ZIONSOLUTIONS LLC
An EnergySolutions Company

10 CFR 50.4(b)(1)
10 CFR 2.390

February 7, 2018 ZS-2018-0007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Zion Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-39 and DPR-48
NRC Docket Nos. 50-295 and 50-304

Subject: License Termination Plan, Revision 2

References:

1) Gerard van Noordennen, Zion*Solutions*, Letter to U.S. Nuclear Regulatory Commission, "License Amendment Request for the License Termination Plan", dated December 19, 2014
2) Gerard van Noordennen, Zion*Solutions*, Letter to U.S. Nuclear Regulatory Commission, "License Termination Plan Update of the Site-Specific Decommissioning Costs", dated February 26, 2015
3) John B. Hickman, U.S. Nuclear Regulatory Commission, Letter to John Sauger, Zion*Solutions,* "Request for Additional Information Related to the License Termination Plan for Zion Nuclear Power Station, Units 1 and 2," dated December 10, 2015
4) Gerard van Noordennen, Zion*Solutions*, Letter to U.S. Nuclear Regulatory Commission, "License Termination Plan Request for Additional Information", dated March 8, 2016
5) John B. Hickman, U.S. Nuclear Regulatory Commission, Letter to John Sauger, Zion*Solutions,* "Request for Additional Information Related to the License Termination Plan for Zion Nuclear Power Station, Units 1 and 2," dated May 31, 2016
6) Gerard van Noordennen, Zion*Solutions*, Letter to U.S. Nuclear Regulatory Commission, "License Termination Plan Request for Additional Information", dated July 20, 2016
7) John B. Hickman, U.S. Nuclear Regulatory Commission, Letter to John Sauger, Zion*Solutions,* "Request for Additional Information Related to the License Termination Plan for Zion Nuclear Power Station, Units 1 and 2," dated November 30, 2016
8) Gerard van Noordennen, Zion*Solutions*, Letter to U.S. Nuclear Regulatory Commission, "License Termination Plan Request for Additional Information", dated July 20, 2017

NMSS01

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The Zion Station License Termination Plan (LTP) was submitted to the U.S. Nuclear Regulatory Commission (NRC) for review on December 19, 2014 as documented in Reference 1. Following initial NRC review, an initial Request for Additional Information (RAI), as documented in Reference 3, was received on December 10, 2015. A response to that request was submitted on March 8, 2016 as documented in Reference 4. A second RAI was received on May 31, 2016 as documented in Reference 5. A response to that RAI was provided on July 20, 2016 as documented in Reference 6. A third RAI was received on November 30, 2016 as documented in Reference 7. A response to that RAI was provided on July 20, 2017 as documented in Reference 8. After further discussions with the NRC Staff, a revised response to the third RAI will be submitted under separate cover. This letter provides revision 2 of the LTP which has incorporated all the agreed upon RAI responses.

Portions of this submittal provide the NRC with financial information to aid in the review of the License Termination Plan. Zion*Solutions* considers this proprietary financial information to be confidential and requests NRC to withhold it from public disclosure under 10 CFR 2.390(a)(4).

This submittal contains a Zion*Solutions* Proprietary Financial Information Affidavit pursuant to 10 CFR 2.390. The Affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). All documents within the scope of this affidavit are marked as "withhold from public disclosure under 10 CFR 2.390."

The LTP has been revised to incorporate changes resulting from responses to the RAIs. LTP Revision 2 (pdf) is provided in Enclosure 1 and replaces in entirety LTP Revision 1 provided in Reference 8. Enclosure 2 contains the Microsoft Word™ (Word) version of the LTP with the "track change" option enabled to assist in the review process. Chapter 7 of the LTP contains the proprietary financial information Zion*Solutions* is providing to the NRC and seeks to have withheld from public disclosure in its entirety. Enclosure 3 of this submittal contains a redacted version of LTP Chapter 7 for public disclosure. Enclosure 4 contains a preflight report for Enclosures 1, 2 and 3.

There are no regulatory commitments made in this submittal. If you should have any questions regarding this submittal, please contact Gerard van Noordennen at (860) 462-9707.

Respectfully,

John Sauger
Executive Vice President & CNO

Enclosures:

Enclosure 1:

A. Proposed License Termination Plan, Revision 2 (pdf)
B. Proposed License Termination Plan, Revision 2 (Word files)
C. Redacted Version of LTP Chapter 7

Enclosure 2: Preflight Report for Enclosure 1-A and 1-C to this submittal

cc: John Hickman, U.S. NRC Senior Project Manager (1 hard copy & 3 CDs)
Regional Administrator, U.S. NRC, Region III (1 hard copy & 1 CD)
Service List (Cover letter only, no enclosures)

Zion Nuclear Power Station, Unit 1 and 2 License Transfer Service List

cc:

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Zion*Solutions*, LLC PROPRIETARY FINANCIAL INFORMATION AFFIDAVIT

Affidavit of John Sauger, Executive Vice President &CNO, Zion*Solutions*, LLC.

LTP Chapter 7, contained in Enclosures 1 and 2 of this submittal, consists of proprietary financial information that Zion*Solutions*, LLC considers confidential. Release of this information would cause irreparable harm to the competitive position of Zion*Solutions*, LLC. The basis for this declaration is:

i. This information is owned and maintained as proprietary by Zion*Solutions*, LLC,

ii. This information is routinely held in confidence by Zion*Solutions*, LLC and not disclosed to the public,

iii. This information is requested to be held in confidence by the NRC by this petition,

iv. This information is not available in public sources,

v. This information would cause substantial harm to Zion*Solutions*, LLC if it were released publicly, and

vi. The information to be withheld is being transmitted to NRC in confidence.

I, *John Sauger*, being duly sworn, state that I am the person who subscribes my name the foregoing statement, I am authorized to execute the Affidavit on behalf of Zion*Solutions*, LLC, and that the matters and facts set forth in the statement are true to the best of my knowledge, information, and belief.

Name: John Sauger

Title: Executive Vice President & CNO

Company: Zion*Solutions*, LLC

SUBSCRIBED AND SWORN TO BEFORE ME

THIS 31 DAY of January, 2018

Notary Public

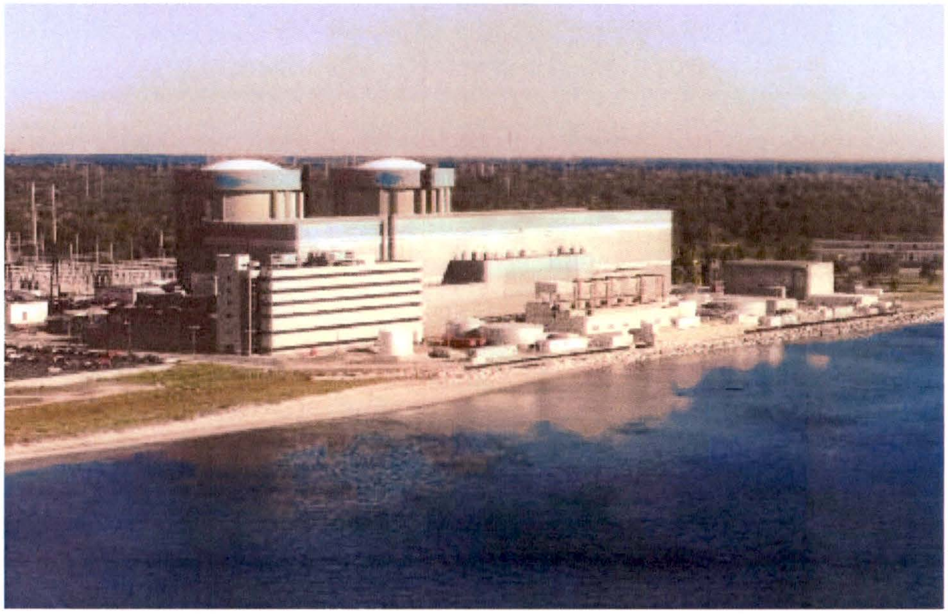
Notary Public - State of Utah
JILL WADSWORTH
Commission #696798
My Commission Expires
September 5, 2021

ZION STATION RESTORATION PROJECT

LICENSE TERMINATION PLAN

REVISION 2

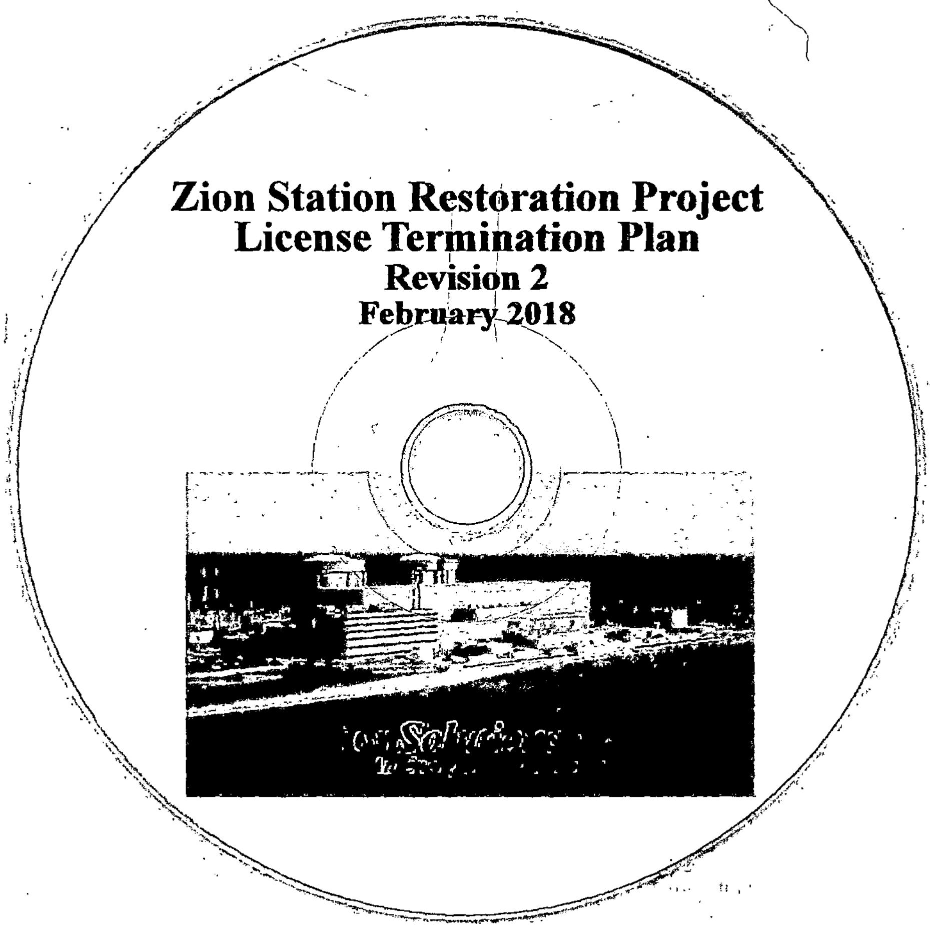
FEBRUARY 2018



ENCLOSURE 1

Proposed License Termination Plan, Revision 2

PDF and WORD files on attached CD



ENCLOSURE 2

Preflight Report

for

Enclosure 1-A and 1-C

This document serves as preflight report for Enclosure 1A and 1C to the ZS-2018-0007 submittal. The following files do not pass pre-flight criteria or do not meet NRC criteria, but text is word searchable with clarity/legibility of high quality.

No.	File Name	Preflight Status	Reason
1	1_Zion LTP Ch 1 Rev 2 012218	Failed	Document contains logos and color maps < 300 ppi, clear and legible
2	2_Zion LTP Ch 2 Rev 2 012218	Failed	Document contains logos, maps, and graphs < 300 ppi, clear and legible
3	3_Zion LTP Ch 3 Rev 2 020518	Failed	Document contains logos and images < 300 ppi, clear and legible
4	4_Zion LTP Ch 4 Rev 2 020518	Passed	
5	5_Zion LTP Ch 5 Rev 2 020518	Failed	Document contains logos < 300 ppi, clear and legible
6	6_Zion LTP Ch 6 Rev 2 020518	Failed	Document contains logos, maps, and graphs < 300 ppi, clear and legible
7	7_Zion LTP Ch 7 Rev 2 012318 NRC	Passed	
8	8_Zion LTP Ch 8 Rev 2 012218	Failed	Document contains logos and color maps < 300 ppi, clear and legible
9	9_Zion LTP Ch 7 Rev 2 012318 Redacted	Passed	

ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 1, REVISION 2
GENERAL INFORMATION

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LIST OF FIGURES

LIST OF ACRONYMS AND ABBREVIATIONS

AEC	Atomic Energy Commission
AMSL	Above Mean Sea Level
ALARA	As Low As Reasonably Achievable
AMCG	Average Member of the Critical Group
ComEd	Commonwealth Edison
DQO	Data Quality Objectives
DCGL	Derived Concentration Guideline Level
DSAR	Defueled Safety Analysis Report
DUST-MS	Disposal Unit Source Term - Multiple Species
FSS	Final Status Survey
FSAR	Final Safety Analysis Report
HSA	Historical Site Assessment
ICMP	Illinois Coastal Management Program
ISFSI	Independent Spent Fuel Storage Installation
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
NRC	Nuclear Regulatory Commission
ODCM	Off-Site Dose Calculation Manual
QAPP	Quality Assurance Project Plan for Characterization and FSS
PWR	Pressurized Water Reactors
RESRAD	RESidual RADioactive materials
SFP	Spent Fuel Pool
ZNPS	Zion Nuclear Power Station

1 GENERAL INFORMATION

The Zion Nuclear Power Station (ZNPS) consists of two (Units 1 and 2) Pressurized Water Reactors (PWR). The station 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.

ZNPS was previously operated by Commonwealth Edison (ComEd) until it was permanently shut down on February 13, 1998. In 2000, the license was transferred from ComEd to Exelon Nuclear Generation, LLC (Exelon). On January 25, 2008, Exelon and Zion*Solutions*, LLC submitted an *Application for License Transfers and Conforming Administrative License Amendments* (Reference 1-1) to the Nuclear Regulatory Commission (NRC) requesting that the NRC consent to the transfer of Exelon's Facility Operating Licenses for ZNPS to Zion*Solutions*. On September 1, 2010, the licenses were transferred from Exelon to Zion*Solutions*. Zion*Solutions* is now the current licensee and the submitter of this License Termination Plan (LTP) (*"Issuance of Conforming Amendments Relating to Transfer of Licenses for Zion Nuclear Power Station, Units 1 and 2"* [Reference 1-2]).

The following provides the licensee name, address, license numbers and docket numbers for ZNPS:

Zion*Solutions*, LLC

Zion Station
101 Shiloh Boulevard
Zion, IL 60099
License No. DPR-39 & DPR-48
Docket No. 50-295 & 50-304 and 72-1037

As an end-state for the decommissioning of ZNPS, all of the spent nuclear fuel will be stored in the Independent Spent Fuel Storage Installation (ISFSI) which will be maintained under amended Part 50 Licenses.

1.1 Purpose

The objective of decommissioning the ZNPS is to reduce the level of residual radioactivity to levels that permit the release of the site for unrestricted use and allow for the termination of the 10 CFR Part 50 licenses, excluding the ISFSI area. This LTP satisfies the requirement of 10 CFR 50.82(a)(9). This LTP was written following the guidance in Regulatory Guide 1.179, *"Standard Format and Contents for License Termination Plans for Nuclear Power Reactors"* (Reference 1-3) and in NUREG-1700, *"Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans"* (Reference 1-4). This LTP is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require NRC approval.

1.2 Decommissioning Objective

The decommissioning objective is to conduct remediation and survey operations such that Zion*Solutions* can submit a request to the NRC for the unrestricted release of the site (other than

the remaining licensed ISFSI facility) in accordance with Subpart E of 10 CFR Part 20 and meeting the unrestricted release requirements of 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use." The LTP documents the process that will be used to demonstrate that the dose from residual radioactivity that is distinguishable from background radiation does not exceed 25 mrem/yr to the Average Member of the Critical Group (AMCG) from all appropriate pathways over a 1,000 yr period and, to also demonstrate that the residual radioactivity has been reduced to levels that are "As Low As Reasonably Achievable" (ALARA).

1.3 Facility

1.3.1 Site Description

The ZNPS site is located in Northeast Illinois on the west shore of Lake Michigan. Figure 1-1 shows the geographical location of the plant relative to nearby towns, cities, river and the lake. The site is approximately 40 miles north of Chicago, Illinois, and 42 miles south of Milwaukee, Wisconsin. The site is in the extreme eastern portion of the city of Zion, in Lake County, Illinois, on the west shore of Lake Michigan approximately 6 miles NNE of the center of the city of Waukegan, Illinois and 8 miles south of the center of the city of Kenosha, Wisconsin. The US Census Bureau listed the estimated 2011 population of Zion as 24,508.

The site is located at longitude 87 degrees 48.1 minutes west and latitude 42 degrees 26.8 minutes north. The site occupies portions of Sections 22, 23, 26, and 27 in Township 46 North, Range 23 East.

The owner-controlled site is approximately 331 acres and within the owner-controlled area is an approximate 87 acre, fence-enclosed nuclear facility. The Zion Station *"Defueled Safety Analysis Report"* (DSAR) (Reference 1-5) states that the licensee maintains "exclusion-area" control over approximately 250 acres. However during site characterization (see Chapter 2 of this LTP), the total acreage surveyed as impacted and non-impacted open land survey units totaled 331 acres. No other major changes to the site boundary have been made. Figure 1-2 shows the main plant structures and the boundaries of the owner-controlled property.

The topography of the site and its immediate environs is relatively flat; elevations vary from the Lake Michigan shoreline (low water level is 577.4 feet Above Mean Sea Level [AMSL]) to approximately 20 feet above the level of the lake. The elevation of the developed portion of the site is 591 feet. A series of low, parallel beach ridges separated by marshy depressions crosses the site. This topography represents recessional beach lines deposited along the Lake Michigan shoreline subsequent to the most recent period of glaciations. The beach ridges are composed primarily of sand. Figure 1-3 provides a topographical map of the site and the surrounding area.

1.3.2 Current/Future Land Use

The 87 acre, fence-enclosed nuclear facility is within an area zoned for industrial use in accordance with *"The City of Zion, Illinois, Comprehensive Plan 2010"* (Reference 1-6). This industrially developed area, which currently includes the major buildings (Containment Buildings, Turbine Building, Spent Fuel Building, Auxiliary Building), the switchyard, parking areas, rail lines, haul paths and the ISFSI, will continue as an industrial-zoned property throughout the decommissioning and for future use. The ISFSI as well as the Commonwealth

Edison switchyard to the west of the plant will remain after the decommissioning of ZNPS is completed. Most of the overall site that lies to the west of the 87 acre developed plant site is designated as wetland according to Chapter 9 of the *"Lake County Illinois, Regional Framework Plan"* (Reference 1-7) and the site property is also identified as an environmentally sensitive area to be protected from development in the City of Zion Comprehensive Plan 2010. None of the land within approximately 4 miles of the Zion property is zoned as agricultural.

The site is bounded by park areas, including the City of Zion's Park District, Hosah Park as well as the Illinois State Beach Park to the north, and another Illinois State Beach Park parcel to the south. These park areas, which are adjacent to Lake Michigan to the east, include various boardwalks, paved walking trails, picnic and camping areas. The parks promote recreational use, including boating, fishing, camping and bird watching. The *"Illinois Coastal Management Program, Issue Paper - Illinois Beach State Park and North Point Marina Including the Dead River and Kellogg Creek Watersheds"* (Reference 1-8) concluded that the presence of the surrounding recreational property has limited residential, commercial or agricultural land use in the land surrounding ZNPS.

Upon license termination, Zion*Solutions* will be returning ownership and management of the property back to Exelon, which will make the ultimate determination about future use of the property after the completion of the decommissioning activities described in this LTP.

1.3.3 Meteorology and Climatology

Zion's climate is continental with cold winters, warm summers, and frequent short fluctuations in temperature, humidity, cloudiness, and wind direction. The average temperature in the summer is 72°F and the average temperature in the winter is 24°F. Because the eastern edge of Zion is bounded by Lake Michigan, inland lake breezes can cool the air along the lake shore by 10°F to 15°F in the summer and can warm the air by as much as 20°F in the winter. The average annual rainfall is 32.0 inches and the average annual snowfall is 41.0 inches.

1.3.4 Geology and Seismology

As stated in the Commonwealth Edison Company, *"Zion Nuclear Power Station Final Safety Analysis Report"* (FSAR) (Reference 1-9), and *"Environmental Report, Zion Nuclear Power Station"* (Reference 1-10), the near-surface geology of northeastern Illinois is comprised of unconsolidated deposits which range from 90 to 150 feet 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 4,000 feet. These sedimentary bedrock layers dip gently toward the east at the rate of about 10 feet per mile and rest on Precambrian basement rock.

The upper portion of surface deposits in the vicinity of the ZNPS is comprised of unconsolidated sand, silt and peat derived from Lake Michigan shore deposits. This is overlaid on a mixture of unconsolidated material ranging from sand, clay to boulders deposited as glacial till, outwash, loess and lake sediments.

As stated in the Conestoga-Rovers and Associates, *"Hydrogeologic Investigation Report, Fleetwide Assessment, Zion Station"*, Revision 1 (Reference 1-11), the surface deposits in the

vicinity of the ZNPS structures are comprised of three layers, or units, of irregular thickness. The uppermost layer is identified as the upper sand unit and ranges from about 30 to 35 feet in thickness. Immediately below the upper sand unit lies a layer comprised predominantly of silt and clay. This layer, identified as the silt-clay unit, ranges from about 20 to 40 feet 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 layer ranges from about 30 to 50 feet in thickness.

The upper bedrock layer is the Niagara Dolomite, a consolidated layer of carbonaceous marine sediments laid down in the Silurian Period. It is about 200 feet thick in the vicinity of the ZNPS site.

There is no indication of faulting beneath the site. The area within 100 miles of the site is considered to be one of minor seismic activity. Few events of moderate significance have occurred in the region in the last 150 to 200 years.

Near Des Plaines, approximately 25 miles southwest of the site, a highly complex faulted zone exists which appears to bear no relationship to the regional structure. The zone is roughly circular and covers an area of 25 square miles. Within the faulted zone, the bedrock generally has been up-thrown. Some faulting also exists in southern Wisconsin and the closest known fault in Southern Wisconsin is approximately 45 miles from the site and has a northeast orientation.

Recent minor earthquakes which occurred near the site include a small 2.4-magnitude earthquake on January 30, 2010 which occurred 2 miles east of McHenry and approximately 30 miles West of the Zion plant and a previous earthquake of 3.8 magnitude which occurred about 2 miles northwest of Lily Lake, approximately 70 miles from ZNPS.

1.3.5 Surface Water Hydrology

ZNPS is located on the shores of Lake Michigan. The lake is 307 miles long from north to south and has an average width of 70 miles. In the general vicinity of the site, the 30-foot depth contour of the lake is 1.2 miles, and the 60-foot depth contour is 2.0 miles from the shore.

Lake Michigan is used by recreational boaters. The nearest marina/public boat launch to ZNPS is located approximately 2.5 miles north of the site. There are also several fishing charter services that are located approximately 3 miles north of the site. Lake Michigan is also used for commercial barge and ship traffic, however these activities typically do not ordinarily operate within 5 miles of the site.

Water from Lake Michigan is used for municipal and domestic water supplies. There are multiple potable water intakes located in Lake Michigan in the vicinity of ZNPS. The nearest lake source water supply is located approximately 1 mile north of the site with the intake located approximately 3,000 feet out in the lake.

Other surface water features near the site include Kellogg Creek (1.25 mile north), Dead River (3 miles south), and Bull Creek (0.2 mile south). Kellogg Creek is a perennial stream that drains to the north property of the Illinois Beach State Park through a bluff/ravine system that is moderately to severely eroded. The creek has a reduced natural function as it has been channelized since early industrial development. Bull Creek is also a perennial stream that drains

to the south property of the Illinois Beach State Park. Similar to Kellogg Creek, the bluff/ravine system for Bull Creek is severely eroded along most of the length. Bull Creek becomes Dead River once it begins to cross the sand plain. The Dead River is an unaltered natural tributary to Lake Michigan that flows through an extensive high quality coastal wetland complex, which is a rare habitat type in the Illinois Lake Michigan watershed.

1.3.6 Ground Water Hydrology

The groundwater table in the area is close to the ground surface and has a flat gradient to the east and south. Shallow groundwater movement in the area is to the east towards Lake Michigan. The upper sand unit is a high permeability unit that is directly connected to Lake Michigan, which is a regional discharge feature and which generally allows unrestricted lateral groundwater flow. Vertical groundwater flow is limited by the underlying silt-clay unit which has a low permeability and is approximately 30 feet thick.

1.3.7 Environs and Natural Resources

Of the 331 acre ZNPS site, about 87 acres is enclosed within the perimeter security fence. This is called the "Radiological-Restricted Area". The remainder, which lies mostly to the west of the ComEd Switchyard, is an open marshy area. This area is undeveloped except for overhead transmission lines and dirt roads maintained by ComEd.

The land area immediately west of the site up to the rail track owned by Chicago & Northwestern is zoned light industrial by the City of Zion. This area is about five blocks long, extending from 29th Street on the south to Shiloh Boulevard on the north and is about four blocks wide in the east-west direction centered on Deborah Avenue. It is currently occupied by several warehouses and associated truck shipping operations, an industrial cleaning-service company, several auto service garages, a salvage yard, a former manufacturing facility and a number of vacant lots.

The center of the community of Zion is approximately 1.6 miles from the plant location on the site. There are no schools or hospitals within one mile of the site and there are no residences within 2,000 feet of any ZNPS structures.

There are no potable wells on or near the site. Water service is provided through the City of Zion municipal water supply which draws water from Lake Michigan via a water intake about one mile north of the site.

A significant factor which affects land use in the near vicinity of ZNPS is the Illinois Beach State Park. The park has been expanded since the construction of ZNPS. The present day north and south Illinois Beach State Park properties are part of a state-owned costal management area which extends from Winthrop Harbor about 3 miles north of the ZNPS site to about 3 miles south of the ZNPS site. The area from Winthrop Harbor Marina on the north to the southern end of the Illinois Beach State Park has been incorporated into the Illinois Coastal Management Program (ICMP). The ICMP has identified this area as a unique public resource requiring special attention for preservation, protection and restoration of areas impacted by shoreline erosion, invasive species and damage caused by previous industrial activities.

1.4 Operational Background

Key station milestones are presented in Table 2-1 of Chapter 2 of this LTP. Several key significant milestones are reproduced below:

- ComEd dockets application for construction: July 1967
- Operating license issued: April 6, 1973 for Unit 1 and November 14, 1973 for Unit 2
- Commercial operations achieved: December 1973 for Unit 1 and September 1974 for Unit 2
- Final reactor operation: February 21, 1997 for Unit 1; September 19, 1996 for Unit 2
- Cessation of operations: February 13, 1998
- All fuel removed from the reactor and placed in the Spent Fuel Pool (SFP): April 27, 1997 for Unit 1 and February 25, 1998 for Unit 2
- Decommissioning operations begin: October 1, 2010
- Spent fuel and Greater Than Class C (GTCC) waste transferred to ISFSI: 2015

1.5 Plan Summary

1.5.1 General Information

The LTP describes the process used to meet the requirements for terminating the 10 CFR Part 50 license and to release the site for unrestricted use. The LTP has been prepared in accordance with the requirements in 10 CFR 50.82(a)(9) and is submitted as a supplement to the DSAR. The LTP submittal is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require prior NRC approval. The subsections below provide a brief summary of the other seven chapters of the LTP.

1.5.2 Site Characterization

LTP Chapter 2 discusses the site characterization that has been conducted to determine the extent and range of radioactive contamination on site prior to remediation, including structures that will remain at the time of license termination, soils, and ground water. Based on the results of the site characterization, Zion*Solutions* will plan remediation and Final Status Surveys (FSS) in areas determined to be impacted by the operation of ZNPS.

The Zion *"Historical Site Assessment"* (HSA) (Reference 1-12) provided the initial foundation for further site characterization and the basis for dividing the site into survey units. The survey units were evaluated against the criteria specified in NUREG-1575, *"Multi-Agency Radiation Survey and Site Investigation Manual"* (MARSSIM) (Reference 1-13) for classification. Data from subsequent characterization may be used to change the original classification of an area, within the requirements of this LTP, up to the time of Final Status Survey (FSS), as long as the classification reflects the level of residual activity existing prior to any remediation in the area.

1.5.3 Identification of Remaining Site Dismantlement Activities

LTP Chapter 3 identifies the remaining site dismantlement and decontamination activities. The information provided in Chapter 3 includes:

- A description of the areas and equipment that need further remediation,
- A summary of radiological conditions that may be encountered,

- Estimates of associated occupational radiation dose,
- An estimate of the types and quantities of radioactive material generated for release and disposal, and,
- Descriptions of proposed control mechanisms to ensure areas are not re-contaminated.

Zion*Solutions* is decommissioning ZNPS in accordance with the DECON alternative described in NUREG-0586 *"Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, Supplement 1, Volume 1"* (Reference 1-14). The decommissioning activities will be conducted in accordance with the Zion*Solutions* Health and Safety Program, Radiation Protection Program, Radioactive Waste Program, Off-Site Dose Calculation Manual (ODCM), and plant administrative, work control and decommissioning implementation procedures. These are established programs that are routinely inspected by the NRC.

Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during plant operations. The radiological risk associated with decommissioning activities is bounded by previously analyzed radiological risk for former operating activities that occurred during major maintenance and outage activities.

The information provided in Chapter 3 supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify resources needed during the remaining dismantlement activities.

1.5.4 Remediation Plans

LTP Chapter 4 discusses the various remediation techniques that may be used during decommissioning to reduce residual contamination to levels that comply with the release criteria in 10 CFR 20.1402. This chapter also discusses the ALARA evaluation and the impact of remediation activities on the Radiation Protection Program.

The selected remediation methods used are dependent upon the contaminated material and extent of contamination. The principal materials that may be subject to remediation are structural surfaces. Very limited soil contamination is expected and no groundwater or surface water contamination has been identified to date. Remediation techniques that may be used for structural surfaces include scabbling and shaving, chipping, sponge and abrasive blasting, standard and pressure washing, wiping, grit blasting, mechanical fracturing and cutting, and other methods. Surface and subsurface soil with activity levels in excess of the appropriate Derived Concentration Guideline Level (DCGL) will be removed and disposed as radioactive waste. Soil remediation equipment will include, but not be limited to, back and track hoe excavators. Remediation of soils will include the use of established excavation safety and environmental control procedures as well as appropriate work package instructions to ensure adequate erosion, sediment and air emission controls during soil remediation.

1.5.5 Final Radiation Survey Plan

LTP Chapter 5 presents the Final Status Survey (FSS) Plan which will be used to develop the site procedures, survey packages and instructions to perform the FSS of the Zion Station site. The FSS Plan describes the final survey process used to demonstrate that the ZNPS facility and site comply with radiological criteria for unrestricted use specified in 10 CFR 20.1402 (e.g. annual dose limit of 25 mrem to AMCG plus ALARA for all dose pathways).

The FSS Plan describes the development of the survey plan, survey design and Data Quality Objectives (DQO), survey method and instrumentation, data collection and processing, data assessment and compliance, and the Zion*Solutions* ZS-LT-01, "*Quality Assurance Project Plan (for Characterization and FSS)*" (QAPP) (Reference 1-15). The FSS Plan addresses only ZNPS structures and land areas that are identified as impacted. The adjacent areas that were classified as non-impacted and the ISFSI, which will still remain a licensed area, will not be subject to FSS.

1.5.6 Compliance with the Radiological Criteria for License Termination

LTP Chapter 6 presents the radiological information and methods used to demonstrate compliance with the radiological criteria for license termination and release of the site for unrestricted use. Chapter 6 discusses the site-specific inventory of radionuclide, future land use scenarios, exposure pathways, computational models used for dose modeling, sensitivity analysis, DCGLs, the derivation of area factors, the basis for the selected exposure compliance scenario and evaluation of alternative exposure scenarios.

LTP Chapter 6 utilizes radiological information from Chapter 2 and establishes the allowable contamination and radioactivity concentration levels that Chapter 4 remediation methods will work towards and be verified by the FSS discussed in Chapter 5. Zion*Solutions* applied the Disposal Unit Source Term - Multiple Species (DUST-MS) model and the RESidual RADioactive materials (RESRAD) v7.0 model to determine the radiological release criteria for remaining structures and soils to enable license termination.

1.5.7 Update of the Site-Specific Decommissioning Costs

LTP Chapter 7 provides an updated estimate of the remaining decommissioning costs for releasing the site for unrestricted use. This chapter also compares the estimated remaining cost with the funds currently available in the decommissioning trust fund.

1.5.8 Supplement to the Environmental Report

LTP Chapter 8 updates the environmental report for ZNPS with new information and any significant environmental impacts associated with the site's decommissioning and license termination activities. This section of the LTP is prepared pursuant to 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G). In accordance with 10 CFR 51.53(d), Zion*Solutions* considers Chapter 8 as a supplement to the Environmental Report addressing the actual or potential environmental impacts associated with the execution of the described decommissioning activities.

LTP Chapter 8 compares the described decommissioning attributes to those identified in NUREG-0586, which provides a generic environmental assessment for the decommissioning of a reference nuclear facility. The environmental assessment performed by Zion*Solutions* determined that the environmental effects for decommissioning ZNPS are minimal and there are no adverse effects outside the bounds of NUREG-0586. Additionally, the conclusions contained in the United States Atomic Energy Commission (AEC) "*Final Environmental Statement related to operation of Zion Nuclear Power Station Units 1 and 2*", - December 1972 (AEC Environmental Statement) (Reference 1-16), used as the original basis for the decommissioning

environmental assessment of radiological and non-radiological effects of decommissioning, are still valid.

1.6 Regulatory Notifications of Changes

Zion*Solutions* is submitting the LTP as a supplement to the DSAR. Accordingly, Zion*Solutions* will update the LTP in accordance with 10 CFR 50.71(e). Once approved, Zion*Solutions* may make changes to the LTP, without prior NRC approval, in accordance with the criteria in 10 CFR 50.59, 10 CFR 50.82(a)(6), and 10 CFR 50.82(a)(7).

Zion*Solutions* is also submitting a proposed amendment to the ZNPS licenses that adds a license condition that establishes the criteria for determining when changes to the LTP require prior NRC approval. Changes to the LTP require prior NRC approval when the change:

- Require Commission approval pursuant to 10 CFR 50.59.
- Result in significant environmental impacts not previously reviewed.
- Detract or negate the reasonable assurance that adequate funds will be available for decommissioning
- 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 NRC a minimum 14 day notification prior to implementing the change in classification.
- Increase the derived concentration guideline levels and related minimum detectable concentrations (for both scan and fixed measurement methods).
- Increase the radioactivity level, relative to the applicable derived concentration guideline level, at which an investigation occurs.
- Change the statistical test applied to one other than the Sign test.
- Increase the Type I decision error

The LTP will also be updated every two years. The contact for LTP information, including any submitted changes and updates, is:

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1.7 References

1-1 Letter from T.S. O'Neill (Exelon Generation Company) and J. Christian (Zion*Solutions*) to NRC, "Application for License Transfers and Conforming Administrative License Amendments" – January 2008

1-2 Letter from J.B. Hickman (U.S. Nuclear Regulatory Commission) to J. Christian (Zion*Solutions*), "Issuance of Conforming Amendments Relating to Transfer of Licenses for Zion Nuclear Power Station, Units 1 and 2" – September 2010

1-3 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" – January 1999

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

1-5 Zion Station, "Defueled Safety Analysis Report" (DSAR) – September 2014

1-6 The City of Zion, Illinois, "Comprehensive Plan 2010" – January 1992

1-7 Lake County Illinois, "Regional Framework Plan, Chapter 9, Land Use" – February 13, 2007

1-8 Illinois Coastal Management Program, "Illinois Beach State Park and North Point Marina Including the Dead River and Kellogg Creek Watersheds" – 2011

1-9 Commonwealth Edison Company, "Zion Nuclear Power Station - Final Safety Analysis Report" (FSAR) – November 1970

1-10 Commonwealth Edison Company, "Environmental Report - Zion Nuclear Power Station" – May 1971, Supplement 1 – November 1971, Supplement II – December 1971, Supplement III – February 1972, Supplement IV – April 1972, Supplement V – May 1972

1-11 Conestoga-Rovers and Associates, "Hydrogeologic Investigation Report, Fleetwide Assessment, Zion Station, Zion Illinois", Revision 1 – September 2006

1-12 "Zion Station Historical Site Assessment" (HSA) – September 2006

1-13 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" – August 2000

1-14 U.S. Nuclear Regulatory Commission NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities", Supplement 1, Volume 1" – November 2002

1-15 Zion*Solutions* ZS-LT-01, Revision 6, "Quality Assurance Project Plan (for Characterization and FSS)" (QAPP)

1-16 United States Atomic Energy Commission, Directorate of Licensing, "Final Environmental Statement related to the Operation of Zion Nuclear Power Station Units 1 and 2", Docket Nos. 50-295 and 50-304 – December 1972

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Figure 1-1 Zion Nuclear Power Station Geographical Location

Figure 1-2 Zion Nuclear Power Station Owner Controlled Area

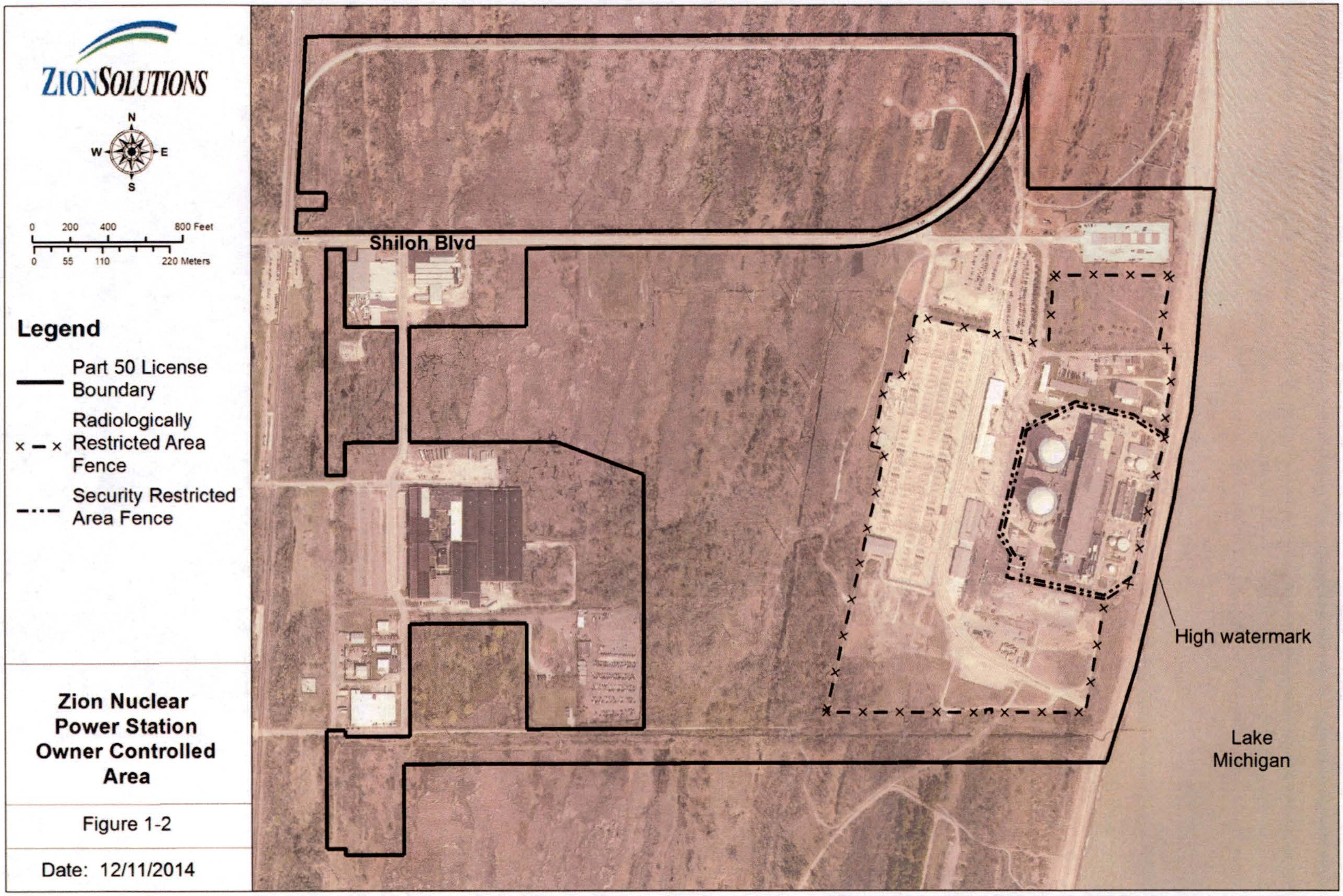
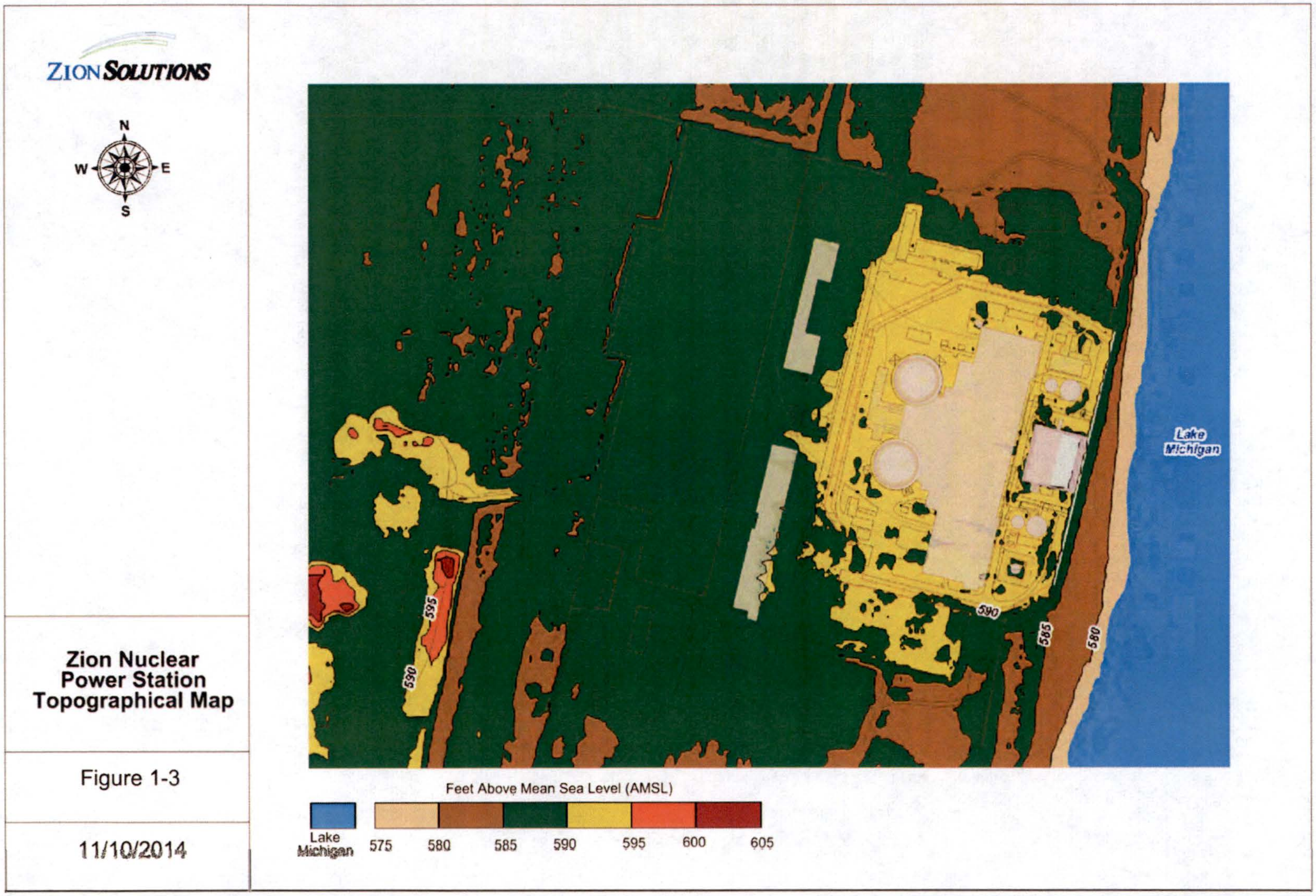


Figure 1-3 Zion Nuclear Power Station Topographical Map



Zion Nuclear Power Station Topographical Map

Figure 1-3

11/10/2014

ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 2, REVISION 2
SITE CHARACTERIZATION

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LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ComEd	Commonwealth Edison
CR	Condition Reports
CRA	Conestoga-Rovers & Associates
CVS	Contamination Verification Survey
DCGL	Derived Concentration Guideline Level
DOE	Department of Energy
DQO	Data Quality Objective
ENC	Engineering and Construction
EPA	Environmental Protection Agency
ESCSG	Energy Solutions Commercial Services Group
Exelon	Exelon Generation Company
FOV	Field of View
FSS	Final Status Survey
GPS	Global Positioning System
HSA	Historical Site Assessment
HTD	Hard to Detect
IDNS	Illinois Department of Nuclear Safety
ISOCS	*In Situ* Object Counting System
IRSF	Interim Radioactive Waste Storage Facility
ISFSI	Independent Spent Fuel Storage Installation
LCPWD	Lake County Public Water District
LER	Licensee Event Reports
LTP	License Termination Plan
MARLAP	Multi-Agency Radiological Laboratory Analytical Protocols Manual
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimal Detectable Concentrations
MDCR	Minimum Detectable Count Rate
MGD	million gallons per day
MWe	Megawatts electric
MWhr	Megawatt hour
MWth	Megawatts thermal
NaI	Sodium Iodide
NGET	Nuclear General Employee Training
NIST	National Institute of Standards and Technology

NPDES	Illinois Environmental Protection Agency – National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
ODCM	Off-site Dose Calculation Manual
PWST	Primary Water Storage Tank
QAPP	Quality Assurance Project Plan for Characterization and FSS
QA/QC	Quality Assurance and Quality Control
RCA	Radiologically Controlled Area
REMP	Radiological Environmental Monitoring Program
RGPP	Radiological Groundwater Protection Program
ROC	Radionuclides of Concern
ROR	Radiological Occurrence Reports
SAFSTOR	SAFeSTORe
SFP	Spent Fuel Pool
S/G	Steam Generator
SST	Secondary Condensate Storage Tank
TSD	Technical Support Document
VCC	Vertical Concrete Cask
WDHS	Wisconsin Department of Health Services
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

2. SITE CHARACTERIZATION

In accordance with the requirements of 10 CFR 50.82 (a)(9)(ii)(A) and the guidance of Regulatory Guide 1.179, *"Standard Format and Contents for License Termination Plans for Nuclear Power Reactors"* (Reference 2-1), this chapter provides a description of the radiological characterization performed at the Zion Nuclear Power Station (ZNPS) site. The purpose of site characterization is to ensure that the Final Status Survey (FSS) will be conducted in all areas where contamination existed, remains, or has the potential to exist or remain. The results of the characterization survey, including the *"Zion Station Historical Site Assessment"* (HSA) (Reference 2-2), demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected. There are areas known to potentially contain contamination that were inaccessible during the initial characterization which will be surveyed during continuing characterization as access is gained (see section 2.5).

The site characterization incorporates the results of investigations and surveys conducted to quantify the extent and nature of contamination at the ZNPS site. In addition, the results of site characterization surveys and analyses have been and continue to be used to identify areas of the site that will require remediation, as well as to plan remediation methodologies, develop waste classification and volumes, and estimate costs.

The characterization survey was designed and executed using the guidance provided in NUREG-1575, *"Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM)*"* (Reference 2-3) and NUREG-1757, Volume 2, Revision 1, *"Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report"*, (Reference 2-4). In addition, surveys were designed and executed in accordance with the Zion*Solutions* ZS-LT-02, "*Characterization Survey Plan*" (Reference 2-5), and ZS-LT-01, "*Quality Assurance Project Plan (for Characterization and FSS)*" (QAPP) (Reference 2-6) which describes policy, organization, functional activities, the Data Quality Objectives (DQO) process, and measures necessary to achieve quality data. The information obtained from the characterization provides guidance for decontamination and remediation planning. Materials which were shown to be contaminated with radioactive material at concentrations greater than the unrestricted release criteria have been and will continue to be removed and properly packaged for shipment and disposal.

The site characterization of ZNPS includes the information requirements listed in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans* (Reference 2-7) and NUREG-1757. Measurements and samples were taken in each accessible area, along with the historical information, to provide a clear picture of the residual radioactive materials and its vertical and lateral extent at the site. Using appropriate DQOs, monitoring well water samples, surface soil, sediment, concrete cores and sub-surface soil were collected to provide a profile of the residual radioactivity at the site. Samples were analyzed for the applicable radionuclides with detection limits that provided the level of detail necessary for decommissioning planning. Based upon the volume of characterization data collected and an assessment of the characterization results, the characterization survey is considered adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected.

Section 8.5 of Exhibit C, Lease Agreement, titled "Removal of Improvements; Site Restoration" integral to the *"Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement"* (Reference 2-8) requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least 3 feet below grade. Consequently, the only structures that will remain at license termination are the exposed steel liner and Under-Vessel concrete in the Unit 1 and Unit 2 Containment Buildings (after all other interior concrete is removed) and the below-grade structural concrete outside of the liner, the reinforced concrete basement floor and outer walls of the Auxiliary Building and Turbine Building (after all internal walls and floors are removed), the reinforced concrete floor and walls of the Spent Fuel Pool (SFP) and Fuel Transfer Canal (after the steel liner is removed), the concrete floor and walls of the Crib House, the Waste Water Treatment Facility (WWTF), the Forebay and the Circulating Water Intake and Discharge Piping below the 588 foot elevation.

All systems, and components, as well as all structures above the 588 foot elevation will be removed during the decommissioning process and disposed of as a waste stream. The switchyard, which belongs to Commonwealth Edison, will also remain. As part of the decommissioning process, all reactor fuel and greater than Class C waste was loaded into casks and transferred to an Independent Spent Fuel Storage Installation (ISFSI). The fuel will remain on-site in dry storage within the ISFSI until it is transferred to the Department of Energy (DOE). The ISFSI has been constructed in the southwest corner of the ZNPS site, immediately south of the switchyard.

2.1. Historical Site Assessment

In accordance with guidance provided in MARSSIM, section 3.0, a HSA was issued in August of 1999. Historical information, including any 10 CFR 50.75(g) files, employee interviews, radiological incident reports, pre-operational survey data, spill reports, special surveys (e.g., site aerial surveys, marine fauna and sediment surveys), operational survey records, and Annual Radiological Environmental Reports (including sampling of air, groundwater, estuary water, milk, invertebrates, fish and surface vegetation) to the Nuclear Regulatory Commission (NRC) were reviewed and compiled for this investigation.

2.1.1. Objectives

The HSA was a detailed investigation to collect existing information (from the start of ZNPS activities related to radioactive materials or other contaminants) for the site and its surroundings. The HSA focused on historical events and routine operational processes that resulted in contamination of plant systems, onsite buildings, surface and subsurface soils within the Radiologically Controlled Area (RCA). It also addressed support structures, open land areas and subsurface soils outside of the RCA but within the owner controlled area. The information compiled by the HSA was used to establish initial area survey units and their MARSSIM classifications. This information was used as input into the development of site-specific Derived Concentration Guideline Levels (DCGL), remediation plans and the design of the FSS. The scope of the HSA included potential contamination from radioactive materials, hazardous materials, and other regulated materials.

The objectives of the HSA were to:

- Identify potential, likely, or known sources of radioactive and chemical contaminants based on existing or derived information.
- Distinguish portions of the site that may need further action from those that pose little or no threat to human health.
- Provide an assessment of the likelihood of contaminant migration.
- Provide information useful to subsequent continuing characterization surveys.
- Provide an initial classification of areas and structures as non-impacted or impacted.
- Provide a graded initial classification for impacted soils and structures in accordance with MARSSIM guidance.
- Delineate initial survey unit boundaries and areas based upon the initial classification.

At the time that the HSA was performed, the facility was in a SAFSTOR condition. As noted in the NRC public meeting in June 1998, the decommissioning approach for the facility specified that the units would remain in a SAFSTOR condition through 2010, when decontamination and dismantlement activities would begin for the structures, systems and components not required for maintenance of the nuclear spent fuel. The intended purpose of the Zion HSA was to provide a compilation and "road-map" of data and documents relating to the contaminant makeup of the site. This road-map would aid subsequent detailed site characterizations to be conducted in support of decommissioning planning which were anticipated to begin in approximately 2010. During the SAFSTOR period, no scoping or initial characterization surveys as defined by MARSSIM, sections 5.2 and 5.3 were performed.

2.1.2. Methodology

The objective of the HSA records search and interview process was the identification of those events posing a significant probability of impacting the hazardous material or radiological status of ZNPS site land areas and structures. These included system, structure, or area contamination from system failures resulting in airborne releases, liquid spills or releases, or the loss of control over solid material. Depending upon previous site operations and processes, the potential for residual contamination varies by area. In order to facilitate effective characterization surveys to guide future decontamination activities and provide sufficient data for the design of FSS, land areas and structures are classified based upon their potential for contamination.

Each incident identified that posed a realistic potential to impact the characterization of the site was further investigated. This investigation focused on the scope of contaminant sampling and analysis, any remedial actions taken to mitigate the situation, and any post-remedial action sampling, survey, and analysis in an attempt to identify the "as left" condition of the incident location. Historical records archives provided the source of a vast majority of the documents inspected.

Also included in the research associated with the development of the HSA were:

- Relevant excerpts from written reports and correspondences;

- Personnel interviews, including the use of questionnaires, of current, former and retired plant personnel to confirm documented incidents and identify undocumented incidents; and
- Site inspection, utilizing historic site drawings, photographs, prints, and diagrams to identify, locate, confirm, and document areas of concern.

Information from this research was used in the HSA development, including the compilation of data, evaluation of results, documentation of findings, and the identification of initial Survey Units.

2.1.2.1. Preliminary Classification

The HSA investigation was designed to obtain sufficient information to provide initial classification of the site land areas and structures as impacted or non-impacted. Impacted areas have a potential for contamination (based on historical data) or contain known contamination (based on past or preliminary radiological surveillance). Non-impacted areas are identified through knowledge of site history or previous survey information and are those areas where there is no reasonable possibility for residual radioactive contamination. Areas were classified as impacted from a radiological perspective. Potential chemical hazards incidents on owner-controlled areas were also documented including the confirmed presence of petroleum products, asbestos or other hazardous materials.

If a land area or structure was classified as impacted, then a determination of the initial impacted area classification (e.g. Class 1, Class 2 or Class 3) in accordance with MARSSIM, section 2.2 was made based upon the information obtained.

Initial classification of ZNPS areas was based on historical information and available historical radiological survey data. Classifying a survey area has a minimum of two stages: (1) initial classification and (2) final classification. Initial classification of most areas is performed at the time of identification of the survey area using the information available when the HSA was prepared. Final classification is performed and verified as a DQO during FSS design. Radiological survey data from characterization surveys, operational surveys in support of decommissioning, routine surveillance, and any other applicable survey data may cause an increase in survey area classifications (for example, from Class 3 to Class 2 and from Class 2 to Class 1) until the time of commencement of the FSS.

