PN,ROC_i}}{Base\ Case\ DCGL_{EP/PN,ROC_i}} + \frac{(Elev\ Conc_{EP/PN,ROC_i} - Mean\ Conc_{EP/PN,ROC_i})}{\left[Base\ Case\ DCGL_{EP/PN,ROC_i} \times (\frac{SA_{SU}}{SA_{Elev}})\right]} \qquad \textbf{Equation 6}$$

Where:

$SOF_{EP/PN}$ = SOF for embedded pipe or penetration survey unit within a Basement using Base Case DCGLs

Mean $Conc_{EP/PN,\ ROCi}$ = Mean concentration for the systematic measurements taken during the FSS of embedded pipe or penetration in survey unit for each ROC_i Base Case $DCGL_{B,\ ROCi}$

Elev $Conc_{EP/PN,\ ROCi}$ = Concentration for ROC_i in any identified elevated area (systematic of judgmental)

SA_{Elev} = surface area of the elevated area

SA_{SU} = adjusted surface area of FSS unit for DCGL calculation

The total dose for each basement will then be calculated using Equation 7.

$$SOF_{Basement} = SOF_B + SOF_{EP} + SOF_{PN} + SOF_{CF} \qquad \textbf{Equation 7}$$

Where:

$SOF_{Basement}$ = SOF (sum 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)

$SOF_{Basement}$ = SOF (sum of FSS systematic results plus the dose from any identified elevated areas) for backfilled basements

3.6.3.4 NRC Evaluation and Independent Analysis of DCGLs for Backfilled Basements, Embedded Piping, and Penetrations

This section includes the staff evaluation and analysis of the licensee's DCGLs for backfilled basements, embedded piping, and penetrations. The staff is assessing the licensee's compliance with the requirements set forth in 10 CFR Part 50.82(a)(9)(ii)(D), "Detailed plans for the final radiation survey" because the DCGLs are used in the final radiation surveys.

As the NRC staff noted in the third set of RAIs, while many aspects of the conceptual model assumed in the determination of the dose from residual radioactivity in the basements are conservative (e.g., the assumption that the water in the basements can support a well with enough water for a farm), other aspects of the conceptual model appear to be non-conservative. Because the conceptual model used in the DUST-MS calculations contains both conservatisms and potential non-conservatisms, the overall conservatism of the BFM is not clear. This makes the review and implementation of the BFM model significantly more complex than the review and implementation of a standard DCGL and MARSSIM approach.

In the BFM model, the release of the radionuclides is assumed to be either instantaneous for loose surface contamination or a diffusion controlled release for volumetric contamination. In the BFM model, it is then assumed that the radionuclides will instantly mix over the basement in question and partition the released radionuclides between the aqueous and solid phases (i.e., the backfill) based on the sorption coefficient (Kd value) for the radionuclide. Because many of the ROCs for this site have high Kd values, a significant fraction of the radionuclide is assumed to be sorbed on the backfill. While the assumption of instantaneous release is conservative, the assumptions that the radionuclides will then instantly mix over the entire basement and a significant fraction will then resorb onto the backfill may be non-conservative.

Additionally, the BFM model assumes that the radionuclides are distributed evenly throughout the basement. While the potential dose from the release of the radionuclides to water is generally a function of total inventory and the physical distribution of the radionuclides would not significantly affect the water concentration, the potential dose to an individual who encounters contaminated material in the basement while drilling a well is very dependent on the physical configuration of the residual radioactivity. If the individual inadvertently drills into an elevated area, their dose would be much higher than if they drill into an area with lower activity. The NRC staff thinks that a scenario in which a worker is exposed to the residual radioactivity through drilling, or otherwise performing subsurface work, is more plausible than a scenario in which water in the basement is used to support a farm. The NRC staff finds that the calculation performed by the licensee to evaluate the potential dose due to drilling into the material adequately evaluated the potential dose to an individual who drills through an elevated area. The NRC staff concludes that the licensee therefore adequately demonstrated that the potential dose from drilling into an elevated area is acceptable.

The NRC staff concludes that the use of both the BFM model to evaluate the potential dose from the total inventory in the basement in combination with the use of the drilling scenario to evaluate the potential from elevated areas is an appropriate method for demonstrating that the dose will be less than 25 mrem/yr. The NRC staff also finds that the methodology proposed by the licensee to sum the dose components in the basements, including elevated areas, is acceptable because it is consistent with MARSSIM guidance.

The NRC staff concludes that the use of the operational DCGL values for individual dose components will ensure that the total dose will be less than 25 mrem/yr and that the survey unit FSS is designed adequately. If the assumption that the maximum dose fraction from a basement is less than 0.448 is not found to be true, the licensee will need to evaluate if the FSS surveys were designed and performed adequately to meet the revised dose fraction and revise their operational DCGL values.

The NRC staff disagrees with the licensee's referral to the rubbilized concrete that is intended for use as onsite backfill as "clean" as some of this concrete originated in areas that were potentially contaminated. Additionally, as noted in Section 3.5, the concrete was not surveyed using a process that is consistent with MARSSIM. The NRC staff therefore think that there is uncertainty in the potential dose from the rubblized concrete and there might be a non-negligible dose associated with residual contamination on the rubblized concrete. To address the NRC's concern, the licensee determined the potential dose from the concrete used as backfill based on the maximum allowable MDC of 5000 dpm/100 cm^2 used in surveying the concrete. The NRC staff finds that this approach is an acceptable way of determining the dose. In Chapter 5 of the LTP, the licensee also stated that the actual MDC values were lower than 5000 dpm/100 cm^2 and the dose from the backfill will be recalculated based on the observed MDCs.

The NRC staff has reviewed the dose modeling analyses for the backfilled basements, embedded piping and penetrations for unrestricted release as part of the review of the licensee's LTP, using the Consolidated Decommissioning Guidance, Volume 2, and NUREG/CR-5512, Vol. 3 Residual Radioactive Contamination from Decommissioning Parameter Analysis (NRC 1999, 2002), and finds that the proposed DCGLs are acceptable for use as release criteria to be used in the FSS, which is a requirement in 10 CFR 50.82(a)(9)(ii)(D) Detailed plans of final radiation surveys. These DCGLs are acceptable as release criteria provided that the combined dose from all sources remains within the 25 mrem/yr limit.