2.1.2.2. Documents Reviewed

Records maintained to satisfy the requirements of 10 CFR Part 50.75(g)(1) provided a major source of documentation for the HSA records review process. During the conduct of the HSA for ZNPS, many record types were evaluated including paper, microform, and electronic media. In total, over 29,000 records were reviewed for applicability for the HSA. A complete listing of documents reviewed is provided in Appendix 3 of the HSA. A summary of the types of records reviewed include:

- License and Technical Specification reports,
- Annual operational and environmental reports,
- Environmental investigations performed by independent entities,

- Regulatory actions against the site,
- Documentation from interviews conducted with currently employed and retired/separated site personnel,
- Radiological and environmental survey documents,
- Site inspection and surveillance documents associated with identified events,
- Annual Environmental and Operational documents,
- Licensee Event Reports (LERs),
- Radiological Occurrence Reports (RORs),
- Condition Reports (CRs),
- Quality Control /Quality Assurance findings.

2.1.2.3. Licenses, Permits and Authorizations

ZNPS was operated in accordance with several Federal and State of Illinois licenses and permits. The NRC Operating Licenses for Units 1 and 2 and supporting Technical Specifications allowed Commonwealth Edison and subsequent license holders to use any quantity of radioactive material at the site, to support operations during its operating lifetime, and to implement decommissioning activities.

The US Environmental Protection Agency (EPA) and applicable State of Illinois agencies maintain files on a variety of environmental programs that are applicable to ZNPS. These include permit applications and monitoring results with information on specific waste types and quantities, sources, type of site operations, and operating status of the facility or site.

The following denotes the licenses and permits relevant to the development of the HSA:

- US Nuclear Regulatory Commission – Docket Number 50-295, Facility Operating License Number DPR-39 (for Unit 1) (Reference 2-9)
- US Nuclear Regulatory Commission – Docket Number 50-304, Facility Operating License Number DPR- 48 (for Unit 2) (Reference 2-10)
- Illinois Environmental Protection Agency – National Pollutant Discharge Elimination System (NPDES) Permit Number IL0002763 (Reference 2-11)

2.1.2.4. Personnel Interviews

Interviews with current or previous employees were performed to collect first-hand information about the site and to verify or clarify information gathered from the records that were reviewed. The personnel interviews included a combination of questionnaires completed by a majority of the participants as well as individual and group interviews with several of the participants. Key personnel were chosen due to their knowledge of and association with the systems and source terms being investigated for the assessment. A number of the personnel interviewed possessed site knowledge and experience that ranged from the site construction period to shutdown.

Two types of interview questionnaires were used in the conduct of the Zion HSA. The first type of questionnaire was designed for permanent site production personnel. The second type of interview questionnaire was used during the personnel exit process (permanent and contractor staff). Typically, individuals were provided the questionnaire as a part of the standard station exit process. These exit interviews/questionnaires were obtained more as a quality check on information obtained during permanent staffing interviews. During the conduct of the Zion HSA, over 300 personnel interviews occurred with current and previous Zion personnel. With few exceptions, the personnel observations were corroborated by either the observations of other interviewees or documentation discovered during the records search.

2.1.3. Operational History

The ZNPS is located 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. The site is in the extreme eastern portion of the city of Zion, (Lake County) Illinois, on the west shore of Lake Michigan approximately 6 miles NNE of the center of the city of Waukegan, Illinois, and 8 miles south of the center of the city of Kenosha, Wisconsin.

The station is comprised of two essentially identical pressurized water reactors with supporting facilities. Each unit's primary coolant system consists of a pressurized water reactor system designed by the Westinghouse Corporation and is comprised of the reactor vessel and four heat transfer loops. Each loop contains a reactor coolant pump, steam generator, and associated piping and valves. In addition, each unit includes a pressurizer, a pressurizer relief tank, interconnecting piping, and the instrumentation necessary for operational control. All major components of each unit's reactor coolant system are located in their respective containment building. The design reactor thermal power level was 3250 Megawatts thermal (MWth). The corresponding electrical output was approximately 1,085 Megawatts electric (MWe) for both Units 1 and 2.

The initial construction of the station was authorized on December 26, 1968. Unit 1 and Unit 2 achieved initial criticality on June 19, 1973 and December 24, 1973 respectively. Next, Unit 1 was synchronized to the grid for the first time on June 28, 1973 and Unit 2 on December 26, 1973. Finally, Unit 1 and Unit 2 began commercial operation on December 31, 1973 and September 19, 1974 respectively. Between the two units, Zion operated for approximately 248,238,983 effective MWhrs over the course of its operating lifetime.

On January 15, 1998, Commonwealth Edison (ComEd) announced the permanent shutdown of both Zion reactors. The shutdown decision was based on the corporation's economic determination that neither Zion reactor would be able to produce competitively priced electricity in a deregulated marketplace over the facility's remaining useful life.

On February 13, 1998, ComEd certified the permanent cessation of operation of ZNPS Units 1 and 2 to the NRC. On March 9, 1998, ComEd certified to the NRC that all fuel assemblies had been permanently removed from both ZNPS reactor vessels and placed in the SFP. Both units at ZNPS were subsequently placed in a SAFSTOR condition (a period of safe storage of the stabilized and defueled facility) until eventual final decommissioning and dismantlement.

Upon docketing of the certification for permanent cessation of operation and permanent removal of fuel from the reactor vessels, the 10 CFR Part 50 license no longer authorizes operation of the reactors or emplacement or retention of fuel in the reactor vessels. In addition, the operating licenses scheduled to expire in April 2013 for Unit 1 and November 2013 for Unit 2 continue to remain in effect until the NRC notifies Zion*Solutions* that the licenses have been terminated.

The reactors at Zion remained in a SAFSTOR condition until September of 2010. At this point, the license for the facility was transferred from Exelon Generation Company (Exelon) (the licensee at that time) to Zion*Solutions* LLC. This was accomplished to allow Zion*Solutions* to begin the process of the physical decommissioning of the ZNPS.

A synopsis of the operational history is provided in Table 2-1.

2.1.4. Incidents

Based on the review of existing plant records (e.g. annual and semi-annual reports, licensee notifications, Occurrence Description Reports, and Personnel Data Questionnaire) approximately 305 incidents with radiological or hazardous material implications occurred between the commencement of plant operation in 1974 and placing both reactor units in a SAFSTOR condition in 1999. A majority of these incidents took place within the "Security-Restricted Area" and, while contributing to the radiological contamination and potential contamination of the structures and soils directly related to the operation of the reactors, were generally contained within the RCA, which is already known to be "impacted". Those incidents occurring outside of the "Security-Restricted Area" have contributed to the "impacted" classification of other supporting structures and surrounding open land areas. These include:

- Spills outside of the RCA or incidents involving potential contamination based on leakage from systems that had been historically contaminated by primary to secondary leaks,
- Loss of control of radioactive materials resulting in the potential for contamination outside of the RCA,
- Spills of plant liquid radioactive effluents resulting in soil contamination,
- Hazardous material spills or losses of control, and
- Contamination of systems not originally designed as radioactive systems.

A synopsis index of these incidents is presented in Table 2-2.

2.1.4.1. Radiological Spills

The HSA indicates that between 1973 and 1997, 64 documented spills occurred at the facility. Of those, 18 spills occurred either inside of Unit 1 or Unit 2 Containment, 21 occurred inside of the Auxiliary Building and two occurred inside of the Fuel Handling Building. These spill incidents, while contributing to the radiological contamination of these buildings, were generally contained within the radiologically controlled drains and waste systems of the structures.

Of the remaining 23 documented spill incidents, 14 occurred either inside the Rad Waste Annex trackways or just outside of the trackway doors in the open land areas between the Containment structures and the Turbine Building. The prevalence of these incidents causes concern for the

potential contamination of ground coverings (concrete and asphalt) as well as surface and subsurface soils in these two areas. The HSA specifically refers to these two areas as the most extensively contaminated open land areas on the site. Substantial spills of spent resins and contaminated water were documented. Most significant spills occurred prior to 1980 which preceded the construction of the Rad Waste Annex. The Annex was constructed to provide an enclosure to the area where spent resins were sluiced and solidified for shipment and disposal. Prior to the construction of the Annex, these activities were conducted in the open on a concrete pad. It is estimated that over 100 gallons of spent resin may have leaked into the soils at these locations.

Other significant spills and releases are summarized as follows;

- Overflow of both Unit 1 and Unit 2 Primary Water Storage Tank (PWST) as well as Unit 1 Secondary Condensate Storage Tank (SST)
- Unmonitored release of potentially contaminated liquids through the normal effluent release pathway to Lake Michigan from the Unit 2 Condenser hotwell, the Turbine Building Fire Sump and the Turbine Building drain system.
- Spill of potentially contaminated water to the 560 ft. elevation of the Turbine Building.

These spills resulted in the affected areas being designated as "impacted".

2.1.4.2. Chemical Spills

Between 1973 and 1997, the HSA documented 67 incidents involving the unplanned spill or release of chemicals and/or potentially hazardous liquids to the environment. These incidents ranged from spills of acids and caustics used in the plant's various systems to the spill of diesel and fuel oil from systems and storage tanks. A majority of the incidents occurred inside of impacted buildings. There were also several instances where caustics and acids exceeding Technical Specification requirements were discharged through the plant WWTF. In addition, significant spills of mercury occurred in the Turbine Building and Crib House.

These spills were controlled and remediated in accordance with the station policies and procedures for identification, control and remediation of hazardous material releases.

2.1.4.3. Loss of Radioactive Material Control

The HSA documents 132 incidents involving the loss of control of radioactive material between 1973 and 1997. Of those incidents, 61 involve the identification of radioactive material in non-posted areas inside of the RCA, such as the Maintenance Shop or the Tool Crib. Twenty-two (22) incidents involve the identification of radioactive material in trash piles or dumpsters located in non-posted areas inside the RCA. Twenty-four (24) incidents involve the identification of radioactive material in the trash segregation area located in the Unit 1 Turbine Building trackway. Four (4) incidents pertained to the loss of control of radioactive material while shipping or receiving radioactive material packages. Seven (7) incidents involved personnel leaving site with radioactive material on their person or in their vehicle and 14 incidents involved the loss of control of radioactive material where the material was

discovered and recovered outside of the RCA. Areas affected by these incidents were classified as "impacted."

2.1.4.4. System Cross-Contamination

Starting in 1981 and expanding dramatically in 1984, the Unit 1 Steam Generators (S/G) exhibited significant primary to secondary leaks. Unit 2 Steam Generators also exhibited primary to secondary leaks, but not to the extent exhibited by Unit 1. On September 10, 1984, Unit 1 was shut down as primary to secondary leakage exceeded the Technical Specification limit of 500 gallons per day. Due to the extensive primary to secondary leakage, secondary systems not originally expected to contain radioactivity became contaminated. The level of contamination varied from system to system.

In addition, many other secondary systems had the potential for cross contamination including open cycle and closed cycle cooling systems, auxiliary systems, and storage tanks. Following the guidance in NRC Bulletin 80-10, *"Contamination of Nonradioactive Systems and Resulting Potential for Unmonitored Release of Radioactivity to Environment"* (Reference 2-12), non-contaminated systems was routinely monitored to identify contamination events in a timely fashion. When normally non-contaminated systems became contaminated, they were evaluated against 10 CFR Part 50 Appendix I criteria. In addition, systems already contaminated were monitored according to plant chemistry and surveillance procedures to measure and trend the levels of activity within the systems.

Based on the HSA investigation, several additional secondary systems besides the Main Steam, Condensate and Feedwater systems were identified as contaminated. These systems included the Auxiliary Boiler, Unit 1 and Unit 2 Oil Separators and the Turbine Building equipment and floor drain systems. The structures associated with these systems are classified as "impacted."

2.1.5. Findings and Conclusions

The ZNPS was designed with multiple boundaries to control and contain the radioactive contents within its many systems, components, and structures. Many of these systems and structures have been impacted due to routine operations and maintenance activities during the operational and post operational history of the plant. Structures classified as "impacted" by the operation of the facility include Unit 1 and Unit 2 Containments, the Auxiliary Building, the Fuel Handling Building, the Turbine Building, the Service Building, the Crib House and system storage tanks located outdoors adjacent to the Crib House. Secondary systems, components and structures that were not originally anticipated to be contaminated have been impacted as the result of system cross contamination between the primary coolant system and secondary steam systems due to the failure of tubes within both unit's S/Gs. In addition to the structures, the soils surrounding these building have also been deemed to be directly impacted by the operation of this facility. This area is defined by the surrounding double-security fence and has been designated as the "Security-Restricted Area".

The area surrounding the "Security-Restricted Area" area has been designated as the "Radiologically-Restricted Area". This area encompasses the ancillary support buildings such as the Engineering and Construction (ENC) Building, the Gate House, the Interim Radioactive Waste Storage Facility (IRSF), the Switchyard and various warehouses and storage buildings.

Based on the findings documented in the HSA, these structures along with the soils within the "Radiologically-Restricted Area" fence has also been deemed as "impacted."

In addition to the area within the "Radiologically-Restricted Area" area, several additional areas have been deemed as "impacted". These include the site parking lot, the open land area along the south site boundary, the beach adjacent to the site and the area along Shiloh Boulevard designated as the West Training area. The parking lot was designated as impacted as it is a major path for radioactive material movement onto and off of the site. The West Training area was the former location of the Zion Training Building. This building housed a Westinghouse training reactor from 1980 to 1987. The reactor was decommissioned in 1988 and the license was terminated. The building itself was dismantled shortly following the permanent shutdown of the facility.

Figure 2-1 illustrates the "Owner-Controlled Area" that is under the management of the licensee. The "Radiologically-Restricted Area" area is illustrated in Figure 2-2 and the "Security-Restricted Area" area is illustrated in Figure 2-3.

Based on current and historic sample results from the ZNPS Radiological Environmental Monitoring Program (REMP), there is no indication that surface waters on or near the facility or the ground water beyond the site have been affected by the licensed operation of the facility. However, further evaluations of the groundwater directly below the licensed facility have been conducted. The initial findings of this study are presented in section 2.3.6. The normal effluent release pathway for the facility is the Circulating Water Discharge Tunnel that discharges to Lake Michigan approximately 870 feet from the lakeshore. During operation, effluent discharges contained measurable amounts of radioactive material resulting from liquid releases conducted in accordance with the license and permit requirements.

There were periods of liquid effluent releases during operation of the plant where it was determined that calculated dose to a maximally exposed individual via the liquid effluent pathway exceeded the design objective of 10 CFR Part 50, Appendix I. However, it was also determined that these liquid effluent releases did not exceed the concentration limits of 10 CFR Part 20 or the EPA fuel cycle dose limit in 40 CFR Part 190. The dose from liquid effluents has already been accounted for in accordance with the regulations governing radioactive effluent from power plants and no remediation is required.

2.1.6. Initial Survey Units and Classification

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.

2.1.6.1. <u>Survey Areas</u>

The entire 331 acre site was divided into survey areas. Survey areas are typically larger physical sections of the site that may contain one or more survey units depending on their classification. Survey area size was determined based upon the specific area and the most efficient and practical size needed to bound the lateral and vertical extent of contamination identified in the area. Survey areas that have no reasonable potential for residual contamination were classified as

"non-impacted". These areas have no radiological impact from site operations and are identified in the HSA. Survey areas with reasonable potential for residual contamination were classified as "impacted."

2.1.6.2. Survey Units

The classified survey areas established by the HSA were further divided into survey units. A survey unit is a portion of a structure or open land area that is surveyed and evaluated as a single entity following FSS. Survey units were delineated to physical areas with similar operational history or similar potential for residual radioactivity to the extent practical. To the extent practical, survey units were established with relatively compact shapes and highly irregular shapes were avoided unless the unusual shape was appropriate for the site operational history or the site topography.

The survey units established by the HSA were used as initial survey units for characterization. Prior to characterization, survey unit sizes for Class 1 open land survey units were adjusted in accordance with the guidance provided in MARSSIM, section 4.6 for the suggested physical area sizes for survey units for FSS.

The decommissioning approach that will be implemented by the ZSRP calls for the complete segmentation, removal and disposal of all impacted systems and above-grade structures. With the exception of structure basement floors and walls that reside 3 feet below grade, and concrete structures that are candidates for the potential reuse of concrete as hard fill, no portion of any structure will remain at site closure and consequently, be subjected to FSS. However, survey units have been established for structures to facilitate other characterization objectives. These objectives include providing survey data for remediation planning, estimating the waste volume contained onsite, and disposition options for the waste.

In addition, the survey units established for structures that are 3 feet below grade are intended for the purpose of characterization planning and do not correspond to the survey units that will be used for the FSS of remaining below grade structures (discussed in section 5.5.2 of Chapter 5).

The non-impacted open land survey units for the site are depicted in Figure 2-4. The impacted Class 3 open land survey units are depicted in Figures 2-5 and 2-6. The impacted Class 1 and Class 2 open land survey units are depicted in Figure 2-7. A summary of the initial survey unit classifications are presented as follows.

2.1.6.2.1. Class 1 Structures

The following is a list of some of the major buildings that were initially classified as impacted Class 1 structures in the HSA. The complete list of all initial structural survey units is provided in Table 2-3. These structures contain the nuclear reactors, primary reactor systems, reactor support systems, radioactive waste systems, and nuclear fuel handling and storage systems. During operations, radioactive material was routinely handled, transferred, and stored within these buildings. A majority of the current radioactive material inventory at Zion resides in these structures:

- Unit 1 Containment Building
- Unit 2 Containment Building
- Fuel Handling Building
- Radioactive Waste Building
- Auxiliary Building

Throughout facility operations, these structures were subjected to spills of radioactive liquids, the spread of loose surface contamination, and airborne radioactive material. Structural surfaces were routinely posted as contaminated areas. The decommissioning approach for these structures involves the complete segmentation, removal, and disposal of all systems and structural material as waste. With the exception of structural floors and walls that reside 3 feet below grade, and concrete structures that are candidates for the potential reuse of concrete as hard fill, no portion of these structures will remain at site closure and are therefore not subjected to FSS. The Class 1 structural survey units for building basements below the 588 foot elevation are depicted in Figures 2-8 and 2-9 for Unit 1 Containment, Figures 2-10 and 2-11 for Unit 2 Containment and Figures 2-12, 2-13 and 2-14 for the Auxiliary Building.

2.1.6.2.2. Class 2 and 3 Structures

The following is a list of some of the major buildings that were initially classified as impacted Class 2 or 3 structures by the HSA. All are located within the "Radiologically-Restricted Area" of ZNPS. The complete initial list of all structures and survey units is provided in Table 2-3. The primary function of these structures is to house the secondary side steam systems or electrical generating systems, or to provide office and/or warehouse space. The primary basis for the initial classification of secondary side systems and structures as impacted is due to a series of primary to secondary side leaks through the S/Gs during plant operations.

- Turbine Building
- Crib House
- Unit 1 Main Steam Valve Houses
- Unit 2 Main Steam Valve Houses
- Waste Water Treatment Facility (WWTF)
- Warehouse/Mechanical Maintenance Training Area
- Station Construction Building
- Illinois Department of Nuclear Safety (IDNS) Building
- Gate House
- North Warehouse
- South Warehouse

These structures did not routinely house radioactive systems or materials during operations. However, it was possible, due to their physical proximity to effluent release pathways, radioactive contamination of secondary side systems, temporary storage and transport of radioactive materials in and through these buildings, and past incidents involving the loss of control of radioactive material, that residual radioactive material could be found in and on, and around these structures. Consequently, this justified their initial classification as impacted Class 2 or Class 3. As with the Class 1 impacted structures, the decommissioning approach calls for the complete segmentation, removal, and disposal of all Class 2 or 3 systems and structural material as waste or salvage. With the exception of structural below-grade foundations and concrete structures that are candidates for the potential reuse of concrete as hard fill, no portion of these structures will remain at site closure and therefore will not be subjected to FSS. The Class 2 structural survey units for building basements below the 588 foot elevation are depicted in Figures 2-15 for the Turbine Building and Figure 2-16 of the Forebay for the Circulating Water system under the Crib House. There are no Class 3 structures that will remain in the final site configuration.

2.1.6.2.3. Class 1 Open Land Areas

The following open land areas have been initially classified as impacted Class 1. The basis for this initial classification is due to a series of documented incidents of the contamination of soil by radioactive material in these areas during facility operations. These incidents include spills of radioactive liquids and resins, radioactive system leakage, and storage of radioactive packages and containers. The complete list of all initial open land survey units is provided in Table 2-4.

- Sludge Drying Bed Area
- WWTF Area
- Unit 1 PWST/SST Area
- Unit 2 PWST/SST Area
- South Yard Area Northeast of the Gate House
- Yard Between Unit 1 Containment and Turbine Building
- Yard Between Unit 2 Containment and Turbine Building
- Soils under and around the Unit 1 Containment, Unit 2 Containment, the Fuel Handling Building and the Auxiliary Building

Based on an assessment of historical incidents and events, it was anticipated that the surface and subsurface soils in these areas could possibly contain residual radioactive material in excess of the unrestricted release criteria. Class 1 open land survey units are illustrated on Figure 2-7.

2.1.6.2.4. Class 2 Open Land Areas

The following open land areas were initially classified as impacted Class 2. Based upon a review of the historical information and operational radiation and contamination surveys performed in these areas as documented in the HSA, there was a potential for residual

radioactive contamination to exceed the unrestricted release criteria. The complete list of all initial open land survey units is provided in Table 2-4.

- Crib House Area
- Southeast Protected Area Yard
- The Gate House Area and the Protected Area Southwest Yard
- North Protected Area Yard
- Soils under and around the Service Building and the Turbine Building

Class 2 open land survey units are illustrated on Figure 2-7.

2.1.6.2.5. Class 3 Open Land Areas

The following open land areas were initially classified as Class 3. Historical information contained in the HSA indicated that the presence of residual radioactivity in concentrations in excess of the unrestricted release criteria was not expected. The complete list of all initial open land survey units is provided in Table 2-4.

- Northeast Corner of the Restricted Area adjacent to the Lake
- Interim Radioactive Waste Storage Area (IRSF)/Fire Training Area
- East Training Area
- North Gate Area
- Switchyard
- In-Processing Building/Station Construction Area
- North Warehouse Area
- South Warehouse Area
- Exclusion Area South of Gate House
- Exclusion Area South of Turbine Building
- Southeast Corner of the Restricted Area
- Construction Parking Area
- Area South of Switchyard
- Owner Controlled Area South of Restricted Area
- West Training Area

Class 3 open land survey units are illustrated on Figure 2-5 and Figure 2-6.

2.1.6.2.6. Non-Impacted Areas

Based on a review of the operating history of the facility, historical incidents, and operational radiological surveys as documented in the HSA, the following areas have been deemed not impacted by licensed activities or materials. The complete list of all initial open land survey units is provided in Table 2-4.

- Northeast Corner of the Exclusion Area
- Power House Area
- Owner Controlled Area North of Shiloh Boulevard
- Owner Controlled Area West and South of the West Training Area
- Owner Controlled Area West of the Switchyard
- Met Tower Area

Non-impacted open land survey units are illustrated on Figure 2-4.

2.2. Characterization Approach

Site characterization of the ZNPS was performed in accordance with the Characterization Survey Plan. It was developed to provide guidance and direction to the personnel responsible for implementing and executing characterization survey activities. The Characterization Survey Plan worked in conjunction with implementing procedures and survey unit specific survey instructions (sample plans) that were developed to safely and effectively acquire the requisite characterization data.

Characterization data acquired through the execution of the Plan was used to meet three primary objectives:

- Provide radiological inputs necessary for the design of FSS,
- Develop the required inputs for this License Termination Plan (LTP), and
- Support the evaluation of remediation alternatives and technologies and estimate waste volumes.

The decommissioning approach for the ZSRP calls for the demolition and removal of all on-site buildings, structures, and components to a depth of at least 3 feet below grade. Consequently, characterization efforts focused on open land areas and remaining structures that will be subjected to FSS. Extensive characterization of equipment, systems or structures that will be removed prior to the performance of final surveys is not required in accordance with NUREG-1757, Appendix O.

The decommissioning approach for ZSRP also calls for the beneficial reuse of concrete from building demolition as clean fill. The only concrete structures that will be considered are those where the probability of residual contamination is minimal. Characterization in this case will consist of an in-situ assessment of the concrete under consideration to ascertain if the structure concrete is an acceptable candidate. Concrete that meets the non-radiological definition of Clean

Concrete Demolition Debris (CCDD) and has been deemed suitable for offsite release in accordance with the site unconditional release process may be used as basement fill.

A significant question that must be answered by the characterization is whether or not a survey unit is classified correctly. The appropriate classification of a survey unit is critical to the survey design for FSS. A classification which underestimates the potential for contamination could result in a survey design that does not obtain adequate information to demonstrate that the survey unit meets the release criteria. In some cases, this can increase the potential for making decisions errors.

As site-specific DCGLs were not yet established for the Zion decommissioning at the time the characterization survey was performed, alternate action levels were selected. The screening DCGLs presented in NUREG-1757 and the concentration values found in NUREG/CR-5512 Volume 3, *"Residual Radioactive Contamination from Decommissioning Parameter Analysis"* (Reference 2-13), Table 6.91 ($P_{crit} = 0.10$) for soils were used as alternate action levels to assess the correct classification of impacted open land or soil survey units.

For structures, the gross screening level that was used during characterization as an action level to evaluate the classification of survey units was the nuclide-specific screening value of 7,100 dpm/100cm^2 total gross beta-gamma surface activity based on Co-60 from NUREG-1757, Appendix H. Use of the Co-60 screening value was conservative as it was anticipated that the radionuclide distribution for surface contamination would be principally Co-60 and Cs-137 and, the more conservative approach was to assume a distribution of 100% Co-60 as the screening value for Cs-137 is significantly greater.

2.2.1. Data Quality Objectives

DQOs were implemented for characterization surveys in a similar manner as anticipated for the FSS. However, the goal of characterization is contamination quantification and delineation of the nuclide suite, whereas the FSS goal is comparison of data against the unrestricted use criteria to demonstrate compliance with 10 CFR 20.1402. Characterization inspections and surveys of sufficient quality and quantity were performed to determine the nature, extent and range of radioactive contamination in each applicable survey unit, including applicable structures, residues, soils and surface water.

Characterization surveys were designed to gather the appropriate data using the DQO process as outlined in MARSSIM, Appendix D. The seven steps in the DQO development process are:

1) State the problem,
2) Identify the decision,
3) Identify inputs to the decision,
4) Define the study boundaries,
5) Develop a decision rule,
6) Specify limits on decision errors, and
7) Optimize the design for obtaining data.

The DQOs for site characterization included identifying the types and quantities of media to collect. No structures located above 3 feet below grade will remain following decommissioning and be subjected to FSS. Consequently, sample collection was focused on the assessment of concrete basement structure materials and surrounding soils. Building concrete was sampled by obtaining concrete core samples. Soils were sampled volumetrically. Sufficient measurements were obtained to determine the mean and maximum activity as well as the sample standard deviation. Direct measurements and scans of concrete and surface soils were also made using the same instruments and Minimal Detectable Concentrations (MDC) as will be employed for FSS. Volumetric samples that exhibited the highest activity were sent to an off-site laboratory for analysis of Hard-to-Detect (HTD) radionuclide(s).

2.2.2. Survey Design

Characterization surveys were designed and performed in accordance with all applicable approved procedures and the Characterization Survey Plan. Survey design incorporated a graded approach based upon the DQOs for each survey unit. 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. Therefore, the characterization surveys that were performed in a Class 1 survey unit focused on bounding the contamination where contamination was potentially present. The survey design was based upon the number of measurements and samples required to identify the lateral and vertical extent of the contamination. Areas classified as non-impacted, Class 2 or Class 3 received surveys developed to include a combination of systematic and biased survey measurement locations and scan areas. Biased survey designs used known information to select locations for static measurements and/or samples. Systematic survey design selected static measurement and/or sample locations at random or by using a systematic sampling design with a random start. The decision of whether to use primarily a biased survey design or a systematic approach was addressed by the DQO process for each survey unit. A biased approach was warranted when the characterization effort was designed to delineate the extent of an area that requires remediation. Alternatively, a systematic approach was warranted if the characterization effort was designed to verify the basis for the classification of a survey unit.

2.2.2.1. Number of Static Measurements and/or Samples

The number of measurements and/or samples that were taken in each survey unit was determined by assessing the sample size necessary to satisfy the DQOs.

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 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. Consequently, the number of systematic static measurements and/or samples was sufficiently robust so that a high degree of confidence was achieved to establish that only diminutive levels of licensee-generated radioactive material resided in these areas.

2.2.2.2. Determination of Static Measurement or Sample Locations

For the characterization of non-impacted and impacted open land areas and Class 2 structural survey units that will be subjected to FSS, sample locations were primarily chosen at random. Sample locations were determined by generating random pairs of coordinates that corresponded to specific locations within the survey unit. The location of biased measurements and/or samples that were taken in each survey unit was determined by the professional judgment of the responsible Radiological Engineer during the survey design process. Consideration was given to locations that exhibited measurable radiation levels above background (i.e. by scanning), depressions, discolored areas, cracks, low point gravity drain points, actual and potential spill locations, or areas where the ground has been disturbed. Historical information from the HSA aided in the selection of biased locations.

2.2.2.3. Scan Coverage

Survey units were scanned to the extent practical in accordance with their classification. The area to be scanned in each survey unit was determined during the survey design process. The area scanned was contingent upon the accessibility of the surface areas in the survey unit and the recommended scan coverage guidelines presented in Table 2-5.

2.2.2.4. Types of Measurements or Samples

The characterization survey of building surfaces consisted of a combination of surface scans (beta and gamma); static beta measurements, material samples and smears. The characterization survey of any concrete and/or asphalt-paved open land areas that will remain and be subjected to FSS consisted of a combination of surface scans (beta and gamma), static beta measurements, and volumetric samples. The survey of the open land areas consisted of gamma scans and the sampling of surface and subsurface soil, sediment and surface water for isotopic analysis. The following is a description of the different types of measurements and samples that were utilized.

2.2.2.4.1. Static Measurements

Static measurements were performed to detect direct levels on structural surfaces of the buildings or on concrete or asphalt paved areas. These measurements were performed using primarily ~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 MDC for static measurements.

2.2.2.4.2. Beta Surface Scans

Scanning was performed in order to locate areas of residual activity above the 7,100 dpm/100cm^2 action level. Beta scans were performed over accessible structural surfaces including, but not limited to; floors, walls, ceilings, roofs, asphalt and concrete paved areas. Floor monitors using large area gas-flow proportional detectors (typically with 584 cm^2) were used for floor and other larger accessible horizontal surfaces. Hand-held beta scintillation and/or gas-flow proportional detectors (typically 126 cm^2) were used for surfaces not accessible by a floor monitor.

Beta scanning was performed 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 surface conditions prevented scanning at the specified distance, the detection sensitivity for the alternate distance was determined, and the scanning technique adjusted accordingly. Scanning speed was calculated *a priori* to ensure that the MDC for scanning was appropriate for the stated objective of the survey.

If not impacted by high ambient noise levels, technicians monitored the audible response of the instrument to identify locations of elevated activity that require further investigation and/or evaluation. Any identified areas of elevated contamination were marked or flagged for further investigation and potential decontamination.

2.2.2.4.3. Gamma Surface Scans

Gamma scans were performed over open land surfaces to identify locations of residual surface activity. Sodium iodide (NaI) gamma scintillation detectors (typically 2" x 2") were typically used for these scans. Zion*Solutions* TSD 11-004, "*Ludlum Model 44-10 Detector Sensitivity*" (Reference 2-14) examines 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.

Scanning was performed 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 and/or sampling.

2.2.2.4.4. Removable Surface Contamination

If applicable, removable beta contamination or smear surveys were performed to verify loose surface contamination is less than the action level of 1,000 dpm/100cm^2. A 100 cm^2 surface area was sampled with a circular cloth or paper filter, using moderate pressure. Smears were then analyzed for the presence of gross beta activity using a proportional counting system or equivalent.

2.2.2.4.5. Concrete Core Sampling

Concrete core boring and the sampling of concrete were used to assess the depth of surficial contamination and the presence of volumetric contamination in concrete walls and floors that will remain and be subjected to FSS. Core bore sampling of concrete was accomplished using a diamond bit core drill. The concrete sample produced by the coring was typically sliced into

½-inch wide "pucks", representing a certain depth into the surface. 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.

2.2.2.4.6. Material Sampling

Samples of soil, sediment, and sludge were obtained from designed judgmental and systematic sample locations as well as other biased locations in areas exhibiting elevated activity that were identified by scanning. Surface soil is defined as the top 15 cm (6-inch) layer of soil while subsurface soil is defined as soil below the top 15 cm layer in 1 m increments. Surface soil was collected using a split spoon sampling system or, by using hand trowels, bucket augers, or other suitable sampling tools.

Subsurface soil was sampled by direct push sampling systems (e.g. GeoProbe®) or by the excavation of test pits. Subsurface soil sampling was performed as necessary to address the DQOs for the survey unit.

An adequate amount of material (ranging from 0.5 liters up to 2 liters) was collected at each location. Sample preparation included the removal of extraneous material and the homogenization and drying of the soil for analysis. Separate containers were used for each sample and each container will be accounted for throughout the analysis process using positive physical control or by a chain-of-custody record.

2.2.3. Instrumentation Selection, Use and Minimum Detectable Concentrations (MDCs)

The radiation detection and measurement instrumentation for characterization was selected to provide both reliable operation and adequate sensitivity to detect the Radionuclides of Concern (ROC) identified for the decommissioning of the ZNPS at levels sufficiently below the established action levels. Detector selection was based on detection sensitivity, operating characteristics, and expected performance in the field. In all cases, the 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.

Commercially available portable and laboratory instruments and detectors were typically used to perform the three basic survey measurements: 1) surface scanning; 2) static measurements; and 3) analysis of material samples.

Instrumentation and nominal MDC values that were employed during characterization are listed in Table 2-6.

2.2.3.1. <u>Instrument Calibration</u>

All data loggers, associated detectors, and all other portable instrumentation that were used for characterization were calibrated on an annual basis using National Institute of Standards and Technology (NIST) traceable sources. The calibration of instruments used for characterization is addressed in section 5.2 of the QAPP.

2.2.3.2. Instrument Use and Control

The receipt, inspection, issue, control, and accountability of portable radiological instrumentation used for characterization was performed in accordance with issue and control procedures for portable radiological instrumentation. The issue and control of instruments used for characterization is addressed in section 5.1 of the QAPP.

2.2.4. Laboratory Instrument Methods And Sensitivities

Gamma spectroscopy was primarily performed by the on-site radiological laboratory. Gas proportional counting and liquid scintillation analysis was performed by an approved vendor laboratory in accordance with approved laboratory procedures. Zion*Solutions* ensured that the quality programs of the contracted off-site vendor laboratories that were used for the receipt, preparation and analysis of characterization samples provided the same level of quality as the on-site laboratory under the QAPP.

In all cases, analytical methods were established to ensure that required MDC values are achieved. The analysis of radiological contaminants used standard approved and generally accepted methodologies or other comparable methodologies. Table 2-7 provides the typical analytical methods employed and the laboratory MDC achieved by the off-site vendor laboratories used for characterization.

2.2.5. Quality Assurance

MARSSIM, section 2.2 discusses the need for a quality system to ensure the adequacy of data used to demonstrate that site conditions are acceptable for unrestricted release. Laboratory quality for sample analysis taken to support characterization and FSS is discussed in NUREG-1576, *"Multi-Agency Radiological Laboratory Analytical Protocols Manual* (MARLAP)*"* (Reference 2-15) and Regulatory Guide 4.15, *"Quality Assurance of Radiological Monitoring Programs (Inception through Normal Operations to License Termination) - Effluent Streams and the Environment"* (Reference 2-16). Further, MARSSIM and MARLAP both indicate that a Quality Assurance Project Plan may be used in addition to, or in lieu of, existing quality systems to ensure data quality is achieved.

The QAPP was prepared and implemented to ensure the adequacy of data being developed and used during the site characterization and FSS process. The QAPP describes policy, organization, functional activities, the DQO process, and measures necessary to achieve quality data. It supplements the quality requirements and quality concepts presented in ZS-QA-10, *"Quality Assurance Project Plan - Zion Station Restoration Project"* (Reference 2-17) which adequately encompass other risk-significant decommissioning activities.

All characterization activities essential to data quality were implemented and performed using approved procedures. The effective implementation of characterization was verified through audit and surveillance activities, including field walk-downs by Zion*Solutions* Characterization/License Termination management and radiological engineering staff and program self-assessments, as appropriate. Corrective actions were prescribed, implemented, and verified when deficiencies were identified. These measures applied to any applicable services provided by off-site vendors, as well as on-site sub-contractors.

The Characterization Survey Plan was developed according to the essential elements of the quality assurance and quality control (QA/QC) program for the decommissioning of the ZNPS and is subject to the QAPP. The QA/QC program elements applicable to characterization are as follows:

- Establishment/implementation of plans, procedures, and protocols for the field operations.
- Actions to ensure that the procedures are understood and followed by the implementing staff.
- Documentation of the data collected.

Details of the QA/QC elements specific to characterization are presented in the QAPP, as well as the procedures and sample plan instructions. The characterization operations and the associated data acquisition and recording was guided and conducted in compliance with these QA/QC requirements. The specific QA/QC program components for site characterization are as follows:

- Personnel qualifications, experience, and training.
- Execution in accordance with approved procedures.
- Proper documentation of survey data and sample analyses.
- Selection of appropriate instruments to perform the surveys.
- Proper instrument calibration and daily functional checks.
- Management oversight of characterization activities relative to the adherence to procedures, protocols, and documentation requirements.

All characterization activities were conducted in accordance with the Characterization Survey Plan, the QAPP, all applicable implementing procedures, and approved sample plan instructions.

2.3. Summary of Characterization Survey Results

The site characterization of the ZNPS site commenced on November 2, 2011 with the characterization of the open land survey units encompassing the proposed site for the future ISFSI facility. At the time this survey was performed, the site-specific Zion*Solutions* characterization plans and procedures were still under development. Consequently, due to schedule constraints, Zion*Solutions* contracted the *Energy*Solutions Commercial Services Group (ESCSG) to perform characterization of the ISFSI location, the location where the Vertical Concrete Cask (VCC) Construction Area was to be located and the pathway for the new rail tracks. These locations are illustrated on Figures 2-17 and 2-18. These surveys were performed on a turnkey basis using ESCSG technicians, procedures and instrumentation. A survey-specific characterization plan was developed and approved for these areas, survey packages were prepared and executed in accordance with ESCSG procedures and a final report "*Characterization of the Zion Station Independent Spent Fuel Storage Facility (ISFSI)* (Reference 2-18) dated July 30, 2012 was prepared by ESCSG and approved by Zion*Solutions*. The results of these surveys were validated and integrated into the subsequent site-specific characterization program and the results are reported in this chapter as valid characterization data for the affected survey units.

In February 2012, the site-specific Zion*Solutions* site characterization program was approved. In addition, all required personnel, equipment and instrumentation necessary to implement the characterization program were procured. Characterization activities, self-performed by Zion*Solutions* in accordance with the Zion*Solutions* characterization program commenced on April 10, 2012 with the acquisition of concrete core samples from the 542 foot elevation of the Auxiliary Building. The initial scheduled site characterization effort concluded on November 11, 2013.

Throughout 2012 and 2013, characterization activities were performed in parallel with radioactive commodity removal and radioactive waste shipment activities at ZNPS. Consequently, the removal and movement of radioactive material directly impacted the ability to access certain structural survey units and open land areas and obtain meaningful characterization survey data. In several specific areas, characterization has been deferred until such time that radiological or physical conditions would allow access for characterization. Zion*Solutions* intends to continue characterization throughout the decommissioning process, including following the submittal of this LTP. This is discussed further in section 2.5. In the case where significant additional characterization data is obtained, this chapter of the LTP will be updated by revision or addendum as a part of the required 2 year update of the approved LTP.

2.3.1. Background Studies

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 by ESCSG as an integral part of the work scope pertaining to the characterization of the ISFSI and VCC 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, Zion*Solutions* conducted a comprehensive background study of non-contaminated concrete by acquiring and analyzing concrete core samples taken from the 559 foot elevation and 594 foot elevation of the Crib House. This study was conducted to support the eventual evaluation of concrete demolition debris as clean hard fill. In July of 2012, an additional study was performed to evaluate background for soils. Additional scanning and acquisition and analysis of volumetric soil samples were performed.

2.3.1.1. *Energy*Solutions Background Study

The primary purpose of the *Energy*Solutions background study was to identify and quantify the levels of natural activity, including fallout, within soils and construction materials. This effort included support of the unconditional release and potential re-use of concrete originating from ZNPS as potential backfill material. The surveys were performed using the same instruments and survey techniques that were to be used during characterization.

This background study was performed through the selection of reference background areas known to be unaffected by plant operations to ensure any measured radioactivity would be of natural origin. It was determined that the best location for measuring background would be outside the restricted area boundary toward the north as based in part on *"Annual Report on the Meteorological Monitoring Program at Zion Nuclear Power Station for 2010"* (Reference 2-19), which demonstrated that winds are predominately from the west and northwest at the site. As a

result, locations north and northwest of the restricted area were less likely to be impacted by airborne particulate and gaseous effluents from past plant operations.

To ensure that the data was relevant, it was necessary to ensure the background reference areas selected were representative of the materials and areas to be surveyed at the facility. This included soils of similar geology and construction materials of similar age and content. Consequently, the reference area(s) selected were located as near to the facility as possible and representative of the construction of the plant. It was determined, based upon this selection criterion, that the area around the former Visitors Center would be representative of background at the site. Figure 2-19 represents the background survey locations used during the *Energy*Solutions background assessment.

Surface scans and direct surface activity measurements were performed for gross beta activity on both asphalt and concrete surfaces within the background reference areas. In addition, gamma walkover scans and direct measurements were performed over soil and vegetation. The scans were distributed evenly over each background reference area. One-minute static measurements were also collected.

In addition to the surface scans and direct measurements, samples were collected for volumetric analysis by gamma spectroscopy analysis. Fifteen (15) samples were taken of asphalt, concrete and surface soils and were analyzed by a qualified off-site vendor laboratory.

To account for any potential difference in geology, subsurface samples were also collected at each surface sampling location to assess for any differences in background if present. Subsurface samples were collected at depths between 6 inches below grade down to a depth of about 4 feet using GeoProbe™. Subsurface soil samples were composited over the depth of the sampled soil column. As with the surface soils, all composite subsurface samples were also analyzed by a qualified off-site vendor laboratory.

The results of the Energy Solutions background study was presented in CS-RS-PN-028, *"Background Reference Area Report - Zion Nuclear Power Station"* (Reference 2-20). A summary of the background survey results from this assessment is provided in Table 2-8. Based upon a review of the sample analysis results for asphalt, concrete and soils, only natural activity expected in background was detected. No other licensed materials were identified in the samples. Additionally, based upon the activities of the daughter products within the decay chain, in approximate equilibrium, the background reference area(s) and samples were deemed representative of background and were not impacted by site activities. A review of both the surface and subsurface soil sample results concluded that there appears to be no difference between the surface and subsurface radionuclide distribution.

2.3.1.2. Crib House Concrete Study

In March of 2012, ZSRP commenced an assessment of the concrete 594 foot and 559 foot elevation floors and lower walls of the Crib House. The DQOs established for this survey were to establish a background threshold range for volumetric concrete at ZNPS, evaluate the basement foundations and floors of the Crib House for the presence of volumetric radiological contamination and to provide a sufficient quantity and quality of uncontaminated concrete media

representative of the "Basement Fill" concrete to an off-site vendor for the derivation of distribution coefficients for the radionuclides of concern.

On each floor (594 foot and 559 foot elevations), 16 six-inch concrete core samples were taken from the floors and 4 six-inch concrete core samples were taken from the lower walls for a total of 40 concrete core samples. Sample locations were selected at random. Prior to acquiring the core samples, the area was scanned to ensure the absence of surficial radioactive contamination at each sample location. Scans were performed with a Ludlum 43-93 100 cm^2 alpha-beta scintillator detector. Gross beta background ranged from 150 cpm to 300 cpm. No activity greater than background was observed at each selected sample location.

All concrete core samples were analyzed by the on-site gamma spectroscopy system for gamma emitting radionuclides. Only natural activity expected in background was detected during the analysis. No other licensed materials were identified in the samples. A summary of the survey results from the assessment of the Crib House concrete is provided in Table 2-9.

2.3.1.3. Zion*Solutions* Soil Background Study

In July of 2012, a survey was performed of non-impacted soils adjacent to the ZNPS owner controlled area with the objective of determining background radioactivity concentrations in soils. The area chosen for the survey was the Zion City Park District's "Hosah Park", located north of the ZNPS at the end of Shiloh Boulevard. The park consists of open land areas covered with native grasses and low lying brush. While there did appear to be evidence of soil disturbance on the property, the evidence suggested that this occurred in the past and the land has been undisturbed for a number of years. Figure 2-20 illustrates the location of the area chosen for the survey.

The survey was designed to determine the radionuclide activity concentrations of key naturally occurring and man-made radionuclides, particularly Cs-137, in surface and subsurface soils. The survey design included surface and subsurface soil samples as well as static gamma measurements. Thirty (30) sample locations were chosen, biased towards soils that appeared to be undisturbed with minimal vegetation. At each location, a static one-minute measurement was taken using a sodium iodide detector, a sample was taken from the soils within the first 15 cm of grade and a composite soil sample was taken from soils between 15 cm below grade to 60 cm below grade. All soil samples were analyzed by a qualified off-site vendor laboratory.

The results of the survey were presented in a report titled *"Determination of Radionuclide Activity Concentrations in Soils in Non-Impacted Soils Adjacent to the Zion Nuclear Station"* (Reference 2-21). In both surface and subsurface soil populations, the only radionuclide identified, with the exception of naturally occurring radionuclides, was Cs-137. In the surface soil sample population, Cs-137 was positively identified in concentrations greater than MDC in 26 of the 30 samples. In the subsurface soil population, Cs-137 was positively identified in concentrations greater than MDC in five (5) of the 30 samples obtained. Based upon the concentrations observed and the distribution, it was postulated that the presence of Cs-137 was due to global fallout. The results of the sample analysis are presented in Table 2-10.

2.3.1.4. Technical Support Document Regarding Cs-137 Global Fallout

In order to establish an action criteria indicative of Cs-137 contamination levels in soil that are distinguishable from background levels, Zion*Solutions* TSD 13-004, *"Examination of Cs-137 Global Fallout In Soils At Zion Station"* (Reference 2-22) was prepared to document Zion*Solutions* soil sample results for Cs-137 in soils to date and compare the results to those anticipated from world-wide fallout. The TSD established the technical basis for the anticipated soil concentrations attributable to fallout and established criteria for investigating soil samples due to observed Cs-137 concentrations.

The soil sample data compiled in the TSD concludes that the majority of the soil samples taken for the background studies were from disturbed soils. The Hosah Park data as well as the data obtained during the ESCSG study corresponded with documented fallout levels from disturbed soil at sites in Massachusetts, New York and Pennsylvania. Consequently, predicted ranges for background concentrations of Cs-137 were established for disturbed soils as well as undisturbed soils based on literature. These ranges are presented in Table 2-11. The upper Cs-137 concentration for each category was used as the action level for the characterization of non-impacted open land survey units. The upper Cs-137 concentration for disturbed, non-drainage soil in Table 2-11 was used as the action level for the characterization of Class 2 and 3 open land area survey units.

2.3.2. Potential Radionuclides of Concern

Zion*Solutions* TSD 11-001, *"Potential Radionuclides of Concern during the Decommissioning of Zion Station"* (Reference 2-23) was prepared and approved in November 2011. The purpose of this document was to establish the basis for an initial suite of potential 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 theoretical 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 is listed in Table 2-12.

2.3.3. Impacted Structures and Systems

The decommissioning approach for the ZSRP requires the demolition and removal of all impacted buildings, structures, systems and components to a depth of at least 3 feet below grade. In addition, all systems and exposed metal below 3 feet below grade will also be removed. The accepted elevation for grade at ZNPS is the 591 foot elevation. The only structures that will remain and be subjected to FSS are the exposed steel lined walls and floor from the Unit 1 and Unit 2 Containment Buildings (after all interior concrete is removed) and the below-grade structural concrete outside of the liner, the reinforced concrete basement floor and outer walls of the Auxiliary Building and Turbine Building (after all internal walls and floors are removed), the reinforced concrete floor and walls of the SFP and Fuel Transfer Canal (after the steel liner is removed), the concrete floor and walls of the Crib House, the WWTF, the Forebay and Circulating Water Intake Piping and Discharge Tunnel below the 588 foot elevation. Consequently, all systems and components and structural surfaces above the 588 foot elevation will be remediated, disassembled and/or demolished, segregated by waste classification and

disposed of as clean demolition debris, clean salvage or radioactive waste. No extensive characterization was or will be performed of equipment, systems or structures that will be removed prior to the performance of FSS.

The current decommissioning approach for ZSRP also calls for the beneficial reuse of concrete from building demolition as clean fill. The only concrete structures that will be considered are those where the probability of the presence of residual contamination is minimal. Only concrete that has been demonstrated to be free of detectable plant-derived radionuclides and hazardous painted surfaces will be used. Characterization in this case will consist of an in-situ assessment of the concrete under consideration to ascertain if the structure concrete is an acceptable candidate. Only concrete that meets the non-radiological definition of CCDD and has been deemed suitable for offsite release in accordance with the site unconditional release process may be used as basement fill.

Radiological surveys of the interiors of structures at the ZNPS are routinely performed to ensure compliance with 10 CFR 20 requirements regarding the posting of areas and to identify radiological conditions for the implementation of controls for the protection of workers in these areas. The radiological information from these surveys will provide the basis for the disassembly and removal of systems and the demolition of impacted structures at the site. When remediation has adequately reduced radiological conditions to levels suitable for controlled demolition, the impacted structures will be demolished, packaged and properly disposed of as waste.

After commodity removal is complete, the structures that will remain at license termination, i.e., 3 feet below grade, will be re-surveyed to determine the concentrations of the residual radioactivity and the extent of additional remediation required, if any, to meet the unrestricted use criteria.

2.3.3.1. Unit 1 and Unit 2 Containments

The Unit 1 and Unit 2 Containment buildings house numerous systems containing primary coolant as well as radioactively contaminated support systems. Physically, both units are basically mirror images of the other. System leakage and maintenance activities over the operating life of both units have resulted in the radiological contamination of most of the interior surfaces of both structures. Some components, equipment, structural steel and concrete have become radioactive due to neutron activation. Based on the building design basis, the operating history as well as the present status of areas that are controlled as contaminated areas, all internal survey units in both the Unit 1 and Unit 2 Containment Buildings are considered to be Class 1 areas.

General area radioactive dose rates within the buildings range from 1 mrem/hr to over 250 mrem/hr. Loose radioactive contamination ranges from <1,000 dpm/100 cm2 to over 110 mrad/swipe. The general radiological conditions within the Unit 1 and Unit 2 Containment buildings are presented per survey unit in Table 2-13.

The basic decommissioning end-state for each Containment building will consist of the walls and floors below 588 foot elevation. In both Containment basements, all concrete will be removed from the interior side of the steel liner above the 565 foot elevation, leaving only the remaining

exposed liner below the 588 foot elevation, 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. The exposed metal liner will be cleaned to levels below DCGLs for basement structures (Chapter 5, Table 5-4). It is anticipated that only activity remaining on the steel liner after concrete removal will be loose dust from the demolition of the concrete. Following completion of a FSS in accordance with section 5.5.4 of this LTP, the basements will be backfilled with clean fill, CCDD, grout, or a combination of the three materials.

The large components inside each structure such as the Reactor Vessel, Pressurizer, Steam Generators, Reactor Coolant Pumps, primary piping and all associated systems will be removed and properly disposed of as radioactive waste. Several sections of pipe that is located below the 588 foot elevation that are embedded in concrete may remain as part of the final configuration of the structure. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of penetrations and embedded piping to remain is provided in Zion*Solutions* Technical Support Document (TSD) 14-016, *"Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State"* (Reference 2-24).

The upper steel containment liner, the refuel cavity and all interior walls above the 588 foot elevation in both units will also be removed and disposed of as radioactive waste. The outer containment shells above the 588 foot elevation will be surveyed and demolished. If the exterior concrete is free of detectable plant-derived radioactive material, it may be used as CCDD or disposed of as clean demolition debris. All waste material will be packaged and sent offsite for disposal at approved disposal facilities for radioactive waste or clean demolition debris.

It is anticipated that the most highly activated concrete in either Containment will be the internal radius of the Bio-Shield concrete that surrounded the vessel as it was closest to the active core region. The concrete inside the liner in the Under-Vessel area is also activated. Unit 1 and 2 Containment have interior concrete walls 23.5 inches thick in the Under-Vessel area and 30 inch thick concrete floors inside the liner. Consequently, the potential for activation outside the liner is minimal.

During the time that initial characterization was performed, all radioactive systems and components were still located inside each Containment. Consequently, ambient radiation dose rates inside the Containments prohibited the direct assessment of concrete and steel structural surfaces below the 588 foot elevation by scanning or direct measurement. Once commodity removal is complete in both of these structures, additional characterization will be performed by scan and direct measurement to identify the lateral and vertical extent of surficial contamination and the extent of any remediation that will be necessary on the structural steel and concrete that will remain in the final configuration of the Containments. All continuing Characterization Plans will be provided to the NRC for information and continuing Characterization Reports will be provided to the NRC for evaluation.

From June of 2012 through January of 2013, a series of concrete core samples were taken in the 568 foot concrete floor, 541 foot In-core tunnel floor and In-core tunnel walls in both Containment buildings. In addition, a single concrete core sample was taken from the reactor centerline location of Unit 1 Bio-Shield. A total of 39 concrete core samples were collected, 20 in Unit 1 Containment and 19 in Unit 2 Containment.

The locations selected for the concrete core sampling were biased toward locations where physical or observed radiological measurements indicated the presence of fixed and/or volumetric contamination of the concrete media. When possible, locations were determined based upon elevated observed contact dose rates or count rates. In addition, visual observations of floor and wall surfaces were used to identify potential locations of surface contamination, such as discoloration or standing water. The goal was to identify, to the extent possible, the locations that exhibited the highest potential of representing the worst case bounding radiological condition for concrete in each survey unit. This judgmental sampling approach also ensured there was sufficient source term in the cores to achieve the sensitivities required to determine the radionuclide distributions of gamma emitters as well as HTD radionuclides.

Sixteen (16) concrete core samples were taken on the 568 foot elevation of both the Unit 1 Containment and the Unit 2 Containment, eight inside the missile shield and eight outside of the missile shield. Three (3) concrete core samples were obtained from each of the In-core tunnel Under-Vessel areas in both Unit 1 and Unit 2 Containments. Two (2) concrete core samples were taken from the 541 foot elevation floor and one was taken from the wall directly under each reactor vessel. A single concrete core sample was obtained from the Unit 1 Bio-Shield. Since the Bio-Shields in both Containments will not remain in the end-state configuration and the concrete is to be removed as waste, this was the only core sample taken of Bio-Shield. No Bio-Shield core was obtained in Unit 2.

A summary of the on-site gamma spectroscopy results for the analysis of the concrete cores taken from the Unit 1 Containment 568 foot elevation is presented in Table 2-14. A summary of the on-site gamma spectroscopy results for the analysis of the concrete cores taken from the Unit 1 Bio-Shield is presented in Table 2-15. A summary of the on-site gamma spectroscopy results for the analysis of the concrete cores taken from the Unit 2 Containment 568 foot elevation is presented in Table 2-16. A summary of the on-site gamma spectroscopy results for the analysis of the concrete cores taken from the Containment In-core tunnel areas are presented in Table 2-17 for Unit 1 Containment and Table 2-18 for Unit 2 Containment. The locations where the core samples were taken are illustrated in Figures 2-21 and 2-22 for Unit 1 Containment and Figures 2-23 and 2-24 for Unit 2 Containment.

Zion*Solutions* TSD 14-028, *"Radiological Characterization Report"* (Reference 2-25) presents additional detail on the concrete sampling methodology and results of the radiological analysis of each concrete core sample obtained from the Containment Building basements. In addition, an in-depth analysis of the results of the concrete core samples taken in the Containment are presented in Zion*Solutions* TSD 13-006, *"Reactor Building Units 1 & 2 End State Concrete and Liner Initial Characterization Source Terms and Distributions"* (Reference 2-26).

With the exception of one (1) core sample where Eu-154 was positively identified, Co-60, Cs-137 and Cs-134 were the only plant-derived gamma emitting radionuclides identified by the analysis of the concrete core samples taken from the 568 foot elevation. The on-site gamma

spectroscopy results identified Co-60, Cs-134, Cs-137, Eu-152 and Eu-154 in the concrete core samples taken from the In-core tunnel areas.

For both Unit 1 and Unit 2 568 foot elevations, the sample analysis indicated that the majority of the radionuclide source inventory resides within the first ½-inch of concrete and that Cs-137 is the dominant radionuclide. Co-60 is significantly more prevalent in Unit 2 verses Unit 1. In addition, the analysis also indicates that the overall source inventory in Unit 2 is much lower than Unit 1.

An examination of the Under-Vessel In-core area profiles show diminishing activation at depth. In Unit 1, the source inventory is dominated by the presence of Cs-137 in the first ½-inch of concrete, which contains approximately 63% of the overall source inventory. In Unit 2, the Cs-137 concentrations are significantly less and the source inventory is dominated by Eu-152, which accounts for approximately 74% of the calculated total source inventory.

Analysis of the core taken from Unit 1 Bio-Shield indicates that Eu-152 and Eu-154, which are indicators of the neutron activation of concrete was not detected in concentrations greater than their respective MDC until a depth of 47.5 inches. This supports the postulation that neutrons were successfully attenuated prior to traversing the Bio-Shield and consequently, there is minimal potential for the activation of concrete in areas outside the Bio-Shield.

The top ½-inch puck from nine (9) of the 20 cores from Unit 1 and the top ½-inch puck from eight (8) of the 19 cores from Unit 2 were sent to Eberline Laboratory for gamma spectroscopy and HTD analyses for radionuclides such as H-3, C-14, Tc-99, Ni-63, Sr-90, and alpha emitters. The results of the analysis are presented in Table 2-19 for Unit 1 Containment and Table 2-20 for Unit 2 Containment. The radionuclide distribution for Unit 1 and Unit 2 Containments that is based on this sample population is presented in Table 2-21. Significant HTD radionuclides identified by the analysis of the concrete core samples include Ni-63, H-3 and Sr-90. The other radionuclides positively detected at concentrations greater than their respective MDC include; C-14, Tc-99, Pu-238, Pu-239/240, Am-241, Am-243 and Cm-243/244.

2.3.3.2. Auxiliary Building

The Auxiliary Building is located between the Unit 1 and Unit 2 Containments and the Turbine Building. The structure is designed to house support systems for the operation of both reactors. Major support systems that are located in the Auxiliary Building include Containment Spray, Residual Heat Removal, Reactor Water Cleanup, Reactor Water Charging and Safety Injection systems. In addition, the Auxiliary Building contains various filters and tanks designed to batch and/or process gaseous, liquid and solid wastes resulting from reactor operation. With the exceptions of the service water, primary de-ionized water, control air, fire protection, nitrogen gas and service air, all of the systems within the Auxiliary Building are radiologically contaminated internally. The structure itself is designed to contain and control leakage from these systems during normal operation as well as unusual events. System leakage and maintenance activities over the operating life of the facility have resulted in the radiological contamination of most of the interior surfaces of the Auxiliary Building. Most of the cubicles that contain major systems are posted as contaminated areas identifying removable radioactive material. Based on the building design basis, the operating history, as well as the present status

of areas that are controlled as contaminated, all internal survey units within the Auxiliary Building are considered to be Class 1 areas.

General area radioactive dose rates within the Auxiliary Building range from 1 mrem/hr to 60 mrem/hr with contact dose rates on specific components in excess of 1000 mrem/hr. Loose radioactive contamination ranges from <1,000 dpm/100 cm^2 to over 250 mrad/swipe. The general radiological conditions within the Auxiliary Building are presented per survey unit in Table 2-13.

The basic decommissioning end-state for the Auxiliary Building will consist of the foundation concrete walls below 588 foot elevation and the 542 foot elevation concrete floor. All systems and components contained within the Auxiliary Building will be disassembled and/or demolished, packaged and properly dispositioned as either a radioactive or non-radioactive waste. The list of penetrations and embedded piping to remain is provided in TSD 14-016.

During the time that initial characterization was performed, all radioactive systems and components were still located inside the Auxiliary Building. Consequently, ambient radiation dose rates inside most of the cubicles on the 542 foot elevation prohibited the direct assessment of concrete surfaces by scanning or direct measurement. Once commodity removal is complete, a contamination verification survey (CVS) will be performed to identify areas requiring remediation to meet the open air demolition limits. Prior to demolition, all structural surfaces that will remain after demolition will be remediated to levels that will ensure that an individual ISOCS measurement will not exceed the DCGLs from Chapter 5, Table 5-4 during FSS. Following completion of a FSS in accordance with section 5.5.4 of this LTP, the basements will be backfilled with clean fill, CCDD, grout, or a combination of the three materials..

In May and June of 2012, a characterization survey was performed of the Auxiliary Building 542 foot elevation and Auxiliary Building exterior walls. The characterization survey consisted of surface scans and the acquisition of a series of concrete core samples taken in the 542 foot elevation concrete floor and exterior lower walls. In March of 2013, two (2) additional concrete cores were taken in the Auxiliary Building elevator shaft and the Hold-Up tank Cubicle floors as these areas became accessible.

During the characterization of the Auxiliary Building basement, extensive beta gamma scan surveys were performed on the floors and lower walls of the 542 foot elevation in an effort to determine the locations representing the worst case radiological condition for concrete in each survey unit. These scans were performed of accessible walls surfaces to the extent practicable while standing on the 542 foot elevation, to a nominal elevation of approximately six feet up the wall from the floor. The scan surveys indicated that, for a majority of the lower wall surfaces on the Auxiliary Building 542 foot elevation, the residual radioactivity on the wall was indistinguishable from ambient background. This was particularly true for all the outer wall surfaces in the east portion of the Auxiliary Building 542 foot elevation, including the Waste Gas Decay Tank area, the Lake Discharge Tank area, the Blowdown Monitor Tank area and the areas adjacent to the Cavity Fill Pump cubicles. Residual contamination at concentrations greater than the ambient background was only detected on the outer walls of the Unit 1 and Unit 2 Pipe Chases, the Unit 1 and Unit 2 ABEDCT cubicles and the outer walls of the HUT cubicles. However, with the exception of the HUT cubicles, the contamination identified on the walls in the Pipe Chases and ABEDCT cubicles was not uniform. The contamination on the walls in

these cubicles was primarily from valve leakage and gland seal spray from primary system pumps.

A total of 20 concrete core samples were collected. The locations selected were biased toward locations where physical or observed radiological measurements indicated the presence of fixed and/or volumetric contamination of the concrete media. When possible, locations were determined based upon elevated observed contact dose rates or count rates from scans. In addition, visual observations of floor and wall surfaces were used to identify potential locations of surface contamination, such as discoloration or standing water. The goal was to identify to the extent possible, the locations that exhibited the highest potential of representing the worst case radiological condition for concrete in each survey unit. This judgmental sampling approach also ensured that there was sufficient source term in the cores to achieve the sensitivities required to determine the radionuclide distributions of gamma emitters as well as HTD radionuclides. The locations where the core samples were taken are illustrated in Figure 2-25. The concrete pucks were analyzed on the on-site gamma spectroscopy system. A summary of the on-site gamma spectroscopy results for the analysis of the concrete cores taken from the 542 foot elevation of the Auxiliary Building is presented in Table 2-22. Co-60, Cs-134 and Cs-137 were the only plant-derived gamma emitting radionuclides identified. Zion*Solutions* TSD 14-028 presents additional detail on the concrete sampling methodology and results of the radiological analysis of each concrete core sample obtained from the Auxiliary Building basement.

Analyses of the concrete core samples taken from the Auxiliary Building 542 foot elevation indicate that there is extensive radiological contamination at depth. This is most likely due to the fact that the 542 foot elevation was routinely flooded with contaminated water during operations. In the first ½-inch of floor, Co-60 concentrations averaged 46 pCi/g with a maximum concentration of 456 pCi/g and Cs-137 concentrations averaged 3,352 pCi/g with a maximum concentration of 25,100 pCi/g. In both Unit 1 and Unit 2 Pipe Tunnel rooms, Cs-137 concentrations of 530 pCi/g and 1,740 pCi/g were observed at depths of 4 and 5 inches respectively. In addition, sample analysis indicated a Cs-137 concentration of 56.80 pCi/g at a depth of 2 inches in the central common area, a Cs-137 concentration of 31.10 pCi/g at a depth of 3.5 inches in the east floor area and a Cs-137 concentration of 63.10 pCi/g at a depth of 2.5 inches in the Unit 1 Equipment Drain Collection Tank room.

The top ½-inch puck from six (6) of the 20 cores from the Auxiliary Building were sent to Eberline Laboratory for gamma spectroscopy and HTD analyses for radionuclides such as H-3, C-14, Tc-99, Ni-63, Sr-90, and alpha emitters. The results of the analysis are presented in Table 2-23. The mixture percentages for the initial suite of radionuclides for the Auxiliary Basement concrete were developed in TSD 14-019, *"Radionuclides of Concern for Soil and Basement Fill Model Source Terms"* (Reference 2-27) using the results of all core sample analyses, including the cores sent to Eberline as well as those analyzed onsite, and is presented in Table 2-24. Significant HTD radionuclides identified by the analysis of the concrete core samples include Ni-63 and H-3. The other radionuclides positively detected at concentrations greater than their respective MDC include; C-14, Tc-99, Sr-90, Ag-108m, Pu-238, Pu-239/240, Am-241 and Am-243.

2.3.3.3. Fuel Handling Building

The Fuel Handling Building is located between the Unit 1 and Unit 2 Containments and adjacent to the Auxiliary Building. The structure is designed for the storage of new and spent fuel. Major support systems that are located in the Fuel Handling Building include the SFP Heat Exchangers and SFP Skimmer Pumps. The SFP is a 63 ft. long by 33 ft. wide by 40 ft. deep pool located in the east half of the building. The pool is filled with borated water and contains storage racks for the storage of spent fuel assemblies. Spent nuclear fuel, highly irradiated reactor components and other highly radioactive debris were stored in the pool. A new fuel storage area and a fuel unloading area are located in the western portion of the building. A cask decontamination pit is located adjacent to the pool. With the exceptions of the service water, de-ionized water, control air, fire protection, nitrogen gas and service air, all of the systems within the Fuel Handling Building are radiologically contaminated internally. The SFP, the decontamination pit, and the equipment cubicles are all posted as "Contaminated Areas." The potential for residual contamination exists throughout the building. Based on the building design basis, the operating history, as well as the present status of areas that are controlled as contaminated, all internal survey units within the Fuel Handling Building are considered to be Class 1 areas.

General area radioactive dose rates within the Fuel Handling Building range from 1 mrem/hr to 25 mrem/hr. Loose radioactive contamination ranges from <1,000 dpm/100 cm^2 to 37,000 dpm/100 cm^2. The general radiological conditions within the Fuel Handling Building are presented per survey unit in Table 2-13.

The spent fuel located in the SFP was packaged into dry cask storage and transferred to the ISFSI facility. All systems, components and materials located in the Fuel Handling Building will be removed and dispositioned as radioactive or non-radioactive waste as appropriate. A majority of the Fuel Handling Building structure is located above the 588 foot elevation. Consequently, the basic decommissioning end-state for the Fuel Handling Building is the complete removal of the current accessible structure. The only portion of the building that resides below the 588 foot elevation is the bottom 12 feet of the SFP and adjoining Transfer Canal. As part of the building demolition, the steel liner will be removed from the SFP and Transfer Canal. Once the liner is removed and the underlying concrete is exposed, additional characterization surveys will be performed to assess the radiological condition of the underlying concrete pad and remaining pool walls. Following an assessment of the results of the survey, a cost benefit analysis will be performed to determine if the concrete will be remediated and abandoned in place or removed. Once commodity removal is complete, a contamination verification survey (CVS) will be performed on the concrete surfaces to identify areas requiring remediation to meet the open air demolition limits. Prior to demolition, all concrete surfaces that will remain will be remediated to levels that will ensure that an individual ISOCS measurement will not exceed the DCGLs from Chapter 5, Table 5-4 during FSS. Following completion of a FSS in accordance with section 5.5.4 of this LTP, any remaining remnant of the SFP and Transfer Canal below the 588 foot elevation will be backfilled with clean fill, CCDD, grout, or a combination of the three materials.

2.3.3.4. Turbine Building

The Turbine Building houses the steam turbines and generators for both reactor units as well as secondary steam systems, circulating water systems, lubrication and fuel oil systems and emergency diesel generators. The steam and support systems in the Turbine Building are designed to be operated as non-radioactive systems. However, from 1981 through the facility shutdown in 1998, both Zion units experienced significant primary system to secondary side leakage from leaking Steam Generator tubes. From 1981 to 1984, primary to secondary side leakage was recorded at levels exceeding 500 gallons per day. Consequently, primary to secondary side leakage has resulted in measurable radioactivity in portions of the secondary system piping, primarily in the high-pressure steam components, the condensate re-heaters, the Auxiliary Boiler and the system, equipment and floor drain systems. In addition, several areas/components adjacent to the fire sump have detectable radioactive contamination. In the late 1970's (prior to fire sump modifications), the Turbine Building Equipment Drain Analysis Tank had exposure rates up to 200 mR/hr on contact.

General area radioactive dose rates within the Turbine Building are <1 mrem/hr. Loose radioactive contamination ranges from <1,000 dpm/100 cm^2 to 18,000 dpm/100 cm^2 located in the Auxiliary Boiler room. The general radiological conditions within the Turbine Building are presented per survey unit in Table 2-13.

The Turbine Building systems and structural surfaces above the 588 foot elevation will be surveyed and released for unconditional use. The concept is to allow a contractor to disassemble, salvage and demolish the Turbine Building as "non-radiological". To allow this process to occur, the "high-risk" systems within the Turbine Building, such as Main Steam, Condensate, Auxiliary Steam, Feedwater and liquid waste systems will be removed and properly dispositioned as radioactive waste. The remaining systems, as well as the Turbine Building structure will be remediated as necessary and surveyed for unconditional use. The drain system embedded in the concrete floor of the 560 foot elevation, which is currently radiologically contaminated, will be surveyed in accordance with Chapter 5, section 5.5.5. The results for the characterization of the Turbine Building floor drains is presented in section 2.3.3.7.

Once the Turbine Building has been verified as meeting the unconditional release criteria, then the building will be gutted and demolished to the 588 foot elevation. Prior to demolition, all concrete surfaces that will remain will be remediated to levels that will ensure that an individual ISOCS measurement will not exceed the DCGLs from Chapter 5, Table 5-4 during FSS. Following completion of a FSS in accordance with section 5.5.4 of this LTP, the basement void below the 588 foot elevation will be backfilled with clean fill, CCDD, grout, or a combination of the three materials.

In November of 2012, a series of concrete core samples were taken in the 560 foot elevation Turbine Building concrete floor as well as the 570 foot elevation Steam Tunnel concrete floors. The locations where the core samples were taken are illustrated in Figures 2-26.

A total of 10 concrete core samples were collected, three (3) in the Turbine Building 560 foot elevation floor, five (5) in the Unit 1 Steam Tunnel floor and two (2) in the Unit 2 Steam Tunnel floor. The locations selected were biased toward locations where physical or observed radiological measurements indicated the presence of fixed and/or volumetric contamination of

the concrete media. When possible, locations were determined based upon elevated observed contact dose rates or count rates. In addition, visual observations of floor and wall surfaces were used to identify potential locations of surface contamination, such as discoloration or standing water. The goal was to identify to the extent possible, the locations that exhibited the highest potential of representing the worst case radiological condition for concrete in each survey unit.

A summary of the gamma spectroscopy results for the concrete cores obtained from the Turbine Building 560 foot elevation and the 570 foot elevation Steam Tunnels are provided in Table 2-25. Zion*Solutions* TSD 14-028 presents additional detail on the concrete sampling methodology and results of the radiological analysis of each concrete core sample obtained from the Turbine Building. Cs-137 was the only plant-derived gamma emitting radionuclides identified. Concentrations for Co-60 were less than the MDC for all samples from the Turbine Building and the Steam Tunnels.

Analyses of the concrete core samples taken from the Turbine Building 560 foot elevation show the presence of Cs-137 at concentrations greater than the MDC of the instrument at two (2) of the three (3) sample locations, and only in the 1st ½-inch of concrete. Observed Cs-137 concentrations ranged from 0.55 pCi/g to 46.7 pCi/g. At depths greater than ½-inch, concentrations for Cs-137 was less than MDC. In the Steam Tunnels, Cs-137 concentrations in the 1st ½-inch on concrete ranged from 6.60 pCi/g to 46.70 pCi/g in Unit 1 and 0.29 pCi/g to 18.60 pCi/g in Unit 2. At depths greater than ½-inch, concentrations for Cs-137 was less than the MDC of the instrument used.

Between March 21, 2013 and March 27, 2013, all accessible surfaces of the 560 foot elevation floor in the Turbine Building were scanned using a Ludlum Model 43-37 floor monitor. The average background of the instrument was 900 cpm. The alarm set-point was set at the observed background plus the Minimum Detectable Count Rate (MDCR) for the instrument. The mean observed count rate was 1,493 cpm with a maximum observed count rate of 3,922 cpm. Three (3) instrument alarms were observed, primarily around a posted radiological area adjacent to the elevator.

2.3.3.5. <u>Service Water Intake and Discharge Structure</u>

The Service Water intake and discharge structure is composed of several different components, including the Crib House, which houses the Circulating Water and Fire Pump motors and impeller housings, the Forebay, which acts as the intake pool for the pumps, the Circulating Water Intake Headers, which funnels the Circulating Water Pump discharge flow to the Main Condenser and the Circulating Water Discharge Tunnels, which directs the heated circulating water back to Lake Michigan. The Circulating Water Discharge Tunnels were also the main authorized effluent release pathway for the discharge of treated and filtered radioactive liquid waste to Lake Michigan. During plant operations and following shut-down, the liquid effluent release pathway was monitored and the results presented in the annual REMP report in accordance with the Off-site Dose Calculation Manual (ODCM).

As previously described in section 2.3.1.2, a scan survey and a total of 40 concrete core samples were taken from the floors and lower walls of the Crib House 594 foot and 559 foot elevations. No activity greater than background was observed by either scan or the isotopic analysis of the concrete samples. The Crib House above the 588 foot elevation will be surveyed for

unconditional release at the same time as the Turbine Building. Once it has been demonstrated that the Crib House is suitable for unconditional release, a "non-radiological" demolition contractor will salvage materials out of the Crib House. The Crib House structural concrete and cinder block will be surveyed to demonstrate that the material is free of plant-derived radionuclides at concentrations greater than background, demolished and used as CCDD.

At the time of LTP submittal, the Forebay is completely underwater. The Forebay walls will be demolished to elevation 588 foot. The Circulating Water Intake piping that runs from the Forebay to the Main Condenser will be isolated and drained. When accessible, the remaining lower portions of the Crib House and the Forebay structure below the 588 foot elevation and the Circulating Water Intake piping will be remediated to levels that will ensure that an individual ISOCS measurement will not exceed the DCGLs from Chapter 5, Table 5-4 during FSS. Following completion of FSS, the Forebay will be backfilled with clean fill, clean concrete debris, grout, or a combination of the three materials. The Circulating Water Intake piping will be filed with grout.

The Circulating Water Discharge Tunnels run under the Forebay and discharges to Lake Michigan approximately 870 feet from the lakeshore. The current decommissioning approach is to leave the Circulating Water Discharge Tunnels in place. The interior surfaces of the Circulating Water Discharge Tunnels will be remediated to levels that will ensure that an individual ISOCS measurement will not exceed the Turbine Building DCGLs from Chapter 5, Table 5-4 during FSS.

2.3.3.6. Support Buildings and Miscellaneous Structures

The major support buildings and miscellaneous structures include the Service Building, Gate House, Waste Water Treatment Facility, IRSF, ENC Building, NGET Building, Unit 1 and Unit 2 Valve Houses, and several warehouses. The location of these buildings and structures are illustrated on Figures 2-2 and 2-3. The general radiological conditions within these buildings are presented per survey unit in Table 2-13.

As previously stated, the Asset Sale Agreement requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least 3 feet below grade. All support buildings and miscellaneous structures are categorized as structures located above the 588 foot elevation. Consequently, all support buildings and miscellaneous structures within the "security-restricted" and "radiologically-restricted" areas at ZNPS will be demolished and dispositioned as a radioactive or non-radioactive waste stream. Several minor structures such as the Switchyard as well as all roadways and rail lines, will remain at license termination as requested by Exelon. The switchyard was characterized as a Class 3 open land survey unit. The result of the switchyard characterization is presented in section 2.3.5.

2.3.3.7. Embedded or Buried Pipe

The End State will also include a range of buried piping, embedded piping and penetrations. Buried piping is defined as pipe that runs through soil. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a

concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of buried piping, penetrations and embedded piping to remain is provided in Zion*Solutions* TSD 14-016.

At the time of LTP submittal, the interior surfaces of most of these sections of pipe were not accessible. As decommissioning progresses and access is achieved, radiological surveys will be performed to assess if any remediation is necessary, to confirm the radiological distribution inside of the pipe and to assess the dose from residual radioactivity remaining in the pipes.

For embedded pipe and penetrations, the pipe interiors will be remediated to levels less than the embedded pipe and penetration specific DCGLs discussed in sections 5.2.4 and 5.2.5 of this LTP. As decommissioning progresses and access is achieved to the interior of these pipe sections, FSS will be performed in accordance with section 5.5.6 of this LTP.

For pipe buried in soil, the pipe interiors will be remediated to levels less than the site-specific DCGLs presented in Table 5-10 of this LTP. FSS surveys will be performed as described in section 5.7.1.8.

In April of 2013, several sediment samples were collected from the floor drain system embedded in the Turbine Building 560 foot elevation. Three (3) sediment samples were collected from the floor drains in the south portion, six (6) sediment samples were collected from the floor drains in the central portion, and seven (7) sediment samples were collected from the floor drains in the north portion of the Turbine Building 560 foot elevation. In addition, two (2) large area wipes were collected and analyzed from the drain pipe interiors of floor drains in the central and north sections. The locations where the sediment samples were taken are illustrated in Figure 2-27. All sediment and wipe samples were weighed and analyzed using the on-site gamma spectroscopy system. The gamma spectroscopy results for the sediment samples from the Turbine Building 560 foot elevation drain system piping are provided in Table 2-26.

Analysis of the sediment samples indicates the presence of Cs-137 in concentrations greater than the instrument MDC in 12 of the 16 sediment samples, ranging from 0.11 pCi/g to 17.7 pCi/g. Co-60 was present in three (3) of the 16 samples, ranging from 0.28 pCi/g to 0.44 pCi/g. The large area wipe taken in drain pipes in the central portion of the 560 foot elevation indicated Cs-137 at a concentration of 77.9 pCi/g. The analysis of the large area wipe taken in the north section did not show Cs-137 or Co-60 at concentrations greater than the instrument MDC.

2.3.4. Non-Impacted Open Land Areas

Based upon the information compiled in the Zion HSA, several large outlying open land survey units received an initial classification as "non-impacted." Non-impacted areas have no reasonable potential for residual contamination because historical information indicates there was no known impact from site operations. These include the outlying open land areas of the site as well as contiguous areas that have no impact from site operations based upon the location(s) of licensed operations, site use, topography, site discharge locations, and other site physical characteristics. These areas are not required to be surveyed for demonstrating compliance beyond any characterization surveys performed to provide a basis for the classification.

For ZNPS, the non-impacted open land areas includes most of the surrounding Exelon owned land outside of the footprint of the 87 acre, fence-enclosed "Radiologically-Restricted Area" as well as tracts of land that are owned by the Town of Zion and adjacent businesses. With the exception of operational surveys performed at the perimeter of the RCA to establish compliance with non-occupational exposure limits, no radiological survey data was available to support this basis.

From June to September 2013, characterization surveys were performed in the non-impacted open land areas of the site. To facilitate data collection, review and evaluation, the non-impacted open land areas were surveyed as designated survey units. A sample plan was prepared for each survey unit in accordance with procedure ZS-LT-100-001-001, "*Characterization Survey Package Development*" (Reference 2-28). The objective of the survey was to perform a sufficient radiological characterization to establish the empirical basis for the "non-impacted" classification and to establish a reasonable assurance that the survey units in question were free of detectable radioactive material resulting from the operation of the reactors.

Within each of the survey units specified, the survey focused primarily on surface (0 to 15 cm) soils. Subsurface (15 to 100 cm) soil samples were included in the survey design only if the analysis of surface soil samples indicated the presence of detectable plant-derived radioactivity. The sample and static measurement locations were based on a random design to ensure an unbiased survey.

The characterization survey of each survey unit consisted of both qualitative evaluations and quantitative analysis results. The qualitative evaluation consisted of static measurements using the Canberra *In Situ* Object Counting System (ISOCS). Investigative and verification gamma scans using a Ludlum Model 2350-1 and a Model 44-10 NaI detector were also performed. MDC and gamma scanning sensitivities were estimated based on the assumed geometry and the potential plant-derived gamma-emitting radionuclides that may be present. Quantitative analysis results were obtained from radionuclide specific analysis of surface soil media using a calibrated counting geometry. Analysis times were set to achieve the required MDCs that were based on the expected Cs-137 background due to global fallout as set forth in TSD 13-004 and reproduced in Table 2-11. The results of the ISOCS and soil sample measurements were compared to the appropriate background value commensurate with the soil type. Any measurements exceeding these values would be indicative of the presence of plant-derived radioactivity.

The minimum number of random-based direct measurements was adjusted to approximate one (1) measurement location for every 2,000 m^2 of land area. This survey frequency was in accordance with the Characterization Survey Plan and procedure ZS-LT-100-001-001. Surface scanning using a NaI detector was performed in the vicinity of each static measurement location and for investigations of elevated results. Surface soil samples were taken at 25% of the measurement locations.

Specific ISOCS measurement locations were determined by generating random pairs of coordinates that corresponded to specific locations within each survey unit. If a randomly selected location was found to be either inaccessible or unsuitable (e.g. a portion of the surface area in the instruments FOV was covered in standing water), then the location was adjusted to the closest adjacent suitable location.

The ISOCS detector was oriented downward and perpendicular to the ground. The exposed face of the detector was positioned at a height of 3 meters above the ground. With the 90-degree collimation shield installed, this orientation corresponded to a nominal FOV of 28 m^2. The ISOCS geometry evaluated soil activity to a depth of 15 cm (6 inches below grade) over the geometric FOV. Measurement count times were adjusted to achieve a MDC of 0.40 pCi/g for Cs-137.

The action level was set at any detectable activity greater than the actual measurement MDC for any potential plant-derived gamma emitting radionuclide or the Cs-137 background values. If any measurement would have identified a plant-derived gamma emitting radionuclide that was not found in background or, if the observed activity exceeded the expected background activity concentration, then an investigation would have been performed.

2.3.4.1. Surface Soils

The owner-controlled non-impacted open land areas at Zion totaled 874,041 square meters of surface area. The non-impacted surface area was broken into 11 survey units in accordance with the area descriptions, sizes and boundaries presented in the HSA. The non-impacted survey units are illustrated by Figure 2-4. Of the 874,041 square meters, 182,127 square meters was deemed as "inaccessible". In this context, "inaccessible" is defined as an area where personnel or vehicle transit was inhibited by the presence of standing water, marsh or wet-lands, thick underbrush, trees or natural grasses where clearing would be prohibitive. The total surface area deemed "accessible" was 691,913 square meters.

Of the 691,913 square meters of surface area, 9,378 square meters were scanned by a Model 2350 paired with a Model 44-10 NaI detector. Alarm set-points for the instrument were set at the observed background plus the MDCR of the instrument. With an average background of 4,337 cpm, the average observed scan result was 4,677 cpm. Twenty-four (24) instrument alarms were logged with a maximum observed scan reading of 9,550 cpm. All alarms were investigated and soil samples were taken at locations where the elevated reading was verified.

Two hundred and thirty-six (236) static measurements were taken with the ISOCS. This equated to a coverage area of 6,608 square meters using a 28 square meter FOV. Of the total measurements taken, 75 measurement results indicated the presence of Cs-137 in concentrations greater that the MDC of the instrument. No other potential plant-derived radionuclides were positively identified. Cs-137 concentrations identified by ISOCS measurements averaged 0.17 pCi/g with a maximum observed Cs-137 concentration of 0.34 pCi/g.

One hundred and sixty-six (166) surface soil sample were taken to verify ISOCS results or as investigations. Of the total number of surface soil samples taken and analyzed, Cs-137 was identified at concentrations greater than the MDC of the instrument in 106 surface soil samples. No other potential plant-derived radionuclides were positively identified. The average Cs-137 concentration observed in the analysis of the surface soil samples was 0.24 pCi/g with a maximum observed concentration of 0.57 pCi/g.

Additional detail on the survey and sampling methodology and results of the radiological analysis of each measurement and soil sample obtained during the characterization of non-impacted open land survey units are presented in Zion*Solutions* TSD 14-028. The scope and

extent of the surveys performed in the non-impacted classified open land areas were sufficient to demonstrate that the non-impacted classification was appropriate for the open land areas in question and that it is highly unlikely, based upon the findings and conclusions of the HSA, combined with the results of the characterization, that plant-derived radioactivity resides in these areas. Based upon the results of the characterization surveys performed of the non-impacted open land areas, it can be concluded that a non-impacted classification for these areas are appropriate. Cs-137 was the only radionuclide positively identified that could potentially be classified as plant-derived. However, the concentrations observed are well within the range of activity defined as background due to global fallout illustrated in Table 2-11. The locations of ISOCS measurements and surface soil samples are illustrated on Figure 2-28. A summary of the findings of the survey for each individual survey unit are presented in Table 2-27.

2.3.4.2. Subsurface Soils

The survey design for the characterization of the non-impacted open land survey units required the acquisition of subsurface soil samples only if an investigation was required prompted by the discovery of plant-derived radionuclides in concentrations greater than background in a surface soil sample or an ISOCS measurement. In addition, a subsurface soil sample would be taken if physical indications would suggest the potential presence of a burial area. Based upon the results of the characterization survey, no investigation was prompted by the discovery of plant-derived radionuclides in concentrations greater than background. In addition, no suspected burial areas were identified during the walkdowns of the survey units. Consequently, no subsurface soil samples were acquired in the non-impacted survey units.

2.3.5. Impacted Open Land Areas

The Zion HSA identified the open land area inside the single fenced "Radiologically-Restricted Area" as "impacted" by reactor operations. The approximate area of the footprint is 87 acres. In addition, a majority of the open land area inside the double-fenced "Security-Restricted Area" was primarily classified as Class 1.

Characterization of the impacted open land areas began in November 2011 with the characterization of the Class 3 open land survey units encompassing the proposed site for the future ISFSI facility, the "non-impacted" location where the Vertical Concrete Cask Construction Area was to be located and the pathway for the new rail tracks. The remaining balance of the "impacted" open land areas were surveyed between November 2012 and October 2013.

Initial survey units were established by the HSA. A sample plan was prepared for each survey unit in accordance with procedure ZS-LT-100-001-001. Survey techniques were employed to determine the lateral and vertical extent of any contamination identified and the radionuclide concentrations in the soil.

A reference grid was established in each impacted survey unit. The reference grid consisted of a grid coordinate system that can be accessed using a Global Positioning System (GPS) system. Survey units were scanned in accordance with their initial classification or the anticipated classification based on historical knowledge. Recommended scan coverage guidelines for characterization for each class of survey unit is presented in Table 2-5.

A combination of random and judgmental survey locations were established in each survey unit in accordance with the survey unit classification. The size of the judgmental sample population was determined using the recommended judgmental sample population sizes stated in procedure ZS-LT-100-001-001 and presented in Table 2-28. The locations of judgmental measurements and/or samples were determined by the professional judgment of the responsible Radiological Engineer. Consideration was given to locations that exhibit measurable radioactivity, depressions, discolored areas, low point gravity drain areas, actual and potential spill locations, or areas where the ground had been disturbed. The number of random-based survey locations was also determined by area classification. Random-based measurements were not required in an open land areas designated as Class 1. In a Class 2 survey unit, a minimum of 15 random-based locations were recommended. In Class 3 open land survey units, the minimum number of random-based survey locations were adjusted to approximate one (1) measurement location for every 2,000 m^2 of land area.

A surface soil sample was taken at each selected judgmental and random-based survey location. Sub-surface soil samples were acquired at a minimum of 10% of the surface soil sample locations identified. Additional subsurface soil samples were acquired if surface soil sample analysis or surface scans indicated elevated activity. Depending on the location, subsurface soil samples were taken from 0.15 cm to depths ranging from 1 to 3 m below grade and composited over 1 m intervals.

Each surface soil and composite subsurface soil sample was analyzed by the on-site radiological laboratory by gamma-spectroscopy for plant-derived gamma emitting radionuclides. Analysis count times were adjusted as necessary to achieve an isotopic MDC equal to or less than 0.18 pCi/gm for Cs-137 and Co-60. In a majority of the survey units, the surface and subsurface composite samples were initially analyzed without drying the sample media. If the initial on-site gamma spectroscopy analysis indicated that no plant-derived radionuclides were detected at a concentration greater than their respective MDC, then the sample was dried and reanalyzed, again using the on-site gamma spectroscopy system. If the initial on-site gamma spectroscopy analysis indicated the presence of plant-derived radionuclides greater than their respective MDC, then a portion of the sample was segregated into a separate container for further analysis and the remainder of the sample was dried and reanalyzed, again using the on-site gamma spectroscopy system. Several soil samples that exhibited concentrations of plant-derived gamma emitting radionuclides greater than MDC were shipped to an approved off-site vendor for full isotopic analysis (HTD radionuclides).

2.3.5.1. Surface Soils and Paved Areas

The "Radiologically-Restricted Area" is illustrated in Figure 2-2. In this area, there are 22 Class 3 open land survey units. The survey unit boundaries are illustrated on Figure 2-6. Seven (7) Class 3 open land survey units reside outside of the "Radiologically-Restricted Area". These areas include the "West Training Area" along Shiloh Boulevard, the beach outside of the fenced area along the lakefront (broken into three (3) Class 3 survey units), survey unit 10220C along the south boundary and two (2) areas initially classified as non-impacted in the HSA but reclassified as Class 3 due to current and expected decommissioning activities. These areas include the VCC area and the site parking lot. The Class 3 survey units situated outside of the "Radiologically-Restricted Area" are illustrated in Figure 2-5.

The Class 3 open land survey units at Zion have a total surface area of 401,042 square meters. 49,116 square meters of the total Class 3 area was deemed as "inaccessible" due to obstacles. The total surface area deemed "accessible" was 351,926 square meters. Of this area, 104,350 square meters or approximately 30% of the surface area were scanned using a Model 2350 paired with a Model 44-10 NaI detector. This area included asphalt paved surfaces. Alarm set-points for the instrument were set at the observed background plus the MDCR of the instrument. With an average background of 3,713 cpm, the average observed scan result was 3,753 cpm. Three hundred-seventeen (317) instrument alarms were logged with a maximum observed scan reading of 12,193 cpm. All alarms were investigated and soil samples were taken at locations where an elevated reading was verified.

In the "West Training Area", forty-six (46) static measurements were taken with the ISOCS. This equated to a coverage area of 1,288 square meters using a 28 square meter FOV. Of the total measurements taken, seven (7) measurement results indicated the presence of Cs-137 in concentrations greater that the MDC of the instrument. No other potential plant-derived radionuclides were positively identified. Cs-137 concentrations identified by ISOCS measurements averaged 0.18 pCi/g with a maximum observed Cs-137 concentration of 0.25 pCi/g.

Seven hundred-eighteen (718) surface soil sample were taken at judgmental, random-based and investigation locations. The locations of random and judgmental surface soil samples taken in Class 3 open land survey units located outside of the "Radiologically-Restricted Area" are illustrated on Figure 2-29. The locations of random and judgmental surface soil samples taken in Class 3 open land survey units located inside of the "Radiologically-Restricted Area" are illustrated on Figure 2-30. Of the total number of surface soil samples taken and analyzed, Cs-137 was identified at concentrations greater than the MDC of the instrument in 142 surface soil samples and Co-60 was identified at concentrations greater than the MDC of the instrument in one (1) sample. No other potential plant-derived radionuclides were positively identified. The average Cs-137 concentration observed in the analysis of the surface soil samples was 0.12 pCi/g with a maximum observed concentration of 1.14 pCi/g. The one sample where Co-60 was positively identified had a Co-60 concentration of 0.13 pCi/g.

Additional detail on the survey and sampling methodology and results of the radiological analysis of each measurement and soil sample obtained during the characterization of Class 3 open land survey units are presented in Zion*Solutions* TSD 14-028. Based upon the results of the characterization surveys performed of the Class 3 open land areas, strong evidence is provided to conclude that minimal residual plant-derived radioactivity is present in these areas. A summary of the findings of the survey for each individual Class 3 open land survey unit are presented in Table 2-29.

The "Security-Restricted Area" is illustrated in Figure 2-3. In this area, there are 13 Class 1 open land survey units and five (5) Class 2 open land survey units. The survey unit boundaries are illustrated on Figures 2-7. The area of each survey units complies with the recommended survey unit sizes for open land survey units specified in MARSSIM, section 4.6. Additional detail on the survey and sampling methodology and results of the radiological analysis of each measurement and soil sample obtained during the characterization of Class 2 and Class 1 open land survey units are presented in Zion*Solutions* TSD 14-028.

The Class 2 open land survey units at Zion total 39,747 square meters of surface area. 9,085 square meters of the total Class 2 area was deemed as "inaccessible", mostly due to the presence of buildings (Turbine Building, Crib House, Service Building and Gate House). The total surface area deemed "accessible" was 30,622 square meters. Of this area, 22,559 square meters or approximately 74% of the surface area were scanned by a Model 2350 paired with a Model 44-10 NaI detector. This area included asphalt paved surfaces. Alarm set-points for the instrument were set at the observed background plus the MDCR of the instrument. With an average background of 3,005 cpm, the average observed scan result was 3,012 cpm. One-hundred-eighteen (118) instrument alarms were logged with a maximum observed scan reading of 11,264 cpm. All alarms were investigated and soil samples were taken at locations where elevated readings were verified.

Fifty-nine (59) surface soil samples were taken at judgmental, random-based and investigation locations in Class 2 open land survey units. The locations of random and judgmental surface soil samples are illustrated on Figure 2-31. Of the total number of surface soil samples taken and analyzed, Cs-137 was identified at concentrations greater than the MDC of the instrument in 12 surface soil samples. No other potential plant-derived radionuclides were positively identified. The average Cs-137 concentration observed in the analysis of the surface soil samples was 0.14 pCi/g with a maximum observed concentration of 0.21 pCi/g. A summary of the findings of the survey for each individual Class 2 open land survey unit are presented in Table 2-30.

There are thirteen (13) open land survey units classified as Class 1 at Zion totaling 24,759 square meters of surface area. Of the 13 Class 1 open land survey units, the surface soils in four (4) survey units, totaling 7,740 square meters of surface area were deemed as "inaccessible". A majority of the inaccessible surface area is obstructed by the presence of buildings, namely the two Containment Buildings, the Fuel Handling Building and the Auxiliary Building. However, due to the fact that all the asphalt and concrete ground coverings inside of the "Security-Restricted Area" will be disposed of as waste, surface soil was also deemed inaccessible if the area was paved. In addition, stored radioactive material in the vicinity of some areas prevented the ability to perform surface scanning due to high background. These surfaces were deemed as "inaccessible" as well.

The Class 1 surface area deemed "accessible" was 17,019 square meters. Of this area, 9,444 square meters or approximately 55% of the surface area was scanned by a Model 2350-1 paired with a Model 44-10 NaI detector. Alarm set-points for the instrument were set at the observed background plus the MDCR of the instrument. With an average background of 2,690 cpm, the average observed scan result was 3,361 cpm. Five-hundred forty-six (546) instrument alarms were logged with a maximum observed scan reading of 35,962 cpm. All alarms were investigated and soil samples were taken at locations where an elevated reading was verified.

Ninety-four (94) surface soil samples were taken at locations biased toward observed elevated scan measurements and known spill locations. The locations of surface soil samples taken in Class 1 open land survey units are illustrated in Figure 2-32. Of the 111 surface soil samples taken, Co-60 was positively identified in concentrations greater than the instrument MDC in 14 samples and Cs-137 was positively identified in concentrations greater than the instrument MDC

in 58 samples. No other potential plant-derived gamma-emitting radionuclides were positively identified. The average Co-60 concentration observed in the analysis of the surface soil samples was 0.13 pCi/g with a maximum observed concentration of 1.04 pCi/g. The average Cs-137 concentration observed in the analysis of the surface soil samples was 0.12 pCi/g with a maximum observed concentration of 3.39 pCi/g. A summary of the findings of the survey for each individual Class 1 open land survey unit are presented in Table 2-31.

Nine (9) surface soil samples taken from Class 1 open land survey units were sent to Eberline Laboratory for gamma spectroscopy and HTD analyses for radionuclides such as H-3, C-14, Tc-99, Ni-63, Sr-90, and alpha emitters. The results of the analysis are presented in Table 2-32. Co-60 and Cs-137 were the only potential plant-derived radionuclide identified in a concentration greater than the analysis MDC. No other plant-derived radionuclides were positively identified by these analyses.

2.3.5.2. Subsurface Soils

The HSA was consulted to identify those survey areas where the potential existed for subsurface radioactivity. Such areas include, but are not limited to, areas under buildings, building floors/foundations, or outside components where leakage was known or suspected to have occurred in the past and on-site storage areas where radioactive materials have been identified. Soil data from both the HSA and any pertinent surface characterization data were used to establish locations and potential depth for any potential sub-surface radioactivity.

A total of 723 composited subsurface soil samples were collected and analyzed during the site characterization. Inside the "Security-Restricted Area", subsurface soil samples were typically taken to a depth of approximately 3 m below grade. In the Class 3 survey units, subsurface soil samples were typically taken to a depth of approximately 1 m below grade.

Two-hundred eighty-three (283) subsurface soil samples were taken in Class 1 open land survey units. Cs-137 was positively identified in concentrations greater than the instrument MDC in 14 samples and Co-60 was positively identified in concentrations greater than the instrument MDC in one (1) sample. No other potential plant-derived gamma-emitting radionuclides were positively identified. The average Cs-137 concentration observed in the analysis of the surface soil samples was 0.18 pCi/g with a maximum observed concentration of 0.70 pCi/g. The one sample where Co-60 was positively identified had a Co-60 concentration of 0.10 pCi/g. A summary of the results of the subsurface soil sample analysis by Class 1 open land survey unit are presented in Table 2-31.

One (1) subsurface soil sample taken from a Class 1 open land survey unit was sent to Eberline Laboratory for gamma spectroscopy and HTD analyses for radionuclides such as H-3, C-14, Tc-99, Ni-63, Sr-90, and alpha emitters. The results of the analysis are presented in Table 2-32. Cs-137 was the only potential plant-derived radionuclide identified in a concentration greater than the analysis MDC. No other plant-derived radionuclides were positively identified by these analyses.

Sixty (60) subsurface soil samples were taken in Class 2 open land survey units. Cs-137 was positively identified in concentrations greater than the instrument MDC in three (3) samples. No other potential plant-derived gamma-emitting radionuclides were positively identified. The

average Cs-137 concentration observed in the analysis of the surface soil samples was 0.11 pCi/g with a maximum observed concentration of 0.15 pCi/g. A summary of the results of the subsurface soil sample analysis by Class 2 open land survey unit are presented in Table 2-32.

Three-hundred eighty (380) subsurface soil samples were taken in Class 3 open land survey units. Cs-137 was positively identified in concentrations greater than the instrument MDC in 32 samples. No other potential plant-derived gamma-emitting radionuclides were positively identified. The average Cs-137 concentration observed in the analysis of the surface soil samples was 0.09 pCi/g with a maximum observed concentration of 0.15 pCi/g. A summary of the results of the subsurface soil sample analysis by Class 2 open land survey unit are presented in Table 2-30.

In October of 2013, a single soil sample was acquired from the soil under the Turbine Building 560 foot floor slab. The basis for the selection of the location assumed groundwater flow from west toward Lake Michigan to the east. The Turbine Building is located between Lake Michigan and four potential sources of radiological contamination (the two Containment Buildings, the Auxiliary Building and the Fuel Handling Building). Other considerations for the selection of the sample location included the accessibility for the GeoProbe equipment and the thickness of the concrete floor slab that must be drilled through in order to access the underlying soils.

A hole was drilled through the concrete floor slab, exposing the underlying soil. The soil consistency found was hard-packed clay. A single sample of the soil was acquired, representing a depth of approximately 38 feet below grade. The sample was analyzed on the on-site gamma spectroscopy system. No plant-derived radionuclides at concentrations greater than the instrument MDC were detected.

The assessment of potential subsurface soil contamination is not currently complete. Soil in difficult to access areas such as under building foundations and surrounding buried structures will be deferred until later in the decommissioning process, when access will be more readily available.

2.3.6. Surface and Groundwater

Section 8.5 in Chapter 8 of this LTP contains a summary description of the geology, hydrogeology and hydrology of ZNPS and environs. The information contained in this section was derived directly from Zion*Solutions* TSD 14-003, Conestoga Rovers & Associates (CRA) Report, "*Zion Hydrogeologic Investigation Report*" (Reference 2-29) and presents a summary of studies that have been performed to investigate radiological groundwater contamination resulting from the operation of ZNPS. TSD 14-003, documents the results of hydrogeologic investigations completed by Conestoga-Rovers & Associates (CRA), Exelon, and Zion*Solutions* from August 2006 through September 2013.

2.3.6.1. Area Groundwater Use

ZNPS is connected to the Zion municipal water supply and does not use groundwater in its operations. The City of Zion provides municipal water to City residents and the surrounding area. The City purchases water from the Lake County Public Water District (LCPWD). The LCPWD obtains its water from Lake Michigan by means of an intake pipe located approximately

1 mile north of the Site and extending 3,000 feet into the Lake. The City of Zion municipal code requires all improved properties to be connected to the City's water supply. The code states that it is "unlawful for any person to construct, permit or maintain a private well or water supply system within the City which uses groundwater as a potable water supply". There is an exception for some existing wells constructed prior to March 2, 2004. However, it is unlikely that any private well or water supply system exists within the City which uses groundwater as a potable water supply.

2.3.6.2. Groundwater Flow

The shallow groundwater in the upper sand unit flows to the east toward Lake Michigan. The sheet pile wall installed along the lakeshore and the building foundations restrict the groundwater flow in the upper sand unit, which causes the groundwater to flow around ZNPS. The shallow water table intercepts the stormwater drainage ditches in the west area of the ZNPS property, but does appear to affect the flow of groundwater to the east and toward Lake Michigan. If the buildings and sheet pile wall are left intact (not perforated for flow) during the decommissioning, then a localized stagnation of groundwater around these barriers will occur since groundwater is prevented from flowing through these structures toward the Lake.

2.3.6.3. Previous Investigations

In 2006, CRA prepared a report titled *"Hydrogeologic Investigation Report, Fleetwide Assessment, Zion Station"*, Revision 1 (Reference 2-30) to determine whether groundwater at and in the vicinity of ZNPS has been adversely impacted by any releases of radionuclides. With the exception of H-3 and total strontium, no radionuclides were detected above the MDC during the 2006 investigation.

In 2011, the Wisconsin Department of Health Services (WDHS) conducted environmental radiation monitoring around the ZNPS site. This investigation was documented in a report titled, *"Zion Environmental Radioactivity Survey-2001"* (Reference 2-31). In this report, the WDHS concluded:

- Air particulate analysis shows no evidence of influence by ZNPS on air quality.
- The average yearly exposure of ambient gamma radiation is at background levels and is comparable to other areas within Wisconsin.
- The surface water samples showed no unusual concentrations of gross beta, gross gamma, tritium, and strontium.
- The gamma isotopic analysis for surface water indicated radioisotopes below their respective minimum detectable concentration.
- The gamma isotopic analysis on vegetation detected only a small amount of the naturally occurring elements K-40 and Be-7.
- The gamma isotopic analysis for soil detected naturally occurring K-40 and Cs-137, which is attributable to fallout from previous atmospheric nuclear tests (these were also detected in previous years).

- Doses of radiation as a result of gaseous and liquid effluent are less than the limits allowed for an average individual as stated in Federal Regulations.

2.3.6.4. On-Going Investigations

Ongoing monitoring of surface water and groundwater at ZNPS include REMP, Radiological Groundwater Protection Program (RGPP) and NPDES Monitoring.

2.3.6.4.1. Radiological Environmental Monitoring Program (REMP)

The REMP at ZNPS was initiated in 1973. The REMP includes the collection of various media samples including air, surface water, groundwater, fish, sediment, vegetation, clams, and crabs. The samples are analyzed for beta and gamma emitters, H-3, Sr-90 and other radiological constituents. The samples were collected at established locations, identified as stations, so that trends in the data could be monitored. The data for the REMP is collected quarterly and reported annually. The data has consistently supported a conclusion that the operation of ZNPS has and continues to have no adverse environmental impact on the environment.

2.3.6.4.2. Radiological Groundwater Protection Program (RGPP)

Since the fall of 2006, routine monitoring has been completed through the sampling and analysis of groundwater from 11 permanent monitoring wells and surface water from one established sampling location. The location of groundwater monitoring wells and the surface water sample location is presented in Figure 2-33.

2.3.6.4.3. National Pollutant Discharge Elimination System (NPDES)

ZNPS has an NPDES permit that was issued by the Illinois EPA. The NPDES permit covers discharge limitations, monitoring and reporting requirements for station discharges to Lake Michigan through two approved outfalls (#001 and #002). In addition, ZNPS has a General NPDES Permit for Storm Water Discharges from Construction Site Activities. The locations of the outfalls are presented in Figure 2-33.

2.3.6.5. Summary of Analytical Results in Groundwater

H-3 was detected in monitoring well MW-ZN-01S (located just south of the WWTF) 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. It was noted that following the initial sampling in June of 2006 the tritium analysis method was changed to include distillation of the samples for tritium. It should also be noted that even though 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 (located south of the Service Building parking lot) and MW-ZN-06S (located near the access road west of the Switchyard and up-gradient of ZNPS). 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 are very near the detection limit. These concentrations were less than the EPA drinking water standard of 8 pCi/L. Both of these locations are up-gradient from the groundwater flow direction and should not be impacted by ZNPS activities.

Between 2006 and 2013, other 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 K-40, Th-228, Ra-226 and Ac-228. These naturally occurring radionuclides are expected to be within the soils and detected during sample analyses. Appendix D of TSD 14-003 presents the 2006 through 2011 analytical data for groundwater analysis. Table 6.2 of TSD 14-003 presents the 2012 through 2nd Quarter 2014 analytical results for groundwater analysis.

2.3.6.6. Summary of Analytical Results in Surface Water

Between 2006 and 2014, H-3 and site-derived radionuclides has not been detected in surface water at ZNPS at concentrations greater than the respective detection limit. Gross Beta was detected within background range. K-40 was detected in a surface soil sample in May 2006 at a concentration of 106.8 pCi/L. Appendix D of TSD 14-003 presents the 2006 through 2011 analytical data for surface water. Table 6.2 of TSD 14-003 presents the 2012 through 2nd Quarter 2014 analytical results for surface water analysis.

2.4. Hazardous Material Characterization

The decommissioning approach for ZSRP calls for the beneficial reuse of concrete from building demolition as clean fill. A sampling plan was developed and implemented for testing of paints and coatings on structures and concrete. Zion*Solutions* evaluated the use of Clean Construction or Demolition Debris (CCDD) for the end-state of the basements and submitted a "Request for Concurrence for Basement Fill End-State" to the Illinois EPA in August of 2014. This request included the results of a sampling plan for concrete candidate fill material to be used for the basement fill end-state. On October 3, 2014, Zion*Solutions* received a "Letter of Concurrence" from the Illinois EPA for the use of CCDD for the basement fill end-state. In accordance with Section 3.160(b) of the Illinois Environmental Protection Act, CCDD is defined as uncontaminated broken concrete without protruding metal bars, bricks, rock, stone, reclaimed or other asphalt pavement or soil generated from construction or demolition activities. Only concrete debris that meets the definition of CCDD will be considered for use as clean hard fill and only when surveys have demonstrated that the concrete is free of detectable residual radioactivity. Concrete debris that is surveyed and found contaminated with detectable residual radioactivity will be disposed of as low-level radioactive waste.

Integral to the decommissioning process, Zion*Solutions* has undertaken a comprehensive asbestos abatement project. Asbestos containing material that is identified on equipment and

structures is remediated prior to demolition. Lead and PCB surveys are also performed on structures and remediation actions identified prior to demolition.

A sampling program has been developed to assess potential impacts to the environment. Non-radiological site characterization activities focus on areas of interest that have been identified based on the HSA and current site conditions. Characterization activities have included surface and subsurface soil sampling as well as groundwater sampling, with primary constituents of concern including, but not limited to: volatile organic compounds, semi-volatile organic compounds, polychlorinated biphenyls, metals, and dioxins. A preliminary list of areas of interest has been developed and will be evaluated for potential impacts to the environment. The areas of interest include the following:

- Fire Training, IRSF, and Mechanical Maintenance Training Warehouse Area
- Underground Fuel Oil Tanks (2 locations)
- Crib House
- East Yard Above Ground Storage Tanks (4 locations)
- Aboveground 150,000 gallon Fuel Oil Tank
- Transformer Areas
- Unit 1 Containment Tendon Area
- Used Oil Storage Pad
- Aboveground Diesel Fuel Tank north of ENC
- North Parking Lot
- Site Warehouses
- Oil/Water Separators
- Wastewater Treatment Facility
- Compactor and Dumpster Areas

Initial characterization activities have been completed at the Fire Training, IRSF, Mechanical Maintenance Training Warehouse Area, the Crib House, and the above ground 150,000 gallon Fuel Oil Tank. Environmental sampling completed for the areas of interest identified elevated poly-nuclear aromatic hydrocarbon (PAH) results in surface soils in the fuel oil tank area. A remediation plan will be required to address the impacted soils to reach closure of this area under IEPA regulations. Characterization activities will be completed at the remaining areas of interest as site decommissioning activities allow. Site closure activities will comply with applicable USEPA and IEPA regulatory requirements.

2.5. Continuing Characterization

The survey of many inaccessible or not readily accessible subsurface soils or surfaces has been deferred. All continuing Characterization Plans will be provided to the NRC for information and all continuing Characterization Reports will be provided to the NRC for evaluation. The

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

- The underlying concrete of the SFP/Transfer Canal below the 588 foot elevation after the steel liner has been removed. Continuing characterization will consist of scanning of the exposed concrete surfaces and the acquisition of concrete core sample(s) at the location of highest activity. The number and location of the additional concrete core sample(s) will be determined by DQO during survey design.
- The concrete walls and floor of the Under-Vessel areas in Unit 1 and Unit 2 Containments. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The floors and walls of the Hold-Up Tank (HUT) cubicle. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The floor of the Auxiliary Building 542 foot elevation Pipe Tunnel floors. Continuing characterization will consist of the acquisition of additional concrete core sample(s). The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The floor and lower walls of the 542 foot elevation of the Auxiliary Building to augment the existing characterization data. The number and location of the additional concrete core samples will be determined by DQO during survey design.
- The subsurface soils in the "keyways" between the Containment Buildings and the Turbine Building once subsurface utilities have been removed and the removal of subsurface structures in this area create access (e.g., Waste Annex Building). Continuing characterization will consist of the scanning of soils exposed by the demolition and building removal, the acquisition of soil sample(s) of the exposed soil and the acquisition of additional subsurface soil samples using test pits or soil borings. The number and location of the additional subsurface soil samples will be determined by DQO during survey design.
- The soils under the basement concrete of the Containment Buildings, the Auxiliary Building and the SFP/Transfer Canal once commodity removal and building demolition have progressed to a point where access can be achieved. Continuing characterization will consist of soil borings at the nearest locations along the foundation walls that can be feasibly accessed, angled soil bores to access the soils under the concrete, and deep cores from building floors, but not entirely through the foundation, at bias locations to assess migration potential from building interiors to soils under basement concrete. The number and location of the additional subsurface soil samples will be determined by DQO during survey design. Additional investigations and sampling will be performed in accordance with a sample plan if activity is positively identified.
- When the interior surfaces become accessible, several potentially contaminated embedded and buried pipe systems that will be abandoned in place. Continuing characterization will

consist of direct measurements on pipe openings and the acquisition of sediment and/or debris samples (if available) for analysis.