3.6.4 Soil DCGLS

This section of the SER discusses the licensee's approach and proposed DCGLs for soil. The staff evaluation and analysis of the licensee's DCGLs is in Section 3.6.4.3, NRC Evaluation and Independent Analysis of DCGLs of Soil DCGLs and Area Factors. The staff is assessing the licensee's compliance with the requirements set forth in 10 CFR Part 50.82(a)(9)(ii)(D) Detailed plans for the final radiation survey because the DCGLs are used in the final radiation surveys.

3.6.4.1 Scenarios, Parameters, Uncertainty Analysis for Soil DCGLs

The licensee developed DCGLs for surface soil (0 - 0.15 m) and subsurface soil (0 – 1 m). The licensee applies the Resident Farmer scenario in the development of DCGLs. Although agriculture use of the property is unlikely given the zoning restrictions and soil classification, the licensee applies the Resident Farmer scenario to ensure that the critical group and exposure scenario produce a conservative and bounding compliance dose calculation. In this scenario a hypothetical adult farmer is assumed to live on the site and grow a portion of his/her food on the site, using the water for irrigation and drinking.

The Resident Famer is exposed through the following pathways: direct exposure to external radiation, inhalation dose from airborne radioactivity, direct ingestion of soil, ingestion of plants grown in contaminated soil and irrigated with site water, ingestion of meat and milk from livestock drinking well water and consuming fodder irrigated with well water, and drinking water. The radon exposure pathway and the aquatic food pathway are not included.

The licensee uses RESRAD, version 7.0 to calculate the soil dose factors and DCGLs.

The licensee assumed site-specific values for the following parameters:

- Kd values for site soil (sand)
- Area of contaminated zone
- Length parallel to aquifer flow
- Thickness of contaminated zone
- Density of contaminated zone
- Contaminated zone total porosity
- Contaminated zone field capacity
- Contaminated zone hydraulic conductivity
- Precipitation
- Density of saturated zone
- Saturated zone total porosity

- Saturated zone field capacity
- Saturated zone hydraulic conductivity
- Saturated zone hydraulic gradient
- Unsaturated zone thickness
- Unsaturated zone density
- Unsaturated zone total porosity
- Unsaturated zone effective porosity
- Unsaturated zone field capacity
- Unsaturated zone hydraulic conductivity

Table 10. Adjusted Soil Derived Concentration Guideline Levels (Adjusted to Account for Insignificant Contributor to Dose) (Table 6-40, LTP Rev 2)

Radionuclide	Surface DCGL pCi/g	Subsurface DCGL pCi/g
Co-60	4.26	3.44
Cs-134	6.77	4.44
Cs-137	14.18	7.75
Ni-63	3572.10	763.02
Sr-90	12.09	1.66

* To convert pCi/g to Bq/g multiply by 0.037

3.6.4.2 Soil Elevated Areas and Area Factors

The licensee generated area factors for areas from 1.0 m^2 up to the full source area of 64,500 m^2, by adjusting the area of the contaminated zone in the RESRAD model. The $DCGL_{EMC}$ calculation for soils will use Base Case DCGLs ($DCGL_{SS}$ and/or $DCGL_{SB}$), Area Factors (also found in Tables 5-16 and 5-17 from LTP Chapter 5) and the EMC comparison in accordance with LTP Chapter 5, Section 5.10.4. For soil in Class 1 open land FSS units, any areas of elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated.

Table 11. Area Factors for Surface Soil DCGLs, LTP, Rev 2 Table 6-41 (ML18052A851)

Area (m^2)	Cs-137	Co-60	Cs-134	Ni-63	Sr-90
1	15	12.3	13.3	8060	890
3	6.46	5.24	5.73	2730	313
10	3.06	2.47	2.72	823	103
30	2.10	1.68	1.86	275	40.2
100	1.62	1.29	1.44	82.6	16.4

Table 12. Area Factors for Subsurface Soil DCGLs, [LTP Table 6-42] (ML18052A851)

Area (m^2)	Cs-137	Co-60	Cs-134	Ni-63	Sr-90
1	20.4	11.0	15.2	6409	1500
3	9.26	4.91	6.92	2170	523

10	4.48	2.36	3.35	651	164
30	3.23	1.70	2.42	217	57.2
100	2.59	1.37	1 .95	65.1	17.6

3.6.4.3 NRC Evaluation and Independent Analysis of Soil DCGLs and Area Factors

The NRC staff has concluded that the choice of the Resident Farmer scenario is reasonable as it reflects the critical group under several circumstances. The critical group is defined in 10 CFR 20.1003 as, "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances." Additionally, the use of the Resident Farmer scenario is consistent with the guidance of NUREG-1757, and NUREG/CR-5512. Since the exposure pathways associated with the Resident Farmer scenario cover all the potential routes of exposures, it results in more restrictive DCGLs (lower concentration levels allowed to be left on-site) than other scenarios. The licensee states that the aquatic pathway from an onsite pond is not credible due to engineering as well as cost issues of construction and the site's proximity to Lake Michigan which negates any foreseeable need. This is a reasonable justification for eliminating the aquatic pathway. The included pathways reflect a subsistence farming practice and are feasible considering the physical, geological, and hydrogeologic characteristics of the site. Therefore, the Resident Farmer scenario is considered acceptable for consideration in developing soil DCGLs.

During the review, the NRC staff requested additional information regarding the soil DCGLs and soil area factors (ML16211A200). The NRC staff requested additional information about the site-specific Kds. Specifically, it was unclear if choosing the minimum site-specific Kd for Sr-90 is conservative in calculating the soil DCGLs given that primary pathway for dose in the surface and subsurface soil scenarios for Sr- 90 is the plant pathway (water independent), as opposed to the drinking water pathway which peaks later in time. As a result of the RAIs, the licensee agreed to apply the maximum site-specific Kds for both Sr-90 and Ni-63 in the final Surface and Subsurface DCGL modeling as opposed to the minimum Kds originally used. The staff reviewed the revised DCGLs and area factors for soil provided by the licensee in its response to request for additional information (ML16211A200).

An Area Factor (AF) is defined as the ratio of the corresponding DCGL for a smaller area to the DCGL for the entire site (64,500m^2). The denominator is therefore expected to be the site-wide DCGLs. The NRC staff notes that the denominator used to calculate the AFs was not the same as the soil DCGL value assumed in the LTP. The soil DCGLs were created from the RESRAD Summary Reports in TSD 14-010, Rev. 0 Attachment 10. In these Summary Reports the contaminated fraction for meat, milk, and plants are listed as +1. (These parameter values were later adjusted to -1 for smaller area sizes in calculating the AFs.) TSD 14-011 states that using the different base case value in AF calculations is appropriate because the AFs are relative values and it is conservative because using the soil DCGL values in the LTP as the denominator would have resulted in higher calculated AFs. Since the AF is multiplied by the $DCGL_W$ to obtain the $DCGL_{EMC}$, a higher AF would have been less conservative than a lower AF. Therefore, the staff finds the use of the alternative value for the denominator in calculating the AF to be appropriate since it resulted in a lower AF.