- The Containment basements after concrete removal. Continuing characterization of the steel liner will consist of beta gamma scans and swipe samples.

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

The decision to defer the characterization of a soil or structure was based on one or more of the following conditions:

- ALARA considerations (e.g., the area is either a high radiation or high contamination area and additional data would likely not change the survey area or area classification of the location or surrounding areas),
- Safety considerations (e.g., difficulty of access to the upper reaches of the Containments due to height),
- Historical data shows that the area could be classified without further characterization,
- Access for characterization would require significant deconstruction of adjacent systems, structures or other obstacles where the removal could result in an unsafe condition or interfere with continued operation of operating systems, or
- The ability to use engineering judgment in assigning the area classification based on physical relationship to surrounding areas and the likelihood of the area to have radiological conditions represented by the conditions in these adjacent areas.

Continuing characterization surveys as specified will be performed when these areas become accessible. Characterization survey plans will be generated for each using the approved characterization survey procedures. All additional characterization data will be collected as necessary, evaluated, documented in a Characterization report and stored with previous characterization survey data in a survey history file for the survey unit. This data will be used along with existing data to update the types of radionuclides present and update the variability in the radionuclide mix for both gamma-emitting and HTD radionuclides. In addition, as the decommissioning progresses, data from operational events caused by equipment failures or personnel errors, which may affect the radiological status of a survey unit(s) will be captured. These events will be evaluated and, when appropriate, stored in the characterization database. This additional characterization data will be used in validating the initial classification and in planning for the FSS.

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

As decommissioning proceeds, areas will, as necessary, be decontaminated to remove loose surface decontamination (as well as fixed contamination) to levels that will meet the conditions for controlled demolition or unrestricted release conditions for demolition. When a structure is ready for demolition, a documented survey and a formal turnover will be made by the Radiological Protection group for the group preforming demolition, validating that the radiological conditions in the structure are suitable.

Following the demolition and/or remediation, and when an area is believed to be ready for FSS, a "turnover assessment" will be performed. If the results of this assessment indicate that the FSS acceptance criteria will be met, then physical and administrative control of the area will be transferred to Characterization/License Termination group personnel for preparation, design, and performance of the FSS. Otherwise, additional remediation may be required. This assessment may include a "turnover survey," primarily for Class 1 and 2 areas within the "Security-Restricted Area".

The "turnover survey" process, together with any additional characterization and remediation survey performed, represent at least one, but possibly several, opportunities to collect additional survey data prior to conducting the FSS. For each type of survey (characterization, remediation, turnover, and FSS), a documented survey plan will be developed using the DQO process. These survey plans will contain the appropriate data assessment to ensure that several objectives are met. These objectives include;

- Appropriate instrument selection to ensure the proper sensitivity relative to the applicable action level and/or release criteria,
- Appropriate instrument quality control measures to ensure operability,
- Appropriate survey techniques to ensure that the field measurement techniques are consistent with the calibration methodologies,
- Appropriate sample collection and analysis to determine spatial variability and variability in radionuclide ratios,
- Data analysis criteria to identify follow-up actions such as remediation and the collection of additional samples and,
- Appropriate classification of the survey area.

2.6. References

2-1 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" – January 1999

2-2 "Zion Station Historical Site Assessment" (HSA) – September 2006

2-3 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," – August 2000

2-4 U.S. Nuclear Regulatory Commission NUREG-1757, Volume 2, Revision 1, "Consolidated Decommissioning Guidance - Characterization, Survey, and

Determination of Radiological Criteria, Final Report" – September 2003

2-5 Zion*Solutions* ZS-LT-02, Revision 3, "Characterization Survey Plan"

2-6 Zion*Solutions* ZS-LT-01, Revision 6,"Quality Assurance Project Plan (for Characterization and FSS)" (QAPP)

2-7 U.S. Nuclear Regulatory Commission NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans"

2-8 "Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement" – December 2007

2-9 U.S. Nuclear Regulatory Commission Docket Number 50-295, "Facility Operating License Number DPR-39 (for Unit One)"

2-10 U.S. Nuclear Regulatory Commission Docket Number 50-304, "Facility Operating License Number DPR-48 (for Unit Two)"

2-11 Illinois Environmental Protection Agency – National Pollutant Discharge Elimination System (NPDES) Permit Number IL0002763

2-12 U.S. Nuclear Regulatory Commission IE Bulletin 80-10, "Contamination of Nonradioactive Systems and Resulting Potential for Unmonitored Release of Radioactivity to Environment", – May 1960

2-13 Sandia National Laboratories, NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination From Decommissioning Parameter Analysis" – October 1999

2-14 Zion*Solutions* Technical Support Document 11-004, Revision 0, "Ludlum Model 44-10 Detector Sensitivity"

2-15 U.S. Nuclear Regulatory Commission NUREG-1576, "Multi-Agency Radiological Laboratory Analytical Protocols Manual" (MARLAP) – August 2001

2-16 U.S. Nuclear Regulatory Commission Regulatory Guide 4.15, "Quality Assurance or Radiological Monitoring Programs (Inception Through Normal Operations to License Termination) – Effluent Streams and the Environment" – July 2007

2-17 Zion*Solutions* ZS-QA-10, Revision 9, "Quality Assurance Project Plan - Zion Station Restoration Project"

2-18 Energy Services Commercial Services Group Report, "Characterization of the Zion Station Independent Spent Fuel Storage Facility (ISFSI)" – July 2012

2-19 "Annual Report on the Meteorological Monitoring Program at Zion Nuclear Power Station for 2010", Murray and Trettel, Inc. – February 2011

2-20 Energy Services Commercial Services Group Report CS-RS-PN-028, "Background Reference Area Report - Zion Nuclear Power Station", – February 2012

2-21 Zion*Solutions* Report, "Determination of Radionuclide Activity Concentrations in Soils in Non-Impacted Soils Adjacent to the Zion Nuclear Station", – July 2012

2-22 Zion*Solutions* Technical Support Document 13-004, Revision 0, "Examination of Cs-137 Global Fallout in Soils at Zion Station"

2-23 Zion*Solutions* Technical Support Document 11-001, Revision 1, "Potential Radionuclides of Concern during the Decommissioning of Zion Station"

2-24 Zion*Solutions* Technical Support Document 14-016, Revision 0, "Description of Embedded Piping and Penetrations to Remain in Zion End State"

2-25 Zion*Solutions* Technical Support Document 14-028, Revision 0, "Radiological Characterization Report"

2-26 Zion*Solutions* Technical Support Document 13-006, Revision 0, "Reactor Building Units 1 & 2 End State Concrete and Liner Initial Characterization Source Terms and Distributions"

2-27 Zion*Solutions* Technical Support Document 14-019, Revision 2, "Radionuclides of Concern for Soil and Basement Fill Model Source Terms"

2-28 Zion*Solutions* Procedure ZS-LT-100-001-001, Revision 2, "Characterization Survey Package Development"

2-29 Zion*Solutions* Technical Support Document 14-003, Revision 3, Conestoga Rovers & Associates (CRA) Report, "Zion Hydrogeologic Investigation Report"

2-30 Conestoga-Rovers and Associates, "Hydrogeologic Investigation Report, Fleetwide Assessment, Zion Station, Zion Illinois", Revision 1 – September 2006.

2-31 Wisconsin Department of Health Services, "Zion Environmental Radioactivity Survey" – 2001

2-32 International Standard ISO 7503-1, Part 1, "Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters" – August 1998

2-33 U.S. Nuclear Regulatory Commission Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants" – December 1975

Table 2-1 Operational History

DATE(S)	EVENT
02/10/67	ComEd announced intent to build Unit 1
07/11/67	ComEd announced intent to build Unit 2
07/12/67	ComEd dockets applications and PSAR to build both Units 1 and 2
11/25/70	ComEd dockets FSAR for both Units 1 and 2
04/06/73	ComEd receives license to operate Unit 1
11/14/73	ComEd receives license to operate Unit 2
06/19/73	Unit 1 Reactor achieves initial criticality
12/24/73	Unit 2 Reactor achieves initial criticality
12/31/73	Commercial operation begins for Unit 1
09/17/74	Commercial operation begins for Unit 2
03/05/76 to 06/19/76	Unit 1 Spring 1976 Refueling Outage
06/25/76	License received to operate both Units 1 and 2 at 100% power
01/07/77 to 03/27/77	Unit 2 Spring 1977 Refueling Outage
09/09/77 to 12/04/77	Unit 1 Fall 1977 Refueling Outage
02/04/78 to 04/12/78	Unit 2 Spring 1978 Refueling Outage
09/21/78 to 11/02/78	Unit 1 Fall 1978 Refueling Outage
03/09/79 to 04/18/79	Unit 2 Spring 1979 Refueling Outage
10/06/79 to 02/19/80	Unit 1 Fall 1979 Refueling Outage
05/02/80 to 07/26/80	Unit 2 Summer 1980 Refueling Outage
01/15/81 to 04/22/81	Unit 1 Spring 1981 Refueling Outage
09/11/81 to 12/01/81	Unit 2 Fall 1981 Refueling Outage
02/13/82 to 07/07/82	Unit 1 Spring 1982 Refueling Outage
02/24/83 to 05/27/83	Unit 2 Spring 1983 Refueling Outage
09/07/83 to 02/09/84	Unit 1 Fall 1983 Refueling Outage
03/27/84 to 07/10/84	Unit 2 Spring 1984 Refueling Outage
01/30/85 to 06/18/85	Unit 1 Spring 1985 Refueling Outage
09/05/85 to 02/04/86	Unit 2 Fall 1985 Refueling Outage

Table 2-1 Operational History (continued)

DATE(S)	EVENT
09/04/86 to 03/20/87	Unit 1 Fall 1986 Refueling Outage
02/03/87 to 02/04/87	Unit 2 Reactor Shutdown for 2D Main Steam Isolation Valve Oil Leak
04/25/87 to 05/05/87	Unit 1 Shutdown due to H-2 leak from Generator
03/25/87 to 08/08/87	Unit 2 Spring 1987 Refueling Outage
05/10/87 to 05/15/87	Unit 1 Shutdown due to High Vibrations in Generator
10/02/87 to 10/06/87	Unit 2 Reactor Forced Outage for Pressurizer Spray Valve Leakage
02/24/88 to 05/09/88	Unit 1 Spring 1988 Refueling Outage
07/13/88 to 07/16/88	Unit 1 Reactor Trip due to S/G Steam-flow/Feed-flow mismatch
07/23/88 to 07/26/88	Unit 1 Reactor Offline due to Extraction Steam leak in 16 heaters
08/09/88 to 08/12/88	Unit 2 Reactor Shutdown for 2A & 2D S/G Handhole Repairs
10/08/88 to 10/09/88	Unit 2 Reactor Trip during PT-1
10/12/88 to 12/28/88	Unit 2 Fall 1988 Refueling Outage
10/25/88 to 11/04/88	Unit 1 Reactor Shutdown due to Station Blackout
01/15/89 to 02/01/89	Unit 2 Reactor Shutdown to Repair 2A Steam Generator Tube Leak
01/27/89 to 01/29/89	Unit 1 Turbine Trip During EM Troubleshooting
02/06/89 to 03/03/89	Unit 1 Outage for Primary S/G Manway Gasket Replacement
02/20/89 to 02/22/89	Unit 2 Reactor Shutdown Due to Excess Pzr Spray Valve Leakage
03/08/89 to 03/09/89	Unit 1 Reactor Shutdown due to Inoperable RPI's / Sola Transformer
04/22/89 to 04/24/89	Unit 2 Reactor Shutdown due to Excess RCS Leakage – 2SS-9351A
08/21/89 to 08/23/89	Unit 1 Forced Outage due to loss of EHC Fluid from Broken Piping
08/27/89 to 08/31/89	Unit 1 Reactor Shutdown due to Inoperable S/G Safety Valves
09/07/89 to 01/25/90	Unit 1 Fall 1989 Refueling Outage
12/01/89 to 12/04/89	Unit 2 Reactor Forced Outage due to RCS Leak
01/18/90 to 01/19/90	Unit 2 Reactor Forced Outage due to Loss of EHC Control
01/27/90 to 01/28/90	Unit 1 Reactor Trip due to 1D S/G High Level alarm
03/01/90 to 06/13/90	Unit 1 and Unit 2 Reactor Forced Outage due to Diesel Failure
03/21/90 to 08/30/90	Unit 2 Spring 1990 Refueling Outage

Table 2-1 Operational History (continued)

DATE(S)	EVENT
08/13/90 to 08/17/90	Manual Trip of Unit 1 Turbine due to Operator Error
09/07/90 to 09/22/90	Unit 2 Turbine/Reactor Trip on loss of Condenser Vacuum
09/22/90 to 11/03/90	Unit 2 Turbine/Reactor Trip on 2W Main Transformer Failure
11/06/90 to 11/12/90	Shutdown of Unit 1 due to inoperable Diesel Generators
11/11/90 to 11/13/90	Unit 2 Turbine Trip Block Diaphragm Rupture
11/18/90 to 11/19/90	Unit 2 Heater Drain Tank Rupture Disk Repair
12/04/90 to 05/14/91	Unit 1 Reactor Shutdown due to excessive RCS Leakage
01/04/91 to 01/11/91	Unit 2 2A & 2B Safety Injection Pump Failure
03/21/91 to 06/11/91	Fire in the Unit 2 System Auxiliary Transformer
06/08/91 to 06/10/91	Unit 1 Reactor Shutdown required by abnormal 1C RCP Indications
06/11/91 to 06/13/91	Unit 2 Seal Leakoff Flow Transmitter Failed
07/28/91 to 07/30/91	Unit 2 Heater Drain Recirculation Line Rupture
09/27/91 to 11/11/91	Unit 2 1992 Surveillance Outage
12/07/91 to 01/02/92	Hydrogen Leak from Unit 1 Main Generator
02/27/92 to 08/13/92	Unit 1 1992 Refueling Outage
04/04/92 to 06/19/91	Unit 2 System Auxiliary Transformer Replacement
09/17/92 to 10/03/92	Shutdown of Unit 1 Reactor to repair the 1A Aux Feedwater Pump
11/12/92 to 02/22/93	Unit 2 1992 Refueling Outage
02/24/93 to 02/25/93	Unit 2 Main Turbine Overspeed Test
07/07/93 to 07/09/93	Unit 1 Reactor Trip due to trip of 1B RCP
07/10/93 to 07/11/93	Unit 1 Heater Drain Tank Disk Rupture
10/07/93 to 04/18/94	Dual Unit 2 Service Water/Refueling Outage
10/21/93 to 04/03/94	Dual Unit 2 Service Water/Refueling Outage
04/03/94 to 06/14/94	Unit 1 Reactor Trip due to fire at Generator Bus Duct
07/02/94 to 07/16/94	Unit 1 Reactor Trip due to fire at under Main Generator Bus Duct
10/21/94 to 10/23/94	Excessive Secondary Leak on 1A S/G Handhold
01/06/95 to 04/19/95	Unit 2 Refueling Outage

Table 2-1 Operational History (continued)

DATE(S)	EVENT
05/11/95 to 05/14/95	Unit 1 1995 Maintenance Outage
12/07/95 to 12/10/95	Unit 2 EHC Leak Due To Cracked Weld Near Accumulators
01/13/96 to 02/03/96	Unit 2 Maintenance Outage to repair Main Condenser Tubes
02/19/96 to 03/16/96	Unit 1 Shutdown to locate a loose item in the 1B S/G
03/16/96 to 03/22/96	Unit 1 Reactor Scram due to "1C" S/G Hi Level alarm
04/17/96 to 04/26/96	Automatic Scram of Unit 1 Reactor caused by Trip Of "C" Loop Flow
05/19/96 to 05/23/96	Shutdown of Unit 2 due to inoperable Diesel Generators
08/18/96 to 08/22/96	Automatic Trip of Unit 1 due to Failed Limit Switch On "1D" MSIV
08/27/96 to 09/18/96	Unit 1 Forced Outage To Repair Block Valve 1MOV-RC8000B
9/19/96	Last Day of Power Operations for Unit 2
9/19/96 to 01/15/98	Unit 2 Extended Refuel Outage
02/21/97	Last day of power operations for Unit 1
02/21/97 to 02/28/97	Unit 1 Forced Outage due to "1C" Core Spray Pump Inoperability
03/01/97 to 03/30/97	Unit 1 Maintenance Outage
03/31/97 to 01/15/98	Unit 1 Spring Refuel Outage
01/15/98	Decision to Permanently Shutdown Zion Unit 1 and Unit 2
02/13/98	Certification of Permanent Cessation of Operations for both Units
03/09/98	Certification of Permanent Defueled Status for both Units
03/10/98 to 01/25/08	Zion Units 1 and 2 maintained in a SAFSTOR condition
02/14/00	Exelon (formally ComEd) submits PSDAR to NRC.
01/25/08	Exelon submits application to NRC to transfer license to *ZionSolutions*
02/02/09	Zion NRC Licenses formally transferred to *ZionSolutions* pending confirmatory orders
09/01/09	Confirmatory orders issued allowing Zion*Solutions* to begin decommissioning
10/13/10	Physical Decommissioning of Zion Units 1 and 2 commences

Table 2-2 Historical Incidents/Occurrence

Date	Incident/Occurrence	Citation
07/24/73	RCS spill from pressurizer spray valves.	(ROR – No #)
12/27/73	Approximately 12,000 gallons of S/G water of pH 9.7 was released to the lake. Also approximately 2800 pounds of boric acid was also released to the lake.	(NRC IR 74-17/74-17)
02/1974 to 10/1974	During this period, forty-six liquid releases were made to the discharge canal from the waste neutralization tank. Two releases had detectable tritium and Co-58. In addition, two liquid releases were made from the Unit 2 Condenser Hotwell to the discharge canal which exhibited detectable tritium.	(NRC IR 74-14/74-13)
02/07/74	Spill of approximately 15 ft^3 of spent resin from "2B" CVCS mixed resin bed demineralizer	(ROR 74-01)
04/25/74	Valve leak in Unit 2 Containment leading to >10 gpm RCS leak.	(NRC IR 74-07/74-07)
07/26/74	Overflow of both Unit 1 and Unit 2 PWSTs. Unit 1 overflowed approximately 500 gallons and Unit 2 overflowed approximately 100 gallons. The top one-inch of soil in both spill areas was collected and disposed of as radioactive material.	(NRC IR 74-13/74-12 and 74-14/74-13)
07/30/74	A Rad Waste shipment to Sheffield arrived with 2 spots of liquid and some resin beads on the floor of the tractor trailer.	(NRC IR 74-11/74-10)
08/27/75	Spent resin spill in the Rad Waste Annex Truckbay 592 ft. elevation	(ROR 75-30)
09/08/75	Contaminated tools were discovered in the storeroom.	(ROR 75-34)
09/12/75	1000 to 2000 gallons of RWST water sprayed through Core Spray into Unit 1 containment.	(NRC IR 75-13/75-12)
01/22/76	Approximately 300 gallons of RCS water spilled from "2A" S/G manway.	(ROR 76-001)
01/26/76	Approximately 3,000 gallons of RCS water spilled from an open check valve on loop "2D".	(ROR 76-003)
03/18/76	Legal overexposure of 8.05 Rem occurred in the Unit 1 Incore area.	(NRC IR 76-12/)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
03/24/76	The Unit 1 RWST overflowed through the vent line contaminating the Unit 1 Auxiliary Building adjacent to the Unit 1 Containment between the 640 ft. to the 592 ft. and into the soil. Contamination levels on the wall were reading up to 3000 cpm/ft^2 and direct reading on the soil indicated up to 10,000 cpm.	(ROR 76-018)
04/20/76	Contaminated water leaking from the "1" Boric Acid Evaporator tank leaked into the "1B" Charging pump room and into the Laundry Drain Tank area on the 560 ft. elevation of the Auxiliary Building	(ROR 76-032)
05/05/76	A rope contaminated was discovered in the Tech Staff area of the Service Building	(ROR 76-39)
06/24/76	Contaminated clothing with readings up to 4,500 cpm was discovered in the Shift Engineer office.	(ROR 76-044)
08/14/76	Due to leakage from valve 0VC8264, a gaseous release occurred which exceeded the tech spec limit of 60,000 uCi/sec. The maximum release rate was 100,000 uCi/sec, with a total activity released of 31.8 Ci.	(LER 2-76-029 and NRC IR 76-31/76-27)
09/19/76	The PRT rupture disc failed due to a trip of Unit 2 Reactor. Approximately 2,500 gallons of PRT water spilled into Unit 2 Containment.	(NRC IR 76-28/76-24)
09/19/76	A fire occurred when operating the "2A" Diesel Generator	(NRC IR 76-28/76-24)
10/01/76	The Turbine Building Floor Drain Analysis Tank (TBFDAT) was found with 5K cpm direct and adjacent pipes up to 175 cpm/ft2	(ROR 76-53)
10/02/76	Contaminated tools were discovered in the Tech Staff area of the Service Building	(ROR 76-51)
10/02/76	Contaminated tools were discovered in the storeroom.	(ROR 76-52)
10/20/76	An unmonitored release (~ 5000 gallons) occurred from the TBFDAT to the fire sump.	(ROR 76-56 and LER 1-76-084)
02/15/77	Contaminated protective clothing with readings up to 2,500 cpm was discovered outside of the Gate House	(ROR 77-07)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
03/07/77	Radiological surveys performed on the 560 ft. and 592 ft. elevations of the Unit 2 Turbine Building identify loose surface contamination.	(ROR – No #)
06/03/77	Fire hoses were found draining the TBFDAT to the Fire Sump.	(ROR 77-27 and ROR 77-28)
06/30/77	Water which had collected in the Unit 1 Electrical Conduit Sump in the Turbine Building was discharged through the normal discharge path to Lake Michigan. The water in this sump is from ground seepage and has been assumed to be non-radioactive and discharged as such. It was discovered after the release that it was contaminated with low level radioactivity. The activity released was estimated to be 0.00614 Ci of beta/gamma emitters and 0.0274 Ci of tritium.	(LER 2-77-038)
06/1977	Release via an unauthorized unmonitored effluent pathway of about 5000 gallons from the Turbine Building Floor Drain Analysis Tank via the Turbine Building Fire Sump. Samples indicated concentrations of 2E-5 uCi/ml of fission products and an H-3 concentration of 2E-3 uCi/ml. This incident also resulted in the contamination of the Unit 1 and Unit 2 Oil Separators. The leakage containing H-3 from a Feedwater pump was being collected and routed via a drain to the Waste Neutralization tank.	(NRC IR 77-15/77-18)
10/19/77	Contaminated tools were discovered in the IM "Hot Shop" in an un-posted area.	(ROR 77-47)
11/09/77	Contaminated line was discovered in a desk located in the MM Shop.	(ROR 77-61)
11/12/77	Carpet in the control room was identified as radiologically contaminated.	(ROR 77-63)
02/25/78	A pipe wrench with contamination up to 500 cpm direct was discovered in the MM Shop tool crib.	(ROR 78-20)
02/25/78	A document that was discovered to be radiologically contaminated was discovered in the Tech Staff area of the Service Building. A follow-up survey of the area identified 35 additional contaminated documents in the area.	(ROR 78-61)
03/23/78	Contaminated shoe covers and coveralls with readings up to 1,000 cpm direct were discovered in a station pickup truck.	(ROR 78-36)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
04/15/78	Pipes by the TBFDAT and the Turbine Building Equipment Drain Analysis Tank (TBEDAT) found to have exposure rates up to 200 mR/h on contact.	(ROR 78-15)
06/06/78	Contamination up to 200,000 cpm direct was found on the ground outside the Rad Waste annex.	(ROR 78-43)
06/20/78	Spill of approximately 16 gallons of spent resin spill in the Rad Waste Annex area. Incident contaminated a truck and a concrete pad	(DVR 22-1-78-107)
07/23/78	During the transfer of S/G blowdown resin to an HN-150 cask, the cask was overfilled, causing contaminated resin to fall on the cask, trailer, and onto the ground outside the loading area.	(ROR 78-46)
02/01/79	During the transfer of the 2A mixed bed to an HN-200 cask, an overflow occurred, spilling approximately 25 gallons of the water/resin mixture onto the cask exterior, truck-bed, and onto the pavement around truck. Dose rates on the spilled mixture ranged from 180 - 310 mR/h.	(ROR 79-01)
03/05/79	A contaminated trailer was found parked outside the Fuel Building trackway door. Although wrapped in plastic, the trailer still had contamination up to 300,000 cpm and a hot spot of 120 mR/h. The trailer was successfully decontaminated and released on 10/25/1979.	(ROR 79-08)
03/30/79	A spill occurred during the filling of a resin cask for solidification and shipment. An estimated 25 gallons of line flush water containing some resins overflowed the cask, contaminating it, the trailer, and the concrete pad outside of the Rad Waste Truckbay where the transfer took place. The occurrence was blamed on failure of a high liquid level alarm together with a frozen overflow line designed to handle such an occurrence.	(NRC IR 79-04/79-04)
04/09/79	Surveys conducted outside the fenced area by the Rad Waste loading area noted soil contamination up to 3000 cpm.	(ROR 79-12)
06/05/79	After the Rad Waste loading area had been remediated, a large amount of plastic was found in the area with up to 20,000 cpm removable contamination. A wall in the area was found with up to 25,000 cpm/ft^2 contamination.	(ROR 79-25)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
08/07/79	During the transfer of resin into an HN-600 cask, a hose became disconnected from a sight-glass, spraying a resin/water mixture into the area. Due to the ventilation configuration, most of the spill was contained in the Rad Waste Truckbay and Auxiliary Building, but some radioactive material spread to the area outside of the Truckbay door.	(ROR 79-36A)
10/26/79	pH of WWTF discharge to Lake Michigan was observed at 8.1 which are above the limit of 8.0.	(LER 1-79-085)
12/1979	A spill occurred during the filling of a resin cask for solidification and shipment. An estimated 25 gallons of line flush water containing some resins overflowed the cask, contaminating it, the trailer, and the concrete pad outside of the Rad Waste Annex where the transfer took place. The occurrence was blamed on failure of a high liquid level alarm together with a frozen overflow line designed to handle such an occurrence. Temperature at the time of the transfer was said to be 10-15 degrees F. A rather extensive cleanup was performed. The tractor was cleaned and released; the trailer moved to the Fuel Handling Building loading area, where it was stored until final disposition.	(NRC IR 79-04/79-04)
12/27/79	The WWTF effluent pH ranged from 4.1 to 4.77 (lower than the Tech spec minimum of 6.0). A total of 5,760 gallons of the acidic solution was discharged to the condenser cooling water, where it was further diluted prior to discharge to Lake Michigan.	(LER 1-79-094)
03/20/80	A hose from the Auxiliary Building Floor Drain Analysis Tank (ABFDAT) and Equipment Drain Analysis Tank (ABEDAT) to a portable demineralizer ruptured and spilled approximately 20 gallons of contaminated water from the 617 ft. elevation to the 562 ft. elevation. Contamination ranged from 1,000 to 2,000 cpm/ft^2.	(ROR 80-16)
03/21/80	While using a portable demineralizer system in the Rad Waste Annex, a hose failure resulted in the spill of approximately 30 to 40 gallons of radioactive water to the floor.	(NRC IR 80-05/80-04)
04/17/80	Spent resin was spilled in the non-posted portion of the Fuel Handling Building 617 ft. elevation mezzanine	(ROR 80-20)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
05/1980	Due to a valve connection error, a freeze seal that was applied was applied to a RCS pipe system failed, causing the spill of approximately 2,000 gallons of refueling cavity water to the basement of the Unit 2 Containment.	(NRC IR 80-12/80-12)
05/21/80	Contamination up to 1,000 cpm/ft^2 was discovered on the cask overflow liner outside of the posted "Contaminated Area".	(ROR 80-31)
06/09/80	A spill of contaminated water occurred during the solidification of an HN-100 demineralizer. The spill contaminated the floor in the Fuel Handling Building outer track-way (Car-Shed) up to 35,000 cpm/ft^2.	(ROR 80-35)
07/08/80	The WWTF effluent pH ranged from 4.0 to 6.0 (lower than the Tech spec minimum of 6.0). A total of 317,430 gallons of the acidic solution was discharged to the condenser cooling water, where it was further diluted prior to discharge to Lake Michigan.	(LER 1-80-033)
07/20/80	The WWTF effluent pH ranged from 4.0 to 6.0 (lower than the Tech spec minimum of 6.0). Approximately 70,200 gallons of the acidic solution with a pH of 5.86 was discharged to the condenser cooling water, where it was further diluted prior to discharge to Lake Michigan.	(LER 1-80-035)
10/24/80	A basic solution from the Waste Neutralization Tank drained to the Fire Sump and the resulting effluent pH to the WWTF was 8.49. Approximately 8,600 gallons of caustic water was discharged to the condenser cooling water, where it was further diluted prior to discharge to Lake Michigan.	(LER 1-80-045)
10/30/80	A contaminated metal pipe with readings up to 100,000 cpm was found in a dumpster.	(ROR 80-64)
11/01/80	The WWTF effluent pH was measured at 4.45 with a tech spec allowed range of 6 - 8. Approximately 27,000 gallons of acidic wastewater was discharged to the condenser cooling water, where it was diluted by a factor of 8200 to 1.	(LER 1-80-047)
11/02/80	A contaminated black hose and plastic with readings up to 1,500 cpm was discovered outside the Unit 2 Containment Emergency hatch.	(ROR 80-56)
11/18/80	A contaminated vent duct (up to 3,000 cpm/ft^2) was found in a dumpster.	(ROR 80-74)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
11/20/80	A contaminated vent duct (up to 800 cpm/ft^2) was found in a dumpster.	(ROR 80-77)
01/15/81 to 01/31/82	Primary to Secondary leakage discovered on the "1B" S/G and "1C" S/G prior to outage. Leakage observed at 45 gpd. After outage completion, leakage again noted in "1B" S/G at ~ 4 to 32 gpd.	(NRC IR 81-03 and NRC IR 81-09/81-05)
02/08/81	Contaminated water was sprayed into the Rad Waste Annex when a hose from a liner ruptured.	(ROR 81-25)
02/26/81	A rug contaminated with readings up to 50,000 cpm direct was discovered adjacent to the old Service Building elevator and a chair contaminated with readings up to 300 cpm direct was discovered in the MM Shop office area.	(ROR 81-31)
03/21/81	An individual carrying personal clothing alarmed the Gate House portal monitor. The individual refused to bag the clothing and return to Radiation Protection. The individual left the clothing in the Gate House and left site.	(ROR 81-33)
04/05/81	Four bags containing contaminated materials (up to 80,000 cpm) was found in a dumpster.	(ROR 81-35)
04/14/81	A bag containing contaminated Out-Of-Service cards (up to 200,000 cpm) was found in a dumpster.	(ROR 81-39)
04/14/81	A contaminated individual was inadvertently released from site resulting in the low level contamination of passenger vehicles and the individual's home with identified readings up to 700 cpm direct. The total offsite activity released was estimated to be less than 0.5 uCi.	(ROR 81-37, NRC IR 81-15/81-11 and ComEd response)
04/15/81	A contaminated hammer with readings up to 500,000 cpm direct was discovered in the Gate House	(ROR 81-40)
04/16/81	Three bags containing contaminated materials (up to 2,500 cpm) was found in a dumpster.	(ROR 81-38A)
08/20/81	A contaminated tool with readings up to 1,200 cpm fixed was discovered in the Tech Staff office area of the Service Building.	(ROR 81-49)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
08/21/81	A dry active waste (DAW) shipment from Zion was found to be leaking water upon its arrival at the Richland Washington disposal site. Approximately 2 pints of water containing 20 to 80 pCi of activity was found on the truck bed. The DAW container was found to have been damaged.	(NRC IR 81-20/81-16)
09/1981	In September 1981, the leakage from "1B" S/G averaged 118 gpd.	(NRC IR 81-20/81-16)
09/11/81	During Unit 2 outage, a fuel assembly was damaged (missing spring clip, torn grid strap and a loose rod). The fuel assembly was removed and repaired. Subsequent debris from the failure was observed in "2B" and "2C" S/Gs.	(NRC IR 81-26/81-22)
10/1981	In 4th qtr of 1981, primary to secondary leakage observed was "1A" S/G (~12 gpd), "1B" S/G (~240 gpd), "1C" S/G (~20 gpd), and "1D" S/G (~19 gpd)	(NRC IR 81-26/81-22)
10/27/81	A "bellows" contaminated with readings up to 1,000 cpm direct was discovered in the EM Shop	(ROR 81-57)
10/29/81	A contaminated teletector with loose contamination up to 20,000 cpm/ft^2 was discovered in the respirator issue room.	(ROR 81-54)
12/01/81	Noted RCS leak from Unit 2 RVLIS fitting upon reaching operating pressure. Due to leak, the upper reactor head was saturated with steam and condensed reactor coolant.	(NRC IR 81-26/81-22)
12/01/81 to 01/15/82	Due to increasing activity in the secondary system of Unit 1 due to primary to secondary leaks in all four S/G at flow rates exceeding 500 gpd, several areas in the Unit 1 Turbine Building were posted as Radiologically Controlled Areas	(NRC IR 81-29/81-27)
12/19/81 to 01/19/82	Primary to secondary leakage observed was "1B" S/G (218 to 431 gpd) and "1C" S/G (8 to 56 gpd)	(NRC IR 81-29/81-27)
02/28/82	Contaminated protective clothing was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 82-35)
03/03/82	A health physicist intentionally released approximately 20 people who had alarmed the gate house portal monitor. One individual was noted to have 500 cpm on a shoe.	(ROR 82-53 and ROR 82-54)
03/09/82	A pair of contaminated rubber gloves reading up to 300 cpm was discovered during a daily survey of a trash pile at the dumpster area.	(ROR 82-88B)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
03/21/82	Contaminated tools and scrap materials with direct readings up to 900 cpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 82-30 and ROR 82-32)
03/24/82	Leakage from the Unit 1 refueling cavity occurred due to gasket failure of a nuclear instrument well cover. The leakage increased beyond the capacity of the installed containment and cavity sump systems. The reactor cavity filled up and overflowed about 2,000 gallons onto the containment floor.	(NRC IR 82-04/82-04)
03/25/82	A portable demineralizer in the Rad Waste Annex overflowed, resulting in the spill of an indeterminate amount of radioactive water spilled to floor	(ROR 82-40)
03/25/82	Legal overexposure of 3.88 Rem occurred in the Unit 1 Incore area.	(NRC IR 82-09/)
04/04/82	A bag of contaminated trash reading 200 cpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 82-47)
04/18/82	Contaminated protective clothing with readings up to 300 cpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 82-56)
04/21/82	Contaminated gloves reading up to 300 cpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 82-60)
04/29/82	A bag containing contaminated material (200 cpm) was found in a dumpster during daily survey.	(ROR 82-63)
05/09/82	A contaminated surgeon's cap with readings up to 1,500 cpm was discovered on the ground between the Gate House and the NRC Building	(ROR 82-64)
05/25/82	A bag containing contaminated material (400 cpm) was found in a dumpster during daily survey.	(ROR 82-69)
06/08/82	A piece of radiologically contaminated waste was discovered in the office area by the control room.	(ROR 82-71)
06/17/82	A piece of contaminated pipe reading 45,000 dpm direct was discovered in the dumpster	(ROR 82-73)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
06/17/82	A barrel from the Unit 1 steam tunnel was transferred to the clean trash compactor and found to have contaminated oil/dirt/grease in bottom. (up to 62,500 dpm). Some material spilled onto the asphalt and was subsequently cleaned.	(ROR 82-74)
06/25/82	Unit 2 Reactor tripped from full power due to a ground on the Main Power Transformer. Fire-fighting exercises to the north of the Turbine Building resulted in the Main Power Transformer being covered with a residue of the fire-fighting chemical ("purple K"). A sudden rain interacted with the chemical causing a conducting solution and the busses arced to ground.	(NRC IR 82-14/82-13)
07/17/82 to 08/31/82	RCS to CC water system leak in the Unit 1 Pzr Steam Space Sample Cooler	(NRC IR 82-19/82-17)
08/11/82	A fire occurred in the "0" Diesel Generator due to a failure of the turbo charger lube oil filter gasket. The oil sprayed on an exhaust manifold where it ignited. An additional 10-15 gallons of lube oil spilled on the floor.	(NRC IR 82-19/82-17)
11/17/82	Trash bags containing contaminated material (up to 100K dpm) were discovered at the trash compacting area. While moving the bags, material spilled which contaminated the asphalt. The surface was subsequently decontaminated. An investigation concluded that the material originated from steam tunnel work areas.	(ROR 83-81)
01/31/83	A severe oil leak occurred on the "2B" Diesel Generator	(NRC IR 83-02/83-02)
04/14/83	Unit 1 shutdown due to severe weather when 4 of 6 incoming 4KV lines were lost. This resulted in a surge on the Unit 1 Main Transformer resulting in an insulator for one phase of the east transformer to explode.	(NRC IR 83-04/83-04)
07/13/83	RCS leakage of ~ 1.6 gpm was identified (source from BIT inlet valves)	(LER 1-83-020)
08/04/83	Hydrazine spill by Unit 1 Hydrazine Tank on the Turbine Building 560 ft. elevation	(NTS 29520193 CAT4-0673)
08/17/83	Fire in the "1A" Diesel Generator room due to the failure of the turbocharger oil filter gasket which allowed oil to spray directly onto the turbocharger manifold.	(NRC IR 83-17/83-18)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
10/16/83	Contaminated slings reading up to 50,000 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 83-67)
11/02/83	High noble gas levels in Unit 1 resulted in the contamination of 65 persons	(NRC IR 83-21/83-22 and 83-27/83-28 and ROR 83-97)
11/09/83	A bag containing contaminated material (20,000 cpm) was found in a dumpster during daily survey.	(ROR 83-78)
11/22/83	An individual alarmed the gate house portal monitor due to contaminated tape stuck to shoe. The investigation led to a probable location of source as outside the east DAW Building.	(ROR 83-87)
01/15/84	A contaminated flashlight with readings up to 3,000 cpm direct was discovered at the Gate House.	(ROR 84-07)
01/20/84	While performing repairs in the seal table room, a fitting broke causing an RCS leak of ~ 18 gpm. Approximate 700 gallons of RCS spilled before the leak was stopped.	(LER 1-84-005, and NRC IR 83-26/83-27)
03/14/84	The Hot Lab and office areas were contaminated during a Na-24 moisture carryover test.	(ROR 84-20)
05/1984	Caustic accidentally entered the RW system, releasing radioactive contaminants which had plated out on the piping and caused higher levels of contaminants in the liquid waste effluents.	(LER 1-85-035)
05/09/84	Contaminated material with readings up to 1,250,000 dpm direct was discovered in the dumpster	(ROR 84-41)
05/11/84	Drained down ~ 3000 gallons of 30% NaOH from the Unit 2 Spray Add tank. The majority of the caustic ended up in the ABEDAT room through vents in the Containment Spray pumps.	(Letter re: high Na concentration in Unit 2 RCS)
05/12/84	Materials with loose surface contamination was discovered in the east DAW Building	(ROR 84-42)
05/30/84	Found a fire extinguisher in the Turbine Building with fixed contamination up to 1,250 dpm	(ROR 84-47)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
06/16/84	A contractor who set off the gate house portal monitor allegedly decontaminated his shoe in a puddle outside the gate house.	(ROR 84-54)
08/14/84	The Unit 2 Containment exceeded the Technical Specification limit of 120 degrees F. Actual maximum temperature was 120.48 degrees.	(LER 2-84-020)
08/26/84	A "teletector" located in the IM Shop was discovered to be internally contaminated.	(ROR 84-76)
09/10/84	Unit 1 was shutdown due to excessive primary to secondary leakage of > 500 gpd (calculated at 1,050 gpd). Unusual Event declared	(NRC Log, LER 1-84-029, NRC IR 84-17/84-18)
09/14/84	A pipe wrap contaminated with readings up to 7,500 cpm direct was discovered in the MM Shop.	(ROR 84-104)
09/27/84	Contaminated material with readings up to 7,500 dpm direct was discovered in the dumpster	(ROR 84-80)
10/06/84	Contaminated material with readings up to 7,500 dpm direct was discovered in the dumpster	(ROR 84-88)
11/12/84	RCS leakage reached 10-12 gpm due to packing leaks from valve 2FCV-121. The valve is located in the "2B" Charging Pump room.	(NRC IR 84-23/84-22)
12/07/84	A U.S. Flag was found to be contaminated with readings up to 2,500 cpm direct. The contamination was later found to be naturally occurring materials.	(ROR 84-102)
12/10/84	Discovered contaminated material with readings up to 375,000 dpm in an uncontrolled storage cage on the Unit 1 Turbine Building 617 ft. elevation	(ROR 84-103)
01/03/85	Radiological surveys discovered contaminated material up to 2,500 cpm fixed in the trash segregation area located in Unit 1 Turbine Building Trackway. Investigations concluded that the material originated in the secondary sample room	(ROR 85-02)
01/19/85	During testing, a fuel oil supply line leaked an indeterminate amount of fuel oil to the Turbine Building floor	(LER 1-85-002)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
02/05/85	Contaminated trash reading up to 2,500 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 85-18)
03/02/85	Contaminated trash reading up to 2,000 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 85-40)
02/16/85 to 05/17/85	2 component cooling water leaks identified in relief valves, the first one on 02/16/85 on 1CC-9427 (total of ~ 1710 gallons of water), and the second on 05/17/85 on 1CC-9428. A total of ~ 10,000 gallons of component coolant water was discharged to the containment floor	(NRC IR 85-12/85- 13 and NRC IR 85-20/85-21)
03/16/85	Contaminated trash reading up to 3,700 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 85-46)
04/06/85	Contaminated material with readings up to 18,750 dpm fixed was discovered in the Tech Staff area of the Service Building	(ROR 85-49)
04/24/85	Contaminated tools with fixed contamination up to 5,000 dpm direct was discovered in the EM Shop	(ROR 85-53)
05/13/85	A radiological survey of Unit 1 Steam Tunnel identified contaminated material up to 1250 dpm by direct frisk.	(ROR 85-61)
08/23/85	A contaminated Rad Waste cask with 200,000 dpm fixed and 3,750 dpm smearable was identified as it was being unloaded inside the north security gate.	(ROR 85-74)
09/14/85	A contaminated ladder with readings up to 7,500 dpm direct was discovered leaning against the Unit 1 Discharge Valve house.	(ROR 85-80)
09/14/85	Contaminated hoses with readings up to 75,000 dpm direct were discovered in the Crib House.	(ROR 85-81)
09/14/85	Contaminated white crystals (up to 2500 dpm) were found under the evergreens outside the Service Building and in the storage shed near the WWTF. Isotopic analysis results showed all activity was due to naturally occurring potassium in the crystals (fertilizer).	(ROR 85-82)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
12/10/85	Contaminated plywood reading up to 18,500 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 85-101)
12/17/85	Contaminated trash reading up to 7,500 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 85-104)
12/18/85	Contaminated wood pallet reading up to 1,000 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 85-106)
01/30/86	Contaminated rubber gloves with readings up to 12,500 dpm direct was discovered in a trash can outside of the Auxiliary Building elevator.	(ROR 86-05)
01/31/86	During a resin sluice, a hose broke spilling contaminated liquid into the Rad Waste Annex 592 ft. elevation floor. Floor was contaminated to levels up to 675,000 dpm/100cm^2 smearable.	(ROR 86-06)
02/11/86	Contaminated protective clothing with readings up to 6,250 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 86-08)
05/16/86	Observed RCS leakage in excess of Technical Specification limit. Source was Pzr spray valve 1PCV-RC06	(NRC IR 86-11/86-10)
06/09/86	Flooding was noted in the Unit 2 Tendon tunnels	(Zion RP/Decon Log)
06/26/86	Pieces of contaminated metal with readings up to 1,250 dpm direct was discovered in the dumpster	(ROR 86-18)
09/16/86	Contaminated protective clothing with readings up to 7,500 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 86-23)
09/30/86	Spill from the Unit 1 SST following weld repairs on the tank.	(Zion RP/Decon Log)
11/17/86	Transportation incident with the shipment of the 1A RCP motor to Westinghouse as the package was damaged in-route. No loss of contamination control occurred.	(NRC IR 86-28/86-28)
12/04/86	Contaminated material with readings up to 12,500 dpm was discovered in the clean trash staging area in the Unit 2 Turbine Building Trackway	(ROR 86-46)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
02/09/87	A spill of approximately 60 gallons of fuel oil occurred in the Turbine Building	(Zion RP/Decon Log)
03/27/87	Minor fire in the Turbine Building 560 ft. elevation	(NRC IR 87-06/87-06)
03/27/87	Contaminated trash with readings up to 10,000 dpm was discovered in the clean trash staging area in the Unit 1 Turbine Building Trackway	(ROR 87-10)
04/29/87	Discovered a contaminated step stool with readings up to 2,500 dpm direct in an outside trash pile	(ROR 87-15)
05/21/87	Service water flood into the "2A" and "2B" Diesel Fuel Oil Storage tank rooms as well as the Turbine Building 560 ft. elevation. Approximately 181,000 gallons of water was discharged	(NRC IR 87-09/87-11 and NRC IR 87-16/87-16)
05/31/87	Contaminated metal with direct readings up to 8,750 dpm was discovered in the scrap metal dumpster	(ROR 87-21)
06/21/87	Found contaminated protective clothing and tape (up to 2500 dpm direct) near Primary Storage Tank Pumps	(ROR 87-25)
08/04/87	Contaminated protective clothing with readings up to 10,000 dpm direct was discovered in a trash pile north of the Crib House.	(ROR 87-36)
10/01/87	Observed RSC leakage in Unit 2 Containment from Pzr Spray valve.	(NRC IR 87-26/87-27)
10/09/87	Discovered contaminated material with readings up to 25,000 dpm in Tech Staff storage cage on the Unit 1 Turbine Building 617 ft. elevation	(ROR 87-41)
03/1988 to 12/1988	Report denotes a 1200 gallon spill of diesel fuel in March 1988, a 500 gallon diesel fuel spill in April 1988 and a 300 gallon turbine oil spill in May of 1988. In addition, a Bulk Acid Tank rupture in December of 1988 resulted in the discharge of 92% sulfuric acid to the WWTF for processing.	(1988 WWTF Report)
03/01/88	An oil spill was identified in Unit 1 Turbine Building that deposited oil along the wall and on the floor of the insulator workshop in the south Condenser Bay	(Zion RP/Decon Log)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
04/01/88	Contaminated tool with fixed contamination up to 12,500 dpm direct was discovered in the MM tool crib	(ROR 88-33)
04/16/88	Discovered contaminated drums of waste oil with smearable levels up to 2,500 dpm on the outside of the drums located in the posted "High Radiation Area" between the Unit 2 Containment and the Fuel Handling Building	(ROR 88-47)
05/1988	Spill of approximately 300 gallons of oil in the Turbine Building.	(1988 WWTF Report)
05/03/88	Radiological surveys identified surface contamination up to 10,000 cpm fixed on materials found in an outside trash pad. An investigation concluded that the material originated in the Unit 2 Steam Tunnel Valve House.	(ROR 88-54)
05/04/88	Diesel fuel oil spill in the Unit 1 Containment Spray cubicle in the Auxiliary Building.	(Zion RP/Decon Log)
05/27/88	Contaminated material, up to 375,000 dpm direct was discovered in a trash pile located outside of the RCA.	(ROR 88-59)
10/22/88	Hydrazine spill by the Unit 1 Chemical Addition area on the Turbine Building 560 ft. elevation.	(Zion RP/Decon Log)
10/27/88	License termination order for the Westinghouse Training reactor housed in the West Training Center received	(NRC License # R-119, Docket # 50-87)
11/19/88	Contaminated tools with readings up to 25,000 dpm fixed were discovered in the contractor office trailer located adjacent to the Unit 2 trackway.	(ROR 88-85)
12/07/88	Discovered a contaminated hand radio with fixed readings up to 40,000 dpm direct in the IM Shop.	(ROR 88-88)
12/19/88	Bulk Acid Tank rupture caused 1,228 gallons of 92% sulfuric acid to be sent to the WWTF for processing.	(1988 WWTF Annual Report)
12/19/88	Discovered spill of sulfuric acid in the Unit 2 Turbine Building Trackway. The acid flowed down an floor penetration and deposited acid on the 560 ft. elevation floor	(NRC IR 88-23/88-23)
02/12/89	The Rad Waste conveyor system broke down during the removal and shipment of some of the drums. As a result, some drum with spent resins was spilled outside.	(Zion RP/Decon Log)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
02/21/89	Rad Waste conveyor system broke during the removal and shipment of 55 gallon drums of waste. Spent resin was spilled outside the Truckbay door when several barrels broke open	(Zion RP/Decon Log)
03/16/89	Contaminated insulating blankets (up to 37,500 dpm direct) were found in the Unit 2 Turbine Building trackway trash area. These blankets were determined to have come from the "2C" East MSR Reheat Stop Valve.	(ROR 89-21)
03/22/89	Discovered contaminated wood with fixed contamination up to 37,500 dpm direct in the Unit 2 Turbine Building Trackway trash segregation area	(ROR 89-24)
05/08/89	Unplanned release of radioactive noble gas and spill of approximately 300 gallons of RCS water during a fill of the CVCS demineralizer.	(NRC IR 89-20/89-18)
07/18/89	Discovered contaminated tool with fixed contamination up to 20,000 dpm direct in the Shift Engineer office.	(ROR 89-24)
07/28/89	Discovered contaminated metal with readings up to 75,000 dpm direct in the dumpster trash area.	(ROR 89-31)
10/01/89	Identified flooding in Unit 1 Containment through the open manway openings in all four S/G	(Zion RP/Decon Log)
01/03/90	Observed RCS leakage of ~ 20 gpm from valve 1MOVRH8702	(LER 1-90-001)
01/16/90	The overflow of the "0A" Lake Discharge Tank (LDT) resulted in a substantial contamination of the Auxiliary Building 542 ft. elevation floor.	(ROR 90-02)
01/23/90	Zion became aware of a condenser cooling water leakage from Unit 2 discharge piping approximately 60 feet off shore. It was determined that a 20 foot section of the 14 foot diameter pipe had shifted slightly, resulting in the top portion of the discharge piping's bell spigot joint to be misaligned. This translated to a four inch gap at one construction joint and an 8 inch gap on the other. The estimate at the time was that 10% of the discharge volume was escaping the normal discharge location of 870 feet into Lake Michigan. The intent was to continue operations in this mode until late March 1990.	(NRC IR 90-03/90-03)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
03/05/90	A loss of Main Turbine oil of approximately 3,000 gallons from the Unit 2 Turbine Oil reservoir was identified and observed as an oil slick in the Screen-house afterbay. The leak was isolated and the oil was removed from the afterbay. No detectable oil was observed in Lake Michigan by the Coast Guard.	(NRC IR 90-03/90-03 and ESD Spill Summary)
03/06/90	Black sand was discovered on the beach adjacent to the site. Analysis indicated that the "black sand" from lake shore was not oil/grease, but ferrotitanium. Oil and grease were less than 40 ug/g.	(SMAD black sand analysis)
03/22/90	A contaminated wire with readings up to 3,750 dpm direct was discovered under the stairs in on the Service Building 592 ft. elevation	(ROR 90-05)
03/29/90	Water leakage and system overflow in both Unit 1 and Unit 2 Containment Spray pump cubicles	(Zion RP/Decon Log)
04/08/90	Contaminated chairs with readings up to 7,500 dpm direct was discovered in the Gate House	(ROR 90-48)
04/02/90	Contaminated material up to 7,500 dpm/100cm2 smearable was discovered in the Fuel Handling Building Car-Shed	(ROR 90-07)
04/26/90	Instruments contaminated with readings up to 150,000 dpm direct was discovered in the IM Shop	(ROR 90-18)
04/26/90	A contaminated instrument component with readings up to 10,000 dpm fixed was discovered in the trash awaiting compaction at the Crib House	(ROR 90-19)
05/04/90	A contaminated clipboard with readings up to 18,750 dpm fixed was discovered in the contractor trailer.	(ROR 90-24)
05/07/90	During the refueling of Unit 2, a piece of grid strap was observed falling from a fuel assembly during movement. The cladding on the assembly remained intact	(LER 2-90-006)
07/02/90	The overflow of the “0A” Lake Discharge Tank (LDT) resulted in a substantial contamination of the Auxiliary Building 542 ft. elevation floor.	(ROR 90-35)
07/31/90	Discovered a bucket containing contaminated concrete with readings up to 57,500 dpm fixed in the outside trash area. It was identified that the bucket originated from the 642 ft. elevation of the Turbine Building.	(ROR 90-38)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
08/22/90	Spill of approximately 200 gallons of radiologically contaminated water from the vendor laundry trailer located at the north missile door entrance to Auxiliary Building. The water spilled onto concrete into the adjacent subsurface soil. The affected area was approximately 80 feet by 80 feet.	(ROR 90-41)
09/10/90	Notification of flooding in the Crib House. Flooding reached levels approximately 2 feet above the basement floor. An unusual event was declared.	(NRC IR 90-21/90-23)
09/22/90	The Unit 2 Main Power Transformer exploded and burned, resulting in a spill of approximately 300 gallons of transformer oil to the pavement adjacent to the transformer.	(NRC IR 90-21/90-23)
09/29/90	Discovered nitrogen tanks with fixed contamination up to 5,000 dpm direct in the Unit 1 Turbine Building trackway	(ROR 90-43)
10/16/90	Both LDTs overflowed resulting in the discharge of approximately 16,000 gallons of contaminated water to the Auxiliary Building 542 ft. elevation floor.	(Zion RP/Decon Log)
02/25/91	Discovered contaminated fittings and flanges with readings up to 25,000 dpm direct on a truck leaving the Protected Area.	(ROR 91-04)
04/17/91	Due to a leaking valve, the SFP overflowed. Water accumulated in and overflowed the ABEDAT (500 to 1000 gallons) causing the backup of water through the bedplate drains in the Unit 1 and Unit 2 Containment Spray pump rooms and the Unit 1 Safety Injection pump room. The areas outside the rooms and the level below were contaminated.	(NRC IR 91-10/91-10)
06/26/91	Break in a temporary PVC pipe for a temporary demineralizer skid created a flood on the Turbine Building 560 ft. elevation.	(Zion RP/Decon Log)
08/01/91	The Boric Acid Mix Tank (BAMT) overflowed, contaminating Auxiliary Building areas from the 617 ft. elevation to the 562 ft. elevation.	(Zion RP/Decon Log)
05/13/92	Approximately 4,200 gallons of RCS water was inadvertently sprayed into Unit 2 Containment through the "2A" Containment Spray header	(NRC IR 92-10/92-10)
05/17/92	A grass fire was reported adjacent to the West Training area.	No citation

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
05/19/92 to 07/06/92	A weld failure in the RWST system resulted in a leak of approximately 1 gpm to the Unit 1 Equipment Drain Tank Collector room.	(NRC IR 92-14/92-14)
07/15/92	A failure of the "2A" Diesel Generator was attributed to zebra mussels in the lube oil system. Zebra mussels were later also found in "2B" Diesel Generator coolers.	(LER 2-92-004 and Zion RP/Decon Log)
10/24/92	Overflow of the "0B" Blow Down Monitor Tank (BDMT) resulted in the contamination of approximately 5,000 ft2 of floor on the Auxiliary Building 542 ft. elevation.	(NTS 2952019 208401)
11/04/92	During a cleanup of the two oil/water separators, the station discovered oil containing PCBs (5 ppm on Unit 1 and 11 ppm on Unit 2). These were the only positive results for PCBs. After the separators were cleaned, water and sludge samples were taken and indicated <0.5 ppb and < 1.0 mg/kg PCB respectively.	(NTS 29555492 CAT3-087)
03/29/93	An oil leak was identified that leak approximately 80 gallons of oil to the floor near the "2C" Feed Water pump	(NTS 30420193 CAT4-0258)
06/01/93	Unit 1 and Unit 2 oil separator sludge found to be contaminated	(NTS 29520193 CAT4-0456)
07/15/93	Underground oil storage tanks for the CS Diesel was identified as leaking	(NTS 29520193 CAT4-0586)
11/18/93	A temporary hose rupture in the Circ. Water bay resulted in the spill of sodium hypochlorite.	(NTS 29520193 CAT4-1339)
11/18/93	Mercury spill occurred outside the Unit 2 Turbine Building trackway adjacent to the Unit 2 Main Transformer on the asphalt.	NTS 30420193 CAT4-1340)
11/27/93	Overflow of the Unit 1 ABEDT resulted in back-leakage into the Containment Spray and Safety Injection cubicles in the Auxiliary Building	(NTS 29520193 CAT4-1381)
12/04/93	Contaminated scaffolding with readings up to 20K dpm fixed was used to repair the temporary Auxiliary Boiler located outside of the Service Building.	(NTS 29520093 CAT3-202)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
01/06/94	A mercury spill occurred while removing a hydrogen purity meter. The material spilled at the base of the panel on 617 ft. elevation of the Unit 1 Turbine Building. While transporting the device to the IM shop, mercury was spilled on the Turbine Building elevator, on the 592 ft. elevation outside the elevator, inside and outside of the MM shop, outside of the north Service Building elevator, hallway to the IM shop, and in the IM shop. All areas were cleaned and released.	(NTS 29520194 CAT4-0019)
01/15/94	Oil was observed leaking from the 1E and 1W Main Transformers.	(NTS 29520194 CAT4-0110)
01/17/94	Oil spill from the Unit 1 Turbine Oil reservoir seeped through the floor to the 560' elevation of the Turbine Building	(NTS-29520194 CAT4-0111)
02/16/94	Contaminated material found in decontamination technician trailer	(NTS 29520094 CAT3-038)
03/16/94	A contractor laborer was found to have contaminated clothing with readings up to 24,000 dpm fixed in a hotel room off-site. The hotel room and vehicle were surveyed and not contaminated.	(NTS 29510094008, NTS 29520094 CAT3-074 and NRC IR 94-08/94-08)
04/03/94	A fire occurred in the Unit 1 Main Generator Lead Box which subsequently cause a trip of the Unit 1 Reactor	(LER 1-94-005, NRC IR 94-09/94-09 and NRC IR 94-12/94-12)
03/07/94	A failure of the "2B" Diesel Generator was attributed to zebra mussels in the lube oil and jacket water coolers.	(LER 2-94-002)
03/22/94	Two abandoned transformers were found in the owner-controlled fields west of the facility. One had fallen over and was leaking oil onto the ground. The oil was tested and did not contain PCBs.	(NTS 29520194 CAT4-0790)
05/31/94	While performing a flush of the Turbine Oil seals, approximately 200 gallons of turbine oil overflowed into the Unit 1 Main Generator	(NTS 29520194 CAT4-1206)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
07/02/94	A fire occurred in the three bus duct phases for Unit 1 Main Generator. The fire resulted in a shutdown of Unit 1 Reactor.	(LER 1-94-010 and NRC IR 94-17/94-17)
08/19/94	Oil was observed leaking from the #9 cooler on the 1W Main Transformer.	(NTS 29520194 CAT4-1844)
08/24/94	The SFP overflowed, resulting in the spill of approximately 1,250 gallons of borated water through the Unit 1 Containment Spray pump and the Unit 1 Safety Injection pump bedplates.	(NRC IR 94-19/94-19 and (NTS 29520094 CAT3-204)
09/03/94	A contaminated cart with readings up to 55,000 dpm fixed was discovered in the Radiation Protection office.	(NTS 29520094 CAT3-211)
09/06/94	Oil was observed leaking from the #7 cooler on the 1W Main Transformer.	(NTS 29520194 CAT4-1985)
03/09/95	Contaminated tool with fixed contamination up to 53,000 dpm fixed found in the MM tool crib	(NTS29520095 CAT3-044)
04/1995	"Cleansweep Project" implemented at Zion site. Project implemented the comprehensive survey of site facilities and grounds to identify uncontrolled radioactive material. The following findings were identified; The survey was conducted between 04/25/95 through 05/04/95. 61 items found with fixed contamination up to 80,000 dpm direct, and 3 items with smearable contamination up to 6,000 dpm/100cm2. Eleven items were discovered in the Bechtel warehouse, one in the east MM training area, one in the decontamination trainer, one in the paint shed and one item on the 4th floor of the Service Building	(NTS29520095 CAT3-044)
04/05/95	The Hydrazine Addition Tank overflowed approximately 6 gallons of 35% solution hydrazine to the floor in the Turbine Building during Auxiliary Boiler layup.	(NTS 29520095 CAT3-071)
04/21/95	Spill of caustic liquid in the Unit 1 Containment Spray pump cubicle.	(NTS 29520195 CAT4-1310)
04/29/95	Resin spill in the "2A" De-Borating demineralizer cubicle left approximately 6 inches of spent resin on the floor of the cubicle.	(Zion RP/Decon Log)
05/10/95	Contaminated materials discovered in the IM Shop	(NTS 29520095 CAT3-090)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
06/12/95	A mercury spill occurred inside the Unit 1 Screen Wash panel in the Crib House.	(NTS 29520195 CAT4-1629)
08/15/95	Contamination discovered on the internals of the Auxiliary Boiler	(NTS 29520195 CAT4-2208
09/11/95 to 10/25/95	Contaminated resin (up to 7.15E^{-7} uCi/g Co-60 and 4.94E^{-7} uCi/g Cs-137) found in 3 barrels of used resin from non-rad systems (only one barrel was identified as Unit 1 stator water). Additional contaminated resin was also found in cardboard drums in the same area.	(NTS 29520095 CAT3-160 and NTS29520195 CAT4-3148)
09/22/95	Fire in the Unit 1 Refueling Cavity	(NTS 29520195 CAT4-2663)
11/13/95	Diesel fuel spill on the floors of both Unit 1 and Unit 2 Containment Spray pump cubicles	(Zion RP/Decon Log and NTS 29520195 CAT4-3351)
12/04/95	Contaminated tools and rigging slings with fixed contamination up to 109,000 dpm fixed was discovered in the MM tool crib.	(NTS 29520095 CAT3-219)
01/22/96	The Lake Discharge Tank (LDT) overflowed, resulting in the contamination of approximately 11,000 ft^2 of the Auxiliary Building 542 ft. elevation being contaminated.	(Zion RP/Decon Log)
01/22/96	Three contaminated clipboards with readings up to 31,000 dpm fixed were discovered in the Chemistry Offices.	(NTS 29520096 CAT3-010)
02/02/96	Approximately 100 gallons of hydrazine spilled near the "2D" Condensate and Condensate Booster Pump	(NTS 30420196 CAT4-0296)
02/18/96	Approximately 1 gallon of hydrazine and approximately 100 gallons of an ammonia/hydrazine mixture spilled on the 560 ft. elevation of the Unit 2 Turbine Building.	(PIF 96-4846 and spill log 96-007)
04/15/96	Barrels which contained oil cleaned up from a spill of Turbine oil was knocked over by high winds, resulting in a spill of oil to the concrete storage pad.	(NTS 29520196 CAT4-0850)
04/20/96	The "0B" LDT overflowed, resulting in approximately 560 gallons of slightly contaminated water discharged to the Auxiliary Building 542 ft. elevation floor.	(NRC IR 96-07/96-07)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
05/22/96	Contaminated ISI block reading 2,000 dpm fixed discovered on the Turbine Building 617 ft. elevation by the generator seal oil unit.	(NTS 29520096 CAT2-008)
06/27/96	Five clothing bins with fixed contamination were discovered stored in an un-posted area on the 592 ft. elevation of Unit 2 Turbine Building.	(NTS 2950096 CAT2008 and PIF 96-1405)
07/15/96	The sludge removed from the Unit 1 and Unit 2 Oil Separators was discovered to be contaminated.	(Zion RP/Decon Log)
07/18/96	An unspecified volume of hydraulic oil was spilled into the Spent Fuel transfer canal.	(PIF 96-1670)
07/27/96	Radiological survey of Turbine Building identified fixed contamination up to 3,800 dpm direct on the MS Stop valve	(Zion RP/Decon Log)
08/17/96	The "0B" LDT overflowed, resulting in approximately 7,000 gallons of slightly contaminated water discharged to the Auxiliary Building 542 ft. elevation floor.	(NRC IR 96-10/96-10)
08/24/96	Approximately 3,000 gallons of demineralized water was inadvertently sprayed into the Unit 2 Containment 568 ft. elevation.	(NRC IR 96-14/96-14)
10/01/96	Scaffolding, delivered to the Zion site was discovered to be contaminated with readings up to 100,000 dpm fixed on various pieces.	(PIF 96-2936)
10/12/96 to 12/06/96	An out-of-service component (2RH-8734A) resulted in the spill of approximately 400 gallons of contaminated water to the floor of the Auxiliary Building 542 ft. elevation floor.	(NRC IR 96-17/96-17)
10/17/96	A spill of approximately 10 gallons of oil occurred on the "2C" RCP pump deck and on the 568 ft. elevation under the pump.	(PIF 96-3505)
10/23/96	An oil leak was identified from the #1 cooler on the 2E Main Transformer.	(PIF 96-3672)
10/25/96	Contaminated scaffolding with readings up to 2,000 dpm fixed was discovered adjacent to the missile doors to the Auxiliary Building near Unit 2 Containment.	(PIF 96-3724)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
10/28/96	Received a notice of violation from the North Shore Sanitary District for discharges with too high levels of Total Suspended Solids and ammonia. Additional violations for same parameters were identified on 11/20/1996.	(PIF 96-4585 and PIF 96-4994)
10/30/96	Spill of approximately 30 gallons of ethylene glycol occurred in the Auxiliary Building heating coil room.	(PIF 96-3816)
11/01/96	Miscellaneous tools with fixed contamination were discovered in the IM Shop on the Service Building 592 ft. elevation.	(PIF 96-3954 and PIF 97-0310)
11/16/96	Approximately 1,000 gallons of turbine oil was spilled into the Fire Sump and pumped to the WWTF	(PIF 96-4665)
11/18/96	Spill of approximately 10 gallons of ethylene glycol occurred in the Auxiliary Building heating coil supply plenum room.	(Spill incident log 96-005)
11/19/96	Approximately 145 gallons of glycol spilled in the bermed area above Unit 2 VCT valve aisle.	(Spill incident log 96-006)
11/19/96	Contaminated material was discovered in the IM Shop	(PIF 96-4351, NTS 29520096 CAT2-0802)
12/06/96	Approximately 100 gallons of turbine oil was spilled into the Fire Sump and pumped to the WWTF	(PIF 96-4264)
12/25/96	Spill of ethylene glycol on the 617 ft. and 642 ft. elevations of the Auxiliary Building	(PIF 96-4986)
01/29/97	Approximately 27 gallons of 40% NaOH from the CS system entered the Unit 2 RWST during CS pump testing.	(PIF 97-0497)
02/01/97	Approximately 1 gallon of diesel fuel spilled at the CS Day Tank fill location.	(PIF 97-0672 and Spill/Incident Log 97-006)
02/24/97	Spill of approximately 1.5 gallons of hydrazine on the Unit 1 Turbine Building 560 ft. elevation.	(PIF 97-0997 and NTS 29520097SCAQ 0997
03/21/97	A LSA barrel containing non-contaminated sand and water was discovered outside the RCA.	(PIF 97-1461)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
03/29/97	A mercury spill (~ 7 grams) occurred in the Crib House by the Unit 1 Traveling Screens	(Spill/Incident Log 97-006)
04/02/97	A grass fire occurred in the open land areas adjacent to the reactor site.	(NTS 29520197 CAQS-1632)
04/03/97	During semi-annual air quality checks, found 2 Service Air valves (on the 560' Turbine Building E-23 and F-8) leaking considerable amount of oil/water mixture.	(NTS 29520197 CAQD-2143 and PIF 97-2143)
04/08/97	Individual left site after successfully passing Gate House portal monitor. Upon return to the site, a single particle reading 110,000 dpm contact was found in pants pocket.	(NTS 29520097 SCAQ-1725 and PIF 97-1725)
04/12/97	The Fuel Handling Building roof leaked rainwater into the SFP	(NTS 29520197 CAQS-1810)
04/18/97	An empty LSA barrel was discovered in a dumpster north of the Construction Building.	(PIF 97-1928 & NTS 29520097 SCAQ-1928)
04/22/97	Four rusty 55 gallon drums and one 30 gallon drum were found on property east of the BAT warehouse and south of Shiloh Blvd. The barrels were in poor condition and contained a black non-radioactive solid.	(PIF 97-2040)
04/23/97	A spill of approximately 10 to 15 gallons of hydraulic oil spilled in the Unit 2 A/C MSIV room	(Spill/Incident Log 97-007)
05/01/97	An oil spill of approximately 15 gallons occurred on the ground south of the Crib House south doors. The area covered approximately 4 feet by 10 feet. Some of the material went into the storm drains.	(NTS 29520197 CAQD-2166 and PIF 97-2166)
05/03/97	A contaminated posting stanchion with a reading of 135,000 dpm fixed was discovered on the Unit 1 Turbine Building 617 ft. elevation	(NTS 29520097 SCAQ-2192 and PIF 97-2192)
05/06/97	A contaminated stanchion (40K dpm fixed) was found in the contractor fab shop.	(NTS 29520197 CAQD-2298 and PIF 97-2298)

Table 2-2 Historical Incidents/Occurrence (continued)

Date	Incident/Occurrence	Citation
05/07/97	A contaminated stanchion (10K dpm fixed) was found in the contractor fab shop.	(NTS 29520197 CAQD-2297 and PIF 97-2297)
05/14/97	Contaminated materials with readings up to 10,000 dpm fixed was discovered in the West Training Center	(NTS 29520197 CAQD-2407)
05/24/97	Flash fire in the Unit 1 Refueling Cavity during the cleaning of stud plugs	(NTS 29520097 SCAQ-2561, PIF 97-2561, and Tech Alert #97-21)
05/27/97	A spill of approximately 5 gallons of diesel fuel occurred in the Auxiliary Boiler Storage Tank area.	(NTS 29520197 CAQD-2577 and PIF 97-2577)
07/26/97	Contaminated materials (up to 1,600 dpm fixed) found in a dumpster. The material is believed to have originated from work on the Auxiliary Boiler	(NTS 29520197 CAQD-0536 and PIF Z1997-00906
08/05/97	A spill of sodium hypochlorite occurred on concrete by hypochlorite tank.	(Spill incident log 97-009)
09/30/97	A spill of approximately 1 gallon of sodium hypochlorite occurred on concrete by hypochlorite tank.	(Spill incident log 97-010)
10/09/97	A spill of approximately 10 gallons of transformer oil occurred near the north gate on gravel.	(Spill incident log 97-011)
10/17/97	A spill of approximately 3 gallons of diesel fuel occurred in the Turbine Building	(Spill incident log 97-013)
12/11/97	Approximately 13 grams of mercury was spilled in the Cold Lab.	(Spill/Incident Log 97-016)
12/18/97	An oil spill of approximately 1 quart occurred into the southwest corner of the fore-bay.	(Spill/Incident Log 97-017)
12/11/98	A small diesel oil spill occurred outside the Gate House	(PIF 96-4731)

Table 2-3 Initial List of Structural Survey Units

Survey Unit ID #	Survey Unit Description
01000	Unit 1 Containment (All Interior Surfaces are Class 1)
100	568 Ft Elevation
01	Vessel Bio-Shield
02	Inside Missile Barrier – "A" Loop
03	Inside Missile Barrier – "B" Loop
04	Inside Missile Barrier – "C" Loop
05	Inside Missile Barrier – "D" Loop
06	Outside Missile Barrier – "A" Loop
07	Outside Missile Barrier – "B" Loop
08	Outside Missile Barrier – "C" Loop
09	Outside Missile Barrier – "D" Loop
10	Incore Area 543 Ft Elevation
200	592 Ft and 603 Ft Elevation
01	Cavity Area
02	Inside Missile Barrier – "A" Loop
03	Inside Missile Barrier – "B" Loop
04	Inside Missile Barrier – "C" Loop
05	Inside Missile Barrier – "D" Loop
06	Outside Missile Barrier – "A" Loop
07	Outside Missile Barrier – "B" Loop
08	Outside Missile Barrier – "C" Loop
09	Outside Missile Barrier – "D" Loop
10	Incore Drive Area/Seal Table Room 603 Ft Elevation
300	617 Ft Elevation
01	Cavity Area
02	"A" and "C" Loop Area

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
03	"B" and "D" Loop Area
04	Outside Missile Barrier – "A" Loop
05	Outside Missile Barrier – "B" Loop
06	Outside Missile Barrier – "C" Loop
07	Outside Missile Barrier – "D" Loop
400	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
02000	Unit 2 Containment (All Interior Surfaces are Class 1)
100	568 Ft Elevation
01	Vessel Bio-Shield
02	Inside Missile Barrier – "A" Loop
03	Inside Missile Barrier – "B" Loop
04	Inside Missile Barrier – "C" Loop
05	Inside Missile Barrier – "D" Loop
06	Outside Missile Barrier – "A" Loop
07	Outside Missile Barrier – "B" Loop
08	Outside Missile Barrier – "C" Loop
09	Outside Missile Barrier – "D" Loop
10	Incore Area 543 Ft Elevation
200	592 Ft and 603 Ft Elevation
01	Cavity Area
02	Inside Missile Barrier – "A" Loop
03	Inside Missile Barrier – "B" Loop
04	Inside Missile Barrier – "C" Loop
05	Inside Missile Barrier – "D" Loop

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
06	Outside Missile Barrier – "A" Loop
07	Outside Missile Barrier – "B" Loop
08	Outside Missile Barrier – "C" Loop
09	Outside Missile Barrier – "D" Loop
10	Incore Drive Area/Seal Table Room 603 Ft Elevation
300	617 Ft Elevation
01	Cavity Area
02	"A" and "C" Loop Area
03	"B" and "D" Loop Area
04	Outside Missile Barrier – "A" Loop
05	Outside Missile Barrier – "B" Loop
06	Outside Missile Barrier – "C" Loop
07	Outside Missile Barrier – "D" Loop
400	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
03000	Fuel Handling Building (All Interior Surfaces are Class 1)
100	592 Ft Elevation
01	Trackway/Cars Shed/Open Area
02	Spent Fuel Pit Heat Exchangers
03	Spent Fuel Pit Pump Rooms
04	Decon Pit
200	617 Ft Elevation
01	New Fuel Storage Area and 602 Ft Elevation Mezzanine
02	Spent Fuel Pool
03	Cross-town Area and SPING Gallery

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
300	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
04000	Radwaste Building (All Interior Surfaces are Class 1)
100	579 Ft Elevation
01	Crystallizer
200	592 Ft Elevation
01	Dry Active Waste Storage Area
02	Drum Fill Area
03	Radwaste Annex Truck Loading Zone
300	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
05000	Auxiliary Building (All Interior Surfaces are Class 1)
100	542 Ft Elevation
01	Unit 1A RHR Pump Room
02	Unit 1B RHR Pump Room
03	Unit 2A RHR Pump Room
04	Unit 2B RHR Pump Room
05	Unit 1 Pipe Chase
06	Unit 2 Pipe Chase
07	Hold Up Tanks
08	Central Area
09	South Area
10	North Area
11	West Area

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
12	Gas Decay Tanks
13	Unit 1 Equipment Drain Collection Tank
14	Unit 2 Equipment Drain Collection Tank
15	HUT Recirculation and BAE Feed Pumps
16	Aux Building Equipment Drain Tank and Pumps
17	Aux Building Sump A
18	Aux Building Sump B
200	560 Ft Elevation
01	Unit 1A Safety Injection Pump
02	Unit 2A Safety Injection Pump
03	Unit 1B Safety Injection Pump
04	Unit 2B Safety Injection Pump
05	Unit 1A RHR Heat Exchanger
06	Unit 2A RHR Heat Exchanger
07	Unit 1B RHR Heat Exchanger
08	Unit 2B RHR Heat Exchanger
09	Unit 1 Containment Spray Pump
10	Unit 2 Containment Spray Pump
11	South Area
12	Central Area
13	North Area
14	West Area
15	Equipment Drain Tank
16	Chemical Drain Tank
17	Unit 1 Vertical Pipe Chase
18	Unit 2 Vertical Pipe Chase

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
19	Blowdown Heat Exchangers
20	Waste Gas Compressor Area
21	Unit 1 RHR Heat Exchanger Pipe Chase
22	Unit 2 RHR Heat Exchanger Pipe Chase
300	579 Ft Elevation
01	Unit 1A Centrifugal Charging Pump
02	Unit 2A Centrifugal Charging Pump
03	Unit 1B Centrifugal Charging Pump
04	Unit 2B Centrifugal Charging Pump
05	Unit 1C Reciprocating Charging Pump
06	Unit 2C Reciprocating Charging Pump
07	Unit 1 Refueling Water Storage Tank
08	Unit 2 Refueling Water Storage Tank
09	South Area
10	Central Area
11	North Area
12	West Area
13	Unit 1 Horizontal Pipe Chase
14	Unit 2 Horizontal Pipe Chase
15	#1 HUT
16	#0 HUT
17	#2 HUT
18	Anion/Cation Valve Aisle
19	Unit 1 Vertical Pipe Chase
20	Unit 2 Vertical Pipe Chase
400	592 Ft Elevation
01	Unit 1 Letdown Heat Exchanger

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
02	Unit 2 Letdown Heat Exchanger
03	Unit 1 Seal Water Heat Exchanger
04	Unit 2 Seal Water Heat Exchanger
05	Unit 2 Future Diesel Generator
06	Primary Sample Room
07	Calibration Facility
08	South Area
09	Central Area
10	North Area
11	West Area
12	Unit 1 Vertical Pipe Chase
13	Unit 2 Vertical Pipe Chase
14	Unit 1 BAT and Pumps
15	Unit 2 BAT and Pumps
500	617 Ft Elevation
01	Unit 1 Volume Control Tank
02	Unit 2 Volume Control Tank
03	Unit 1 Pipe Chase
04	Unit 2 Chase
05	Radiation Protection Offices
06	Unit 1 Containment Purge Air Equipment Room
07	Unit 2 Containment Purge Air Equipment Room
08	Tool Crib
09	Waste Evaporator Monitor Tank
10	Radwaste Evaporator
11	Spent Resin Storage Area
12	Laboratory

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
13	West Area
14	East Area
600	630 Ft Elevation
01	Unit 1 Cable Spreading Room
02	Unit 2 Cable Spreading Room
03	Unit 1 Cable Penetration Room
04	Unit 2 Cable Penetration Room
700	642 Ft Elevation
01	642 Ft Elevation
800	654 Ft Elevation
01	654 Ft Elevation
900	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
06000	Turbine Building (All Interior Surfaces are Class 2)
100	560 Ft Elevation
01	Oil Room
02	South Area
03	Central Area
04	North Area
200	570 Ft Elevation
01	Unit 1 Diesel Fuel Oil Storage 1B
02	Unit 2 Diesel Fuel Oil Storage 2B
03	Unit 1 Diesel Fuel Oil Storage 1A
04	Unit 2 Diesel Fuel Oil Storage 2A
05	Unit 1 Diesel Fuel Oil Storage 0A
06	Unit 2 Future Diesel Fuel Oil Storage 0B

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
07	Unit 1 Steam Pipe Tunnel
08	Unit 2 Steam Pipe Tunnel
300	592 Ft Elevation
01	Unit 1 Diesel Generator 1A
02	Unit 2 Diesel Generator 1B
03	Unit 1 Diesel Generator 0
04	Unit 2 Diesel Generator 2A
05	Unit 1 Diesel Generator 2B
06	Battery and Charger Room
07	Technical Support Center
08	Secondary Sample Room
09	Auxiliary Heating Boiler
10	South Area
11	Central Area
12	North Area
400	609 Ft Elevation
01	609 Ft Elevation
500	617 Ft Elevation
01	Switchgear Rooms 17, 18 &19
02	Switchgear Rooms 27, 28 &29
03	Air Conditioning Equipment Room
04	QA Records Area
05	South Area
06	Central Area
07	North Area
600	625 Ft Elevation
01	625 Ft Elevation

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
700	642 Ft Elevation
01	Unit 1 Nonessential Switchgear and Rod Drive MG Set
02	Unit 2 Nonessential Switchgear and Rod Drive MG Set
03	Main Control Room Complex
04	Control Room Annex South
05	Control Room Annex North
06	Unit 1 Auxiliary Electric Equipment Room
07	Unit 2 Auxiliary Electric Equipment Room
08	South Area
09	Central Area
10	North Area
800	656 Ft Elevation
01	656 Ft Elevation
900	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
07000	Service Building (All Interior Surfaces are Class 2)
100	592 Ft Elevation
01	East
02	West
200	608 Ft Elevation
01	East
02	West
300	624 Ft Elevation
01	East
02	West
400	636 Ft Elevation

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
01	East
02	West
500	648 Ft Elevation
01	East
600	660 Ft Elevation
01	East
700	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	East Roof
02	West Roof
03	Exterior Walls
08000	Crib House (All Interior Surfaces are Class 2)
100	552 Ft Elevation
01	552 Ft Elevation
200	594 Ft Elevation
01	594 Ft Elevation
300	Exterior Surfaces & Roof (All Exterior Surfaces are Class 2)
01	Roof
02	Exterior Walls
09000	Outbuildings
100	WWTF
01	1st Floor Interior Surfaces (Class 2)
02	2nd Floor Interior Surfaces (Class 2)
03	Roof (Class 2)
04	Exterior Surfaces (Class 2)
200	North Valve House
01	1st Floor Interior Surfaces (Class 2)
02	Roof (Class 2)

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
03	Exterior Surfaces (Class 2)
300	North Chlorination/Dechlorination Building
01	1st Floor Interior Surfaces (Class 2)
02	Roof (Class 2)
03	Exterior Surfaces (Class 2)
400	Warehouse/Mechanical Maintenance Training Center
01	1st Floor Interior Surfaces (Class 3)
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
500	Interim Radwaste Storage Facility (IRSF) Building
01	1st Floor Interior Surfaces (Class 2)
02	Mezzanine (Class 2)
03	Roof (Class 2)
04	Exterior Surfaces (Class 2)
600	Fire Training Building
01	1st Floor Interior Surfaces (Class 3)
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
700	Contractor Break Building
01	1st Floor Interior Surfaces (Class 3)
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
800	East Training Center
01	1st Floor Interior Surfaces (Class 3)
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
900	In-Processing Building

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
01	1st Floor Interior Surfaces (Class 3)
02	2nd Floor Interior Surfaces (Class 3)
03	Roof (Class 3)
04	Exterior Surfaces (Class 3)
110	Station Construction Building
01	1st Floor Interior Surfaces (Class 3)
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
120	Illinois Department of Nuclear Safety (IDNS) Building
01	1st Floor Interior Surfaces (Class 2)
02	Roof (Class 2)
03	Exterior Surfaces (Class 2)
130	Security Offices
01	1st Floor Interior Surfaces (Class 2)
02	Roof (Class 2)
03	Exterior Surfaces (Class 2)
140	Gate House
01	1st Floor Interior Surfaces (Class 2)
02	2nd Floor Interior Surfaces (Class 2)
03	Roof (Class 2)
04	Exterior Surfaces (Class 2)
150	North Warehouse
01	1st Floor Interior Surfaces (Class 3)
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
160	South Warehouse
01	1st Floor Interior Surfaces (Class 3)

Table 2-3 Initial List of Structural Survey Units (continued)

Survey Unit ID #	Survey Unit Description
02	Roof (Class 3)
03	Exterior Surfaces (Class 3)
170	South Valve House
01	1st Floor Interior Surfaces (Class 2)
02	Roof (Class 2)
03	Exterior Surfaces (Class 2)
180	South Chlorination/Dechlorination Building
01	1st Floor Interior Surfaces (Class 2)
02	Roof (Class 2)
03	Exterior Surfaces (Class 2)
190	West Training Building
01	1st Floor Interior Surfaces (Class 3)
02	2nd Floor Interior Surfaces (Class 3)
03	Roof (Class 3)
04	Exterior Surfaces (Class 3)

Table 2-4 Initial List of Open Land Survey Units

Survey Unit ID #	Survey Unit Description	Initial Classification	Approximate Survey Unit Area (m^2)
12000	Security Restricted Area Grounds		
12101	WWTF Sludge Drying Bed Area	Class 1	2,036
12102	WWTF Facility	Class 1	2,024
12103	Unit 2 PWST/SST Area	Class 1	2,034
12104	Area Around the North Half of Unit 2 Containment	Class 1	1,940
12105	Area Around the South Half of Unit 2 Containment	Class 1	1,938
12106	Area Around the North Half of Fuel Handling and Auxiliary Buildings	Class 1	1,936
12107	Area Around the South Half of Fuel Handling and Auxiliary Buildings	Class 1	1,934
12108	Area Around the North Half of Unit 1 Containment	Class 1	1,932
12109	Area Around the South Half of Unit 1 Containment	Class 1	1,931
12110	Yard Between Unit 1 Containment and Turbine	Class 1	1,740
12111	South Yard Area Northeast of Gate House	Class 1	1,964
12112	Unit 1 PWST/SST Area West	Class 1	1,658
12113	Unit 1 PWST/SST Area East	Class 1	1,693
12201	North Protected Area Yard	Class 2	9,610
12202	Gate House and Southwest Yard	Class 2	7,562
12203	Soils Around the Service Building and South East Yard	Class 2	7,569
12204	Crib House Area	Class 2	5,909
12205	Area Around the Turbine Building	Class 2	9,085
10200	Radiological Restricted Area Grounds		
10201	NE Corner of Restricted Area - Lakeshore	Class 3	8,530
10202	IRSF/Fire Training Area	Class 3	7,799
10203	East Training Area	Class 3	11,761
10204	North Gate Area	Class 3	7,230
10205	Switchyard	Class 3	55,432

Table 2-4 Initial List of Open Land Survey Units (continued)

Survey Unit ID #	Survey Unit Description	Initial Classification	Approximate Survey Unit Area (m^2)
10206	Station Construction Area	Class 3	10,539
10207	North Warehouse Area	Class 3	10,284
10208	South Warehouse Area	Class 3	12,381
10209	Restricted Area South of Gate House	Class 3	5,664
10210	Restricted Area South of Turbine Building	Class 3	5,048
10211	Southeast Corner of Restricted Area - Lakeshore	Class 3	6,472
10212	North Restricted Area Fence line - Lakeshore	Class 3	36,714
10213	Power House Area	Class 3	44,471
10214	Construction Parking Area	Class 3	29,681
10215	Area Northwest of Switchyard	Non-Impacted	26,008
10216	Area West Northwest of Switchyard	Non-Impacted	31,171
10217	Area Southwest of Switchyard	Non-Impacted	45,984
10218	Area Near South of Switchyard	Class 3	17,822
10219	Area Far South of Switchyard	Class 3	12,185
10220	Adjacent to South Restricted Area - lakeshore	Class 3	46,964
10221	Owner Controlled Area South of Restricted Area	Class 3	27,297
10300	Owner Controlled Area Grounds (included in Final Safety Analysis Report [FSAR])		
10301	West Training Area	Class 3	55,942
10302	Northwest Corner of FSAR Area	Non-Impacted	35,162
10303	Southwest Corner of FSAR Area	Non-Impacted	61,955
10304	Southern Area of FSAR	Non-Impacted	34,387
10305	Area West of Survey Unit #10217	Non-Impacted	121,535
10306	Area West of Survey Unit #10216	Non-Impacted	85,268
10400	Owner Controlled Area Grounds (not included in FSAR)		
10401	Northeast Corner of Owner Controlled Property	Non-Impacted	76,118
10402	MET Tower Area	Non-Impacted	107,618

Table 2-4 Initial List of Open Land Survey Units (continued)

Survey Unit ID #	Survey Unit Description	Initial Classification	Approximate Survey Unit Area (m^2)
10403	Area North of West Training	Non-Impacted	139,282
10404	Northwest Corner of Owner Controlled Property	Non-Impacted	100,075

Table 2-5 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.

Table 2-6 Instrument Types and Nominal MDC

Detector Model [b]	Meter Model	Application	Typical Detection Sensitivity	
			MDC_{scan} (dpm/100cm^2)	MDC_{static} [a] (dpm/100cm^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 (Reference 2-32).
b Functional equivalent instrumentation may be used
c MDC Requirements per Regulatory Guide 4.8 (Reference 2-33)

Table 2-7 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

Table 2-8 Energy*Solutions* Background Study Results

Asphalt Results					
Measurements	Results in dpm/100cm2				
	Min	Max	Average	St. Dev	95% UCL[a]
25	101	393	257	77	284

a 95% Upper Confidence Level on the average based on 25 measurements or 24 (n-1) degrees of freedom.

Nuclide	Results in pCi/g		
	Average	Std. Dev	95% UCL[a]
K-40	**4.64E+00**	**9.26E-01**	**5.06E+00**
Cs-137	*1.23E-03*	*1.05E-02*	*6.02E-03*
Tl-208	**1.57E-01**	**5.56E-02**	**1.82E-01**
Pb-212	**1.77E-01**	**3.81E-02**	**1.95E-01**
Bi-214	**1.19E-01**	**4.01E-02**	**1.37E-01**
Pb-214	**1.55E-01**	**3.68E-02**	**1.72E-01**
Ac-228	**1.60E-01**	**6.55E-02**	**1.90E-01**
Th-234	*8.60E-02*	*3.65E-01*	*2.52E-01*
U-235	*3.51E-03*	*4.11E-02*	*2.22E-02*

a 95% Upper Confidence Level on the average based on 15 measurements or 14 (n-1) degrees of freedom.
b Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Concrete Results					
Measurements	Results in dpm/100cm2				
	Min	Max	Average	St. Dev	95% UCL[a]
25	141	541	356	100	391

a 95% Upper Confidence Level on the average based on 25 measurements or 24 (n-1) degrees of freedom.

Nuclide	Results in pCi/g		
	Average	Std. Dev	95% UCL[a]
K-40	**5.55E+00**	**2.63E+00**	**6.74E+00**
Cs-137	*1.21E-02*	*1.82E-02*	*2.03E-02*
Tl-208	**2.60E-01**	**8.74E-02**	**3.00E-01**
Pb-212	**3.01E-01**	**1.15E-01**	**3.54E-01**
Bi-214	**3.00E-01**	**6.51E-02**	**3.30E-01**
Pb-214	**3.03E-01**	**6.57E-02**	**3.32E-01**
Ac-228	**3.08E-01**	**1.03E-01**	**3.55E-01**
Th-234	*3.48E-01*	*3.27E-01*	*4.96E-01*
U-235	*8.78E-02*	*9.81E-02*	*1.32E-01*

a 95% Upper Confidence Level on the average based on 15 measurements or 14 (n-1) degrees of freedom.
b Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Table 2-8 Energy*Solutions* Background Study Results (continued)

Surface Soil Results					
Surface Scanning					
Measurements	Results in cpm (gross)				
	Min	Max	Average	St. Dev	95% UCL
8,955	2,520	9,240	5,192	832	5,206
Static Measurements					
Measurements	Results in cpm (gross)				
	Min	Max	Average	St. Dev	95% UCL[a]
30	4,530	6,362	5,718	543	5,886

a 95% Upper Confidence Level on the average based on 30 measurements or 29 (n-1) degrees of freedom.

Nuclide	Results in pCi/g		
	Average	Std. Dev	95% UCL[a]
K-40	**1.40E+01**	**2.00E+00**	**1.49E+01**
Cs-137	*6.61E-03*	*2.15E-02*	*1.64E-02*
Tl-208	**4.45E-01**	**8.78E-02**	**4.85E-01**
Pb-212	**5.30E-01**	**8.94E-02**	**5.70E-01**
Bi-214	**5.44E-01**	**1.10E-01**	**5.94E-01**
Pb-214	**5.43E-01**	**9.21E-02**	**5.85E-01**
Ac-228	**5.20E-01**	**1.02E-01**	**5.66E-01**
Th-234	*6.05E-01*	*4.68E-01*	*8.18E-00*
U-235	*9.83E-02*	*1.16E-01*	*1.51E-01*

a 95% Upper Confidence Level on the average based on 15 measurements or 14 (n-1) degrees of freedom.
b Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Sub-Surface Soil Results

Nuclide	Results in pCi/g		
	Average	Std. Dev	95% UCL[a]
K-40	**1.36E+01**	**2.89E+00**	**1.49E+01**
Cs-137	*-1.95E-03*	*1.83E-02*	*6.35E-03*
Tl-208	**4.42E-01**	**9.56E-02**	**4.85E-01**
Pb-212	**4.98E-01**	**1.68E-01**	**5.74E-01**
Bi-214	**6.06E-01**	**1.76E-01**	**6.86E-01**
Pb-214	**6.10E-01**	**1.92E-01**	**6.97E-01**
Ac-228	**4.87E-01**	**1.14E-01**	**5.39E-01**
Th-234	*5.74E-01*	*4.87E-01*	*7.95E-00*
U-235	*5.26E-02*	*7.70E-02*	*8.76E-02*

a 95% Upper Confidence Level on the average based on 15 measurements or 14 (n-1) degrees of freedom.
b Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Table 2-9 Crib House Concrete Core Sample Analysis Results

Concrete Core Sample Analysis Results			
Nuclide	Results in pCi/g		
	Average	Max	Std. Dev
K-40	**1.39E+01**	**2.18E+01**	**2.86E+00**
Co-60	*1.20E-01*	*1.44E-01*	*1.84E-02*
Cs-137	*7.00E-02*	*9.24E-02*	*1.07E-02*
Bi-212	**1.82E-01**	**1.90E-01**	**1.20E-02**
Pb-212	**2.38E-01**	**4.20E-01**	**6.13E-02**
Bi-214	**4.00E-01**	**6.53E-01**	**9.16E-02**
Pb-214	**4.36E-01**	**7.07E-01**	**1.30E-01**
Ra-226	**1.46E+00**	**2.64E+00**	**4.89E-01**
Ac-228	**6.93E-01**	**4.79E-01**	**9.76E-02**
U-235	**8.30E-02**	**1.60E-01**	**3.59E-01**

a Sample population consisted of 40 samples with 32 taken on the floor and 8 taken on lower walls
b Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Table 2-10 Hosah Park Background Assessment Sample Analysis

Surface Soil Sample Analysis Results

Nuclide	**Results in pCi/g**		
	Average	**Max**	**Std. Dev**
K-40	**6.96E+00**	**8.95E+00**	**9.01E-01**
Co-60	*-3.19E-02*	*2.87E-02*	*1.88E-01*
Sr-90	*-6.03E-04*	*5.26E-02*	*2.41E-02*
Cs-137	**2.11E-01**	**6.51E-01**	**1.48E-01**
Th-228	**1.72E-01**	**4.30E-01**	**1.18E-01**
Th-230	**3.45E-01**	**2.07E+00**	**3.55E-01**
Th-232	**1.53E-01**	**5.11E-01**	**1.04E-01**
U-234	**2.03E-01**	**1.74E+00**	**3.23E-01**
U-235	*1.40E-02*	*1.37E-01*	*3.24E-02*
U-238	*2.11E-01*	*1.86E+00*	*3.24E-01*

Subsurface Soil Sample Analysis Results

Nuclide	**Results in pCi/g**		
	Average	**Max**	**Std. Dev**
K-40	**6.62E+00**	**8.59E+00**	**9.49E-01**
Co-60	*3.72E-04*	*3.79E-02*	*1.38E-02*
Sr-90	*4.40E-03*	*5.30E-02*	*2.16E-02*
Cs-137	**2.64E-02**	**2.41E-01**	**6.00E-02**
Th-228	**1.26E-01**	**4.50E-01**	**1.14E-01**
Th-230	*3.28E-01*	*8.11E-01*	*2.09E-01*
Th-232	*1.21E-01*	*4.05E-01*	*9.54E-02*
U-234	*1.25E-01*	*7.36E-01*	*1.57E-01*
U-235	*1.05E-02*	*1.10E-01*	*2.43E-02*
U-238	*1.31E-01*	*6.65E-01*	*1.47E-01*

a Bold values indicate concentration(s) greater than MDC. Italicized values indicate MDC value.

Table 2-11 Investigative Levels for Cs-137 Based on Background Studies

Condition and Depth	Measured Range (pCi/g)	Range for 95% Distribution (pCi/g)
Drainage Areas Surface 0-10 cm		
Undisturbed	0.00 to 2.80	0.45 to 3.63
Disturbed	0.00 to 1.67	0.35 to 2.86
Non-Drainage Areas Surface 0-10 cm		
Undisturbed	0.23 to 0.66	0.15 to 0.77
Disturbed	0.27 to 0.34	0.23 to 0.42

Table 2-12 Initial Suite of Radionuclides

Radionuclide	Half Life (years)		Radionuclide	Half Life (years)
H-3	1.24 E 01		Pm-147	2.62 E 00
C-14	5.73 E 03		Sm-146	1.03 E 08
Fe-55	2.70 E 00		Sm-151	9.00 E 01
Ni-59	7.50 E 04		Eu-152	1.33 E 01
Co-60	5.27 E 00		Eu-154	8.80 E 00
Ni-63	9.60 E 01		Eu-155	4.96 E 00
Sr-90	2.91 E 01		Pu-238	8.77 E 01
Mo-93	3.50 E 03		Pu-239/240	2.41 E 04
Nb-94	2.03 E 04		Pu-241	1.44 E 01
Tc-99	2.13 E 05		Np-237	2.14 E 06
Sb-125	2.77 E 00		Am-241	4.32 E 02
Cs-134	2.06 E 00		Am-243	7.38 E 03
Cs-137	3.00 E 01		Cm-244	1.81 E 01

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Table 2-13 Nominal Radiological Data – Structural Survey Units

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
01100	Unit 1 CTMT 568 ft. el.	01101	Unit 1 Vessel Bio-Shield	Class 1	(NOT SURVEYED – INACCESSIBLE)							Not Accessible
		01102	A Loop I/S Missile Barrier	Class 1	5	6	548K	2.2M	5200	10000	1A Reactor Coolant Pump	
		01103	B Loop I/S Missile Barrier	Class 1	6	8	226K	500K	19000	40000	1B Reactor Coolant Pump	
		01104	C Loop I/S Missile Barrier	Class 1	4	7	684K	2.3M	4400	10000	1C Reactor Coolant Pump	
		01105	D Loop I/S Missile Barrier	Class 1	5	10	213K	450K	5000	10000	1D Reactor Coolant Pump	
		01106	A Loop O/S Missile Barrier	Class 1	1.3	5	152K	400K	3000	5000		
		01107	B Loop O/S Missile Barrier	Class 1	2	10	368K	1.5M	3000	1000		
		01108	C Loop O/S Missile Barrier	Class 1	0.2	0.4	150K	490K	3000	1000		
		01109	D Loop O/S Missile Barrier	Class 1	1	5	53K	145K	1000	1000		
		01110	543 ft. el. Incore Tunnel	Class 1	5	10	(not taken)		21K	40K	Incore Sump	Under Reactor Vessel
01200	Unit 1 CTMT 592 ft. el.	01201	Cavity Area	Class 1	100	187	(210 mrad)		500K	1M	Unit 1 Reactor	Cavity filled with water
		01202	A Loop I/S Missile Barrier	Class 1	11	20	(not taken)		1600	40K	1A Steam Generator (S/G)	
		01203	B Loop I/S Missile Barrier	Class 1	16	20	(not taken)		1600	40K	1B Steam Generator (S/G)	
		01204	C Loop I/S Missile Barrier	Class 1	11	22	(not taken)		1600	40K	1C Steam Generator (S/G)	
		01205	D Loop I/S Missile Barrier	Class 1	11	18	(not taken)		1600	40K	1D Steam Generator (S/G)	
		01206	A Loop O/S Missile Barrier	Class 1	0.6	1	84K	345K	1600	3000		
		01207	B Loop O/S Missile Barrier	Class 1	0.2	0.3	49K	136K	1200	2000		
		01208	C Loop O/S Missile Barrier	Class 1	0.4	1	83K	198K	1200	2000		
		01209	D Loop O/S Missile Barrier	Class 1	0.2	0.2	21K	46K	2000	5000		
		01210	603 ft. Seal Table/Incore Drives	Class 1	0.5	1.6	359K	780K	4200	5000		HRA Not Accessible
01300	Unit 1 CTMT 617 ft. el.	01301	Cavity Area	Class 1	.06	1	32K	55K	<1000	<1000	Unit 1 Reactor	Upper Area Only
		01302	"A" & "C" Loop Area	Class 1	37	70	(not taken)		<1000	<1000	1A & 1C S/G	
		01303	"B" & "D" Loop Area	Class 1	13	18	(not taken)		<1000	<1000	1B & 1D S/G, Pressurizer	
		01304	A Loop O/S Missile Barrier	Class 1	0.5	0.7	23K	48K	<1000	<1000		
		01305	B Loop O/S Missile Barrier	Class 1	0.5	0.7	13K	28K	1500	2000		
		01306	C Loop O/S Missile Barrier	Class 1	2	10	30K	80K	1000	1500		
		01307	D Loop O/S Missile Barrier	Class 1	0.4	2	80K	300K	1000	1000		
01400	Unit 1 CTMT Exterior	01401	Roof	Class 2	(NOT SURVEYED – INACCESSIBLE)							Not surveyed – safety
		01402	Exterior Walls	Class 2	40	120	225	272	<300	<300		Accessible walls only
02100	Unit 2 CTMT 568 ft. el.	02101	Unit 2 Vessel Bio-Shield	Class 1	(NOT SURVEYED – INACCESSIBLE)							Not Accessible
		02102	A Loop I/S Missile Barrier	Class 1	8	25	1.1M	2.5M	20K	60K	2A Reactor Coolant Pump	
		02103	B Loop I/S Missile Barrier	Class 1	5	10	600K	2.1M	4000	12K	2B Reactor Coolant Pump	
		02104	C Loop I/S Missile Barrier	Class 1	7	30	500K	1.3M	3500	6000	2C Reactor Coolant Pump	
		02105	D Loop I/S Missile Barrier	Class 1	5	20	550K	1.4M	4000	10K	2D Reactor Coolant Pump	

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
Area ID	Area Description	Unit ID	Unit Description				Total		Loose			
					G/A	Max	G/A	Max	G/A	Max		
		02106	A Loop O/S Missile Barrier	Class 1	0.1	1	90K	270K	<1000	1000		
		02107	B Loop O/S Missile Barrier	Class 1	0.3	10	80K	240K	<1000	1000	Rx Coolant Drain Tank	
		02108	C Loop O/S Missile Barrier	Class 1	0.1	0.2	19K	50K	1000	2000		
		02109	D Loop O/S Missile Barrier	Class 1	0.1	0.1	120K	350K	<1000	1000		
		02110	543 ft. el. Incore Tunnel	Class 1	3	6	(not taken)		25K	900K	Incore Sump	Under Reactor Vessel
02200	Unit 2 CTMT 592 ft. el.	02201	Cavity Area	Class 1	40	250	(not taken)		(110 mrad)		Unit 2 Reactor	Cavity filled with water
		02202	A Loop I/S Missile Barrier	Class 1	3	6	14K	40K	<1000	<1000	2A Steam Generator (S/G)	
		02203	B Loop I/S Missile Barrier	Class 1	5	7.5	14K	20K	<1000	<1000	2B Steam Generator (S/G)	
		02204	C Loop I/S Missile Barrier	Class 1	3.2	6	12K	20K	<1000	<1000	2C Steam Generator (S/G)	
		02205	D Loop I/S Missile Barrier	Class 1	4	7.5	24K	80K	<1000	<1000	2D Steam Generator (S/G)	
		02206	A Loop O/S Missile Barrier	Class 1	0.3	0.3	53K	200K	<300	531		
		02207	B Loop O/S Missile Barrier	Class 1	0.2	0.2	9000	40K	<300	506		
		02208	C Loop O/S Missile Barrier	Class 1	0.3	0.3	3000	9000	<300	1000		
		02209	D Loop O/S Missile Barrier	Class 1	0.3	0.6	900	2000	<300	367		
		02210	603 ft. Seal Table/Incore Drives	Class 1	2.6	6	62K	200K	2000	5000		
02300	Unit 2 CTMT 617 ft. el.	02301	Cavity Area	Class 1	40	250	(not taken)		(110 mrad)		Unit 2 Reactor	Cavity filled with water
		02302	"A" & "C" Loop Area	Class 1	0.8	1	4500	7000	2000	5000	2A & 2C S/G	
		02303	"B" & "D" Loop Area	Class 1	1.4	4	2000	3000	<1000	<1000	2B & 2D S/G, Pressurizer	
		02304	A Loop O/S Missile Barrier	Class 1	0.5	0.7	1000	4000	<1000	1000		
		02305	B Loop O/S Missile Barrier	Class 1	0.7	1.6	10000	30000	1000	2000		
		02306	C Loop O/S Missile Barrier	Class 1	0.5	2.2	2000	5000	<1000	<1000		
		02307	D Loop O/S Missile Barrier	Class 1	10	24	10000	30000	<1000	<1000		
02400	Unit 2 CTMT Exterior	02401	Roof	Class 2	(NOT SURVEYED – INACCESSIBLE)							Not surveyed – safety
		02402	Exterior Walls	Class 2	11	15	764	764	<300	<300		Accessible walls only
03100	Fuel Handling Bldg. 592 ft. el.	03101	Trackway/Cars Shed/Open Area	Class 1	0.2	0.2	7900	37000	<300	<300		
		03102	Spent Fuel Pit Heat Exchangers	Class 1	0.3	0.6	<634	1790	<300	<300		
		03103	Spent Fuel Pit Pump Rooms	Class 1	0.3	0.6	8700	34000	<300	<300		
		03104	Decon Pit	Class 1	5.7	25	(not taken)		3000	10000		Hot Work during Survey
03200	Fuel Handling Bldg. 617 ft. el.	03201	New Fuel Storage & Mezzanine	Class 1	0.2	0.4	1254	2276	<300	<300		
		03202	Spent Fuel Pool	Class 1	<0.2	<0.2	2479	5304	<300	<300	Spent Nuclear Fuel	Filled with water
		03203	Cross-town & SPING Gallery	Class 1	<0.2	0.2	8591	29039	<300	569		
03300	Fuel Bldg. Exterior & Roof	03301	Roof	Class 2	0.022	0.022	2057	2210	<300	<300		
		03302	Exterior Walls	Class 2	11	14	613	660	<300	<300		
04100	Radwaste Bldg. 579 ft. el.	04101	Crystallizer	Class 1	0.3	0.5	17K	96K	306	2709		
04200	Radwaste Bldg. 592 ft. el.	04201	Dry Active Waste Storage Area	Class 1	10	50	(floor covered)		<300	<300		

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
		04202	Drum Fill Area	Class 1	4	40	20K	178K	<300	<300		
		04203	Radwaste Annex Truck Loading	Class 1	1	4.1	5400	22K	<300	426		
04300	Radwaste Bldg. Exterior & Roof	04301	Roof	Class 2	0.34	0.36	1429	1581	<300	<300		
		04302	Exterior Walls	Class 2	300	600	513	597	<300	<300		Max DR @ RW Truckbay
05100	Auxiliary Bldg. 542 ft. el.	05101	1A RHR Pump Room	Class 1	3	10	64K	35K	<1000	<1000	1A RHR Pump	
		05102	1B RHR Pump Room	Class 1	2	5	16K	35K	<1000	<1000	1B RHR Pump	
		05103	2A RHR Pump Room	Class 1	0.7	1.2	101K	800K	<1000	3000	2A RHR Pump	
		05104	2B RHR Pump Room	Class 1	0.7	1	21K	80K	<1000	<1000	2B RHR Pump	
		05105	Unit 1 Pipe Chase	Class 1	5	15	60K	240K	<1000	3000		
		05106	Unit 2 Pipe Chase	Class 1	2	5	300K	1.7M	8000	50000		
		05107	Hold Up Tanks Cubicle	Class 1	(ACCESSED FROM THE 579 FT. ELEVATION)						Hold Up Tanks	
		05108	Central Area	Class 1	<0.2	0.2	3726	31.0K	<300	1000		
		05109	South Area	Class 1	0.2	1	1110	3400	<300	<300		
		05110	North Area	Class 1	0.2	0.6	5018	38K	<300	<300		
		05111	West Area	Class 1	0.2	0.6	4447	20K	<300	<300		
		05112	Gas Decay Tanks	Class 1	<0.2	<0.2	5000	1 mrad	<300	<300		
		05113	Unit 1 EDCT	Class 1	1	1.5	120K	13 mrad	<300	<300	Equip Drain Collection Tk	
		05114	Unit 2 EDCT	Class 1	0.4	0.5	38K	6 mrad	<300	<300	Equip Drain Collection Tk	
		05115	HUT Recirc & BAE Feed Pmps	Class 1	0.5	1	(270 mrad)		50K	300K		Hi Contamination Drains
		05116	EDT and Pumps	Class 1	10	20	15K	70K	<1000	1000	Equipment Drain Tank	
		05117	Sump A	Class 1	0.3	0.4	36K	140K	1000	1000		
		05118	Sump B	Class 1	1.7	2	66K	150K	2000	4000		
05200	Auxiliary Bldg. 560 ft. el.	05201	Unit 1A Safety Injection Pump	Class 1	0.75	1	23K	200K	<300	1696		
		05202	Unit 2A Safety Injection Pump	Class 1	0.4	0.4	45K	400K	<1000	3000		
		05203	Unit 1B Safety Injection Pump	Class 1	0.5	0.5	2000	42K	<300	<300		
		05204	Unit 2B Safety Injection Pump	Class 1	0.4	0.4	41K	400K	<1000	2000		
		05205	Unit 1A RHR Heat Exchanger	Class 1	1.8	5	2000	4000	<300	<300	1A RHR Heat Exchanger	
		05206	Unit 2A RHR Heat Exchanger	Class 1	0.8	1.3	2000	4000	<300	<300	2A RHR Heat Exchanger	
		05207	Unit 1B RHR Heat Exchanger	Class 1	1.9	5	2000	3000	<300	<300	1B RHR Heat Exchanger	
		05208	Unit 2B RHR Heat Exchanger	Class 1	0.5	1	1000	2000	<300	<300	2B RHR Heat Exchanger	
		05209	Unit 1 Containment Spray Pump	Class 1	0.4	1.2	11K	60K	<300	<300		
		05210	Unit 2 Containment Spray Pump	Class 1	0.3	0.5	84K	222K	<300	607		
		05211	South Area	Class 1	0.6	1	17K	61K	<300	417	Laundry Drain Tanks	
		05212	Central Area	Class 1	0.3	0.4	9000	15K	<1000	15000	Component Cooling Pmps	
		05213	North Area	Class 1	0.3	0.5	7013	61K	<300	<300		
		05214	West Area	Class 1	0.3	0.5	26K	227K	<300	455		