The staff has reviewed the dose modeling analyses for unrestricted release as part of the review of the licensee's LTP, using the Consolidated Decommissioning Guidance, Volume 2, Section 5.1.2 (Surface Soil Evaluation Criteria). In determining the dose, the licensee has applied a combination of the conceptual model, exposure scenario, mathematical model and input parameters to calculate a reasonable estimate of dose. The licensee has adequately considered the uncertainties inherent in the modeling analysis in compliance with 50.82(a)(9)(D).

3.6.5 Buried Piping DCGLs

Buried piping may remain onsite at the time of license termination and the residual contamination may remain on the internal surfaces of these pipes. As per the LTP, Buried piping is defined as pipe that runs through soil. The list of buried piping, is provided in Zion*Solutions* TSD 14-016, Rev 0.

As described in Section 3.6.2 of this SER on Radionuclides of Concern, the licensee is currently using the results of auxiliary basement concrete cores to represent the ROC and mixture for buried piping. Buried piping will be characterized as part of the continuing characterization program in accordance with LTP Chapter 2, Section 2.5.

3.6.5.1 Scenarios, Parameters, and Uncertainty Analysis for Buried Piping DCGLs

The licensee evaluated three scenarios in developing the buried piping DCGLs: Excavation, Insitu Unsaturated, Insitu Saturated. The Excavation scenario assumes that all source term within the buried piping is brought to the surface and mixed with the surface soil. The Insitu Unsaturated scenario assumes that all pipes are located at 1 m below the ground surface and the Insitu Saturated scenario assumes that all pipes are located in the saturated zone.

To determine the Buried Piping DCGL, the licensee calculated the pCi/g within a 0.15 m contaminated zone mixing layer assuming 1 dpm/100 cm^2 on the pipe surface. The licensee determined the overall piping DSR by summing the Excavation is added to the maximum of either the Insitu Unsaturated or Insitu Saturated DSR. The DSR (mrem per pCi/g) is used in conjunction with the pCi per dpm/100 cm^2 to find the DCGL (dpm/100 cm^2) equivalent to 25 mrem/yr).

Table 13. Base Case DCGLs for Buried Pipe ($DCGL_{BP}$) [LTP Chapter 5, Table 5-9]

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

The licensee also derived Operational DCGLs for buried pipe based on assigning an a priori fraction of 0.256 of the dose limit (TSD 17-004, Rev. 3). The licensee may revise these Operational DCGLs once the FSS of structures is complete, by incorporating the difference between the *a priori* fraction of dose for the maximum basement (0.448) and the actual fraction of dose for the maximum basement as measured by FSS results.

Table 14. 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

3.6.5.2 Buried Piping Elevated Area Consideration

For buried pipe, areas of elevated activity are defined as any area identified by measurement (systematic or judgmental) that exceeds the Operational DCGL but is less than the Base Case DCGL. Any residual radioactivity greater than the Base Case DCGLs will be remediated. Locations identified by measurement or sample analyses which exceed the Operational DCGL will be subject to additional surveys to bound the elevated area.

3.6.5.3 NRC Evaluation and Independent Analysis of Buried Piping DCGLS

The scenarios assumed for buried piping, while not realistic, are adequately conservative. It is not physically possible for the entire source term from piping to be simultaneously excavated and remain underground, so adding the Excavation and Insitu doses is conservative. Furthermore, since some of the radionuclides provide a dose through the groundwater pathway, while others pose more of a risk when they are closer to the surface and all the pipes are at various elevations (e.g., via plant uptake), it is appropriate to simplify the conceptual model and take the maximum DSR for each radionuclide from either the Insitu Unsaturated scenario or the Insitu Saturated scenario.

As part of the RAIs, the NRC staff asked the licensee to explain the reasons for using different parameter values from those assumed in the Surface DCGL calculation that may not be conservative for the calculation of the Buried Piping DCGLs. As a result, the licensee revised the Buried Pipe DCGLs to include a mixing depth of 0.15 m and unsaturated zone Kds that are consistent with the Surface Soil RESRAD parameters (ML16211A200).

The NRC staff also asked that the licensee consider that the sensitivity analysis conducted for the Surface DCGL model may not result in the same sensitive parameters or relationships as the conceptual model for buried piping. Therefore, staff asked the licensee to verify that appropriate values are chosen for sensitive parameters. In response, the licensee performed a full uncertainty analysis for the buried pipe scenarios. As a result of the uncertainty analysis the following parameters were adjusted: Saturated Zone Hydraulic Gradient for the Insitu Saturated scenario and the Depth of Roots for the Insitu Unsaturated scenario (ML16211A200).

The NRC staff also requested additional justification for the assumed contaminated zone thickness of 0.15 m for the buried pipe models. As a result, the licensee performed a sensitivity analysis of the CZ thickness. For the buried pipe scenario, as the contaminated zone thickness increases the source term concentration decreases as an inverse linear function of the mixing depth and therefore the sensitivity analysis accounted for both effects. With the exception of Sr-90, increasing the thickness of the CZ resulted in equal or lower dose to source ratios. Sr-90 showed an 8% increase in dose at a 1 m source term depth for the Insitu Saturated scenario and a 13% increase at 1 m depth for the Excavation Scenario. Therefore, the licensee increased the Sr-90 DSRs by factors of 1.08 and 1.13 for the Insitu Saturated and Excavation scenarios, respectively. Furthermore, as a result of the RAI process, the licensee removed any use of a $DCGL_{EMC}$ for buried piping. The licensee committed to remediating any buried piping with residual radioactivity greater than the Base Case DCGLs for buried piping.

The staff has reviewed the dose modeling analyses for unrestricted release as part of the review of the licensee's LTP, using the Consolidated Decommissioning Guidance, Volume 2. In developing the DCGLs for buried piping, the licensee has applied an appropriate combination of the conceptual model, exposure scenario, mathematical model and input parameters. The licensee has adequately considered the uncertainties inherent in the modeling analysis consistent with 50.82(a)(9)(D).

3.6.6 Dose from Groundwater

The licensee stated that they have not identified groundwater contamination to date. If groundwater contamination is identified, the potential dose from existing groundwater will be calculated using the BFM groundwater exposure factors.

The NRC staff finds that this approach is acceptable to calculate the groundwater dose, if any, because the BFM groundwater exposure factors provide the dose per exposure to contamination in a well that is used as a water source and are therefore applicable to groundwater outside the basements.

3.6.7 NRC Evaluation for Compliance with Radiological Criteria for License Termination

As described in Section 3.6.1 "Radiological Criteria for License Termination" of this SER, the licensee must demonstrate that the dose from all source terms at the site meets the radiological criteria for license termination using Equation 1.

The licensee assigned a fraction of 0.448 to the basement source term in TSD 17-004, Rev. 3. This is based on the assumption that the maximum basement dose will be no greater than 0.448 of 25 mrem/yr (11.2 mrem/yr), which is the dose contribution the licensee expects from the Auxiliary Basement. In the case that the maximum basement presents a dose fraction greater than 0.448, the licensee has committed to performing additional remediation in order to reduce the basement fraction to be 0.448 or less.

The remaining dose fraction for the compliance equation, assuming fraction for Basements is 0.448, is 0.552. The expected fraction of dose for the groundwater term (f_{GW}) based on the instrument MDC is assumed to be 0.040. The licensee has distributed the remaining dose fraction of 0.512 between soil and buried pipe at 0.256 each. FSS surveys of open land and buried pipe survey units performed prior to the completion of the FSS of structures will be performed using these a priori fractions.

Once the FSS of structures is complete, the soil and buried pipe Operational DCGLs will be revised by incorporating the difference between the a priori fraction of dose for the maximum basement ($f_{Basement}$) and the actual fraction of dose for the maximum basement as measured by FSS results. Should the licensee desire to assign a fraction greater than 0.448 to the basements, it may invalidate the results for final status surveys which have already been completed on the buried piping or soil. However, a lower fraction than 0.448 assigned to the basements would allow a greater fraction for soil and buried piping. Assigning a higher fraction to soil and buried piping would not invalidate the completed FSS results. Revision of any completed release records for any FSS performed prior to the establishment of final soil and buried pipe Operational DCGLs will not be necessary as the Operational DCGLs used will be based on a lower a priori dose fraction.

The NRC staff has reviewed the dose modeling analyses used to generate DCGL values for soil, building surfaces, and buried and embedded piping at the Zion site as part of the review of the LTP, using NUREG-1757, Volume 2. Based upon the analyses above, the staff concludes that the dose modeling is reasonable and is appropriate for the exposure scenarios under consideration. The NRC staff concludes that the DCGL values developed for soil, backfilled basements, and buried and embedded piping at the Zion site are consistent with the 0.25 mSv/yr (25 mrem/yr) annual dose criterion for unrestricted release in 10 CFR 20.1402 as long as the overall dose from all source terms is verified using Equation 1. This conclusion is based on the modeling effort performed by Zion*Solutions* and the independent, confirmatory analyses performed by the NRC staff.

3.7 Groundwater

The NRC staff has evaluated the geologic and hydrogeologic conditions at ZNPS to determine whether operations have resulted in radiological impacts within the groundwater. The evaluation and summary of information below is based upon information provided in the ZNPS LTP, supporting documents referenced in the LTP, and NRCs independent assessment of the LTP and supporting documents.

3.7.1 Geology

The near-surface geology of northeastern Illinois consists of unconsolidated deposits which range from approximately 27 to 56 m [90 to 150 ft] in thickness. The surface deposits are comprised mostly of unconsolidated glacial deposits which rest on a series of sedimentary rock layers that were deposited in the Paleozoic Era. The thickness of the Paleozoic sedimentary rocks in northeastern Illinois is approximately 1219 m [4,000 ft]. These sedimentary bedrock layers dip gently toward the east at the rate of approximately 53 meters per kilometer [10 ft per mile] and rest on Precambrian basement rock.

The surface deposits in the vicinity of the ZNPS consist of three units that are of irregular thicknesses. The uppermost layer is identified as the Upper Sand Unit and ranges from approximately 9 to 11 m [30 to 35 ft] in thickness. Immediately below the Upper Sand Unit lies a layer that is predominantly made up of silt and clay and is identified at the Silty Clay Unit. The Silty Clay Unit ranges from about 6 to 12 m [20 to 40 ft] in thickness. The lower unconsolidated layer, which rests on the upper bedrock layer, is a mixture of sand and glacial deposits. This unit, called the Lower Sand Unit ranges from about 9 to 15 m [30 to 50 ft] in thickness.
In descending order, the aforementioned units are further discussed and were characterized during the various site investigations performed for the ZNPS.

3.7.2 Upper Sand Unit

The Upper Sand Unit consists of dense to very dense granular soils which range in gradation from very fine sand to fine to coarse sand, and which contains some gravel and occasional cobbles and boulders (i.e. shallow granular lake deposits). The Upper Sand Unit ranges from approximately 9 to 11 m [30 to 35 ft] in thickness. This unit includes both native and fill sand that was during the construction of ZNPS. The Upper Sand Unit includes the surficial deposits of the Zion beach-ridge plain and consists of sand and gravel of the Lake Michigan Formation. The Lake Michigan Formation describes Holocene shallow-water, near-shore beach sediments predominantly consisting of medium-grained sand with local lenses of sandy gravel, and containing beds of silt.

3.7.3 Silty Clay Unit

The Silty Clay Unit consists of hard silt, silty clay, clayey silt, and sandy silt which contain some sand and gravel and occasional cobbles and boulders. The Silty Clay Unit ranges from about 6 to 12 m [20 to 40 ft] in thickness. The Silty Clay Unit is consistent with quiet water lacustrine deposits and may be associated with post-glacial Lake Michigan (Nipissing Phase).

3.7.4 Lower Sand Unit

The Lower Sand Unit consists of dense to very dense sands and silty sands which contain some gravel, occasional cobbles and boulders, and layers of hard silty clay, clayey silt, and sandy silt. The Lower Sand Unit ranges from approximately 9 to 15 m [30 to 50 ft] in thickness. The Lower Sand Unit is consistent with recurring sequences of beach and quiet water lacustrine deposits and may be associated with the extreme Lake level fluctuations. As Lake levels rose, beach deposits moved westward with the shoreline and were followed by quiet water silt and clay deposits (a transgressive sequence). As Lake levels fell, the beach moved eastward with the shoreline (a regressive sequence).