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
		05215	Equipment Drain Tank	Class 1	4	10	47K	170K	<300	1329		
		05216	Chemical Drain Tank	Class 1	16	40	124K	700K	<1000	250K		Pump Base contaminated
		05217	Unit 1 Vertical Pipe Chase	Class 1	(ACCESSED FROM THE 579 FT. ELEVATION)							
		05218	Unit 2 Vertical Pipe Chase	Class 1	(ACCESSED FROM THE 579 FT. ELEVATION)							
		05219	Blowdown Heat Exchangers	Class 1	0.1	0.1	<690	2617	<300	<300		
		05220	Waste Gas Compressor Area	Class 1	0.1	0.1	<690	8413	<300	<300		
		05221	Unit 1 RHR HE Pipe Chase	Class 1	30	60	(20 mrad)		12K	30K		Boric Acid on Floor
		05222	Unit 2 RHR HE Pipe Chase	Class 1	5	5	(12 mrad)		7000	18K		
05300	Auxiliary Bldg. 579 ft. el.	05301	Unit 1A Charging Pump	Class 1	0.4	0.5	1902	3606	<300	<300	1A Centrifugal Charging	
		05302	Unit 2A Charging Pump	Class 1	0.2	0.2	12K	50K	<300	<300	2A Centrifugal Charging	
		05303	Unit 1B Charging Pump	Class 1	0.4	0.7	32K	165K	<300	<300	1B Centrifugal Charging	
		05304	Unit 2B Charging Pump	Class 1	0.4	0.4	230K	1.3M	<300	<300	2B Centrifugal Charging	
		05305	Unit 1C Charging Pump	Class 1	1.3	4	70K	342K	<300	<300	1C Reciprocating	
		05306	Unit 2C Charging Pump	Class 1	0.2	0.3	74K	255K	<300	784	2C Reciprocating	
		05307	Unit 1 RWST	Class 1	(NOT SURVEYED – INACCESSIBLE)						Refueling Wtr Storage Tk	
		05308	Unit 2 RWST	Class 1	(NOT SURVEYED – INACCESSIBLE)						Refueling Wtr Storage Tk	
		05309	South Area	Class 1	0.3	0.7	13K	45K	<300	<300		
		05310	Central Area	Class 1	1	2	760K	2.9M	560	1860		
		05311	North Area	Class 1	0.2	0.5	17.5K	81K	520	1392		
		05312	West Area	Class 1	<0.2	0.2	760	1280	<300	<300		
		05313	Unit 1 Horizontal Pipe Chase	Class 1	1.6	5	326K	2.5M	4000	30K		
		05314	Unit 2 Horizontal Pipe Chase	Class 1	0.4	0.5	345K	1.6M	<1000	<1000		
		05315	#1 Hold Up Tank	Class 1	10	20	(250 mrad)		50K	150K		
		05316	#0 Hold Up Tank	Class 1	5	5	(160 mrad)		50K	110K		
		05317	#2 Hold Up Tank	Class 1	5	40	(125 mrad)		50K	150K		
		05318	Anion/Cation Valve Aisle	Class 1	<0.2	0.2	1380	5700	<300	<300		
		05319	Unit 1 Vertical Pipe Chase	Class 1	5	20	(not taken)		50K	500K		
		05320	Unit 2 Vertical Pipe Chase	Class 1	5	20	(not taken)		4000	20K		
05400	Auxiliary Bldg. 592 ft. el.	05401	Unit 1 Letdown Heat Exchanger	Class 1	20	40	380K	1.4M	4000	10K		
		05402	Unit 2 Letdown Heat Exchanger	Class 1	1	2	15K	136K	<1000	<1000		
		05403	Unit 1 Seal Water HE	Class 1	5	15	48K	220K	<300	556	Seal Wtr Heat Exchanger	
		05404	Unit 2 Seal Water HE	Class 1	0.4	0.8	6680	24K	<300	1278	Seal Wtr Heat Exchanger	
		05405	Unit 2 Future Diesel Generator	Class 1	<0.2	<0.2	<300	320	<300	<300		
		05406	Primary Sample Room	Class 1	0.25	0.3	19K	60K	10K	30K		
		05407	Calibration Facility	Class 1	0.5	0.6	<300	674	<300	<300		
		05408	South Area	Class 1	0.2	0.3	710	1900	<300	<300		

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Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
		05409	Central Area	Class 1	0.2	0.4	144K	5.6M	340	556		
		05410	North Area	Class 1	0.2	0.3	463	696	<300	<300		
		05411	West Area	Class 1	0.2	0.5	8000	30K	<300	<300		
		05412	Unit 1 Vertical Pipe Chase	Class 1	(ACCESSED FROM THE 579 FT. ELEVATION)							
		05413	Unit 2 Vertical Pipe Chase	Class 1	(ACCESSED FROM THE 579 FT. ELEVATION)							
		05414	Unit 1 BAT and Pumps	Class 1	0.5	1.5	<300	<300	<300	<300		
		05415	Unit 2 BAT and Pumps	Class 1	<0.2	<0.2	6500	15.9K	<300	<300		
05500	Auxiliary Bldg. 617 ft. el.	05501	Unit 1 Volume Control Tank	Class 1	9	15	153K	740K	<300	632		
		05502	Unit 2 Volume Control Tank	Class 1	1	2	3800	19K	<300	<300		
		05503	Unit 1 Pipe Chase	Class 1	0.6	2	3629	19K	<300	1075	U1 CTMT Hatch	
		05504	Unit 2 Chase	Class 1	0.2	0.5	5600	20K	<300	<300	U2 CTMT Hatch	
		05505	Radiation Protection Offices	Class 1	0.013	0.06	953	5319	<300	<300		
		05506	Unit 1 CTMT Purge Room	Class 1	<0.2	0.3	60	300	<300	<300	CTMT Purge Air Equip	
		05507	Unit 2 CTMT Purge Room	Class 1	10	40	360	1000	<300	<300	CTMT Purge Air Equip	
		05508	Tool Crib	Class 1	0.3	0.3	8911	47K	<300	<300		
		05509	Waste Evaporator Monitor Tank	Class 1	0.3	4	92K	117K	<300	734		
		05510	Radwaste Evaporator	Class 1	0.6	2	69K	195K	<300	367		Areas not accessible
		05511	Spent Resin Storage Area	Class 1	(NOT SURVEYED - LOCKED HI RAD AREA)							Posted LHRA >15 R/hr
		05512	Laboratory	Class 1	0.01	0.014	444	1213	<300	<300		
		05513	West Area	Class 1	<0.2	<0.2	8300	26K	<300	<300		
		05514	East Area	Class 1	0.3	1	552	1204	<300	<300		
05600	Auxiliary Bldg. 630 ft. el.	05601	Unit 1 Cable Spreading Room	Class 1	0.007	0.008	905	1031	<300	<300		
		05602	Unit 2 Cable Spreading Room	Class 1	0.007	0.008	760	1031	<300	<300		
		05603	Unit 1 Cable Penetration Room	Class 1	0.006	0.007	763	849	<300	<300		
		05604	Unit 2 Cable Penetration Room	Class 1	0.006	0.007	530	731	<300	<300		
05700	Auxiliary Bldg. 642 ft. el.	05701	642 ft. elevation	Class 1	0.2	0.3	<658	<658	<300	<300		
05800	Auxiliary Bldg. 654 ft. el.	05801	654 ft. elevation	Class 1	0.2	0.3	<658	989	<300	<300	Vent Filter Banks	
05900	Auxiliary Bldg. Exterior & Roof	05901	Roof	Class 2	0.028	0.035	1886	2114	<300	<300		
		05902	Exterior Walls	Class 2	600	300	764	764	<300	<300		Max DR @ RW Truckbay
06100	Turbine Bldg. 560 ft. el.	06101	Oil Room	Class 2	0.006	0.007	<514	699	<300	<300		
		06102	South Area	Class 2	0.005	0.006	<366	1383	<300	<300		
		06103	Central Area	Class 2	0.008	0.015	3700	27K	<300	<300		
		06104	North Area	Class 2	0.005	0.006	1752	7257	<300	<300		
06200	Turbine Bldg. 570 ft. el.	06201	1B Diesel Fuel Oil Storage	Class 2	0.005	0.006	<528	787	<300	<300		
		06202	2B Diesel Fuel Oil Storage	Class 2	0.006	0.007	<539	721	<300	<300		
		06203	1A Diesel Fuel Oil Storage	Class 2	0.005	0.005	<465	915	<300	<300		

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
		06204	2A Diesel Fuel Oil Storage	Class 2	0.006	0.007	<539	<539	<300	<300		
		06205	0A Diesel Fuel Oil Storage	Class 2	0.004	0.005	<521	<521	<300	<300		
		06206	Unit 2 Future Fuel Oil Storage	Class 2	0.007	0.008	<606	664	<300	<300		
		06207	Unit 1 Steam Pipe Tunnel	Class 2	0.008	0.04	17K	104K	<300	<300		Reclassified to Class 1
		06208	Unit 2 Steam Pipe Tunnel	Class 2	0.006	0.01	1500	6200	<300	<300		Reclassified to Class 1
06300	Turbine Bldg. 592 ft. el.	06301	Unit 1 Diesel Generator 1A	Class 2	0.004	0.004	<433	752	<16	<16		
		06302	Unit 2 Diesel Generator 1B	Class 2	0.004	0.005	<354	638	<16	<16		
		06303	Unit 1 Diesel Generator 0	Class 2	0.005	0.007	<302	505	<16	<16		
		06304	Unit 2 Diesel Generator 2A	Class 2	0.003	0.004	<353	641	<16	<16		
		06305	Unit 1 Diesel Generator 2B	Class 2	0.004	0.004	<575	<575	<16	<16		
		06306	Battery and Charger Room	Class 2	0.005	0.005	<460	807	<16	<16		
		06307	Technical Support Center	Class 2	0.006	0.006	<442	<442	<16	<16		
		06308	Secondary Sample Room	Class 2	0.007	0.011	<312	819	<300	<300		
		06309	Auxiliary Heating Boiler	Class 2	0.006	0.008	3233	18K	<300	<300		
		06310	South Area	Class 2	0.009	0.015	<510	648	<16	<16		
		06311	Central Area	Class 2	0.005	0.008	<670	1048	<300	<300		
		06312	North Area	Class 2	0.006	0.007	<455	1028	<16	<16		
06400	Turbine Bldg. 609 ft. el.	06401	609 ft. elevation	Class 2	0.004	0.005	<353	589	<16	<16		
06500	Turbine Bldg. 617 ft. el.	06501	Switchgear Rooms 17, 18 &19	Class 2	0.006	0.008	<522	738	<16	<16		
		06502	Switchgear Rooms 27, 28 &29	Class 2	0.006	0.007	<654	928	<16	<16		
		06503	Air Conditioning Equipment Rm	Class 2	0.016	0.025	<553	682	<16	<16		
		06504	QA Records Area	Class 2	0.005	0.005	<539	862	<17	<17		
		06505	South Area	Class 2	0.005	0.008	<438	785	<16	<16		
		06506	Central Area	Class 2	0.005	0.006	<596	1028	<16	<16		
		06507	North Area	Class 2	0.005	0.007	<595	983	<16	<16		
06600	Turbine Bldg. 625 ft. el.	06601	625 ft. elevation	Class 2	0.005	0.005	661	862	<16	<16		
06700	Turbine Bldg. 642 ft. el.	06701	Unit 1 Switchgear & Rod Drive	Class 2	0.007	0.009	<690	1171	<16	<16		
		06702	Unit 2 Switchgear & Rod Drive	Class 2	0.005	0.005	<612	873	<16	<16		
		06703	Main Control Room Complex	Class 2	0.005	0.006	<476	<476	<16	<16		
		06704	Control Room Annex South	Class 2	0.004	0.005	<353	<589	<16	<16		
		06705	Control Room Annex North	Class 2	0.004	0.005	<393	1048	<16	<16		
		06706	Unit 1 Aux Electric Equipment	Class 2	0.005	0.006	<416	752	<16	<16		
		06707	Unit 2 Aux Electric Equipment	Class 2	0.005	0.005	<429	674	<16	<16		
		06708	South Area	Class 2	0.007	0.01	<334	600	<16	<16		
		06709	Central Area	Class 2	0.005	0.005	<448	796	<16	<16		
		06710	North Area	Class 2	0.008	0.01	<410	641	<16	<16		

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
06800	Turbine Bldg. 656 ft. el.	06801	656 ft. elevation	Class 2	0.005	0.006	<507	848	<16	<16		
06900	Turbine Bldg. Exterior & Roof	06901	Roof	Class 2	0.015	0.016	2490	2790	<300	<300		
		06902	Exterior Walls	Class 2	10	30	<300	<300	<300	<300		Max DR @ Truckbay
07100	Service Bldg. 592 ft. el.	07101	East	Class 2	0.008	0.012	<596	789	<17	<17		
		07102	West	Class 2	0.009	0.013	312	705	<16	<16		
07100	Service Bldg. 608 ft. el.	07201	East	Class 2	0.008	0.012	<716	804	<16	<16		
		07202	West	Class 2	0.009	0.012	1867	473	<16	<16		
07100	Service Bldg. 624 ft. el.	07301	East	Class 2	0.008	0.012	<716	792	<17	<17		
		07302	West	Class 2	0.013	0.01	222	1067	<17	<17		
07100	Service Bldg. 636 ft. el.	07401	East	Class 2	0.008	0.012	872	2846	<300	<300		
		07402	West	Class 2	0.011	0.012	160	286	<16	<16		
07100	Service Bldg. 648 ft. el.	07501	East	Class 2	0.007	0.01	181	343	<16	<16		
07600	Service Bldg. 660 ft. el.	07601	East	Class 2	0.009	0.013	184	457	<16	<16		
07700	Service Bldg. Exterior & Roof	07701	East Roof	Class 2	0.017	0.018	890	1000	<300	<300		
		07702	West Roof	Class 2	0.044	0.046	2267	2438	<300	<300		
		07703	Exterior Walls	Class 2	20	30	319	461	<300	<300		Elevated DR waste
08100	Crib House 559 ft. el.	08101	552 ft. elevation	Class 2	0.006	0.007	<615	849	<300	<300	Circ Water Pumps	1990 -552 ft flooded
08200	Crib House 594 ft. el.	08201	594 ft. elevation	Class 2	0.006	0.007	<702	<702	<300	<300	Fire Pumps, Pmp Motors	
08300	Crib House Exterior & Roof	08301	Roof	Class 2	0.006	0.007	2412	2504	<300	<300		
		08302	Exterior Walls	Class 2	0.006	0.007	<615	658	<300	<300		
09100	Waste Water Treatment Facility	09101	1st Floor	Class 2	0.008	0.025	<683	<683	<300	<300	Pits & drying area	HSA SU ID# 09101
		09102	2nd Floor	Class 2	0.008	0.025	<683	<683	<300	<300		HSA SU ID# 09201
		09103	Roof	Class 2	0.006	0.007	<677	<677	<300	<300		HSA SU ID# 09301
		09104	Exterior Walls	Class 2	0.008	0.025	<683	<683	<300	<300		
09200	North Valve House	09201	1st Floor	Class 2	0.006	0.007	1049	1826	<300	<300		HSA SU ID# 09102
		09202	Roof	Class 2	0.006	0.007	2803	2861	<300	<300		HSA SU ID# 09202
		09203	Exterior Walls	Class 2	0.006	0.007	737	1266	<300	<300		HSA SU ID# 09302
09300	North Chlor/Dechlor Bldg.	09301	1st Floor	Class 2	0.004	0.005	<606	<606	<300	<300		HSA SU ID# 09103
		09302	Roof	Class 2	0.006	0.007	1165	1276	<300	<300		HSA SU ID# 09203
		09303	Exterior Walls	Class 2	0.004	0.005	<606	<606	<300	<300		
09400	Warehouse/Maint. Training Ctr	09401	1st Floor	Class 3	0.006	0.01	<554	1240	<16	<16		
		09402	Roof	Class 3	0.006	0.007	<623	<623	<300	<300		
		09403	Exterior Walls	Class 3	0.006	0.007	<623	<623	<300	<300		
09500	IRSF	09501	1st Floor	Class 2	0.006	0.007	<680	1964	<300	<300		
		09502	Mezzanine	Class 2	0.006	0.007	<554	575	<300	<300		
		09503	Roof	Class 2	0.006	0.007	927	962	<300	<300		

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	Total G/A	Total Max	Loose G/A	Loose Max		
		09504	Exterior Walls	Class 2	0.006	0.007	<554	853	<300	<300		
09600	Fire Training Building	09601	1st Floor	Class 3	<0.1	<0.1	<607	1306	<300	<300		Fire Pit - 20 dpm α
		09602	Roof	Class 3	0.006	0.007	888	1011	<300	<300		
		09603	Exterior Walls	Class 3	<0.1	<0.1	<607	<607	<300	<300		
09700	Contractor Break Building	09701	1st Floor	Class 3	0.006	0.007	<642	667	<300	<300		HSA SU ID# 09107
		09702	Roof	Class 3	0.006	0.007	<677	<677	<300	<300		HSA SU ID# 09207
		09702	Roof	Class 3	0.006	0.007	<677	<677	<300	<300		HSA SU ID# 09207
		09703	Exterior Walls	Class 3	0.006	0.007	<642	<642	<300	<300		
09800	East Training Center	09801	1st Floor	Class 3	0.006	0.007	<650	1110	<300	<300		Old N-GET Bldg
		09802	Roof	Class 3	0.006	0.007	652	864	<300	<300		
		09803	Exterior Walls	Class 3	0.006	0.007	<650	933	<300	<300		
09900	In-Processing Building	09901	1st Floor	Class 3	0.008	1.3	<717	<717	<300	<300	Laboratory (sources)	Also known as ENC Bldg
		09902	2nd Floor	Class 3	0.006	0.008	<717	<717	<300	<300		
		09903	Roof	Class 3	0.006	0.007	864	953	<300	<300		
		09904	Exterior Walls	Class 3	0.006	0.007	<717	<717	<300	<300		
09010	Station Construction Building	09011	1st Floor	Class 3	0.006	0.006	<569	1188	<300	<300		
		09012	Roof	Class 3	0.006	0.007	<623	714	<300	<300		
		09013	Exterior Walls	Class 3	0.006	0.006	<569	<569	<300	<300		
09030	Security Offices	09031	1st Floor	Class 2	0.04	0.07	<768	1011	<300	<300		HSA SU ID# 09116
		09032	Roof	Class 2	0.048	0.048	1495	1649	<300	<300		HSA SU ID# 09216
		09033	Exterior Walls	Class 2	0.04	0.07	<768	<768	<300	<300		
09040	Gate House	09041	1st Floor	Class 2	0.15	0.5	<749	2209	<300	<300		HSA SU ID# 09117
		09042	2nd Floor	Class 2	0.15	0.5	<749	<749	<300	<300		HSA SU ID# 09217
		09043	Roof	Class 2	0.041	0.042	894	1128	<300	<300		HSA SU ID# 09317
		09044	Exterior Walls	Class 2	0.15	0.5	<749	<749	<300	<300		
09050	North Warehouse	09051	1st Floor	Class 3	0.006	0.007	<663	697	<663	<663		
		09052	Roof	Class 3	0.006	0.007	<623	<623	<300	<300		
		09053	Exterior Walls	Class 3	0.006	0.007	<623	<623	<300	<300		
09060	South Warehouse	09061	1st Floor	Class 3	0.006	0.007	<623	137	<300	<300		RAM stored in warehouse
		09062	Roof	Class 3	0.006	0.007	652	744	<300	<300		
		09063	Exterior Walls	Class 3	0.006	0.007	<609	<609	<300	<300		
09070	South Valve House	09071	1st Floor	Class 2	0.006	0.007	716	1149	<300	<300		HSA SU ID# 09121
		09072	Roof	Class 2	0.006	0.007	2856	3074	<300	<300		HSA SU ID# 09221
		09073	Exterior Walls	Class 2	0.006	0.007	<586	943	<300	<300		
09080	South Chlor/Dechlor Bldg.	09080	1st Floor	Class 2	0.004	0.005	<492	156	<300	<300		
		09081	Roof	Class 2	0.006	0.007	1846	1926	<300	<300		

Table 2-13 Nominal Radiological Data – Structural Survey Units (continued)

Survey				HSA Class	Radiation Levels (mr/hr)		Contamination Levels (dpm/100cm2)				Major Systems & Components	Comments
							Total		Loose			
Area ID	Area Description	Unit ID	Unit Description		G/A	Max	G/A	Max	G/A	Max		
		09082	Exterior Walls	Class 2	0.004	0.005	<492	<492	<300	<300		
09210	Temp Rad Waste Liner Storage	09211	1st Floor	Class 3	0.006	0.007	630	2211	<300	<300	Concrete Liners	Not included in HSA
		09212	Roof	Class 3	0.006	0.007	2266	2558	<300	<300		Not included in HSA
		09213	Exterior Walls	Class 3	0.006	0.007	(not taken)		<300	<300		Not included in HSA
09220	Warehouse #13	09221	1st Floor	Class 3	0.004	0.005	<672	<672	<300	<300	ISFSI Warehouse	Not included in HSA
		09222	Roof	Class 3	0.006	0.007	<492	<492	<300	<300		Not included in HSA
		09223	Exterior Walls	Class 3	0.004	0.005	<672	<672	<300	<300		Not included in HSA

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Table 2-14 Unit 1 Containment 568 Foot Elevation Concrete Core Sample Analysis Summary

Location	Sample ID#	Core Depth (inches)	Co-60 Surface Activity (a) (pC/g)	Co-60 Avg. Subsurface Activity (b) (pC/g)	Cs-137 Surface Activity (a) (pC/g)	Cs-137 Avg. Subsurface Activity (b) (pC/g)
"A" Loop I/S Missile Barrier	B101102-CJFCCV-001	1.5	9.69E+01	1.40E+01	3.46E+04	4.76E+03
"A" Loop I/S Missile Barrier	B101102-CJFCCV-002	5.0	1.91E+02	1.90E+01	4.04E+05	1.65E+04
"B" Loop I/S Missile Barrier	B101103-CJFCCV-001	4.0	2.18E+02	2.55E+01	3.05E+05	3.10E+04
"B" Loop I/S Missile Barrier	B101103-CJFCCV-002	4.5	7.97E+03	3.64E+02	9.86E+04	1.00E+04
"C" Loop I/S Missile Barrier	B101104-CJFCCV-001	4.5	1.25E+02	1.69E+01	4.93E+05	6.71E+04
"C" Loop I/S Missile Barrier	B101104-CJFCCV-002	4.0	4.63E+02	2.86E+01	4.40E+05	2.09E+04
"D" Loop I/S Missile Barrier	B101105-CJFCCV-001	3.5	3.66E+04	6.54E+03	2.56E+03	2.09E+02
"D" Loop I/S Missile Barrier	B101105-CJFCCV-002	5.0	1.04E+03	5.88E+01	3.07E+04	5.79E+03
"A" Loop O/S Missile Barrier	B101106-CJFCCV-001	3.0	1.84E+01	1.20E+00	1.39E+03	6.42E+01
"A" Loop O/S Missile Barrier	B101106-CJFCCV-002	1.0	2.66E+01	6.16E+00	8.03E+01	1.23E+01
"B" Loop O/S Missile Barrier	B101107-CJFCCV-001	1.0	2.04E+01	3.36E+00	3.68E+02	5.21E+01
"B" Loop O/S Missile Barrier	B101107-CJFCCV-002	1.0	1.92E+01	3.56E+00	2.41E+02	3.40E+01
"C" Loop O/S Missile Barrier	B101108-CJFCCV-001	2.5	1.50E+01	2.10E+00	8.70E+03	4.71E+02
"C" Loop O/S Missile Barrier	B101108-CJFCCV-002	1.5	*6.16E+01*	1.16E+01	6.08E+04	8.50E+03
"D" Loop O/S Missile Barrier	B101109-CJFCCV-001	1.0	1.02E+01	1.93E+00	1.94E+02	3.21E+01
"D" Loop O/S Missile Barrier	B101109-CJFCCV-002	1.0	1.18E+02	1.84E+01	7.32E+03	9.08E+02

(a) Represents surface activity of floor following removal of loose contamination
(b) Represents average of activity over entire depth of core sample minus the surface activity
(c) Italicized values indicate MDC value.

Table 2-15 Unit 1 Bio-Shield Concrete Core Samples Gamma Spectroscopy Analysis

Sample ID	**Sample Date**	**Puck**	**Depth** (inches)	**Co-60** [a] (pCi/g)	**Cs-137** [a] (pCi/g)	**Eu-152** [a] (pCi/g)	**Eu-154** [a] (pCi/g)
B101101-CJWCCV-001 – 1st Core Section	01/10/13	1	11.5	*9.15E-02*	**1.33E-01**	*2.26E-01*	*1.22E-01*
			12	*1.46E-01*	*1.06E-01*	*1.76E-01*	*1.42E-01*
B101101-CJWCCV-001 – 2nd Core Section	01/17/13	2	23.5	*1.85E-01*	*1.89E-01*	*1.80E-01*	*1.54E-01*
			24	*1.43E-01*	*1.67E-01*	*2.40E-01*	*1.09E-01*
B101101-CJWCCV-001 – 2nd Core Section	01/17/13	3	35.5	*1.28E-01*	*1.02E-01*	*2.27E-01*	*1.82E-01*
			36	*8.28E-02*	*1.02E-01*	*2.06E-01*	*1.76E-01*
B101101-CJWCCV-001 – 2nd Core Section	01/17/13	4	47.5	*2.76E-01*	*1.95E-01*	**1.53E+00**	*2.48E-01*
			48	*2.67E-01*	*1.49E-01*	**1.25E+00**	*2.29E-01*
B101101-CJWCCV-001 – 3rd Core Section	02/21/13	5	59.5	**1.94E+00**	*2.77E-01*	**2.04E+01**	*8.55E-01*
			60	**1.94E+00**	*2.77E-01*	**2.04E+01**	*8.95E-01*
B101101-CJWCCV-001 – Rebar Metal Piece	02/21/13	Rebar	63	**1.76E+01**	**1.92E-01**	*4.10E-01*	*2.53E-01*
B101101-CJWCCV-001 – 3rd Core Section	02/21/13	6	71.5	**1.50E+01**	*4.22E-01*	**1.55E+02**	**6.49E+00**
			72	**1.52E+01**	*4.12E-01*	**1.63E+02**	**7.04E+00**

a Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Table 2-16 Unit 2 Containment 568 Foot Elevation Concrete Core Sample Analysis Summary

Location	Sample ID#	Core Depth (inches)	Co-60		Cs-137	
			Surface Activity (a) (pC/g)	Avg. Subsurface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Subsurface Activity (b) (pC/g)
"A" Loop I/S Missile Barrier	B102102-CJFCCV-001	3.0	7.56E+02	1.78E+02	1.10E+05	1.66E+04
"A" Loop I/S Missile Barrier	B102102-CJFCCV-002	6.0	9.40E+04	1.65E+03	1.75E+03	5.65E+01
"B" Loop I/S Missile Barrier	B102103-CJFCCV-001	5.5	5.64E+02	2.31E+01	4.57E+04	1.57E+03
"B" Loop I/S Missile Barrier	B102103-CJFCCV-002	4.0	1.91E+02	8.11E+00	6.39E+03	2.27E+02
"C" Loop I/S Missile Barrier	B102104-CJFCCV-001	5.5	3.55E+02	1.22E+01	1.94E+04	6.59E+02
"C" Loop I/S Missile Barrier	B102104-CJFCCV-002	1.5	1.40E+02	4.05E+00	1.45E+04	1.73E+02
"D" Loop I/S Missile Barrier	B102105-CJFCCV-001	4.5	1.11E+03	4.39E+01	2.88E+03	8.77E+01
"D" Loop I/S Missile Barrier	B102105-CJFCCV-002	4.0	1.11E+02	4.05E+00	1.25E+04	3.82E+02
"A" Loop O/S Missile Barrier	B102106-CJFCCV-001	5.0	5.59E+01	1.75E+00	8.13E+02	1.70E+01
"A" Loop O/S Missile Barrier	B102106-CJFCCV-002	1.5	2.71E+01	1.20E+01	1.40E+04	2.93E+03
"B" Loop O/S Missile Barrier	B102107-CJFCCV-001	4.5	6.61E+01	8.16E+00	4.10E+03	7.06E+00
"B" Loop O/S Missile Barrier	B102107-CJFCCV-002	1.5	3.57E+01	3.86E+00	2.09E+03	1.82E+02
"C" Loop O/S Missile Barrier	B102108-CJFCCV-001	2.0	1.30E+01	1.17E+00	2.55E+02	1.77E+01
"C" Loop O/S Missile Barrier	B102108-CJFCCV-002	2.0	1.89E+01	1.64E+00	3.97E+02	2.64E+01
"D" Loop O/S Missile Barrier	B102109-CJFCCV-001	4.0	1.13E+03	4.61E+01	6.07E+03	2.15E+02
"D" Loop O/S Missile Barrier	B102109-CJFCCV-002	2.5	8.62E+01	7.46E+00	4.65E+03	3.09E+02

(a) Represents surface activity of floor following removal of loose contamination
(b) Represents average of activity over entire depth of core sample minus the surface activity

Table 2-17 Unit 1 Containment 541 Foot Elevation Concrete Core Sample Analysis Summary

Location	Sample ID#	Core Depth (inches)	Co-60		Cs-137		Eu-152		Eu-154	
			Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)
Incore Tunnel Floor	B101110-CJFCCV-001	15.5	2.79E+01	5.97E+00	5.27E+01	4.07E+00	6.97E+01	3.34E+01	5.23E+00	1.86E+00
Incore Tunnel Floor	B101110-CJFCCV-002	4.0	3.48E+02	5.91E+01	2.71E+03	3.28E+02	6.30E+01	9.15E+01	4.35E+00	5.72E+00
Incore Tunnel Wall	B101110-CJWCCV-003	3.5	1.12E+01	7.64E+00	3.72E+01	2.61E+00	5.60E+01	6.67E+01	4.19E+00	3.73E+00

(a) Represents surface activity of floor following removal of loose contamination
(b) Represents average of activity over entire depth of core sample minus the surface activity

Table 2-18 Unit 2 Containment 541 Foot Elevation Concrete Core Sample Analysis Summary

Location	Sample ID#	Core Depth (inches)	Co-60		Cs-137		Eu-152		Eu-154	
			Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Sub-surface Activity (b) (pC/g)
Incore Tunnel Floor	B102110-CJFCCV-001	14.0	2.35E+01	9.15E+00	6.74E+02	1.08E+01	1.14E+02	7.71E+01	8.45E+00	4.33E+00
Incore Tunnel Floor	B102110-CJFCCV-002	4.5	1.42E+01	1.37E+01	1.74E+01	9.43E-01	1.13E+02	1.22E+02	8.61E+00	7.38E+00
Incore Tunnel Wall	B102110-CJWCCV-003	5.5	1.35E+01	8.45E+00	5.01E+01	1.99E+00	7.08E+01	6.27E+01	5.12E+00	3.56E+00

(a) Represents surface activity of floor following removal of loose contamination
(b) Represents average of activity over entire depth of core sample minus the surface activity

Table 2-19 Unit 1 Containment Concrete Core Samples – Eberline Laboratory Analysis

Radionuclide	B101102-CJFCCV-002 (pCi/g)	B101103-CJFCCV-002 (pCi/g)	B101104-CJFCCV-001 (pCi/g)	B101105-CJFCCV-001 (pCi/g)	B101105-CJFCCV-002 (pCi/g)	B101106-CJFCCV-002 (pCi/g)	B101107-CJFCCV-002 (pCi/g)	B101108-CJFCCV-002 (pCi/g)	B101109-CJFCCV-001 (pCi/g)	B101110-CJFCCV-001 (pCi/g)	B101110-CJFCCV-002 (pCi/g)
H-3	**1.91E+01**	**2.60E+01**	**3.75E+01**	**2.71E+01**	**1.19E+02**	**8.11E+01**	**5.05E+01**	**7.61E+01**	**5.40E+01**	**1.33E+02**	**2.20E+02**
C-14	*2.50E+00*	**3.62E+00**	**4.40E+00**	**2.46E+01**	**2.71E+00**	**1.51E+00**	**1.12E+00**	**9.52E+00**	*1.09E+00*	*9.89E-01*	**3.30E+00**
Co-60	*8.12E+01*	**7.09E+03**	**8.69E+01**	**3.35E+04**	**9.77E+02**	**4.36E+01**	**1.50E+01**	**8.67E+01**	**8.50E+00**	**4.45E+01**	**3.60E+02**
Ni-63	**1.81E+04**	**2.26E+03**	**6.29E+03**	**2.81E+03**	**1.14E+01**	**3.02E+01**	**5.48E+00**	**2.39E+01**	**1.13E+01**	**1.40E+02**	**2.98E+01**
Sr-90	**4.75E+01**	**7.60E+00**	**1.63E+02**	**1.45E+01**	**7.08E+00**	**3.97E-01**	**5.67E-01**	**8.46E+00**	**4.74E-01**	*3.16E-01*	**6.01E-01**
Nb-94	*8.56E+01*	*1.56E+01*	*5.13E+01*	*2.56E+01*	*7.85E+00*	*6.01E-01*	*3.03E-01*	*9.30E+00*	*2.37E-01*	**1.47E+00**	*1.97E+00*
Tc-99	*2.25E+01*	**1.22E+00**	*3.46E+01*	*9.31E-01*	**9.35E-01**	*2.96E-01*	*2.60E-01*	*1.13E+00*	*2.66E-01*	*2.63E-01*	*1.33E+00*
Ag-108m	*2.39E+02*	*1.72E+01*	*5.72E+01*	*2.05E+01*	*9.25E+00*	*4.50E-01*	*2.35E-01*	*1.19E+01*	*2.09E-01*	*1.45E+00*	*1.91E+00*
Sb-125	*1.92E+02*	*5.54E+01*	*1.23E+02*	*3.88E+01*	*4.78E+01*	*1.86E+00*	*1.65E+00*	*5.62E+01*	*2.96E-02*	NA	NA
Cs-134	*5.47E+01*	**1.61E+01**	*3.68E+01*	*1.53E+01*	**9.47E+00**	*4.45E-01*	*3.31E-01*	**3.92E+01**	*3.54E-01*	*6.39E-01*	*1.61E+00*
Cs-137	**1.45E+05**	**9.44E+04**	**1.22E+05**	**2.72E+03**	**4.27E+04**	**1.38E+02**	**1.60E+02**	**5.91E+04**	**1.43E+02**	**8.83E+01**	**2.84E+03**
Pm-145	*1.49E+04*	*1.38E+04*	*3.49E+03*	*2.75E+02*	*5.26E+03*	**1.42E+01**	**1.74E+00**	*8.49E+03*	*2.02E+00*	**1.63E+01**	*3.94E+02*
Eu-152	*4.52E+02*	*7.31E+01*	*2.52E+02*	*1.25E+02*	*2.29E+01*	*1.82E+00*	*8.56E-01*	*1.95E+01*	*1.06E+00*	**1.64E+02**	**1.08E+02**
Eu-154	*2.26E+02*	*3.58E+01*	*1.29E+02*	*5.16E+01*	*1.48E+01*	*9.41E-01*	*5.26E-01*	*1.76E+01*	*5.44E-01*	**1.29E+01**	**8.69E+00**
Eu-155	*5.39E+01*	*2.37E+01*	*3.54E+01*	*1.70E+01*	*1.68E+01*	*6.24E-01*	*6.11E-01*	*1.89E+01*	*6.41E-01*	**3.69E+00**	*3.52E+00*
Np-237	*9.15E-02*	*9.56E-02*	*7.24E-02*	*9.82E-02*	*7.93E-02*	*2.40E-02*	*3.62E-02*	*1.01E-01*	*2.56E-02*	NA	NA
Pu-238	*1.25E-01*	**2.35E-01**	**3.05E-01**	**3.18E+00**	*1.31E-01*	*5.39E-02*	**1.03E-01**	*1.65E-01*	*9.33E-02*	**5.21E-02**	*1.22E-01*
Pu-239/240	*9.84E-02*	**1.11E-01**	**1.12E-01**	**2.30E+00**	*7.62E-02*	*5.24E-02*	**1.64E-01**	*1.05E-01*	*5.88E-02*	*4.50E-02*	**1.09E-01**
Pu-241	*8.17E+00*	*7.08E+00*	*8.54E+00*	*1.24E+01*	*7.86E+00*	*5.60E-01*	*4.53E-01*	*9.36E+00*	*4.79E-01*	*4.21E-01*	*7.28E+00*
Am-241	**3.58E-01**	**1.39E+00**	**4.25E-01**	**5.15E+01**	**2.68E-01**	**1.67E-01**	8.80E-02	*1.01E-01*	*5.40E-02*	*4.36E-02*	**2.15E-01**
Am-243	*8.37E-02*	**1.36E-01**	*6.05E-03*	**5.33E-01**	*8.36E-02*	*3.80E-02*	*4.49E-02*	*7.63E-02*	*2.96E-02*	*3.04E-02*	*9.33E-02*
Cm-243/244	**8.70E-02**	**3.34E-01**	**1.27E-01**	**1.01E+01**	**1.43E-01**	*4.35E-02*	*4.04E-02*	*7.26E-02*	*6.46E-02*	*6.35E-02*	**1.56E-01**

a Bold values indicate concentration greater than MDC. Italicized values indicate MDC value. NA indicates "no analysis"

Table 2-20 Unit 2 Containment Concrete Core Samples – Eberline Laboratory Analysis

Radionuclide	B102102-CJFCCV-001 (pCi/g)	B102103-CJFCCV-001 (pCi/g)	B102103-CJFCCV-002 (pCi/g)	B102105-CJFCCV-001 (pCi/g)	B102105-CJFCCV-002 (pCi/g)	B102106-CJFCCV-001 (pCi/g)	B102106-CJFCCV-002 (pCi/g)	B102107-CJFCCV-001 (pCi/g)	B102110-CJFCCV-001 (pCi/g)	B102110-CJFCCV-002 (pCi/g)
H-3	**1.27E+02**	**7.61E+01**	**5.57E+00**	**9.18E+00**	**2.06E+02**	**1.12E+01**	**5.19E+01**	**7.26E+00**	**2.41E+02**	**3.29E+01**
C-14	**2.80E+00**	**5.80E+00**	**1.17E+00**	**3.94E+00**	**1.23E+00**	**2.72E+00**	**2.51E+00**	**4.44E+00**	*9.72E-01*	**1.84E+00**
Co-60	**9.96E+02**	**6.91E+02**	**2.59E+02**	**1.09E+03**	**1.83E+02**	**8.01E+01**	**4.35E+01**	**7.57E+01**	**4.95E+01**	**2.27E+01**
Ni-63	**4.35E+03**	**1.23E+03**	**2.72E+02**	**4.85E+03**	**1.08E+02**	**9.98E+00**	**1.07E+02**	**3.11E+02**	**7.31E+01**	**4.73E+01**
Sr-90	**4.84E+00**	**1.41E+00**	**1.46E+00**	**1.76E+01**	**8.33E-01**	*4.17E-01*	**3.79E+00**	**2.08E+00**	**3.71E-01**	*3.35E-01*
Nb-94	*1.92E+01*	*1.77E+01*	*1.77E+01*	*3.50E+00*	*5.97E+00*	*1.06E+00*	*7.04E+00*	*1.25E+00*	*3.22E+00*	**2.48E+00**
Tc-99	**2.86E-01**	**2.22E+00**	*6.85E-01*	*3.97E-01*	*2.54E-01*	*2.95E-01*	**3.96E-01**	*2.80E-01*	*2.60E-01*	*2.50E-01*
Ag-108m	**6.85E+01**	*2.29E+01*	*1.88E+00*	*2.93E+00*	*7.03E+00*	*8.09E-01*	*9.72E+00*	*1.08E+00*	**4.17E+00**	**2.08E+00**
Sb-125	*2.39E+02*	*1.00E+02*	*1.00E+02*	*2.93E+00*	*5.56E+01*	*6.51E+00*	*6.41E+01*	*1.52E+01*	NA	NA
Cs-134	**6.09E+01**	*2.10E+01*	**8.27E+00**	**3.10E+00**	**8.75E+00**	**1.26E+00**	*1.12E+01*	**1.68E+01**	**1.30E+00**	*8.19E-01*
Cs-137	**2.14E+05**	**5.58E+04**	**8.45E+03**	**2.49E+03**	**2.09E+04**	**1.13E+03**	**2.66E+04**	**4.51E+03**	**1.05E+03**	**1.56E+01**
Pm-145	**1.23E+04**	**8.75E+02**	**1.08E+03**	**3.31E+01**	**2.62E+03**	**1.28E+02**	**1.12E+03**	**3.88E+02**	**4.78E+01**	**2.79E+01**
Eu-152	*3.98E+01*	*5.08E+01*	**5.08E+01**	*9.27E+00*	*1.22E+01*	*3.28E+00*	*1.11E+01*	*3.58E+00*	**3.97E+02**	**2.53E+02**
Eu-154	*2.70E+01*	*3.35E+01*	**3.35E+01**	*5.45E+00*	*9.97E+00*	*1.43E+00*	*1.21E+01*	*1.75E+00*	**2.58E+01**	**2.17E+01**
Eu-155	*7.97E+01*	*3.55E+01*	*6.12E+00*	*4.05E+00*	*2.00E+01*	*2.32E+00*	*2.39E+01*	*4.58E+00*	**7.02E+00**	**5.98E+00**
Np-237	*2.74E-02*	*2.85E-02*	*2.85E-02*	*2.96E-02*	*3.68E-02*	*3.84E-02*	*2.90E-02*	*3.17E-02*	NA	NA
Pu-238	*4.15E-02*	*3.99E-02*	*2.59E-02*	*1.64E-01*	*4.71E-02*	*2.85E-02*	*4.43E-02*	*3.67E-02*	*3.18E-02*	*2.71E-02*
Pu-239/240	*4.40E-02*	*5.00E-02*	**4.74E-02**	*1.54E-01*	*4.48E-02*	*2.86E-02*	*4.04E-02*	*2.56E-02*	*2.78E-02*	*2.36E-02*
Pu-241	*3.38E+00*	*4.01E+00*	*2.79E+00*	*6.17E+00*	*4.65E+00*	*3.45E+00*	*4.20E+00*	*3.09E+00*	*3.30E+00*	*2.70E+00*
Am-241	**2.49E-01**	**1.24E-01**	**1.27E-01**	**6.28E+00**	*5.08E-02*	*4.55E-02*	*4.48E-02*	*4.25E-02*	*4.95E-02*	*1.94E-02*
Am-243	*4.86E-02*	*3.28E-02*	*3.92E-02*	*3.80E-01*	*3.58E-02*	*7.19E-02*	*5.63E-02*	*3.94E-02*	**5.18E-02**	*3.61E-02*
Cm-243/244	**9.04E-02**	**4.54E-02**	**5.56E-02**	**1.19E-00**	*3.50E-02*	*3.98E-02*	*4.60E-02*	*3.66E-02*	*4.96E-02*	*2.86E-02*

a Bold values indicate concentration greater than MDC. Italicized values indicate MDC value. NA indicates "no analysis"

Table 2-21 Radionuclide Distributions for Unit 1 and Unit 2 Containments

Unit 1 Containment 568 foot elevation			Unit 1 Containment 541 foot elevation	
H-3	0.10%		H-3	3.25%
C-14	0.01%		C-14	0.04%
Fe-55	0.07%		Fe-55	0.27%
Ni-59	0.01%		Ni-59	0.51%
Co-60	2.88%		Co-60	8.22%
Ni-63	2.04%		Ni-63	3.61%
Sr-90	0.05%		Sr-90	0.01%
Nb-94	0.01%		Nb-94	0.07%
Tc-99	0.01%		Tc-99	0.01%
Ag-108m	0.02%		Ag-108m	0.07%
Sb-125	0.10%		Sb-125	0.09%
Cs-134	0.03%		Cs-134	0.41%
Cs-137	94.30%		Cs-137	27.28%
Eu-152	0.19%		Eu-152	51.89%
Eu-154	0.10%		Eu-154	2.82%
Eu-155	0.03%		Eu-155	1.36%
Np-237	0.00%		Np-237	0.00%
Pu-238	0.00%		Pu-238	0.00%
Pu-239	0.00%		Pu-239	0.00%
Pu-240	0.00%		Pu-240	0.00%
Pu-241	0.01%		Pu-241	0.07%
Am-241	0.01%		Am-241	0.00%
Am-243	0.00%		Am-243	0.00%
Cm-243	0.00%		Cm-243	0.00%
Cm-244	0.00%		Cm-244	0.00%

Table 2-21 Radionuclide Distributions for Unit 1 and Unit 2 Containments (continued)

Unit 2 Containment 568 foot elevation			Unit 2 Containment 541 foot elevation	
H-3	0.06%		H-3	1.13%
C-14	0.00%		C-14	0.01%
Fe-55	1.14%		Fe-55	0.70%
Ni-59	0.24%		Ni-59	1.55%
Co-60	13.70%		Co-60	7.84%
Ni-63	45.02%		Ni-63	12.94%
Sr-90	0.00%		Sr-90	0.00%
Nb-94	0.29%		Nb-94	0.05%
Tc-99	0.00%		Tc-99	0.00%
Ag-108m	0.46%		Ag-108m	0.67%
Sb-125	0.07%		Sb-125	0.03%
Cs-134	0.06%		Cs-134	0.34%
Cs-137	38.90%		Cs-137	4.40%
Eu-152	0.02%		Eu-152	65.28%
Eu-154	0.01%		Eu-154	3.72%
Eu-155	0.02%		Eu-155	1.31%
Np-237	0.00%		Np-237	0.00%
Pu-238	0.00%		Pu-238	0.00%
Pu-239	0.00%		Pu-239	0.00%
Pu-240	0.00%		Pu-240	0.00%
Pu-241	0.00%		Pu-241	0.02%
Am-241	0.00%		Am-241	0.00%
Am-243	0.00%		Am-243	0.00%
Cm-243	0.00%		Cm-243	0.00%
Cm-244	0.00%		Cm-244	0.00%

Table 2-22 Auxiliary Building 542 Foot Elevation Concrete Core Sample Analysis Summary

Location	Sample ID#	Core Depth (inches)	Co-60		Cs-137	
			Surface Activity (a) (pC/g)	Avg. Subsurface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Subsurface Activity (b) (pC/g)
1A RHR Pump Room	B105101-CJFCCV-001	5.5	**6.11E+01**	**1.06E+01**	**3.74E+03**	**3.06E+02**
2A RHR Pump Room	B105103-CJFCCV-001	4.0	**4.56E+02**	**2.04E+02**	**2.51E+04**	**4.92E+03**
2B RHR Pump Room	B105104-CJFCCV-001	1.0	**4.08E-01**	*4.20E-01*	**9.44E+00**	**1.79E+00**
Unit 1 Hot Pipe Chase	B105105-CJFCCV-001	5.0	**7.00E+00**	**8.89E-01**	**9.83E+03**	**2.65E+03**
Unit 2 Hot Pipe Chase	B105106-CJFCCV-001	4.5	**3.86E+00**	**1.51E+00**	**2.77E+03**	**1.05E+03**
Unit 2 Hot Pipe Chase	B105106-CJFCCV-002	1.0	**1.75E+01**	**3.60E+00**	**3.09E+03**	**4.14E+02**
Hold-up Tank Cubicles	B105107- CJFCCV-002	14.0	**1.18E+01**	**3.71E-01**	**1.17E+02**	**2.26E+00**
Central Common Area	B105108-CJFCCV-001	4.5	*4.32E-01*	*1.74E-01*	**1.92E+03**	**2.18E+02**
Central Common Area	B105108-CJFCCV-002	4.0	**2.80E+00**	**1.28E+00**	**1.13E+02**	**1.53E+01**
Central Common Area	B105108-CJFCCV-003	3.0	**3.03E+01**	**1.65E+00**	**1.67E+03**	**7.26E+01**
Central Common Area Elevator Shaft	B105108-CJFCCV-004	4.0	**4.05E+00**	**2.71E-01**	**1.51E+02**	**4.92E+00**
South Common Area	B105109-CJFCCV-001	1.0	**4.22E+00**	**8.22E-01**	**1.43E+03**	**2.13E+02**
North Common Area	B105110-CJFCCV-001	1.0	**8.09E+00**	**1.81E+00**	**2.11E+03**	**3.50E+02**
North Common Area	B105110-CJFCCV-002	1.5	*3.28E-01*	*7.85E-02*	**6.37E+01**	**5.17E+00**
East Common Area	B105111-CJFCCV-001	3.5	**2.06E+00**	**7.66E-01**	**1.13E+03**	**1.99E+02**
East Common Area Wall	B101111-CJWCCV-002	0.5	*1.94E-01*	*2.85E-01*	**9.52E-01**	**4.25E-01**
Unit 1 ABEDCT Room	B105113-CJFCCV-001	5.5	**1.96E+02**	**1.33E+01**	**5.69E+03**	**1.54E+02**
Unit 1 ABEDCT Room	B105113-CJFCCV-002	5.5	**3.76E+01**	**6.98E+00**	**1.79E+03**	**1.50E+02**
Unit 1 ABEDCT Room Wall	B105113-CJWCCV-003	1.0	**6.71E+02**	**1.69E+02**	**1.66E+04**	**3.77E+03**

(a) Represents surface activity of floor following removal of loose contamination
(b) Represents average of activity over entire depth of core sample minus the surface activity
(c) Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Table 2-23 Auxiliary Building Concrete Core Samples – Eberline Laboratory Analysis

Radionuclide	B105101-CJFCCV-001 (pCi/g)	B105103-CJFCCV-001 (pCi/g)	B105105-CJFCCV-001 (pCi/g)	B105106-CJFCCV-002 (pCi/g)	B105108-CJFCCV-003 (pCi/g)	B105113-CJWCCV-003 (pCi/g)
H-3	**1.12E+01**	**1.68E+01**	**6.98E+00**	*2.96E+01*	**5.61E+00**	**4.71E+00**
C-14	*9.56E-01*	**4.88E+00**	*9.98E-01*	**1.48E+00**	**1.19E+00**	**3.10E+00**
Fe-55	*1.20E+01*	*3.16E+01*	*1.81E+01*	*1.12E+01*	*8.17E+00*	*1.73E+01*
Ni-59	*1.59E+01*	*2.08E+01*	*1.81E+01*	*1.74E+01*	*1.67E+01*	*1.75E+01*
Co-60	**2.62E+01**	**2.28E+02**	**3.20E+00**	**1.01E+01**	**1.83E+01**	**2.03E+02**
Ni-63	**1.52E+02**	**3.38E+03**	**2.91E+02**	**9.18E+01**	**8.59E+01**	**1.65E+03**
Sr-90	**1.07E+00**	**5.44E+00**	**1.52E+00**	**4.04E-01**	*3.29E-01*	**8.82E+00**
Nb-94	*2.29E-01*	*8.17E-01*	*6.19E-01*	*2.13E-01*	**3.02E-01**	*6.57E-01*
Tc-99	**1.45E+00**	**9.61E-01**	**6.70E-01**	**4.73E-01**	**9.09E-01**	**5.81E-01**
Ag-108m	*2.06E-01*	**1.66E+00**	*8.06E-01*	*2.53E-01*	*1.82E-01*	*5.96E-01*
Sb-125	*2.27E+00*	*7.79E+00*	*4.39E+00*	*2.46E+00*	*1.99E+00*	*4.57E+00*
Cs-134	**4.53E-01**	*1.47E+00*	*8.63E-01*	**8.11E-01**	*2.90E-01*	**1.23E+00**
Cs-137	**1.35E+03**	**1.33E+04**	**5.29E+03**	**1.88E+03**	**1.05E+03**	**4.19E+03**
Pm-147	**1.43E+01**	*8.17E-01*	*9.63E-01*	**5.30E+00**	**9.57E+00**	**7.19E+00**
Eu-152	*7.57E-01*	*2.58E+00*	*1.05E+00*	*6.75E-01*	*7.10E-01*	*1.80E+00*
Eu-154	*4.00E-01*	*1.38E+00*	*1.19E+00*	*3.95E-01*	*3.21E-01*	*1.05E+00*
Eu-155	*5.76E-01*	*1.76E+00*	*1.16E+00*	*6.33E-01*	*4.84E-01*	*1.05E+00*
Np-237	*2.04E-02*	*2.13E-02*	*2.39E-02*	*2.10E-01*	*1.92E-02*	*3.20E-02*
Pu-238	*1.77E-02*	**3.43E-01**	*1.32E-02*	*1.37E-02*	**1.78E-02**	*1.39E-02*
Pu-239/240	*1.77E-02*	**2.74E-01**	*1.06E-02*	*1.37E-02*	*1.33E-02*	*1.35E-02*
Pu-241	*2.48E+00*	*2.46E+00*	*1.79E+00*	*2.36E+00*	*1.73E+00*	*1.72E+00*
Am-241	*3.03E-02*	**2.29E-01**	*4.38E-02*	*2.72E-02*	*4.29E-02*	*3.17E-02*
Am-243	*2.89E-02*	**1.09E-01**	**6.01E-02**	**3.62E-02**	**3.36E-02**	**3.18E-02**
Cm-243/244	*3.91E-02*	*4.41E-02*	*3.95E-02*	*3.89E-02*	*4.02E-02*	*4.29E-02*

a Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Table 2-24 Radionuclide Distribution for the Auxiliary Building

Auxiliary Building 542 foot elevation	
H-3	0.174%
C-14	0.044%
Fe-55	0.106%
Ni-59	0.498%
Co-60	0.908%
Ni-63	23.480%
Sr-90	0.051%
Nb-94	0.013%
Tc-99	0.016%
Ag-108m	0.017%
Sb-125	0.017%
Cs-134	0.010%
Cs-137	74.597%
Eu-152	0.017%
Eu-154	0.009%
Eu-155	0.008%
Np-237	0.0004%
Pu-238	0.001%
Pu-239	0.0005%
Pu-240	0.001%
Pu-241	0.028%
Am-241	0.001%
Am-243	0.001%
Cm-243	0.0003%
Cm-244	0.0003%

Table 2-25 Turbine Building 560 Foot and 570 Foot Elevation Concrete Core Sample Analysis Summary

Location	Sample ID#	Core Depth (inches)	Co-60		Cs-137	
			Surface Activity (a) (pC/g)	Avg. Subsurface Activity (b) (pC/g)	Surface Activity (a) (pC/g)	Avg. Subsurface Activity (b) (pC/g)
TB 560 el. North @ Grid F8	B206104-CJFCCV-001	5.5	*1.14E-01*	*8.57E-02*	*1.27E-01*	*1.30E-01*
TB 560 el. North @ Grid F11	B206104-CJFCCV-002	5.0	*2.11E-01*	*1.07E-01*	1.65E+00	1.76E-01
TB 560 el. North @ Grid F12	B206104-CJFCCV-003	4.0	*2.73E-01*	*1.10E-01*	4.67E+00	1.37E+00
Unit 1 Steam Tunnel @ Grid L32	B206207-CJFCCV-001	2.0	*1.04E-01*	*1.18E-01*	4.67E+00	2.98E+00
Unit 1 Steam Tunnel @ Grid K31	B206207-CJFCCV-002	2.0	*1.30E-01*	*6.86E-02*	4.52E+01	3.02E+00
Unit 1 Steam Tunnel@ Grid M32	B206207-CJFCCV-003	2.0	*1.06E-01*	*9.61E-02*	1.49E+01	1.03E+00
Unit 1 Steam Tunnel o/s West Valve Room	B206207-CJFCCV-004	2.0	*1.66E-01*	*7.57E-02*	3.97E+01	2.44E+00
Unit 1 Steam Tunnel @ End of Tunnel	B206207-CJFCCV-005	1.5	*6.56E-02*	*1.22E-01*	1.74E+01	1.70E+00
Unit 2 Steam Tunnel o/s West Valve Room	B206208-CJFCCV-001	2.0	*1.50E-01*	*1.17E-01*	6.72E+00	5.95E-01
Unit 2 Steam Tunnel o/s West Valve Room	B206208-CJFCCV-002	5.0	*1.35E-01*	*8.93E-02*	1.86E+01	1.97E-01

(a) Represents surface activity of floor following removal of loose contamination
(b) Represents average of activity over entire depth of core sample minus the surface activity
(c) Italicized values indicate MDC value.

Table 2-26 Turbine Building 560 Foot Elevation Floor Drain System Sediment Samples Gamma Spectroscopy Analysis

Location	Sample ID [b]	Sample Date	Analysis Date	Co-60 [a] (pCi/g)	Cs-137 [a] (pCi/g)
Turbine Bldg. 560 ft. South	S1-00333-CJSMSM-A001	04/01/13	04/22/13	*1.92E-01*	**1.08E-01**
	S1-00333-CJSMSM-A002	04/02/13	04/22/13	*2.61E-01*	**6.37E-01**
	S1-00333-CJSMSM-A004	04/10/13	04/11/13	*1.88E-01*	**6.03E-01**
Turbine Bldg. 560 ft. Central	S1-00333-CJSMSM-B001	04/03/13	04/09/13	*3.18E-01*	**1.24E+01**
	S1-00333-CJSMSM-B002	04/03/13	04/10/13	*2.29E-01*	**5.44E+00**
	S1-00333-CJSMSM-B003	04/03/13	04/09/13	*1.67E-01*	**2.05E+00**
	S1-00333-CJSMSM-B004	04/03/13	04/09/13	**3.02E-01**	**1.04E+01**
	S1-00333-CJSMSM-B006 (LAW)	04/17/13	04/18/13	*8.22E+00*	**7.79E+01**
	S1-00333-CJSMSM-B007	04/17/13	04/18/13	*1.50E-01*	**1.77E+01**
	S1-00333-CJSMSM-B008	04/17/13	04/18/13	*1.43E-01*	**9.46E-01**
Turbine Bldg. 560 ft. North	S1-00333-CJSMSM-C001	04/04/13	04/10/13	**2.77E-01**	**1.38E+00**
	S1-00333-CJSMSM-C002	04/04/13	04/16/13	**4.38E-01**	**4.72E+00**
	S1-00333-CJSMSM-C003	04/04/13	04/11/13	*1.36E-01*	**2.21E-01**
	S1-00333-CJSMSM-C004	04/16/13	04/17/13	*2.67E-01*	*3.38E-01*
	S1-00333-CJSMSM-C005	04/16/13	04/16/13	*1.88E-01*	*1.99E-01*
	S1-00333-CJSMSM-C006	04/16/13	04/17/13	*2.09E-01*	*2.51E-01*
	S1-00333-CJSMSM-C007	04/16/13	04/17/13	*3.11E-01*	*2.74E-01*
	S1-00333-CJSMSM-C008 (LAW)	04/16/13	04/16/13	*2.24E+01*	*2.87E+01*

a. Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.
b. "LAW" indicates Large Area Wipe

Table 2-27 Non-Impacted Open Land Survey Units – Characterization Survey Summary

Survey Unit	**10215**		**10216**		**10217**	
Surface Area	26,007.80 m^2		31,171.10 m^2		50,880.20 m^2	
Description	**Area Northwest of Switchyard**		**Area West of Switchyard**		**Area Southwest of Switchyard**	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	4	4	4	4	7	7
# >MDC	0	0	0	3	0	7
Mean	<0.06 pCi/g	<0.09 pCi/g	<0.04 pCi/g	0.23 pCi/g	<0.05 pCi/g	0.41 pCi/g
Median	<0.05 pCi/g	<0.09 pCi/g	<0.04 pCi/g	0.23 pCi/g	<0.05 pCi/g	0.43 pCi/g
Max	<0.09 pCi/g	<0.10 pCi/g	<0.05 pCi/g	0.26 pCi/g	<0.06 pCi/g	0.52 pCi/g
Min	<0.04 pCi/g	<0.08 pCi/g	<0.01 pCi/g	0.20 pCi/g	<0.05 pCi/g	0.26 pCi/g
Standard Deviation	N/A	N/A	N/A	0.03	N/A	0.09
ISOCS	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	14	14	16	16	27	27
# >MDC	0	1	0	8	0	10
Mean	<0.04 pCi/g	0.05 pCi/g	<0.04 pCi/g	0.19 pCi/g	<0.04 pCi/g	0.20 pCi/g
Median	<0.04 pCi/g	0.05 pCi/g	<0.04 pCi/g	0.27 pCi/g	<0.04 pCi/g	0.21 pCi/g
Max	<0.05 pCi/g	0.05 pCi/g	<0.05 pCi/g	0.27 pCi/g	<0.06 pCi/g	0.30 pCi/g
Min	<0.02 pCi/g	0.05 pCi/g	<0.03 pCi/g	0.07 pCi/g	<0.04 pCi/g	0.04 pCi/g
Standard Deviation	N/A	N/A	N/A	0.07	N/A	0.08
Surface Scans						
% Scanned	1%		1%		1%	
# of Alarms	0		0		0	
Mean Scan	3,995 cpm		4,828 cpm		4,563 cpm	
Max Scan	4,725 cpm		5,389 cpm		5,070 cpm	
Notes	1) 9 of 14 samples in west section of survey unit were relocated as area was deemed as inaccessible. 2) Additional ISOCS measurement (#14) was taken on discovered debris pile.		1) 9 of 16 samples in west section of survey unit were relocated as area was deemed as inaccessible.		1) 3 additional judgmental ISOCS measurement locations were added to account for additional area. 2) 1 additional ISOCS measurement added for discovered abandoned drain pipe.	

Table 2-27 Non-Impacted Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10302**				**10303**				**10304**			
Surface Area	64,739.50	m^2			68,847.10	m^2			34,009.70	m^2		
Description	**Northwest Corner of FSAR Area**				**Southwest Corner of FSAR Area**				**Southern Area of FSAR**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	11		11		25		25		5		5	
# >MDC	0		9		0		13		0		3	
Mean	<0.05	pCi/g	0.30	pCi/g	<0.06	pCi/g	0.17	pCi/g	<0.05	pCi/g	0.26	pCi/g
Median	<0.06	pCi/g	0.27	pCi/g	<0.06	pCi/g	0.15	pCi/g	<0.06	pCi/g	0.23	pCi/g
Max	<0.08	pCi/g	0.46	pCi/g	<0.10	pCi/g	0.39	pCi/g	<0.07	pCi/g	0.35	pCi/g
Min	<0.02	pCi/g	0.12	pCi/g	<0.02	pCi/g	0.06	pCi/g	<0.02	pCi/g	0.21	pCi/g
Standard Deviation	N/A		0.11		N/A		0.10		N/A		0.08	
ISOCS	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	19		19		18		18		17		17	
# >MDC	0		7		0		9		0		8	
Mean	<0.05	pCi/g	0.25	pCi/g	<0.05	pCi/g	0.21	pCi/g	<0.04	pCi/g	0.17	pCi/g
Median	<0.05	pCi/g	0.22	pCi/g	<0.05	pCi/g	0.23	pCi/g	<0.04	pCi/g	0.16	pCi/g
Max	<0.06	pCi/g	0.34	pCi/g	<0.06	pCi/g	0.30	pCi/g	<0.05	pCi/g	0.27	pCi/g
Min	<0.04	pCi/g	0.20	pCi/g	<0.03	pCi/g	0.09	pCi/g	<0.02	pCi/g	0.09	pCi/g
Standard Deviation	N/A		0.06		N/A		0.07		N/A		0.06	
Surface Scans												
% Scanned	1%				1%				1%			
# of Alarms	0				7				4			
Mean Scan	4,951	cpm			5,858	cpm			4,456	cpm		
Max Scan	7,474	cpm			9,550	cpm			5,704	cpm		
Notes	1) 33 designed locations reduced to 22 due to accessibility. 2) 3 of 22 locations scanned with NaI & soil sample only. 3) 6 of 18 ISOCS taken at height of 1 meter due to obstructions – FOV = 3 m^2.				1) 35 designed locations reduced to 18 due to accessibility. 2) 5 of 18 locations scanned with NaI & soil sample only. 3) 17 additional random and 3 additional judgmental soil samples taken.				1) 16 of 17 random locations relocated due to accessibility.			

Table 2-27 Non-Impacted Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	10305				10306				10402			
Surface Area	121,535.20 m²				85,267.80 m²				133,565.00 m²			
Description	**Area West of Survey Unit #10217**				**Area West of Survey Unit #10216**				**MET Tower Area**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	17		17		4		4		41		41	
# >MDC	0		4		0		4		0		31	
Mean	<0.05	pCi/g	0.19	pCi/g	<0.05	pCi/g	0.26	pCi/g	<0.04	pCi/g	0.20	pCi/g
Median	<0.05	pCi/g	0.18	pCi/g	<0.04	pCi/g	0.23	pCi/g	<0.04	pCi/g	0.18	pCi/g
Max	<0.08	pCi/g	0.33	pCi/g	<0.08	pCi/g	0.40	pCi/g	<0.07	pCi/g	0.42	pCi/g
Min	<0.01	pCi/g	0.07	pCi/g	<0.02	pCi/g	0.20	pCi/g	<0.01	pCi/g	0.04	pCi/g
Standard Deviation	N/A		0.14		N/A		0.09		N/A		0.10	
ISOCS	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	61		61		18		18		39		39	
# >MDC	0		10		0		14		0		5	
Mean	<0.04	pCi/g	0.19	pCi/g	<0.04	pCi/g	0.17	pCi/g	<0.05	pCi/g	0.10	pCi/g
Median	<0.04	pCi/g	0.20	pCi/g	<0.04	pCi/g	0.17	pCi/g	<0.05	pCi/g	0.11	pCi/g
Max	<0.06	pCi/g	0.30	pCi/g	<0.06	pCi/g	0.27	pCi/g	<0.08	pCi/g	0.12	pCi/g
Min	<0.03	pCi/g	0.08	pCi/g	<0.01	pCi/g	0.08	pCi/g	<0.03	pCi/g	0.07	pCi/g
Standard Deviation	N/A		0.06		N/A		0.05		N/A		0.02	
Surface Scans												
% Scanned	1%				1%				1%			
# of Alarms	9				0				0			
Mean Scan	5,473	cpm			4,407	cpm			4,835	cpm		
Max Scan	7,343	cpm			5,337	cpm			5,543	cpm		
Notes					1) 43 designed ISOCS measurement locations reduced to 18 due to accessibility. 2) 4 of 18 locations scanned with NaI & soil sample only.				1) 67 designed ISOCS measurement locations reduced to 39 due to accessibility. 2) 30 additional random locations designated for NaI scanning and surface soil samples.			

Table 2-27 Non-Impacted Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10403**				**10404**			
Surface Area	139,282.00 m²				118,735.00 m²			
Description	**Area North of West Training**				**NW Corner of Owner Controlled Property**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	18		18		30		30	
# >MDC	0		12		0		20	
Mean	<0.04	pCi/g	0.18	pCi/g	<0.04	pCi/g	0.20	pCi/g
Median	<0.04	pCi/g	0.17	pCi/g	<0.04	pCi/g	0.20	pCi/g
Max	<0.07	pCi/g	0.39	pCi/g	<0.06	pCi/g	0.57	pCi/g
Min	<0.01	pCi/g	0.05	pCi/g	<0.01	pCi/g	0.02	pCi/g
Standard Deviation	N/A		0.09		N/A		0.14	
ISOCS	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	7		7		0		0	
# >MDC	0		3		0		0	
Mean	<0.04	pCi/g	0.19	pCi/g	N/A	pCi/g	N/A	pCi/g
Median	<0.04	pCi/g	0.19	pCi/g	N/A	pCi/g	N/A	pCi/g
Max	<0.05	pCi/g	0.23	pCi/g	N/A	pCi/g	N/A	pCi/g
Min	<0.03	pCi/g	0.14	pCi/g	N/A	pCi/g	N/A	pCi/g
Standard Deviation	N/A		0.04		N/A		N/A	
Surface Scans								
% Scanned	<1%				1%			
# of Alarms	4				0			
Mean Scan	3,959	cpm			3,391	cpm		
Max Scan	5,310	cpm			6,964	cpm		
Notes	1) Approximately 90% of area was inaccessible to the ISOCS. 2) 70 ISOCS measurements reduced to 7. 3) 4.8% of accessible 13,928 m² of land area scanned. 4) 16 additional random locations designated for NaI scanning and surface soil samples.				1) 60 ISOCS locations were designed. Entire area was inaccessible to the ISOCS. To compensate, additional scan and soil sample locations were evenly distributed in areas that were accessible. The total area scanned equates to 1,200 m², or approximately 1%.			

Table 2-28 Recommended Judgmental Sample Population Size

The following are the recommended judgmental sample population sizes for impacted survey units that will be subjected to FSS. The judgmental sample population should be taken in addition to any random-based sample population. These are recommended minimum sample population sizes.

Area Classification	Structure	Land
Class 1	8 judgmental samples per survey unit	8 judgmental samples per survey unit
Class 2	30 judgmental samples per survey unit	30 judgmental samples per survey unit
Class 3	13 judgmental samples per survey unit	13 judgmental samples per survey unit

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary

Survey Unit	**10201**		**10202**		**10203**	
Surface Area	6,028.20 m²		6,843.90 m²		9,997.80 m²	
Description	**NE Corner Restricted Area - Lakeshore**		**IRSF/Fire Training Area**		**East Training Area**	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	17	17	17	17	20	20
# >MDA	0	0	0	0	0	5
Mean	<0.05 pCi/g	<0.07 pCi/g	<0.04 pCi/g	<0.06 pCi/g	<0.08 pCi/g	0.11 pCi/g
Median	<0.06 pCi/g	<0.06 pCi/g	<0.05 pCi/g	<0.06 pCi/g	<0.08 pCi/g	0.11 pCi/g
Max	<0.09 pCi/g	<0.11 pCi/g	<0.08 pCi/g	<0.10 pCi/g	<0.11 pCi/g	0.16 pCi/g
Min	<0.02 pCi/g	<0.02 pCi/g	<0.02 pCi/g	<0.01 pCi/g	<0.02 pCi/g	0.06 pCi/g
Standard Deviation	N/A	N/A	N/A	N/A	N/A	0.04
Subsurface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	2	2	2	2	2	2
# >MDC	0	0	0	0	0	0
Mean	<0.05 pCi/g	<0.07 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.08 pCi/g
Median	<0.05 pCi/g	<0.07 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.08 pCi/g
Max	<0.06 pCi/g	<0.08 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.10 pCi/g
Min	<0.05 pCi/g	<0.05 pCi/g	<0.05 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.07 pCi/g
Standard Deviation	N/A	N/A	N/A	N/A	N/A	N/A
Surface Scans						
% Scanned	25%		27%		25%	
# of Alarms	0		0		4	
Mean Scan	1,224 cpm		1,681 cpm		2,977 cpm	
Max Scan	1,967 cpm		2,379 cpm		8,706 cpm	
Notes	1) 13 judgmental locations. 2) Subsurface soil samples taken to 1-meter.		1) 13 judgmental locations. 2) Subsurface soil samples taken to 1-meter.		1) 13 judgmental and 2 investigative locations. 2) Subsurface soil samples taken to 1-meter.	

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	10204		10205		10206	
Surface Area	7,228.10 m^2		54,573.00 m^2		10,529.10 m^2	
Description	North Gate Area		Switchyard		Station Construction Area	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	18	18	32	32	19	19
# >MDC	0	1	0	0	0	0
Mean	<0.07 pCi/g	0.11 pCi/g	<0.05 pCi/g	<0.07 pCi/g	<0.04 pCi/g	<0.05 pCi/g
Median	<0.05 pCi/g	0.11 pCi/g	<0.05 pCi/g	<0.07 pCi/g	<0.05 pCi/g	<0.05 pCi/g
Max	<0.46 pCi/g	0.11 pCi/g	<0.08 pCi/g	<0.09 pCi/g	<0.12 pCi/g	<0.08 pCi/g
Min	<0.02 pCi/g	0.11 pCi/g	<0.05 pCi/g	<0.05 pCi/g	<0.02 pCi/g	<0.01 pCi/g
Standard Deviation	N/A	N/A	N/A	N/A	N/A	N/A
Subsurface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	2	2	0	0	2	2
# >MDC	0	0	0	0	0	0
Mean	<0.05 pCi/g	<0.08 pCi/g	N/A pCi/g	N/A pCi/g	<0.05 pCi/g	<0.05 pCi/g
Median	<0.05 pCi/g	<0.08 pCi/g	N/A pCi/g	N/A pCi/g	<0.05 pCi/g	<0.05 pCi/g
Max	<0.06 pCi/g	<0.11 pCi/g	N/A pCi/g	N/A pCi/g	<0.06 pCi/g	<0.05 pCi/g
Min	<0.05 pCi/g	<0.06 pCi/g	N/A pCi/g	N/A pCi/g	<0.05 pCi/g	<0.05 pCi/g
Standard Deviation	N/A	N/A	N/A	N/A	N/A	N/A
Surface Scans						
% Scanned	26%		10%		27%	
# of Alarms	2		1		2	
Mean Scan	2,071 cpm		2,594 cpm		3,386 cpm	
Max Scan	5,437 cpm		6,246 cpm		7,406 cpm	
Notes	1) All samples were dried prior to analysis. 2) Subsurface soil samples taken to 1-meter.		1) Initial survey design called for 25% scan coverage. Scan suspended at 10% due to safety concerns. 2) Additional surface samples taken to compensate for loss of scan coverage.		1) All samples were dried prior to analysis. 2) Subsurface soil samples taken to 1-meter.	

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Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10207**				**10208**				**10209**			
Surface Area	10,273.80	m^2			11,821.00	m^2			5,970.80	m^2		
Description	**North Warehouse Area**				**South Warehouse Area**				**Protected Area South of Gate House**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	19		19		33		33		13		13	
# >MDA	0		0		0		0		0		1	
Mean	<0.05	pCi/g	<0.07	pCi/g	<0.04	pCi/g	<0.05	pCi/g	<0.05	pCi/g	0.12	pCi/g
Median	<0.05	pCi/g	<0.07	pCi/g	<0.04	pCi/g	<0.05	pCi/g	<0.05	pCi/g	0.12	pCi/g
Max	<0.09	pCi/g	<0.12	pCi/g	<0.06	pCi/g	<0.08	pCi/g	<0.11	pCi/g	0.12	pCi/g
Min	<0.02	pCi/g	<0.02	pCi/g	<0.01	pCi/g	<0.01	pCi/g	<0.03	pCi/g	0.12	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	2		2		3		3		2		2	
# >MDC	0		0		0		0		0		0	
Mean	<0.06	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.08	pCi/g	<0.07	pCi/g	<0.07	pCi/g
Median	<0.06	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.09	pCi/g	<0.07	pCi/g	<0.07	pCi/g
Max	<0.07	pCi/g	<0.07	pCi/g	<0.07	pCi/g	<0.10	pCi/g	<0.09	pCi/g	<0.08	pCi/g
Min	<0.05	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.04	pCi/g	<0.06	pCi/g	<0.07	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Surface Scans												
% Scanned	29%				34%				26%			
# of Alarms	6				10				2			
Mean Scan	5,688	cpm			3,274	cpm			2,281	cpm		
Max Scan	12,193	cpm			6,464	cpm			3,006	cpm		
Notes	1) All samples were dried prior to analysis. 2) Subsurface soil samples taken to 1-meter. 3) Elevated scan readings due to the presence of packaged radioactive material in vicinity during scan performance.				1) 8 judgmental and 3 investigative locations. 2) Subsurface soil samples taken to 1-meter.				1) 5 judgmental and 2 investigative locations. 2) Subsurface soil samples taken to 1-meter.			

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10210**				**10211**				**10212A**			
Surface Area	5,593.90	m^2			3,198.30	m^2			12,225.70	m^2		
Description	**Protected Area South of Turbine Building**				**SE Corner Protected Area - Lakeshore**				**North Protected Area Fence - Lakeshore**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	17		17		16		16		32		32	
# >MDC	0		0		0		1		0		6	
Mean	<0.04	pCi/g	<0.05	pCi/g	<0.04	pCi/g	0.02	pCi/g	<0.04	pCi/g	0.11	pCi/g
Median	<0.04	pCi/g	<0.05	pCi/g	<0.04	pCi/g	0.02	pCi/g	<0.04	pCi/g	0.10	pCi/g
Max	<0.05	pCi/g	<0.07	pCi/g	<0.06	pCi/g	0.02	pCi/g	<0.10	pCi/g	0.14	pCi/g
Min	<0.01	pCi/g	<0.02	pCi/g	<0.01	pCi/g	0.02	pCi/g	<0.01	pCi/g	0.09	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		0.02	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	3		3		2		2		32		32	
# >MDC	0		0		0		0		0		1	
Mean	<0.07	pCi/g	<0.08	pCi/g	<0.04	pCi/g	<0.08	pCi/g	<0.04	pCi/g	0.08	pCi/g
Median	<0.07	pCi/g	<0.08	pCi/g	<0.04	pCi/g	<0.08	pCi/g	<0.04	pCi/g	0.08	pCi/g
Max	<0.08	pCi/g	<0.09	pCi/g	<0.05	pCi/g	<0.09	pCi/g	<0.06	pCi/g	0.08	pCi/g
Min	<0.06	pCi/g	<0.07	pCi/g	<0.02	pCi/g	<0.07	pCi/g	<0.01	pCi/g	0.08	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Surface Scans												
% Scanned	26%				30%				25%			
# of Alarms	40				0				9			
Mean Scan	2,686	cpm			2,619	cpm			5,363	cpm		
Max Scan	4,585	cpm			3,327	cpm			9,014	cpm		
Notes	1) High number of scan alarms due to improper set-point –area rescanned with no alarms. 2) 13 judgmental and 1 investigation sample location. 3) All subsurface samples taken to 1-meter depth, sample #01 – taken to 2-meter depth.				1) 13 judgmental and 1 investigative locations. 2) Subsurface soil samples taken to 1-meter.				1) 5 judgmental and 5 investigative locations. 2) Subsurface soil samples taken to 1-meter. 3) All judgmental samples analyzed for H-3, no activity detected greater than MDA.			

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Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10213A**				**10214**				**10218A**			
Surface Area	12,255.30	m^2			33,551.40	m^2			11,559.00	m^2		
Description	**Power House Area**				**Construction Parking Lot**				**ISFSI Area East**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	24		24		31		31		32		32	
# >MDA	0		6		0		4		0		11	
Mean	<0.05	pCi/g	0.17	pCi/g	<0.04	pCi/g	0.09	pCi/g	<0.04	pCi/g	0.06	pCi/g
Median	<0.05	pCi/g	0.10	pCi/g	<0.04	pCi/g	0.10	pCi/g	<0.04	pCi/g	0.05	pCi/g
Max	<0.08	pCi/g	0.29	pCi/g	<0.09	pCi/g	0.11	pCi/g	<0.08	pCi/g	0.12	pCi/g
Min	<0.01	pCi/g	0.09	pCi/g	<0.01	pCi/g	0.07	pCi/g	<0.02	pCi/g	0.03	pCi/g
Standard Deviation	N/A		0.08		N/A		0.02		N/A		0.03	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	24		24		3		3		31		31	
# >MDC	0		2		0		0		0		9	
Mean	<0.04	pCi/g	0.07	pCi/g	<0.05	pCi/g	<0.04	pCi/g	<0.04	pCi/g	0.05	pCi/g
Median	<0.04	pCi/g	0.07	pCi/g	<0.04	pCi/g	<0.04	pCi/g	<0.04	pCi/g	0.05	pCi/g
Max	<0.06	pCi/g	0.08	pCi/g	<0.09	pCi/g	<0.04	pCi/g	<0.07	pCi/g	0.08	pCi/g
Min	<0.01	pCi/g	0.07	pCi/g	<0.03	pCi/g	<0.04	pCi/g	<0.02	pCi/g	0.03	pCi/g
Standard Deviation	N/A		0.01		N/A		N/A		N/A		0.02	
Surface Scans												
% Scanned	25%				10%				50%			
# of Alarms	3				2				29			
Mean Scan	5,459	cpm			2,145	cpm			4,315	cpm		
Max Scan	9,798	cpm			3,798	cpm			7,440	cpm		
Notes	1) Subsurface soil samples taken to 1-meter.				1) Subsurface soil samples taken to 1-meter.				1) Area survey by ESCSG as stand-alone survey. 2) Scan alarms due to bkgd from adjacent radioactive material, area rescanned – no alarms. 3) Several samples analyzed for H-3, Fe-55, Ni-63 and Sr-90, no positive results > MDC.			

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Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	10218B		10218F		10219A	
Surface Area	11,559.00 m²		3,151.70 m²		2,433.20 m²	
Description	ISFSI Area East		Area Near South of Switchyard		Area Far South of Switchyard (Part A)	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	30	30	4	4	25	25
# >MDA	0	16	0	0	0	4
Mean	<0.05 pCi/g	0.08 pCi/g	<0.04 pCi/g	<0.06 pCi/g	<0.06 pCi/g	0.19 pCi/g
Median	<0.04 pCi/g	0.07 pCi/g	<0.04 pCi/g	<0.06 pCi/g	<0.05 pCi/g	0.15 pCi/g
Max	<0.08 pCi/g	0.31 pCi/g	<0.04 pCi/g	<0.09 pCi/g	<0.12 pCi/g	0.34 pCi/g
Min	<0.02 pCi/g	0.03 pCi/g	<0.03 pCi/g	<0.04 pCi/g	<0.01 pCi/g	0.11 pCi/g
Standard Deviation	N/A	0.07	N/A	N/A	N/A	0.11
Subsurface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	31	31	0	0	25	25
# >MDC	0	10	0	0	0	2
Mean	<0.04 pCi/g	0.05 pCi/g	N/A pCi/g	N/A pCi/g	<0.05 pCi/g	0.14 pCi/g
Median	<0.04 pCi/g	0.05 pCi/g	N/A pCi/g	N/A pCi/g	<0.05 pCi/g	0.14 pCi/g
Max	<0.08 pCi/g	0.08 pCi/g	N/A pCi/g	N/A pCi/g	<0.08 pCi/g	0.15 pCi/g
Min	<0.03 pCi/g	0.01 pCi/g	N/A pCi/g	N/A pCi/g	<0.01 pCi/g	0.13 pCi/g
Standard Deviation	N/A	0.03	N/A	N/A	N/A	0.02
Surface Scans						
% Scanned	50%		13%		50%	
# of Alarms	21		0		27	
Mean Scan	4,883 cpm		1,507 cpm		6,468 cpm	
Max Scan	6,897 cpm		1,964 cpm		7,671 cpm	
Notes	1) Area survey by ESCSG as stand-alone survey. 2) Scan alarms due to increase in ambient bkgd from adjacent radioactive material, area rescanned – no alarms. 3) Several samples analyzed for H-3, Fe-55, Ni-63 and Sr-90, no positive results > MDC.		1) Majority of surface area in survey unit occupied by ISFSI warehouse.		1) High number of scan alarms due to improper adjustment of set-point –readjusted and area rescanned with no alarms. 2) Subsurface soil samples taken to 1-meter. 3) Surface & subsurface biased samples analyzed for H-3, no activity detected > MDC.	

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	10219B				10220A				10220B			
Surface Area	7,515.70 m²				8,192.00 m²				8,271.20 m²			
Description	Area Far South of Switchyard (Part B)				SE Corner of Exclusion Area – Lake Shore				SE Corner of Exclusion Area – Inland			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	25		25		28		28		31		31	
# >MDA	0		15		1		4		0		4	
Mean	<0.05	pCi/g	0.24	pCi/g	0.13	pCi/g	0.15	pCi/g	<0.05	pCi/g	0.15	pCi/g
Median	<0.06	pCi/g	0.25	pCi/g	0.13	pCi/g	0.14	pCi/g	<0.05	pCi/g	0.12	pCi/g
Max	<0.09	pCi/g	0.41	pCi/g	0.13	pCi/g	0.24	pCi/g	<0.12	pCi/g	0.27	pCi/g
Min	<0.01	pCi/g	0.09	pCi/g	0.13	pCi/g	0.06	pCi/g	<0.01	pCi/g	0.08	pCi/g
Standard Deviation	N/A		0.09		N/A		0.08		N/A		0.09	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	0		0		28		28		31		31	
# >MDA	0		0		0		1		0		2	
Mean	N/A	pCi/g	N/A	pCi/g	<0.04	pCi/g	0.11	pCi/g	<0.05	pCi/g	0.12	pCi/g
Median	N/A	pCi/g	N/A	pCi/g	<0.04	pCi/g	0.11	pCi/g	<0.04	pCi/g	0.12	pCi/g
Max	N/A	pCi/g	N/A	pCi/g	<0.07	pCi/g	0.11	pCi/g	<0.09	pCi/g	0.12	pCi/g
Min	N/A	pCi/g	N/A	pCi/g	<0.01	pCi/g	0.11	pCi/g	<0.01	pCi/g	0.11	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		0.01	
Surface Scans												
% Scanned	11%				50%				50%			
# of Alarms	3				2				18			
Mean Scan	4,926	cpm			6,766	cpm			5,089	cpm		
Max Scan	6,355	cpm			9,830	cpm			10,221	cpm		
Notes	1) No subsurface soil samples taken in this survey unit. 2) 21 random and 4 investigation surface soil samples taken.				1) Subsurface soil samples taken to 1-meter. 2) Investigation samples taken in "wet-land" area. 3) Investigation samples analyzed for H-3, no activity detected greater than MDC.				1) Subsurface soil samples taken to 1-meter. 2) Investigation samples analyzed for H-3, no activity detected >MDC. 3) Sample location #12 not sampled due to safety concerns.			

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	10220C				10221A				10221B			
Surface Area	25,560.00 m²				6,273.60 m²				6,374.00 m²			
Description	SE Corner of Exclusion Area – Inland				South of Protected Area - Lakeshore				South of Protected Area - Inland			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	55		55		30		30		26		26	
# >MDC	0		41		0		3		0		2	
Mean	<0.05	pCi/g	0.33	pCi/g	<0.04	pCi/g	0.16	pCi/g	<0.04	pCi/g	0.06	pCi/g
Median	<0.05	pCi/g	0.26	pCi/g	<0.04	pCi/g	0.13	pCi/g	<0.05	pCi/g	0.06	pCi/g
Max	<0.18	pCi/g	1.14	pCi/g	<0.08	pCi/g	0.23	pCi/g	<0.07	pCi/g	0.07	pCi/g
Min	<0.01	pCi/g	0.09	pCi/g	<0.01	pCi/g	0.12	pCi/g	<0.01	pCi/g	0.05	pCi/g
Standard Deviation	N/A		0.25		N/A		0.06		N/A		0.01	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	0		0		30		30		3		3	
# >MDC	0		0		0		1		0		0	
Mean	N/A	pCi/g	N/A	pCi/g	<0.03	pCi/g	0.04	pCi/g	<0.04	pCi/g	<0.07	pCi/g
Median	N/A	pCi/g	N/A	pCi/g	<0.04	pCi/g	0.04	pCi/g	<0.02	pCi/g	<0.07	pCi/g
Max	N/A	pCi/g	N/A	pCi/g	<0.05	pCi/g	0.04	pCi/g	<0.08	pCi/g	<0.09	pCi/g
Min	N/A	pCi/g	N/A	pCi/g	<0.01	pCi/g	0.04	pCi/g	<0.02	pCi/g	<0.05	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Surface Scans												
% Scanned	14%				50%				26%			
# of Alarms	9				0				2			
Mean Scan	6,037	cpm			5,053	cpm			3,053	cpm		
Max Scan	8,418	cpm			9,970	cpm			6,464	cpm		
Notes	1) Extensive investigation of Bull Creek, where elevated Cs-137 soil sample was located. 2) Investigation concluded that elevated Cs-137 due to concentration of global fallout. 3) Investigation samples 10 thru 20 taken outside of survey unit.				1) Portion of survey unit surveyed by ESCSG as part of "rail spur" survey. 2) Subsurface soil samples taken to 1-meter.				1) Portion of survey unit surveyed by ESCSG as part of "rail spur" survey. 2) Subsurface soil samples taken to 1-meter. 3) Subsurface sample #12 labeled as "QC" sample when actually taken as random sample.			

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10222**				**10223**				**10224**			
Surface Area	21,777.80	m^2			12,371.40	m^2			14,607.70	m^2		
Description	**North Beach Area**				**Power Block Beach Area**				**South Beach Area**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	30		30		20		20		22		22	
# >MDC	0		0		0		0		0		0	
Mean	<0.05	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.04	pCi/g	<0.07	pCi/g
Median	<0.05	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.06	pCi/g	<0.05	pCi/g	<0.07	pCi/g
Max	<0.08	pCi/g	<0.09	pCi/g	<0.07	pCi/g	<0.08	pCi/g	<0.07	pCi/g	<0.08	pCi/g
Min	<0.02	pCi/g	<0.03	pCi/g	<0.02	pCi/g	<0.02	pCi/g	<0.02	pCi/g	<0.04	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	45		45		21		21		24		24	
# >MDC	0		0		0		0		0		1	
Mean	<0.04	pCi/g	<0.05	pCi/g	<0.04	pCi/g	<0.05	pCi/g	<0.04	pCi/g	0.03	pCi/g
Median	<0.04	pCi/g	<0.05	pCi/g	<0.04	pCi/g	<0.05	pCi/g	<0.04	pCi/g	0.03	pCi/g
Max	<0.06	pCi/g	<0.07	pCi/g	<0.06	pCi/g	<0.08	pCi/g	<0.07	pCi/g	0.03	pCi/g
Min	<0.01	pCi/g	<0.03	pCi/g	<0.02	pCi/g	<0.03	pCi/g	<0.01	pCi/g	0.03	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Surface Scans												
% Scanned	56%				50%				52%			
# of Alarms	102				0				0			
Mean Scan	2,139	cpm			1,788	cpm			2,744	cpm		
Max Scan	4,316	cpm			2,277	cpm			3,400	cpm		
Notes	1) "Black Beauty" sand-blasting sand cause of elevated scans alarms investigation soil samples taken. 2) Analysis of investigation samples indicate natural U & Th 3) Subsurface soil samples taken to 3-meters.				1) Subsurface soil samples taken to 3-meters.				1) Subsurface soil samples taken to 3-meters.			

Table 2-29 Impacted Class 3 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10301**				**VCC**			
Surface Area	55,941.60	m^2			15,364.00	m^2		
Description	**West Training Area**				**Vertical Concrete Cask Construction Area**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	22		22		30		30	
# >MDA	0		11		0		7	
Mean	<0.05	pCi/g	0.17	pCi/g	<0.04	pCi/g	0.04	pCi/g
Median	<0.06	pCi/g	0.14	pCi/g	<0.04	pCi/g	0.04	pCi/g
Max	<0.08	pCi/g	0.39	pCi/g	<0.08	pCi/g	0.08	pCi/g
Min	<0.02	pCi/g	0.06	pCi/g	<0.03	pCi/g	0.03	pCi/g
Standard Deviation	N/A		0.12		N/A		0.02	
Subsurface Soil Samples	**ISOCS**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements			46		30		30	
# >MDC			7		0		3	
Mean			0.18	pCi/g	<0.05	pCi/g	0.04	pCi/g
Median			0.19	pCi/g	<0.04	pCi/g	0.03	pCi/g
Max			0.25	pCi/g	<0.10	pCi/g	0.06	pCi/g
Min			0.11	pCi/g	<0.02	pCi/g	0.03	pCi/g
Standard Deviation			0.05		N/A		0.02	
Surface Scans								
% Scanned	14%				20%			
# of Alarms	23				0			
Mean Scan	5,433	cpm			5,192	cpm		
Max Scan	9,499	cpm			9,240	cpm		
Notes	1) No subsurface soil samples taken in this survey unit. 2) 46 ISOCS measurements acquired.				1) Area survey by ESCSG as stand-alone survey – documented in separate report. 2) Subsurface soil samples taken to 1-meter. 3) Following survey, area was paved for use as VCC Construction Area.			

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Table 2-30 Impacted Class 2 Open Land Survey Units – Characterization Survey Summary

Survey Unit	**12201**		**12202**		**12203**	
Surface Area	9,610.40 m^2		7,573.90 m^2		7,568.70 m^2	
Description	**North Restricted Area Yard**		**Gate House and Southwest Yard**		**Under Service Building and South East Yard**	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	15	15	16	16	15	15
# >MDC	0	2	0	1	0	3
Mean	<0.06 pCi/g	0.15 pCi/g	<0.06 pCi/g	0.12 pCi/g	<0.06 pCi/g	0.15 pCi/g
Median	<0.07 pCi/g	0.15 pCi/g	<0.06 pCi/g	0.12 pCi/g	<0.06 pCi/g	0.18 pCi/g
Max	<0.10 pCi/g	0.17 pCi/g	<0.09 pCi/g	0.12 pCi/g	<0.10 pCi/g	0.18 pCi/g
Min	<0.02 pCi/g	0.12 pCi/g	<0.02 pCi/g	0.12 pCi/g	<0.02 pCi/g	0.08 pCi/g
Standard Deviation	N/A	0.04	N/A	N/A	N/A	0.06
Subsurface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	9	9	9	9	6	6
# >MDC	0	0	0	0	0	1
Mean	<0.06 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.08 pCi/g	<0.04 pCi/g	0.15 pCi/g
Median	<0.06 pCi/g	<0.06 pCi/g	<0.06 pCi/g	<0.07 pCi/g	<0.04 pCi/g	0.15 pCi/g
Max	<0.07 pCi/g	<0.07 pCi/g	<0.10 pCi/g	<0.14 pCi/g	<0.08 pCi/g	0.15 pCi/g
Min	<0.02 pCi/g	<0.05 pCi/g	<0.02 pCi/g	<0.05 pCi/g	<0.02 pCi/g	0.15 pCi/g
Standard Deviation	N/A	N/A	N/A	N/A	N/A	N/A
Surface Scans						
% Scanned	66%		26%		45%	
# of Alarms	4		12		1	
Mean Scan	2,132 cpm		4,805 cpm		2,541 cpm	
Max Scan	3,611 cpm		11,264 cpm		5,677 cpm	
Notes	1) Subsurface soil samples taken to 3-meters.		1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & systems in vicinity. 3) Sample #11 labeled as "QC" when actually random sample. 4) Scans performed under 2 sample plans.		1) Subsurface soil samples taken to 3-meters. 2) ~50% of surface area either obstructed or covered.	

Table 2-30 Impacted Class 2 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**10204**			
Surface Area	5,908.70	m^2		
Description	**Crib House Area**			
Surface Soil Samples	**Co-60**		**Cs-137**	
# of Samples	13		13	
# >MDC	0		6	
Mean	<0.08	pCi/g	0.15	pCi/g
Median	<0.08	pCi/g	0.14	pCi/g
Max	<0.10	pCi/g	0.21	pCi/g
Min	<0.07	pCi/g	0.12	pCi/g
Standard Deviation	N/A		0.04	
Subsurface Soil Samples	**Co-60**		**Cs-137**	
# of Measurements	36		36	
# >MDC	0		2	
Mean	<0.05	pCi/g	0.08	pCi/g
Median	<0.05	pCi/g	0.08	pCi/g
Max	<0.09	pCi/g	0.12	pCi/g
Min	<0.01	pCi/g	0.04	pCi/g
Standard Deviation	N/A		0.06	
Surface Scans				
% Scanned	30%			
# of Alarms	102			
Mean Scan	2,569	cpm		
Max Scan	6,203	cpm		
Notes	1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & contaminated systems in vicinity.			

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Table 2-31 Impacted Class 1 Open Land Survey Units – Characterization Survey Summary

Survey Unit	**12101**				**12102**				**12103**			
Surface Area	2,035.70 m^2				2,024.50 m^2				2,033.60 m^2			
Description	**WWTF Sludge Drying Bed Area**				**WWTF Facility**				**Unit 2 PWST/SST Area**			
Surface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Samples	13		13		13		13		13		13	
# >MDC	0		2		0		4		0		2	
Mean	<0.05	pCi/g	0.19	pCi/g	<0.04	pCi/g	0.32	pCi/g	<0.06	pCi/g	0.39	pCi/g
Median	<0.06	pCi/g	0.19	pCi/g	<0.05	pCi/g	0.25	pCi/g	<0.07	pCi/g	0.39	pCi/g
Max	<0.09	pCi/g	0.21	pCi/g	<0.06	pCi/g	0.70	pCi/g	<0.09	pCi/g	0.46	pCi/g
Min	<0.02	pCi/g	0.17	pCi/g	<0.02	pCi/g	0.09	pCi/g	<0.03	pCi/g	0.33	pCi/g
Standard Deviation	N/A		0.03		N/A		0.28		N/A		0.09	
Subsurface Soil Samples	**Co-60**		**Cs-137**		**Co-60**		**Cs-137**		**Co-60**		**Cs-137**	
# of Measurements	38		38		38		38		39		39	
# >MDC	0		1		0		0		1		0	
Mean	<0.06	pCi/g	0.08	pCi/g	<0.06	pCi/g	<0.08	pCi/g	0.10	pCi/g	<0.06	pCi/g
Median	<0.06	pCi/g	0.08	pCi/g	<0.06	pCi/g	<0.08	pCi/g	0.10	pCi/g	<0.06	pCi/g
Max	<0.13	pCi/g	0.08	pCi/g	<0.10	pCi/g	<0.16	pCi/g	0.10	pCi/g	<0.09	pCi/g
Min	<0.02	pCi/g	0.08	pCi/g	<0.02	pCi/g	<0.04	pCi/g	0.10	pCi/g	<0.03	pCi/g
Standard Deviation	N/A		N/A		N/A		N/A		N/A		N/A	
Surface Scans												
% Scanned	100%				100%				100%			
# of Alarms	27				83				232			
Mean Scan	3,293	cpm			2,767	cpm			3,320	cpm		
Max Scan	35,962	cpm			10,759	cpm			30,660	cpm		
Notes	1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & contaminated systems in vicinity. 3) No HTD analysis performed.				1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & contaminated systems in vicinity. 3) Surface soil sample #s 5, 7 & 8 sent to Eberline for HTD analysis.				1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & contaminated systems in vicinity. 3) Surface soil sample #s 12 & 13 and subsurface sample # 8 sent to Eberline for HTD analysis.			

Table 2-31 Impacted Class 1 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**12104**		**12109**		**12110**	
Surface Area	1940.3 m^2		1930.5 m^2		1739.7 m^2	
Description	**Area Under and North of Unit 2 CTMT**		**Area Under and South of Unit 1 CTMT**		**Yard Between U1 CTMT and TB**	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	1	1	16	16	13	13
# >MDC	0	0	14	16	0	6
Mean	<0.10 pCi/g	<0.05 pCi/g	0.13 pCi/g	0.18 pCi/g	<0.05 pCi/g	0.12 pCi/g
Median	<0.10 pCi/g	<0.05 pCi/g	0.06 pCi/g	0.14 pCi/g	<0.05 pCi/g	0.11 pCi/g
Max	<0.10 pCi/g	<0.05 pCi/g	1.04 pCi/g	0.59 pCi/g	<0.08 pCi/g	0.27 pCi/g
Min	<0.10 pCi/g	<0.05 pCi/g	0.02 pCi/g	0.05 pCi/g	<0.03 pCi/g	0.03 pCi/g
Standard Deviation	N/A	N/A	0.26	0.13	N/A	0.08
Subsurface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	9	9	15	15	24	24
# >MDC	0	0	0	1	0	2
Mean	<0.06 pCi/g	<0.07 pCi/g	<0.05 pCi/g	0.08 pCi/g	<0.07 pCi/g	0.27 pCi/g
Median	<0.06 pCi/g	<0.07 pCi/g	<0.05 pCi/g	0.08 pCi/g	<0.07 pCi/g	0.27 pCi/g
Max	<0.10 pCi/g	<0.09 pCi/g	<0.08 pCi/g	0.08 pCi/g	<0.09 pCi/g	0.30 pCi/g
Min	<0.02 pCi/g	<0.05 pCi/g	<0.03 pCi/g	0.08 pCi/g	<0.03 pCi/g	0.24 pCi/g
Standard Deviation	N/A	N/A	N/A	N/A	N/A	0.05
Surface Scans % Scanned # of Alarms Mean Scan Max Scan	On date, April 16, 2013, survey unit was not scanned due to prohibitive background from adjacent systems and stored RAM. Soil sample locations were biased based upon visual cues & opportune accessibility		Survey unit was not scanned. On date, April 14, 2013, area posted as "Radiation Area" for loading RAM on rail cars. Radiation area will remain until commodity removal is complete. Soil sample locations were biased based upon visual cues & accessibility.		Survey unit was not scanned. On date, April 25, 2013, area posted as "Radiation Area" for loading RAM on rail cars. Radiation area will remain until commodity removal is complete. Soil sample locations were biased based upon visual cues & accessibility.	
Notes	1) Subsurface soil samples taken to 3-meters. 2) No surface soil samples were taken at locations 2 and 3 as surface was covered with concrete.		1) Subsurface soil samples taken to 3-meters. 2) No surface soil samples were taken at locations 1 thru 4 as surface was covered with concrete. 3) SS #5 sent to Eberline for HTD analysis.		1) Subsurface soil samples taken to 3-meters. 2) SB sample #5 – no soil, SB sample #7 limited to 1-meter and SB sample #4 limited to 2-meter due to refusal. 3) No samples collected at 11, 12 & 13 due to potential ACM.	

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Table 2-31 Impacted Class 1 Open Land Survey Units – Characterization Survey Summary (continued)

Survey Unit	**12111**		**12112**		**12113**	
Surface Area	1964.3		1692.6 m²		1,657.60 m²	
Description	**South Yard Area NE of Gate House**		**Unit 1 PWST/SST Area East**		**Unit 1 PWST/SST Area West**	
Surface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Samples	13	13	14	14	15	15
# >MDC	0	10	0	9	0	9
Mean	<0.07 pCi/g	0.15 pCi/g	<0.08 pCi/g	0.82 pCi/g	<0.08 pCi/g	0.12 pCi/g
Median	<0.10 pCi/g	0.14 pCi/g	<0.08 pCi/g	0.23 pCi/g	<0.08 pCi/g	0.10 pCi/g
Max	<0.10 pCi/g	0.20 pCi/g	<0.12 pCi/g	3.39 pCi/g	<0.11 pCi/g	0.22 pCi/g
Min	<0.05 pCi/g	0.12 pCi/g	<0.02 pCi/g	0.07 pCi/g	<0.06 pCi/g	0.06 pCi/g
Standard Deviation	N/A	0.03	N/A	1.19	N/A	0.06
Subsurface Soil Samples	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**	**Co-60**	**Cs-137**
# of Measurements	42	42	39	39	39	39
# >MDC	0	1	0	2	0	0
Mean	<0.05 pCi/g	0.04 pCi/g	<0.05 pCi/g	0.45 pCi/g	<0.07 pCi/g	<0.07 pCi/g
Median	<0.05 pCi/g	0.04 pCi/g	<0.05 pCi/g	0.45 pCi/g	<0.07 pCi/g	<0.07 pCi/g
Max	<0.08 pCi/g	0.04 pCi/g	<0.07 pCi/g	0.70 pCi/g	<0.10 pCi/g	<0.12 pCi/g
Min	<0.03 pCi/g	0.04 pCi/g	<0.02 pCi/g	0.19 pCi/g	<0.03 pCi/g	<0.04 pCi/g
Standard Deviation	N/A	N/A	N/A	0.36	N/A	N/A
Surface Scans						
% Scanned	Survey unit was not scanned. On date, April 1, 2013, area posted as "Radiation Area" for loading RAM on rail cars. Radiation area will remain until commodity removal is complete. Soil sample locations were biased based upon visual cues & accessibility.		100%		100%	
# of Alarms			98		106	
Mean Scan			3,483 cpm		3,943 cpm	
Max Scan			5,225 cpm		7,562 cpm	
Notes	1) Subsurface soil samples taken to 3-meters. 2) SS #s 11, 12 & 13 – asphalt. 3) SS #s 1, 3 7 4, additional 1-meter sample taken. 4) No SS samples collected at 13, 14 & 15 due to safety concerns.		1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & contaminated systems in vicinity. 3) SS #s 1 & 2 – asphalt. 4) SS #s 4 & 6 sent to Eberline for HTD analysis.		1) Subsurface soil samples taken to 3-meters. 2) Elevated scans from RAM & contaminated systems in vicinity.	

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Table 2-32 Surface and Subsurface Soil Samples – Eberline Laboratory Analysis

Radionuclide	L1-12102-CJGSSS-0510 (pCi/g)	L1-12102-CJGSSS-0710 (pCi/g)	L1-12102-CJGSSS-0810 (pCi/g)	L1-12102-CQGSSS-0810 (pCi/g)	L1-12103-CJGSSS-1210 (pCi/g)	L1-12103-CJGSSS-1310 (pCi/g)	L1-12109-CJGS-050-SS (pCi/g)	L1-12112-CJGSSS-0410 (pCi/g)	L1-12112-CJGSSS-0610 (pCi/g)	L1-12103-CJGSSB-0821 (pCi/g)
H-3	*2.51E+00*	*2.73E+00*	*2.54E+00*	*2.48E+00*	*2.50E+00*	*2.51E+00*	*3.96E+00*	*3.56E+00*	*3.61E+00*	*2.69E+00*
C-14	*8.52E-01*	*8.10E-01*	*9.49E-01*	*8.76E-01*	*8.39E-01*	*8.51E-01*	*1.01E+00*	*1.07E+00*	*9.26E-01*	*9.56E-01*
Fe-55	*3.53E+00*	*3.19E+00*	*3.24E+00*	*3.47E+00*	*3.36E+00*	*3.06E+00*	N/A	*7.16E+00*	*6.34E+00*	*3.24E+00*
Ni-59	*7.28E+00*	*7.23E+00*	*6.65E+00*	*6.30E+00*	*7.40E+00*	*6.30E+00*	N/A	*6.12E+00*	*7.71E+00*	*5.69E+00*
Co-60	*4.82E-02*	*4.52E-02*	*3.00E-02*	*3.22E-02*	*4.29E-02*	*4.29E-02*	**2.43E-01**	*8.80E-02*	*1.22E-01*	*3.98E-02*
Ni-63	*6.22E-01*	*7.06E-01*	*6.24E-01*	*6.41E-01*	*6.21E-01*	*6.26E-01*	*1.77E+00*	*8.36E-01*	*7.95E-01*	*6.10E-01*
Sr-90	*2.90E-01*	*3.15E-01*	*2.35E-01*	*2.99E-01*	*2.67E-01*	*3.33E-01*	*4.70E-01*	*3.76E-01*	*3.59E-01*	*2.95E-01*
Nb-94	*3.35E-02*	*2.64E-02*	*2.21E-02*	*2.40E-02*	*2.92E-02*	*2.85E-02*	*5.35E-02*	*7.05E-02*	*7.16E-02*	*2.83E-02*
Tc-99	*4.03E-01*	*4.38E-01*	*4.20E-01*	*3.95E-01*	*4.43E-01*	*4.18E-01*	*2.61E-01*	*2.61E-01*	*2.62E-01*	*6.39E-01*
Ag-108m	*2.95E-02*	*3.05E-02*	*2.03E-02*	*2.21E-02*	*2.97E-02*	*2.91E-02*	*4.89E-02*	*7.14E-02*	*7.52E-02*	*2.62E-02*
Sb-125	*9.26E-02*	*8.40E-02*	*6.11E-02*	*6.35E-02*	*8.03E-02*	*8.44E-02*	*1.68E-01*	*2.19E-01*	*2.90E-01*	*7.13E-02*
Cs-134	*2.92E-02*	*2.91E-02*	*2.24E-02*	*2.41E-02*	*2.78E-02*	*2.85E-02*	*6.04E-02*	7.72E-02	*9.92E-02*	*2.72E-02*
Cs-137	**6.98E-01**	**3.85E-01**	**7.98E-02**	**8.86E-02**	**3.15E-01**	**4.57E-01**	**1.57E-01**	**2.30E+00**	**3.39E+00**	*3.31E-02*
Pm-147	*5.09E-01*	*5.06E-01*	*5.03E-01*	*5.04E-01*	*4.90E-01*	*5.00E-01*	N/A	*2.74E-01*	*2.63E-01*	*5.02E-01*
Eu-152	*1.84E-01*	*2.13E-01*	*1.61E-01*	*1.83E-01*	*1.67E-01*	*1.76E-01*	*1.60E-01*	*6.27E-01*	*4.74E-01*	*2.27E-01*
Eu-154	*1.16E-01*	*9.27E-02*	*7.38E-02*	*7.90E-02*	*8.26E-02*	*1.10E-01*	*8.00E-02*	*2.23E-01*	*2.66E-01*	*9.19E-02*
Eu-155	*7.42E-02*	*8.00E-02*	*6.40E-02*	*6.32E-02*	*7.11E-02*	*7.83E-02*	*1.43E-01*	*1.87E-01*	*2.12E-01*	*7.41E-02*
Np-237	*2.43E-02*	*2.42E-02*	*2.37E-02*	*3.22E-02*	*2.01E-02*	*2.35E-02*	N/A	*7.80E-02*	*7.14E-02*	*2.58E-02*
Pu-238	*5.78E-02*	*3.45E-02*	*3.33E-02*	*3.34E-02*	*3.83E-02*	*4.30E-02*	*9.30E-02*	*7.45E-02*	*3.95E-02*	*4.52E-02*
Pu-239/240	*6.53E-02*	*4.71E-02*	*3.33E-02*	*3.02E-02*	*3.83E-02*	*3.88E-02*	*6.25E-02*	*6.87E-02*	*4.49E-02*	*4.52E-02*
Pu-241	*8.31E+00*	*6.13E+00*	*5.20E+00*	*5.05E+00*	*6.56E+00*	*6.55E+00*	N/A	*4.89E-01*	*5.10E-01*	*7.84E+00*
Am-241	*6.19E-02*	*2.91E-02*	*4.90E-02*	*3.78E-02*	*3.57E-02*	*3.45E-02*	*4.61E-02*	*8.29E-02*	*5.50E-02*	*4.58E-02*
Am-243	*3.33E-02*	*3.90E-02*	*3.80E-02*	*4.10E-02*	*7.03E-02*	*4.30E-02*	N/A	*4.51E-02*	*5.67E-02*	*4.10E-02*
Cm-243/244	*5.74E-02*	*5.40E-02*	*2.80E-02*	*4.09E-02*	*3.44E-02*	*5.40E-02*	N/A	*1.22E-01*	*6.94E-02*	*3.80E-02*

a Bold values indicate concentration greater than MDC. Italicized values indicate MDC value.