3.7.5 Bedrock

The overburden sediments are underlain by the Niagara Dolomite at approximately 30 m below ground surface (bgs) [100 ft bgs]. The Niagara Dolomite is a consolidated layer of carbonaceous marine sediments laid down in the Silurian Period and is approximately 61 m [200 ft] thick in the vicinity of ZNPS. Below the Silurian carbonates lie Pre-Cambrian through Ordovician sedimentary rocks, including shales, carbonates, and sandstone. Crystalline basement rock is located at a depth of approximately 762 m [2,500 ft]. The sedimentary bedrock strata are generally horizontal with a gentle dip to the east.

3.7.6 Hydrogeology

The Upper Sand Unit is a high permeability unit that is in hydraulic communication with Lake Michigan, the regional discharge feature, which generally allows unrestricted lateral groundwater flow, with the exception of the areas around plant structures and a discontinuous sheet pile wall that restricts groundwater flow for the extent of the protected area. These deep structures alter the local flow patterns, however ultimate discharge of groundwater is to Lake Michigan to the east. Lake Michigan acts as a major regional discharge zone for groundwater. The groundwater flow in both unconsolidated deposits and bedrock units in the region is generally toward the lake; however, localized pumping induces variations in flow directions in the bedrock aquifers.

Groundwater at ZNPS is encountered at a depth less than 6 m [20 ft] below ground surface in the shallow granular lake deposits identified above as the Upper Sand Unit. This shallow water-bearing zone is isolated from the underlying regional bedrock aquifers by the low permeability of the glacial silts and clays of the Silty Clay Unit that act as an aquitard with a thickness of approximately 9 to 15 m [30 to 50 ft].

Some of the Station structures are constructed to depths of approximately 18 m bgs [60 ft bgs]. Excavations were completed from grade, through the Upper Sand Unit and into the topmost portion of the Silty Clay Unit. The Silty Clay Unit (underlies the Upper Sand Unit) extends approximately 5 m [15 ft] below the deepest structural feature at ZNPS. Due to the continuity of the Silty Clay Unit across ZNPS, the underlying Lower Sand Unit and the regional Silurian dolomite bedrock aquifer is not known to be hydraulic communication with the Upper Sand Unit. The lack of hydraulic communication across the Silty Clay Unit effectively mitigates potential transport of contaminants between units if impacts from decommissioning were identified.

3.7.7 Groundwater Monitoring

Assessments of any residual radioactivity in groundwater at the ZNPS will be through groundwater monitoring wells. The data collected from the monitoring wells will be used to ensure that the concentration of well water available, based upon the well supply requirements assumed for the resident farmer (i.e., resident farmer's well) in chapter 6 of the LTP, is below the U.S. Environmental Protection Agency (EPA) maximum contaminant levels (MCLs) (e.g., 20,000 pCi/L for tritium (H-3)). This will ensure that the dose contribution from groundwater is a small fraction of the limit in 10 CFR 20.1402. Sampling results have shown that no groundwater contamination has been identified to date.

The Radiological Environmental Monitoring Program (REMP) was initiated in 1973 and included monitoring of public drinking water. Environmental monitoring was conducted by sampling at indicator and control (background) locations in the vicinity of ZNPS to measure changes in radioactivity levels that may be attributable to the station. The two background locations are the Kenosha Water Works located 16.1 km [10.0 miles] north of ZNPS and the Lake Forest Water Works located 20.8 km [12.9] miles south of ZNPS. The two indicator locations are the Lake County Water Works located 2.3 km [1.4 miles] north northwest of ZNPS and Waukegan Water Works located 9.8 km [6.1 miles] south of the ZNPS. If significant changes attributable to Zion Station are measured, these changes are correlated with effluent releases. The sampling results from the intakes of the public water stations were considered the principal pathways used to calculate dose from liquid effluents through ingestion of drinking water. The data has consistently supported a conclusion that the operation of ZNPS has not impacted sources used for public water supplies.

In 2006, the Radiological Groundwater Protection Program (RGPP) was initiated to evaluate the impact of ZNPS operations on groundwater. Fifteen new monitoring wells were installed including 11 permanent and 4 temporary monitoring wells. The fifteen new monitoring wells (MW-ZN-01S, MW-ZN-02S, MW-ZN-03S, MW-ZN-04S, MW-ZN-05S, MW-ZN-06S, MW-ZN-07S, MW-ZN-08S, MW-ZN-09S, MW-ZN-10S, MW-ZN-11S, TW-ZN-100, TW-ZN-101, TW-ZN-102, and TW-ZN-103) were installed within the Upper Sand Unit, which consists of a primarily fine-grained sand that overlies the Upper Silty Clay Unit.

Initially, separate samples were collected from the lower portion and the upper portion of the screened interval within the permanent monitoring wells. The lower sampling interval targets potential releases from deep structural features such as the basement of the Auxiliary Building. The upper sampling interval targeted potential surface and near surface releases such as spills.

Samples of water are collected, managed, transported and analyzed in accordance with approved procedures following EPA methods. Sample locations, sample collection frequencies and analytical frequencies are controlled in accordance with approved station procedures.

Analytical laboratories are subject to internal quality assurance programs, industry crosscheck programs, as well as nuclear industry audits. Station personnel review and evaluate all analytical data deliverables as data are received. Analytical data results are reviewed by both station personnel and an independent hydrogeologist for adverse trends or changes to hydrogeologic conditions.

During decommissioning, ZionSolutions has and will continue to implement a Radiation Protection Program in accordance with the license specifications and the requirements of 10 CFR Part 20. Potential groundwater impacts will continue to be monitored by the routine sampling of the 11 permanent onsite RGPP wells at ZNPS.

3.7.8 Groundwater Sampling and Analysis for Radionuclides of Concern

Tritium was detected in monitoring well MW-ZN-01S in May of 2006. The analysis of groundwater samples collected from both the upper and lower portions of the screened intervals in MW-ZN-01S resulted in H-3 concentrations of 261 ± 124 pCi/L and 506 ± 141 pCi/L respectively. Well MW-ZN-01S was re-sampled in June of 2006, and the analysis of the samples resulted in H-3 concentrations of 220 ± 123 pCi/L and <200 pCi/L for the upper and lower screened intervals respectively. Since the initial sample results noted above, tritium has not been observed in MW-ZN-01S or any other ZNPS monitoring well greater than the MDC of 200 pCi/L. While the level of H-3 was greater than the detection limit, the reported values were significantly less than the EPA drinking water standard of 20,000 pCi/L.

Sr-90 was also detected during the May 2006 sampling event, specifically at monitoring wells MW-ZN-05S and MW-ZN-06S. The Sr-90 concentrations were respectively, 1.93 ± 0.8 pCi/L (MDC: 1.3 pCi/L) and 1.77 ± 0.72 pCi/L, (MDC: 1.15 pCi/L). Based on the uncertainty and associated MDC both samples were very near the detection limit. These concentrations were less than the EPA drinking water standard of 8 pCi/L and both of these locations are up-gradient from the groundwater flow direction and should not be impacted by ZNPS activities. Additional radionuclides have been detected in groundwater at the Facility at concentrations greater than their respective detection limits. These include the natural radionuclides associated with the soils, silts and clays. These radionuclides were potassium-40, thorium-228, radium-226 and actinium-228. These naturally occurring radionuclides are expected to be within the soils and detected during sample analyses.

Gamma-emitting radionuclides associated with licensed plant operations were not detected at concentrations greater than their respective Lower Limits of Detection (LLDs) in any of the groundwater or surface water samples obtained to date.

3.7.9 Area Groundwater Use

The City of Zion provides municipal water to City residents and the surrounding area. The water is obtained from Lake Michigan by means of an intake pipe located approximately 1.6 km [1 mile] north of the Site and extending 914 m [3,000 ft] into the Lake. The City of Zion municipal code requires all improved properties to be connected to the City's water supply. There is an exception for some existing wells constructed prior to March 2, 2004. However, as previously discussed, no impacts to the groundwater have been identified on-site or off-site during operations or decommissioning to date.

3.7.10 Conclusion

The staff reviewed the geologic and hydrogeologic site characterization provided in the LTP and referenced reports and, based on the acceptance criteria in NUREG-1700, determined that it is sufficiently detailed and fulfills, in part, the site characterization requirement of 10 CFR 50.82 (a)(9)(ii)(A)(requiring the LTP to include "[a] site characterization") and the remediation plans requirement of 10 CFR 50.82(a(9)(ii)(C). The information provided will allow the NRC staff to determine the extent and range of radiological contamination to the groundwater prior to license termination. While detectable concentrations of radionuclides have been reported within the groundwater of the ZNPS restricted area, concentrations for radiological constituents have not exceeded EPA MCLs for drinking water. No imminent threats to human health or the environment due to radiological constituents in the groundwater have been identified, although additional assessments, i.e., groundwater monitoring, during decommissioning will be ongoing.

The LTP has identified radiological spills, leaks, and releases with the potential of impacting groundwater. The identified spills, leaks, and releases have not resulted in impacts to the groundwater based on the analyses performed on samples obtained from groundwater monitoring wells.

The NRC will obtain split samples from the groundwater monitoring wells prior to license termination. The samples will be analyzed for all of the primary constituents of concern identified in the LTP.

3.8 Site Specific Cost Estimate

3.8.1 Introduction

Per 10 CFR 50.82(a)(9)(ii)(F), the LTP must include "[a]n updated site-specific estimate of remaining decommissioning costs." On December 19, 2014, ZionSolutions submitted a License Amendment Request (LAR), "License Amendment Request for the License Termination Plan," for Zion Units 1 and 2 (ML15005A330), for NRC review and approval. Appendix 3 contained the License Termination Plan (LTP). However, this initial LTP did not include financial data for consideration. By letter dated February 26, 2015 (ML15061A230), ZionSolutions submitted LTP Chapter 7, "Update of the Site-Specific Decommissioning Costs." (ML15061A232) This submittal updating the LTP reflects, among other things, a net balance in the decommissioning trust fund of $317.1 Million.

Staff's review is limited to evaluation of the site-specific decommissioning costs contained in the LTP Update using the acceptance criteria in NUREG-1700, Revision 2, Section 2.7.1.

3.8.2 Background

Ownership of ZNPS and the responsibility for decommissioning was transferred from Exelon Corporation via NRC Order to ZionSolutions, a subsidiary of EnergySolutions, LLC, (a direct subsidiary of EnergySolutions, Inc.), on May 4, 2009 (ADAMS Accession No. ML090930037), with a provision for the transfer of the ISFSI back to Exelon at partial site release. At the time of the Order, the financial qualifications of ZionSolutions to perform its obligations under the License had been demonstrated by, among other considerations, the provision of additional financial assurance in the form of a $200 million irrevocable letter of credit. On March 18, 2008, ZionSolutions submitted an amended PSDAR for ZNPS (ADAMS Accession No.

ML080840398). The updated PSDAR included the estimated costs and available decommissioning trust funds for active reactor decommissioning and dismantlement. Decommissioning began in October of 2010, and spent fuel and greater than Class C Waste were transferred to the on-site ISFSI in 2015. Later, ZionSolutions was granted a specific exemption from 10 CFR 50.82 to use funds from the nuclear decommissioning trusts for irradiated fuel management, consistent with the ZNPS updated Irradiated Fuel Management Plan and Post Shutdown Decommissioning Activities Report (70 FR 44213, July 30, 2014). Staff notes that EnergySolutions, Inc. (the ultimate parent holding company of ZionSolutions) was acquired by Rockwell Holdco, Inc. and the NRC order approving the indirect transfer was issued on May 8, 2013, (ADAMS Accession No. ML13122A063).

3.8.3 Regulatory Requirements and Criteria

Per 10 CFR 50.82(a)(9)(ii)(F), the LTP must include "[a]n updated site-specific estimate of remaining decommissioning costs." Guidance on meeting 50.82(a)(9)(ii)(F) is provided in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" section 7.

Guidance on the staff's review of the updated site-specific estimate of remaining decommissioning costs in the LTP is provided by section 7 of NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans." As stated therein, the acceptance criteria are:

> The LTP decommissioning cost estimate includes an evaluation of the following cost elements:
>
> > cost assumptions used, including a contingency factor (normally 25%);
> > major decommissioning activities and tasks;
> > unit cost factors;
> > estimated costs of decontamination and removal of equipment and structures;
> > estimated costs of waste disposal, including applicable disposal site surcharges and transportation costs;
> > estimated final survey costs; and
> > estimated total costs.
>
> The LTP focuses on detailed activity by activity cost estimates.
>
> The LTP also compares the funds available for decommissioning with the calculated total cost from the licensee's detailed cost analysis. In addition, Regulatory Guide 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors" (Ref. 10), explains in detail the methods for estimating decommissioning costs, as well as accepted financial assurance mechanisms.
>
> The LTP cost estimate is based on credible engineering assumptions, and the assumptions are related to all major remaining decommissioning activities and tasks and are consistent with the information identified in Sections B3 and B4 of this SRP.
>
> The LTP cost estimate includes the cost of the remediation action being evaluated, the cost of transportation and disposal of the waste generated by the action, and other costs that are appropriate for the specific case. The current version of NUREG-1307, "Report

on Waste Burial Charges" (Ref. 11), provides guidance on estimating waste disposal costs.

Therefore, the cost estimate should focus on the remaining work and include the cost of labor, materials, equipment, energy, and general services. The cost of transportation and disposal of the waste generated by the remedial work should also be included.

3.8.4 Evaluation

3.8.4.1 Evaluation of the Updated Site-Specific Estimate of Remaining Decommissioning Costs

ZionSolutions estimated the remaining decommissioning costs to complete the radiological decommissioning of ZNPS, including contingency, as of September 30, 2014 at $309 Million (year of expenditure dollars). The staff notes that this estimate does not include the costs associated with the disposal of non-radiological materials or structures beyond those necessary to terminate the Part 50 license. Costs associated with the construction or operation of an ISFSI, in consideration of the transfer of the ISFSI back to Exelon upon partial site release, are also not included in the estimate.

According to the update of the LTP, ZionSolutions' decommissioning cost estimate for ZNPS is based upon the schedule of remaining work and relies, in large part, on the experience gained from similar tasks completed at ZNPS since decommissioning started in October, 2010. The current estimate also reflects the contracts currently in place; adding a degree of certainty to work productivity, the cost of labor, and the cost of services to complete decommissioning. The estimate update incorporates the site-specific and special tasks that have been prescribed or implemented as a result of the ongoing decommissioning planning. The elements supporting the basis of the estimate and the sources of information, methodology, site-specific considerations, assumptions, and total costs were presented in the update to the LTP (Section 7 of the Application).

Prior to starting the detailed review of the cost estimate, the NRC staff reviewed the LTP and the updated estimate to confirm the supporting systems/structures necessary to support the safe operation had been identified in the estimate. The validity of the cost estimate is based on a reasonable estimate of the costs to decommission the remaining major project activities, as well as required resources. Consideration was also given to contingency factors presented in the cost estimate.

In the 2015 update to the LTP, the licensee divided the estimated remaining decommissioning costs (in year of expenditure dollars) into the following Resource Categories: Labor; Equipment, Materials and Supplies; Fixed-Price Contracts, Services & Fees; and Radioactive Waste Packaging, Transportation & Disposal. The estimated total remaining cost of decommissioning based on the above factors, was $284.3 Million; $309 Million with the application of a contingency in the amount of $24.7 Million. The staff reviewed the contingency factors and the resource requirement used in Zion's updated site-specific cost estimate and found that they are reasonable.

3.8.4.2 Evaluation of the Decommissioning Funding Plan

Pursuant to 10 CFR 50.82(a)(8)(v):

> After submitting its site-specific DCE required by paragraph (a)(4)(i) of this section, and until the licensee has completed its final radiation survey and demonstrated that residual radioactivity has been reduced to a level that permits termination of its license, the licensee must annually submit to the NRC, by March 31, a financial assurance status report. The report must include the following information, current through the end of the previous calendar year:
>
> (A) The amount spent on decommissioning, both cumulative and over the previous calendar year, the remaining balance of any decommissioning funds, and the amount provided by other financial assurance methods being relied upon;
>
> (B) An estimate of the costs to complete decommissioning, reflecting any difference between actual and estimated costs for work performed during the year, and the decommissioning criteria upon which the estimate is based;
>
> (C) Any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and
>
> (D) Any material changes to trust agreements or financial assurance contracts.

Accordingly, on March 30, 2017, ZionSolutions submitted the Units 1 and 2 Decommissioning Funding Status (DFS) Report (ADAMS ML17094A735). The subsequent NRC staff analysis of the annual DFS report was based on a DTF balance for radiological decommissioning of $191.5 million as of December 31, 2016. Staff applied a real rate of return of 2.0 percent to its analysis through the expected license termination year of 2020 and noted an additional $200 Million line of credit available in the event of shortfalls.

The NRC staff finds the updated site-specific cost estimate for remaining radiological decommissioning costs for Zion Nuclear Power Stations, Units 1 and 2, appears reasonable, and that the DTF balance, as of December 31, 2016, will be sufficient to fund the remaining radiological decommissioning expenses.

Subsequent submissions by ZionSolutions (Responses to the NRC's request for additional information on the license termination plan and a subsequent "License Termination Plan, Revision 2," (Package ML17215A095 and ML18052A857, respectively)), were considered by staff, but did not provide significant financial data for consideration. Revision 1 of the LTP only added the costs for final status surveys to the financial data which represented approximately 2% of the expenses and did not change the staff conclusions. The financial information provided in Chapter 7 was not changed in Revision 2 of the LTP.

3.8.5 Conclusion

The NRC staff finds that decommissioning cost estimate and the decommissioning funding plan associated with ZionSolutions' update of the site-specific decommissioning costs for Units 1 and

2, are adequate and provide sufficient details associated with the funding mechanisms. The NRC staff, therefore, concludes that the licensee's License Termination Plan Update of the Site-Specific Decommissioning Costs, submitted via letter dated February 26, 2015, and supplemented on July 20, 2017, and February 7, 2018, for the Zion Nuclear Power Station, Units 1 and 2, complies with 10 CFR 50.82(a)(9)(ii)(F).

3.9 Environmental Report

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(G)(requiring the LTP to include "[a] supplement to the environmental report, pursuant to § 51.53, describing any new information or significant environmental change associated with the licensee's proposed termination activities"), the licensee is required to provide a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental changes associated with the licensee's proposed license termination activities. The licensee asserted that Chapter 8 "Supplement to the Environmental Report" of Rev. 2 of the LTP fulfilled this requirements. See Section 5 "Environmental Considerations" of this SER for details concerning the staff's environmental review.

3.10 Change Procedure

The licensee has proposed that it be authorized to make certain changes to the NRC-approved LTP without NRC approval if these changes do not:

(1) Require Commission approval pursuant to 10 CFR 50.59; (2) Result in significant environmental impacts not previously reviewed; (3) Detract or negate the reasonable assurance that adequate funds will be available for decommissioning; (4) Decrease a survey unit area classification (i.e., impacted to not impacted; Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3) without providing the NRC a minimum 14 day notification prior to implementing the change in classification; (5) Increase the derived concentration guideline levels (DCGL) and related minimum detectable concentrations (for both scan and fixed measurement methods); (6) Increase the radioactivity level, relative to the applicable DCGL, at which an investigation occurs; (7) Change the statistical test applied other than the Sign test; (8) Increase the approved Type I decision error above the level stated in the LTP Increase the approved Type I decision error above the level stated in the LTP; (9) Change the approach used to demonstrate compliance with the dose criteria (e.g., change from demonstrating compliance using derived concentration levels to demonstrating compliance using a dose assessment that is based on final concentration data); (10) Change parameter values or pathway dose conversion used to calculate the dose such that the resultant dose is lower than in the approved LTP and if a dose assessment is being used to demonstrate compliance with the dose criteria; (11) Reuse concrete from demolished structures, other than from the list of areas specified in Section 2.1.1 of TSD 17-010, "Final Report - Unconditional Release Surveys at the Zion Station Restoration Project, Revision 1", as backfill; (12) Assign a dose for reuse concrete other than the dose values provided along with the LTP (as shown in Table 6-53 (Revision 2) of the LTP) and documented in Section 8 and Table 33 of TSD 14-010, "RESRAD Dose Modeling for Basement Fill Model and Soil DCGL and Calculation of Basement Fill Model Dose Factors and DCGLs, Revision 6"; or (13) Use area-specific surrogate ratios that are less than the maximum surrogate ratios (H-3/Cs-137, Ni-63/Co-60, Sr-90/Cs-137) presented in Table 5-15 (Revision 2) of the LTP.

The licensee will submit changes to the LTP not requiring NRC approval as an update to the final safety analysis report, in accordance with 10 CFR 50.71(e). The staff concludes that authorizing the licensee to make certain changes, during the final site remediation, is acceptable, subject to the above listed conditions.

4.0 STATE CONSULTATION

In accordance with the NRC's regulations, Kay Foster, Bureau Chief, Nuclear Facility Safety, Division of Nuclear Safety, Illinois Emergency Management Agency, was notified of the proposed issuance of the amendment. The State official replied on August 13, 2018, stating that they support approval.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21 (stating criteria for and identification of licensing and regulatory actions requiring environmental assessments), 51.32 (addressing a finding of no significant impact), and 51.35 (proving the requirement to publish finding of no significant impact, and limiting pre-publication Commission action), an environmental assessment and finding of no significant impact was published in the Federal Register on June 29, 2018 (83 FR 30783). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSIONS

The NRC has concluded, based on the considerations discussed above, that there is reasonable assurance that the remainder of decommissioning activities (1) will be performed in accordance with the regulations in 10 CFR Part 50; (2) will not be inimical to the common defense and security or to the health and safety of the public, and (3) will not have a significant effect on the quality of the environment.

7.0 LIST OF CONTRIBUTORS

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8.0 LIST OF ACRONYMS

AEC	Atomic Energy Commission
AF	Area Factor
ALARA	As Low As Is Reasonable Achievable
ANL	Argonne National Laboratory
bgs	below ground surface
BMP	Best Management Practice
Bq/g	Becquerel per gram
Bq/L	Becquerel per liter
CAB	Citizens Advisory Board.
CCS	Continuing Characterization Survey
CFR	Code of Federal Regulations

CFS	Containment Foundation Sump
cm	centimeters
cm^2	square centimeter
COC	Chain of Custody
D&D	Decontamination & Decommissioning
DAW	Dry Activated Waste
DCF	Dose Conversion Factor
DCGL	Derived Concentration Guideline Limit
$DCGL_B$	Base Case DCGL for structural surfaces
$DCGL_{EMC}$	DCGL that represents the same dose to an individual for residual radioactivity in a smaller area within a survey unit.
$DCGL_W$	DCGL for the average residual radioactivity in a survey unit
DOE	U.S. Department of Energy
DP	Decommissioning plan
dpm	disintegrations per minute
dpm/100cm^2	disintegrations per minute per 100 square centimeters
DPR	Decommissioning Project Report
DQA	Data Quality Assessment
DQO	Data Quality Objective
DSAR	Defueled Safety Analysis Report
DTSC	Department of Toxic Substances Control
EA	Environmental Assessment
EMC	Elevated Measurement Comparison
EPA	U.S. Environment Protection Agency
ETD	Easy to detect
FR	Federal Register
Ft^3	cubic foot
FGR	Federal Guidance Report
FSME	Office of Federal and State Materials and Environmental Management Programs
FSS	Final Status Survey
GEIS	Generic Environmental Impact Statement
GTCC	Greater than Class C
HEPA	High Efficiency Particulate Air filter
HPGe	High Purity Germanium
HSA	Historical Site Assessment
HSE	Health, Safety, and Environment
HTD	Hard to Detect
IA	Industrial Area
ICS	Initial Characterization Survey
ISFSI	Independent Spent Fuel Storage Installation
ISOCS	In Situ Object Counting System
kV	kilovolt
LBGR	Lower Boundary of the Gray Region
LLRW	Low-level Radioactive Waste
LTP	License Termination Plan
m^2	square meter
m^3	cubic meter
MARSSIM	Multi-Agency Radiation Survey And Site Investigation Manual
MDC	Minimum Detectable Concentration
MeV	Mega electron Volts
uR/hr	microroentgen per hour

mrem/hr	millirem per hour
mrem/yr	millirem per year
MSL	Mean Sea Level
mSv/yr	milliSievert per year
nC/Kg-hr	nanocoulomb per kilogram per hour
NEI	Nuclear Energy Institute
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
OWS	Oil/Water Separator
PAB	Primary Auxiliary Building
PCB	Polychlorinated Biphenyl
pCi/g	picocurie per gram
pCi/L	picocurie per Liter
PM_{10}	particular matter of 10 microns
PRCC	partial rank correlation coefficient
PSDAR	Post-Shutdown Decommissioning Activities Report
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control
RA	Restricted Area
RAI	Request for Additional Information
RCA	Radiologically Controlled Area
RCRA	Resource Conservation and Recovery Act
REMP	Radiological Environmental Monitoring Program
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
S_o	Sensitivity Threshold
SAFSTOR	A method of decommissioning in which a nuclear facility is placed and maintained in a condition that allows the facility to be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.
SCC	Secondary Component Cooling
SCM	Site Conceptual Model
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SRP	Standard Review Plan
Sv/hr	Sievert per hour
TBD	Technical Basis Documents
TCP	Traffic Control Plan
TEDE	Total Effective Dose Equivalent
TRU	Transuranic
VSP	Visual Sample Plan software
WMB	Waste management building
WWI	Wastewater Impoundments
ZNPS	Zion Nuclear Power Station
ZS	Zion*Solutions*

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