Figure 2-1 Zion Nuclear Power Station Owner Controlled Area

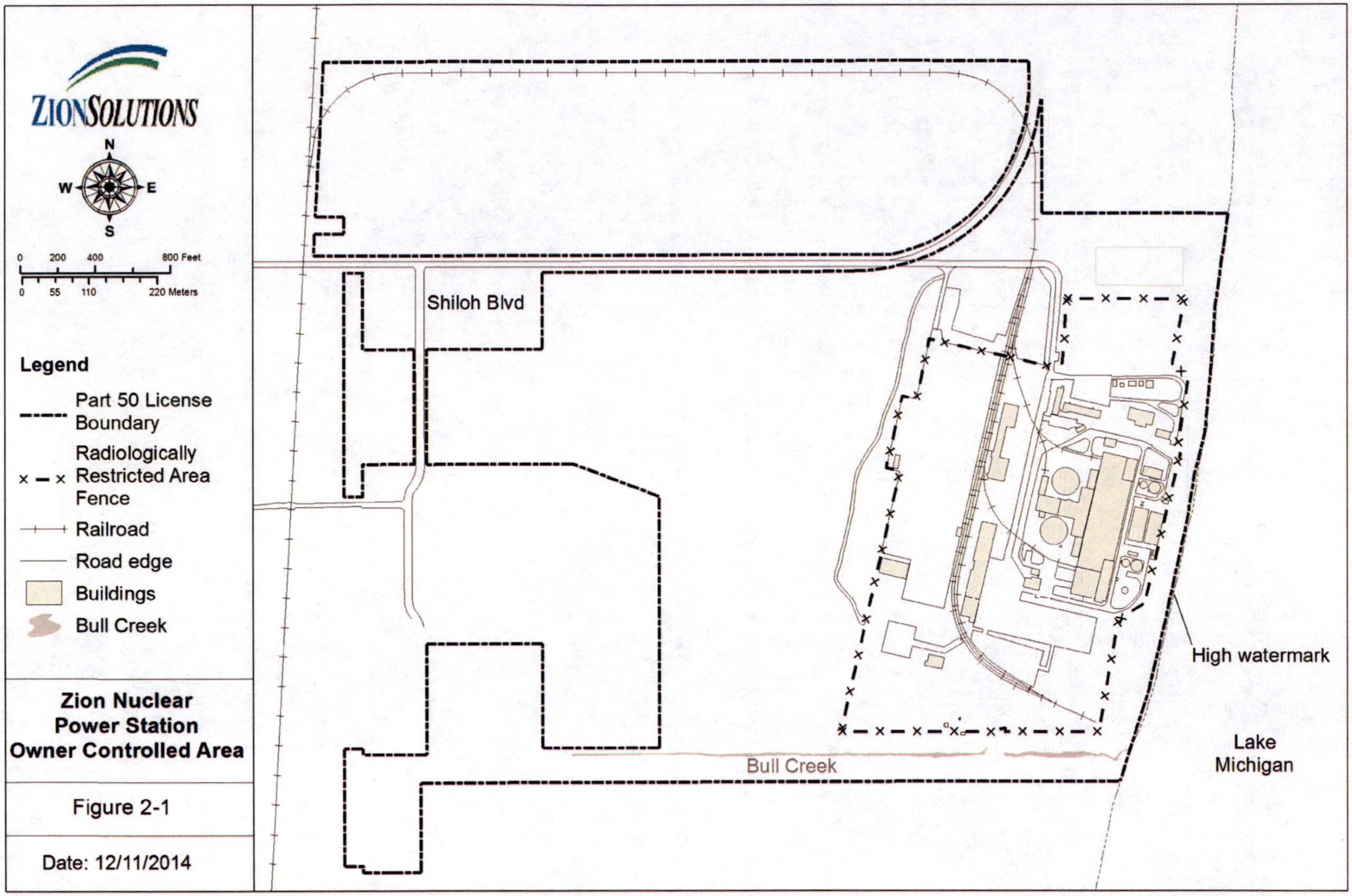


Figure 2-2 Zion Nuclear Power Station Radiologically Restricted Area (RRA)

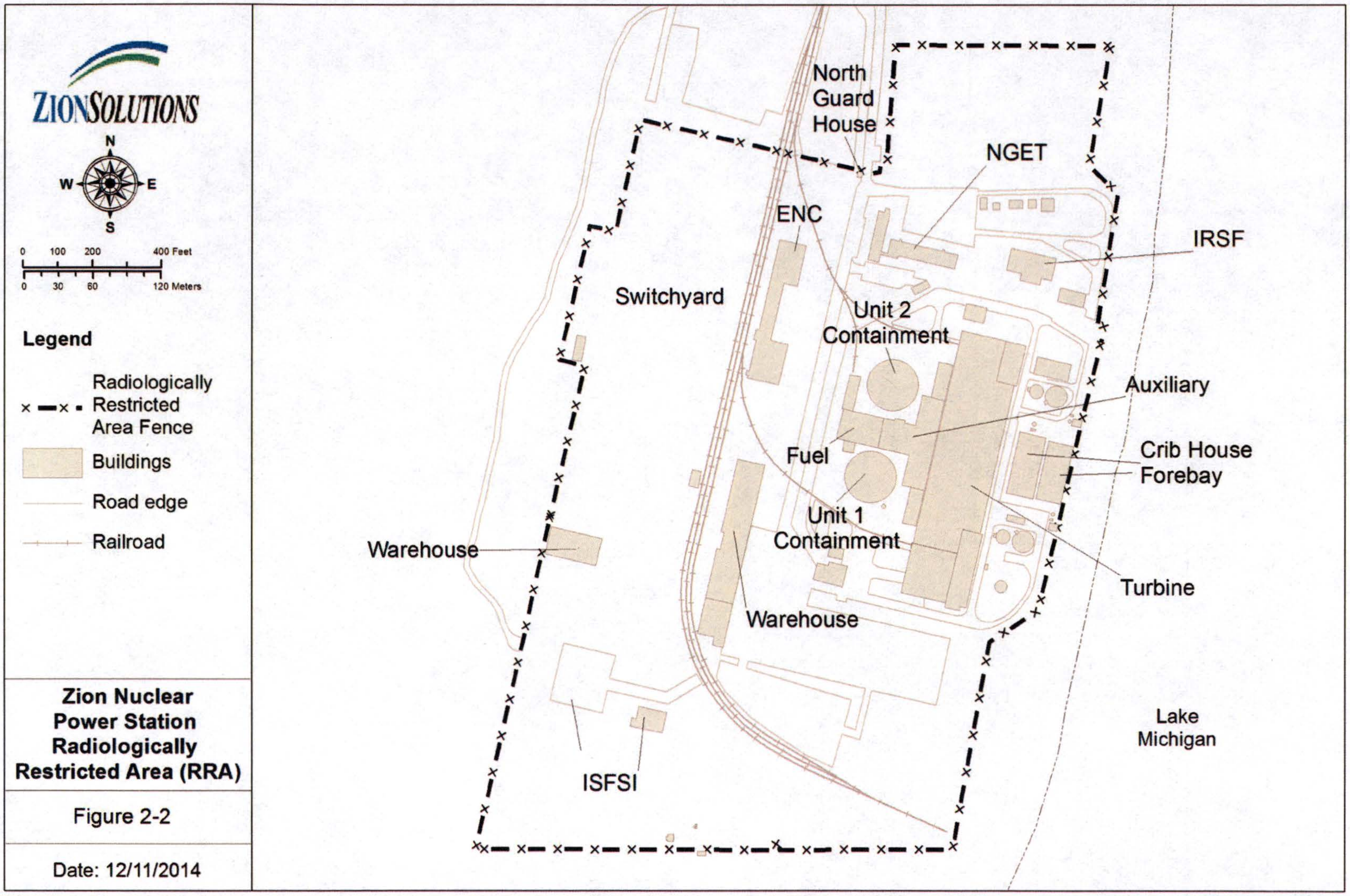


Figure 2-3 Zion Nuclear Power Station Security Restricted Area

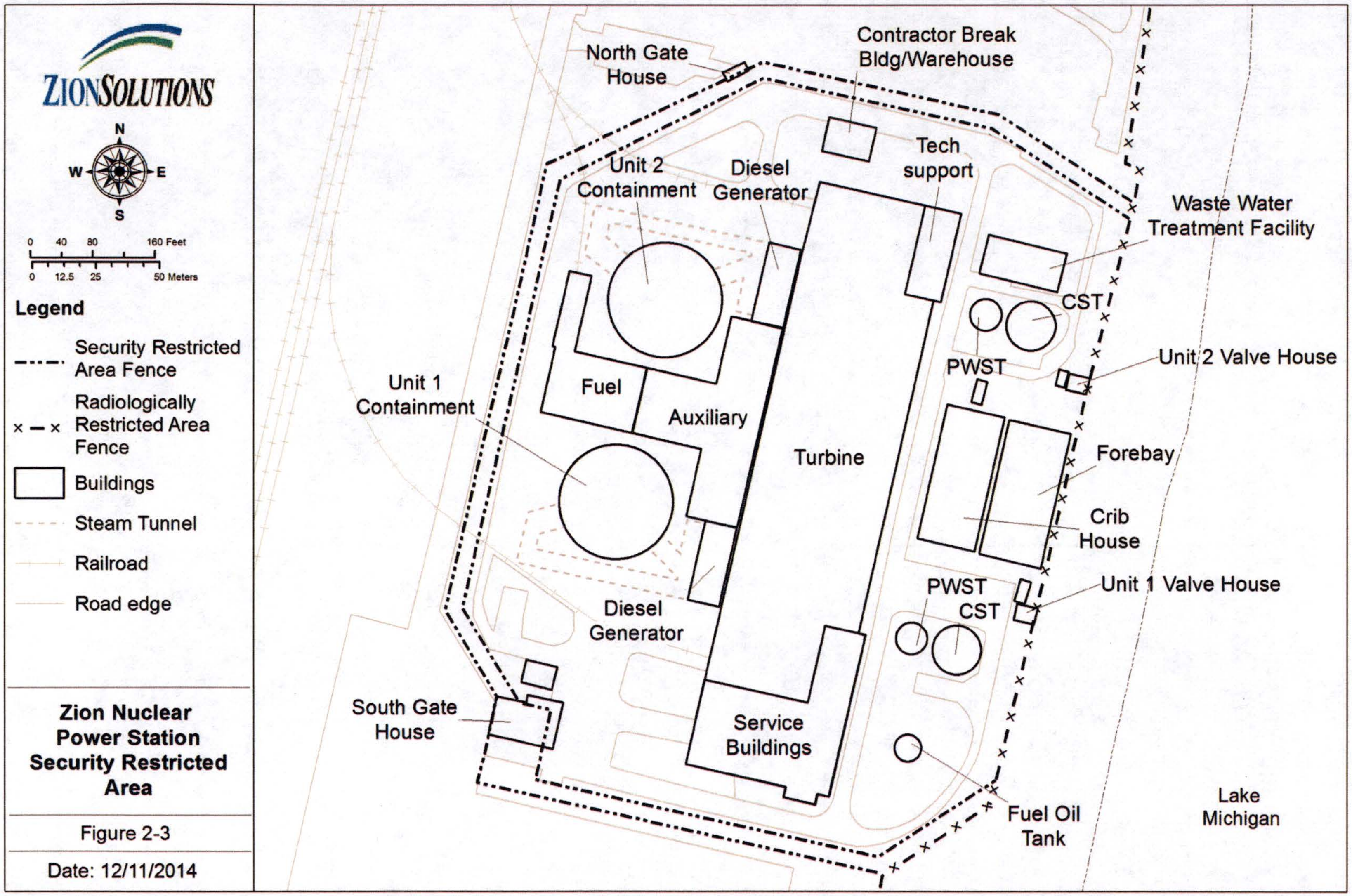


Figure 2-4 Non-Impacted Open Land Survey Units

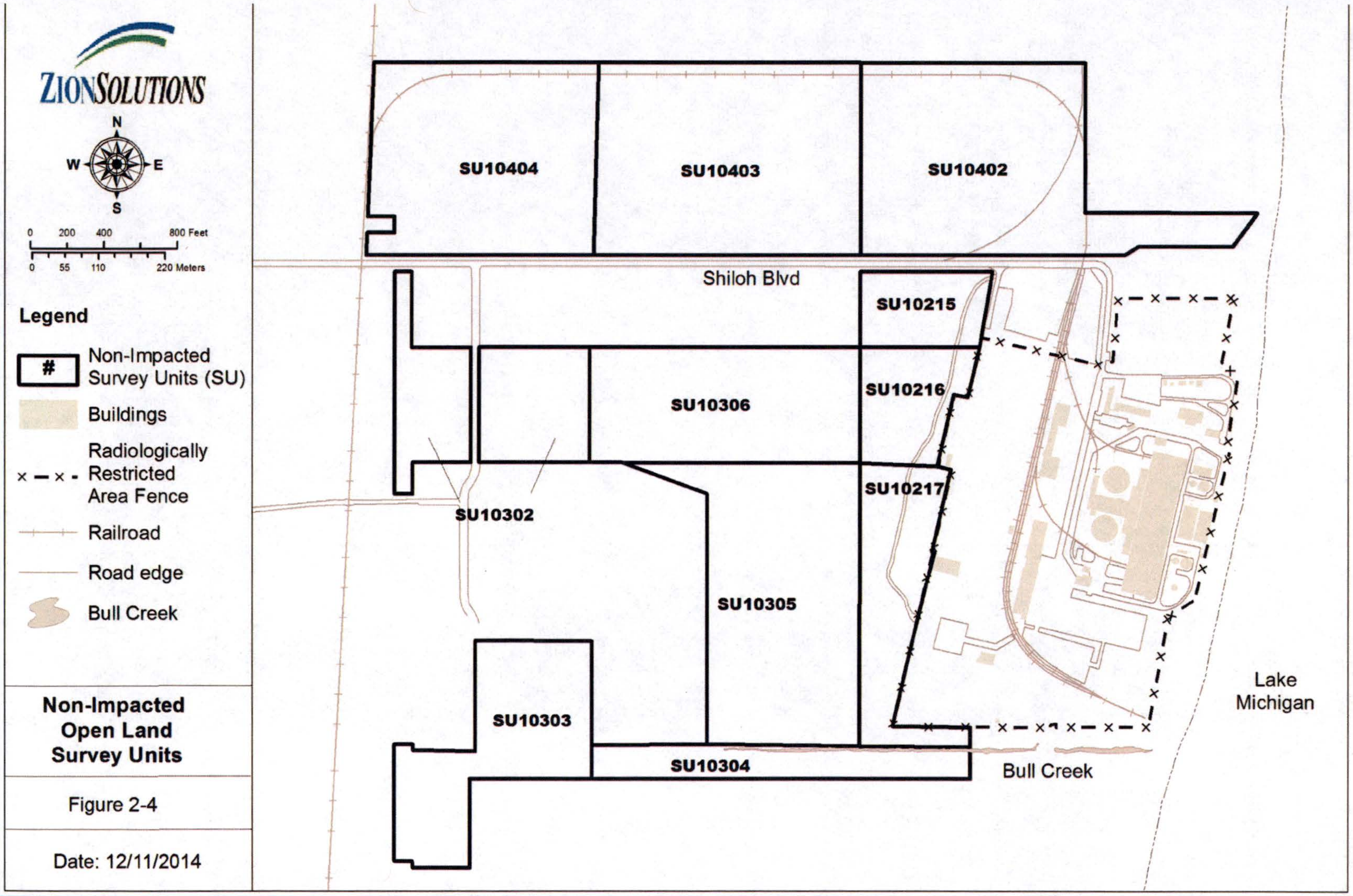


Figure 2-5 Class 3 Open Land Survey Units Outside of "Radiologically-Restricted Area"

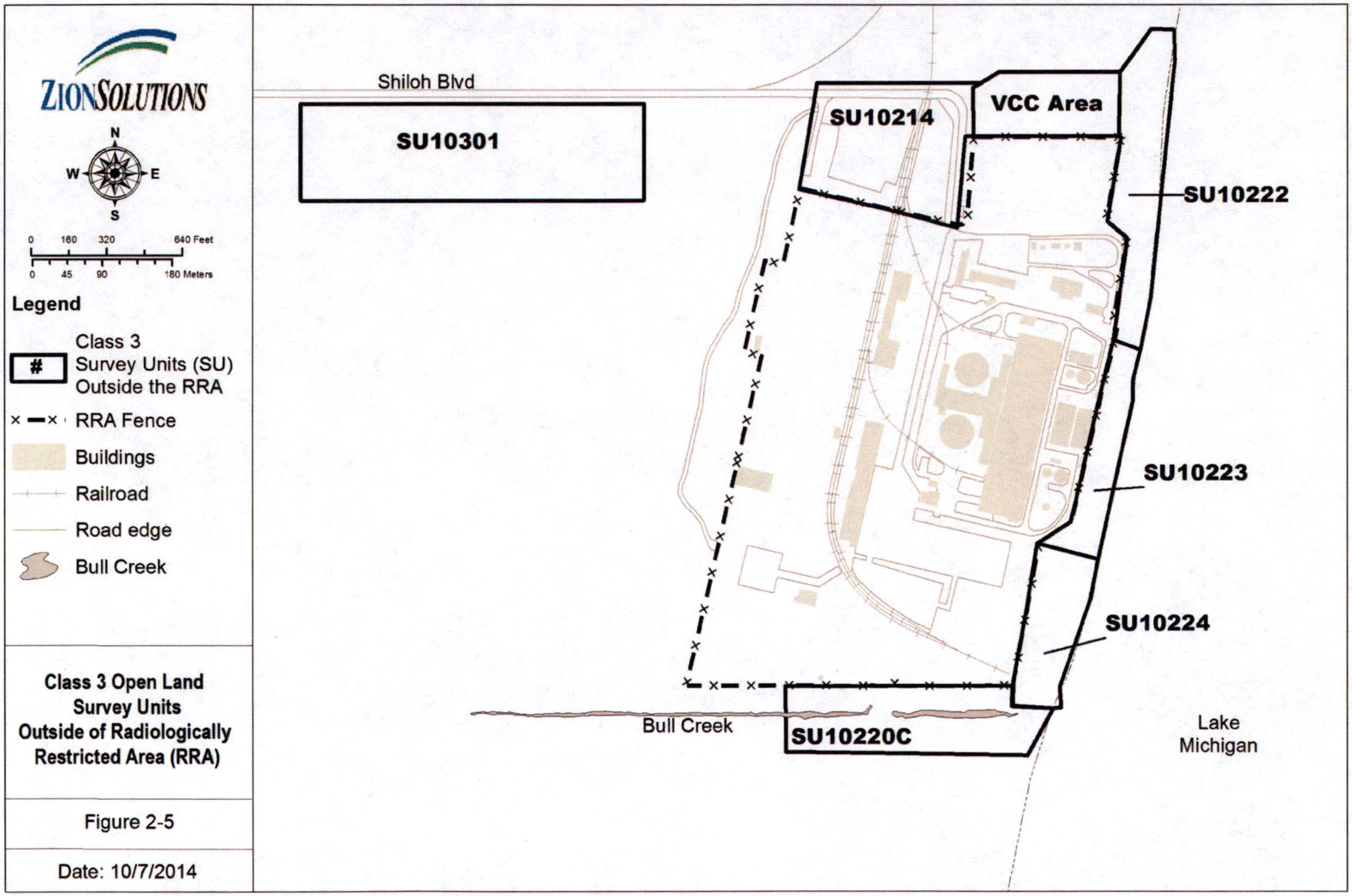


Figure 2-6 Class 3 Open Land Survey Units Inside of "Radiologically-Restricted Area"

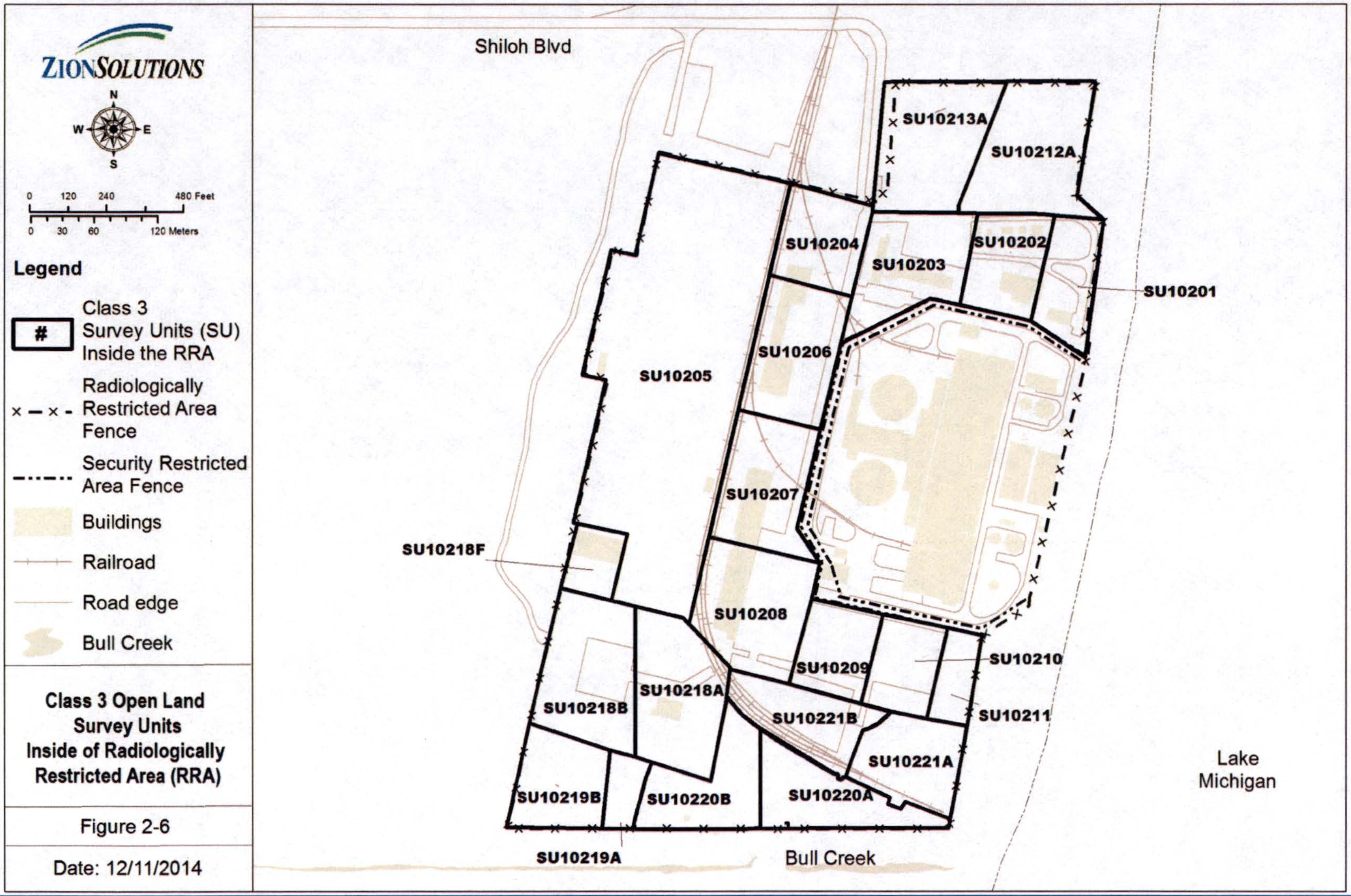


Figure 2-7 Class 1 and Class 2 Open Land Survey Units

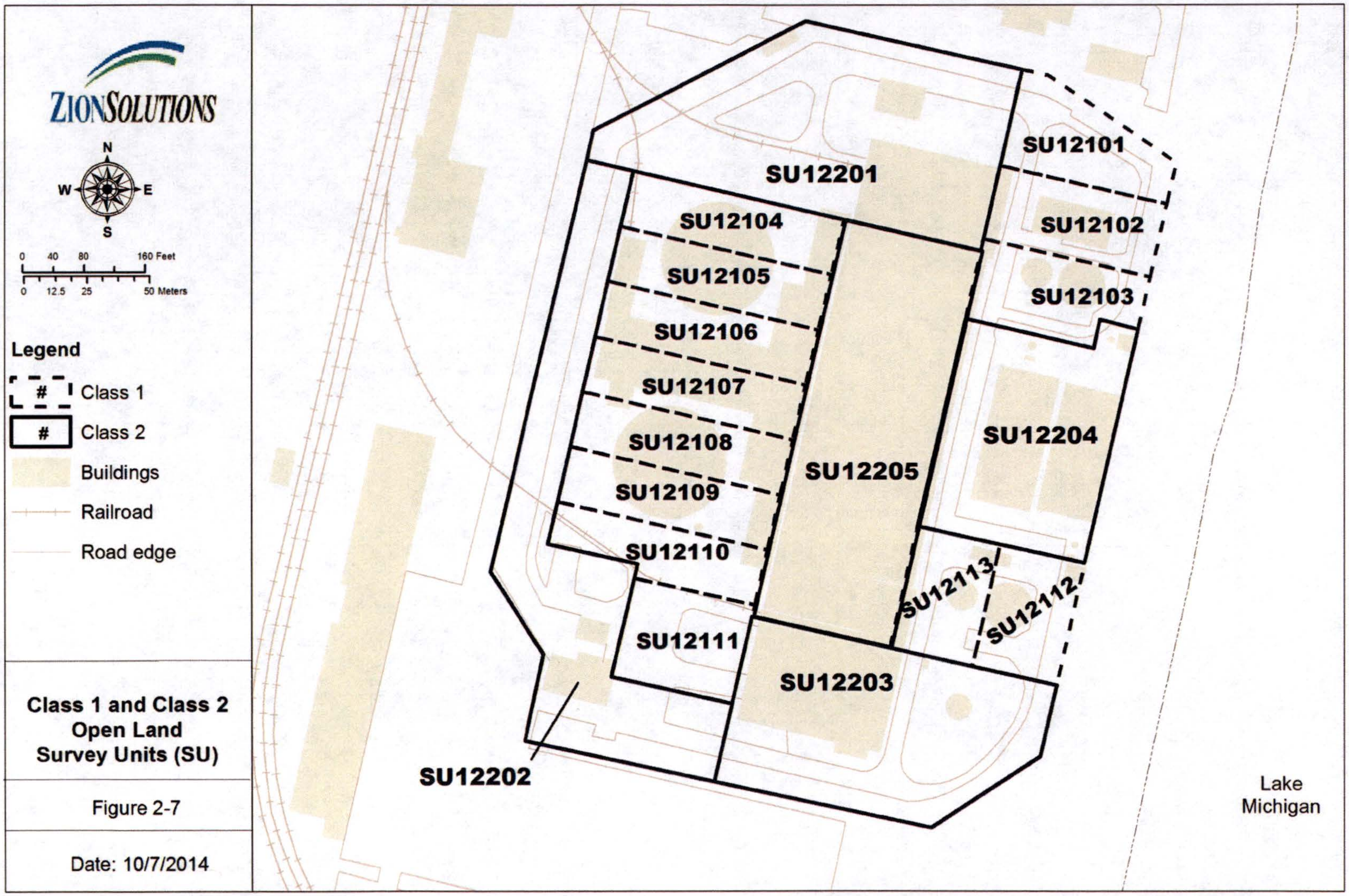


Figure 2-8 Unit 1 Containment 568 foot el. Class 1 Survey Units

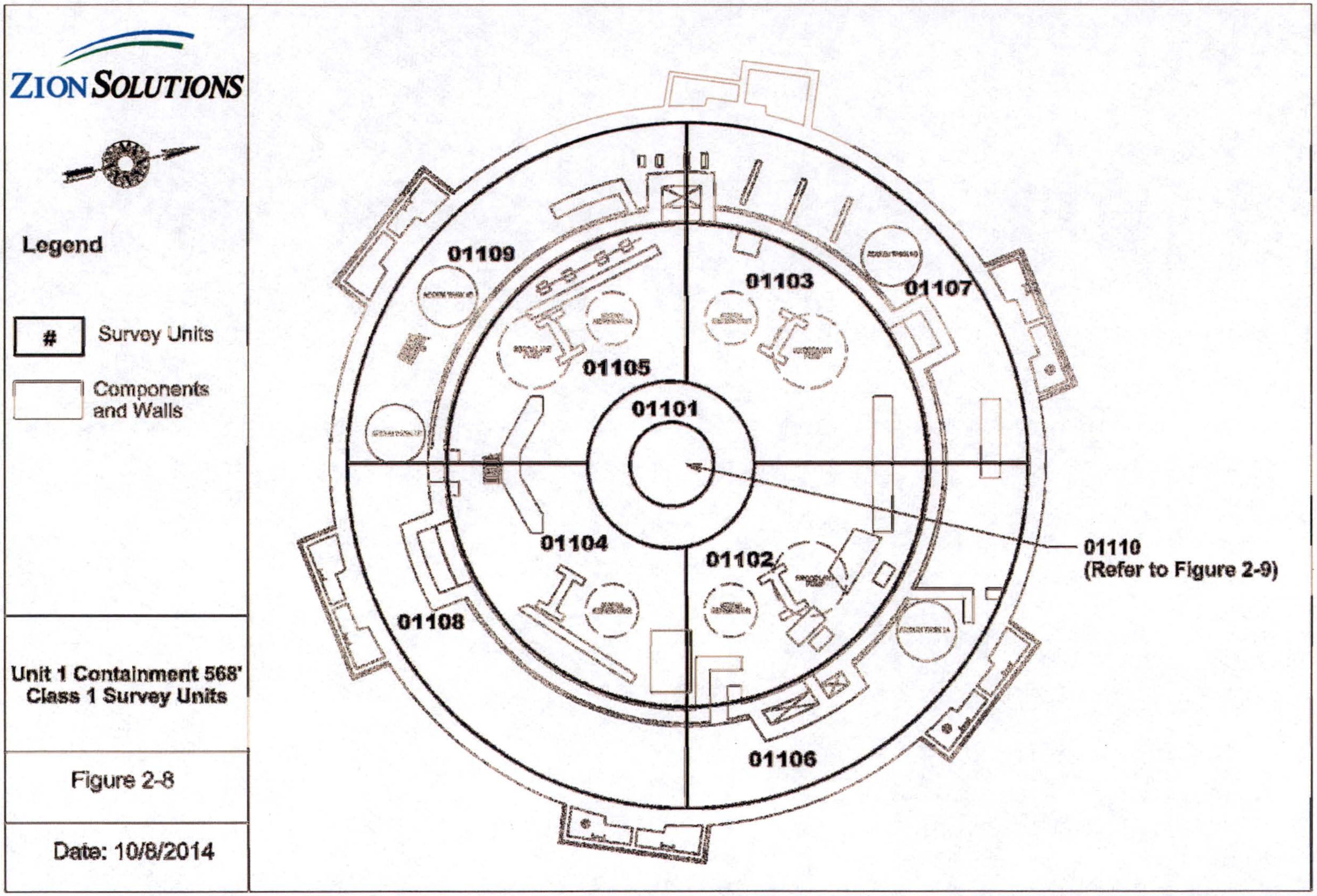


Figure 2-9 Unit 1 Containment 541 foot el. Class 1 Survey Units

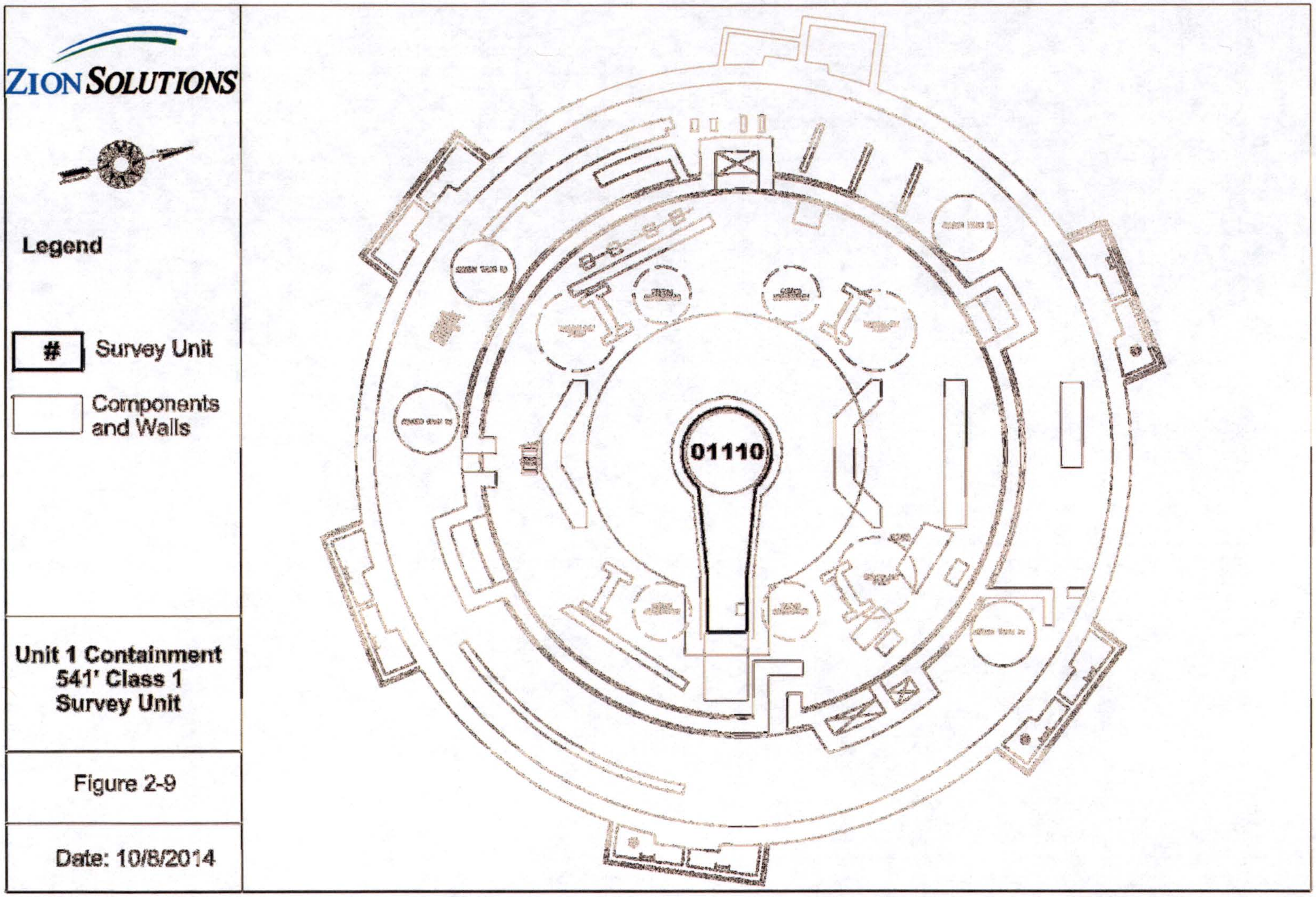


Figure 2-10 Unit 2 Containment 568 foot el. Class 1 Survey Units

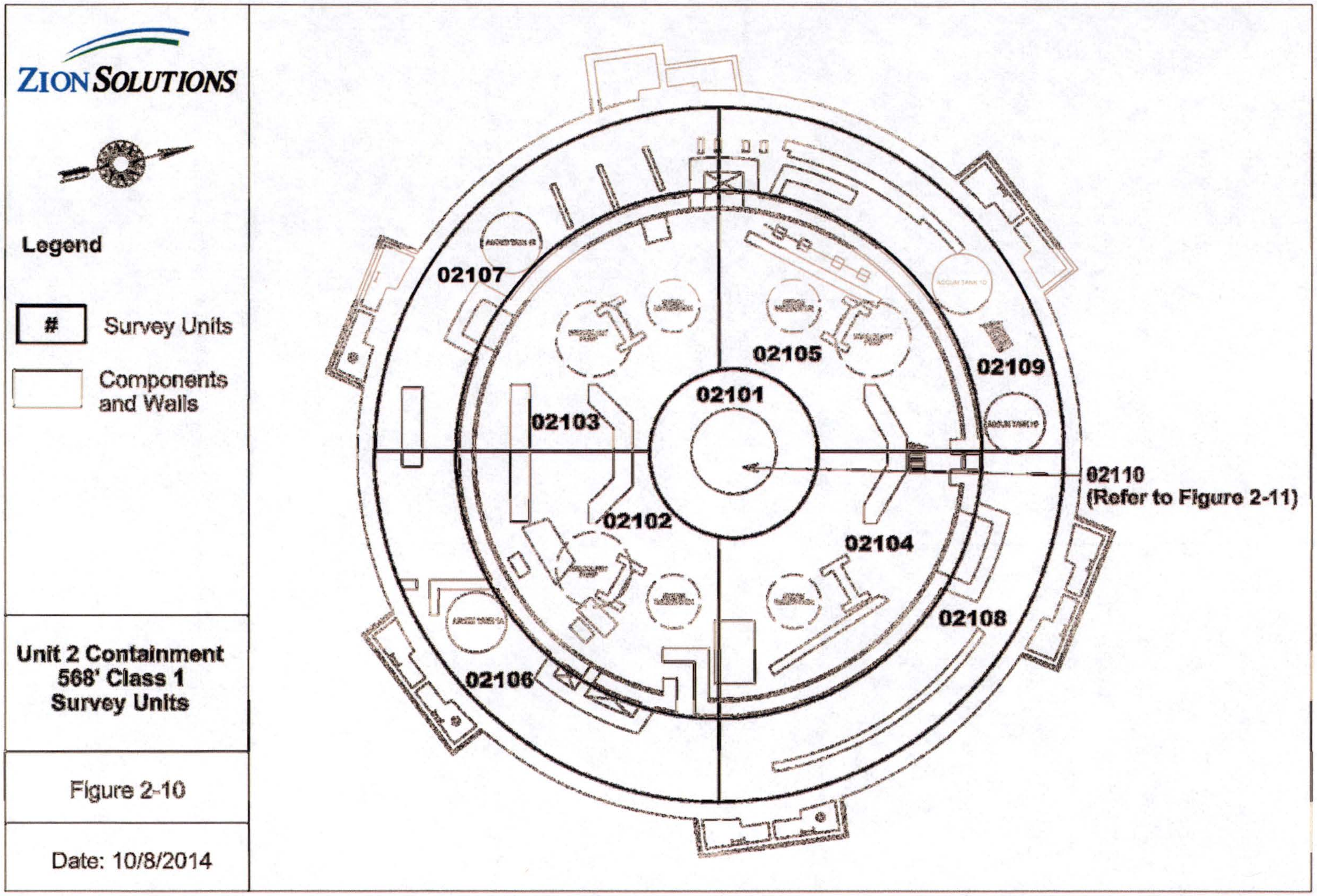


Figure 2-11 Unit 2 Containment 541 foot el. Class 1 Survey Units

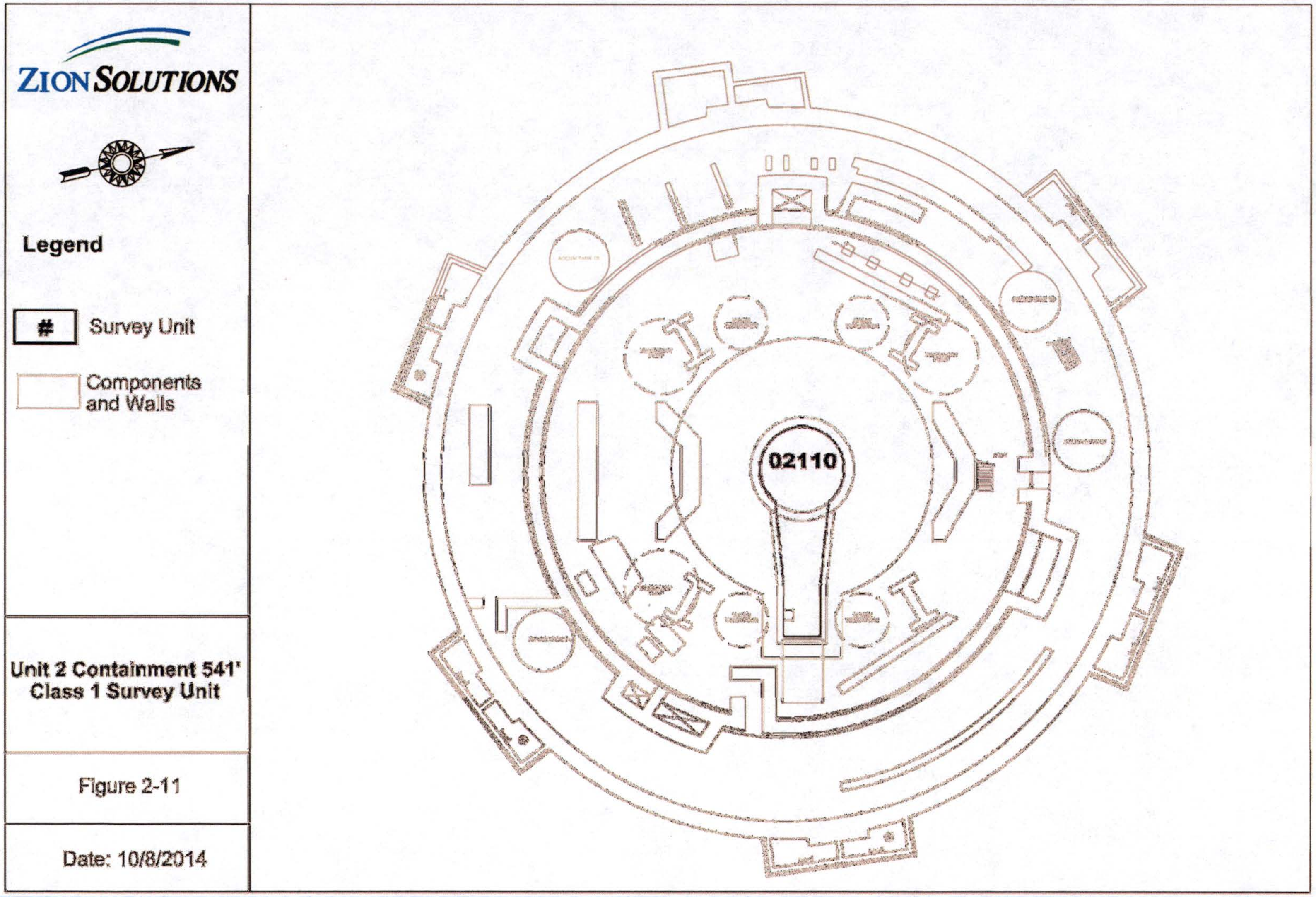


Figure 2-12 Auxiliary Building 542 foot el. Class 1 Survey Units

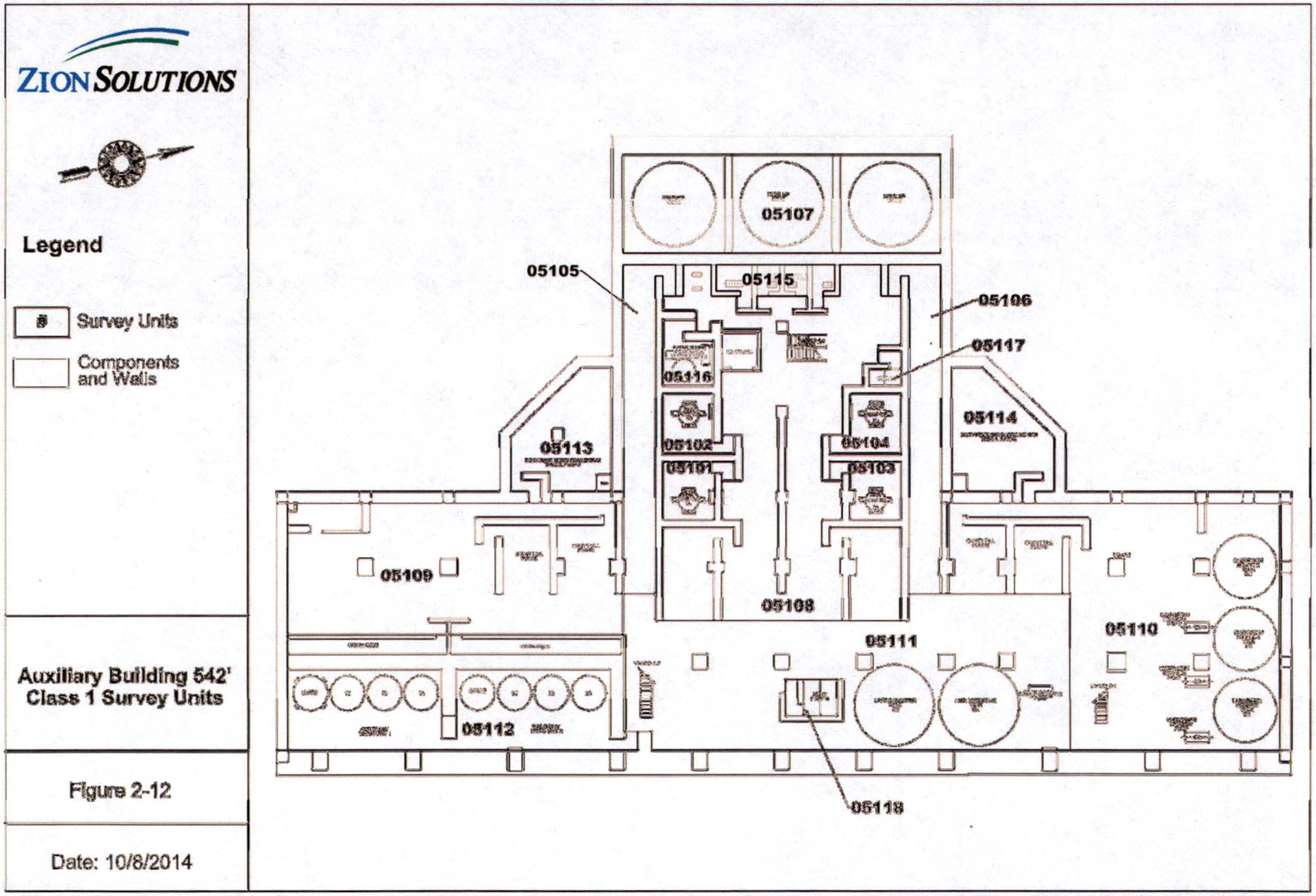


Figure 2-13 Auxiliary Building 560 foot el. Class 1 Survey Units

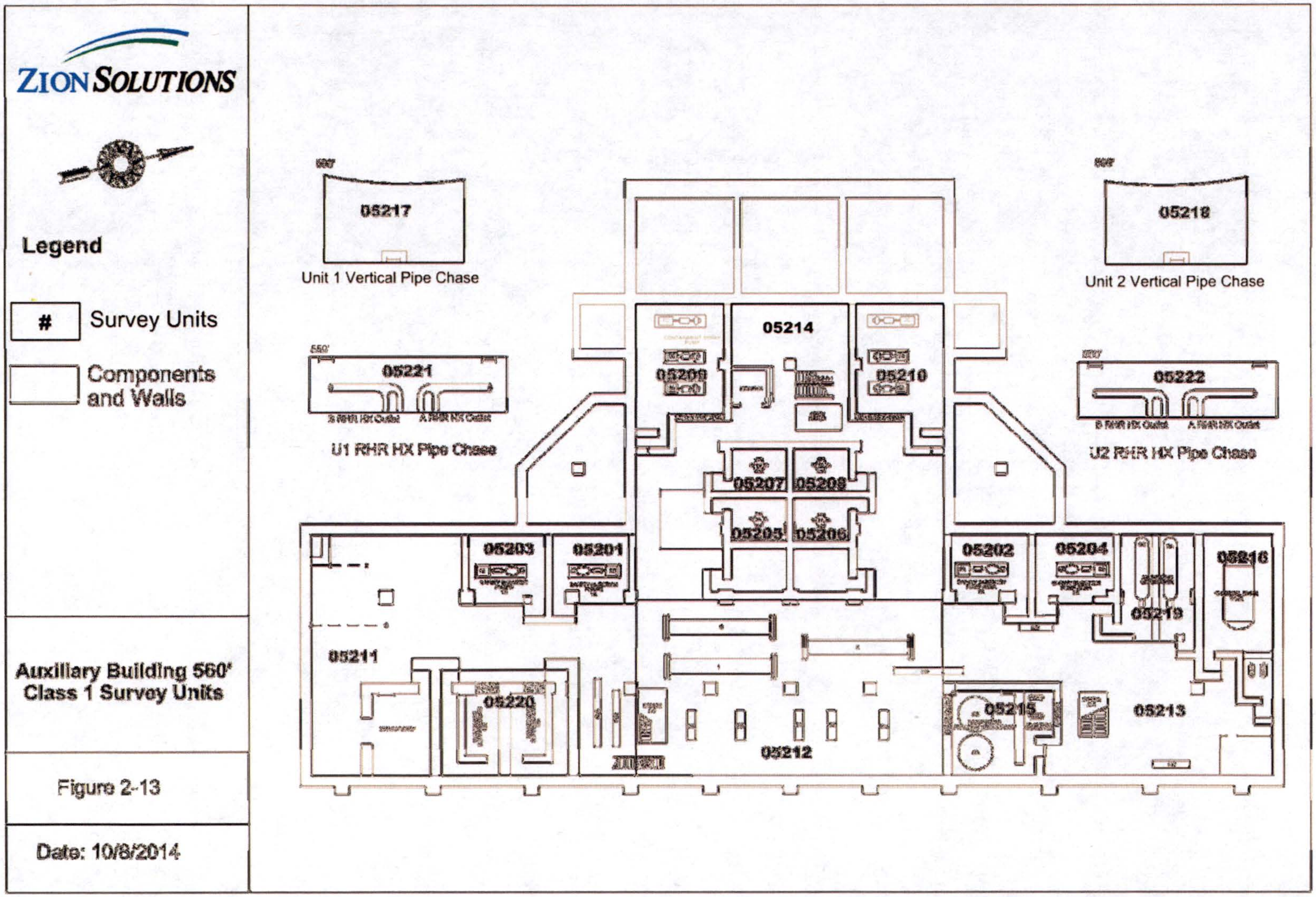


Figure 2-14 Auxiliary Building 579 foot el. Class 1 Survey Units

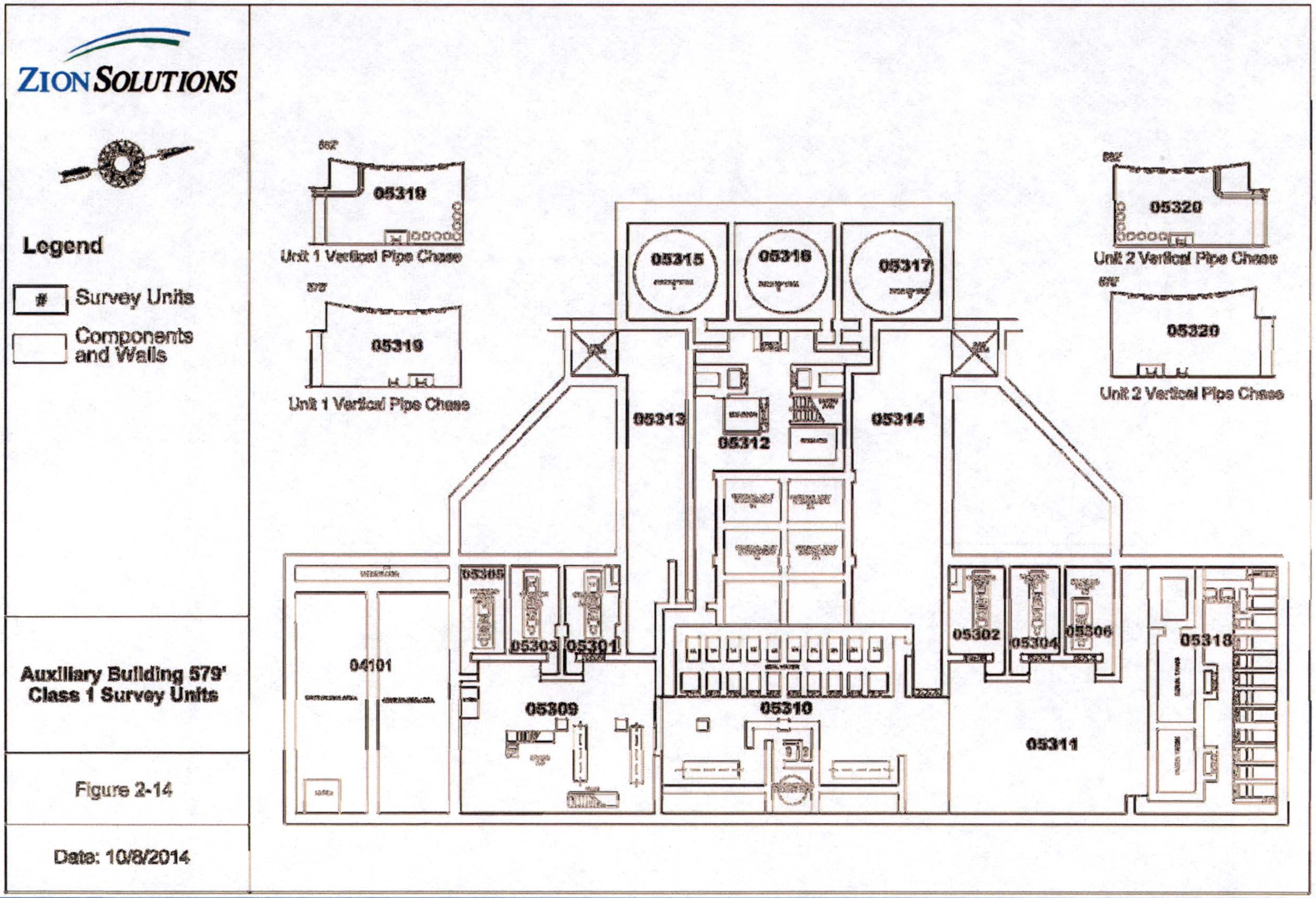


Figure 2-15 Turbine Building 560 and 570 Foot Elevations Class 2 Survey Units

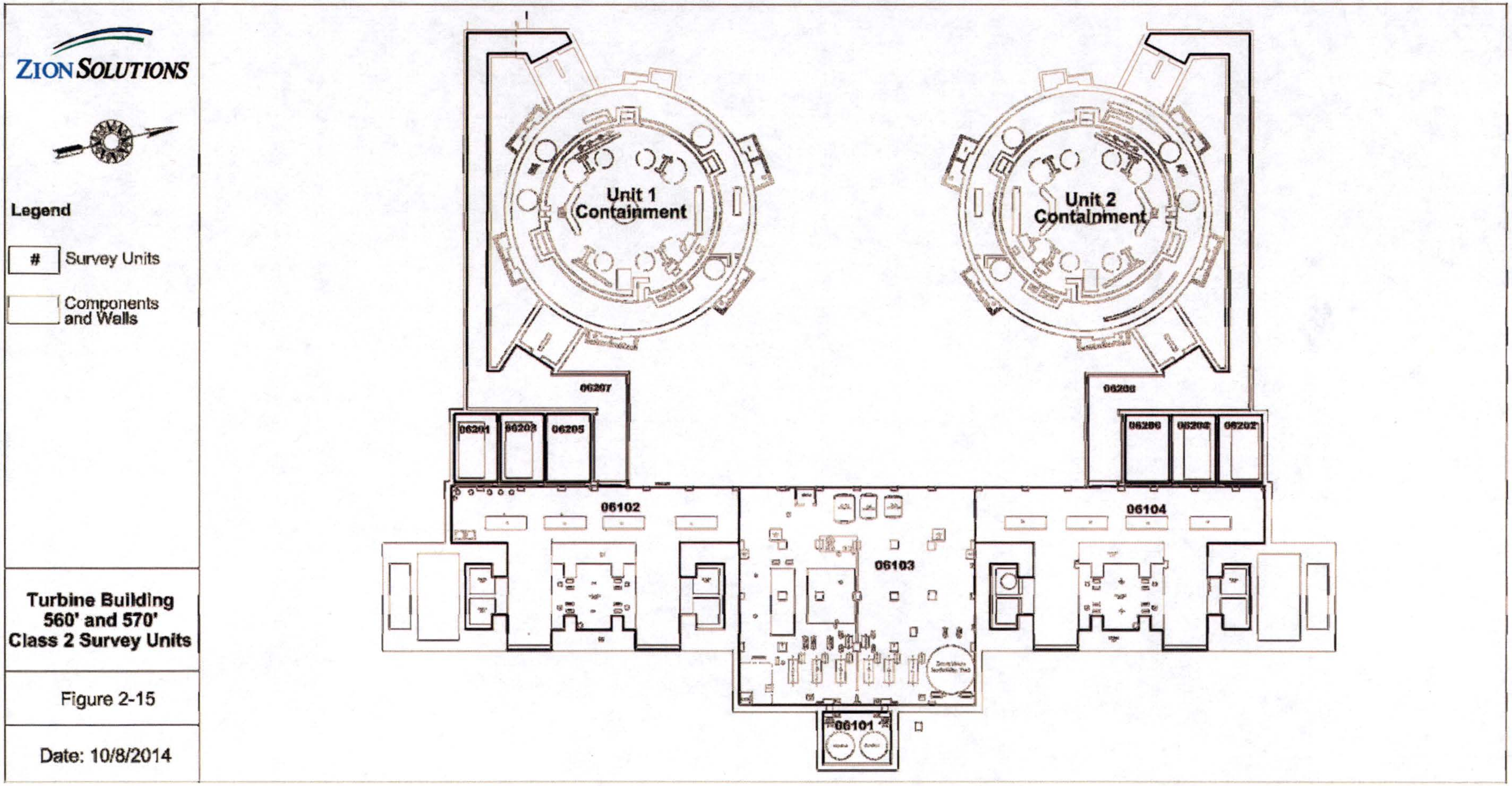


Figure 2-16 Crib House Suction Forebay Class 2 Survey Units

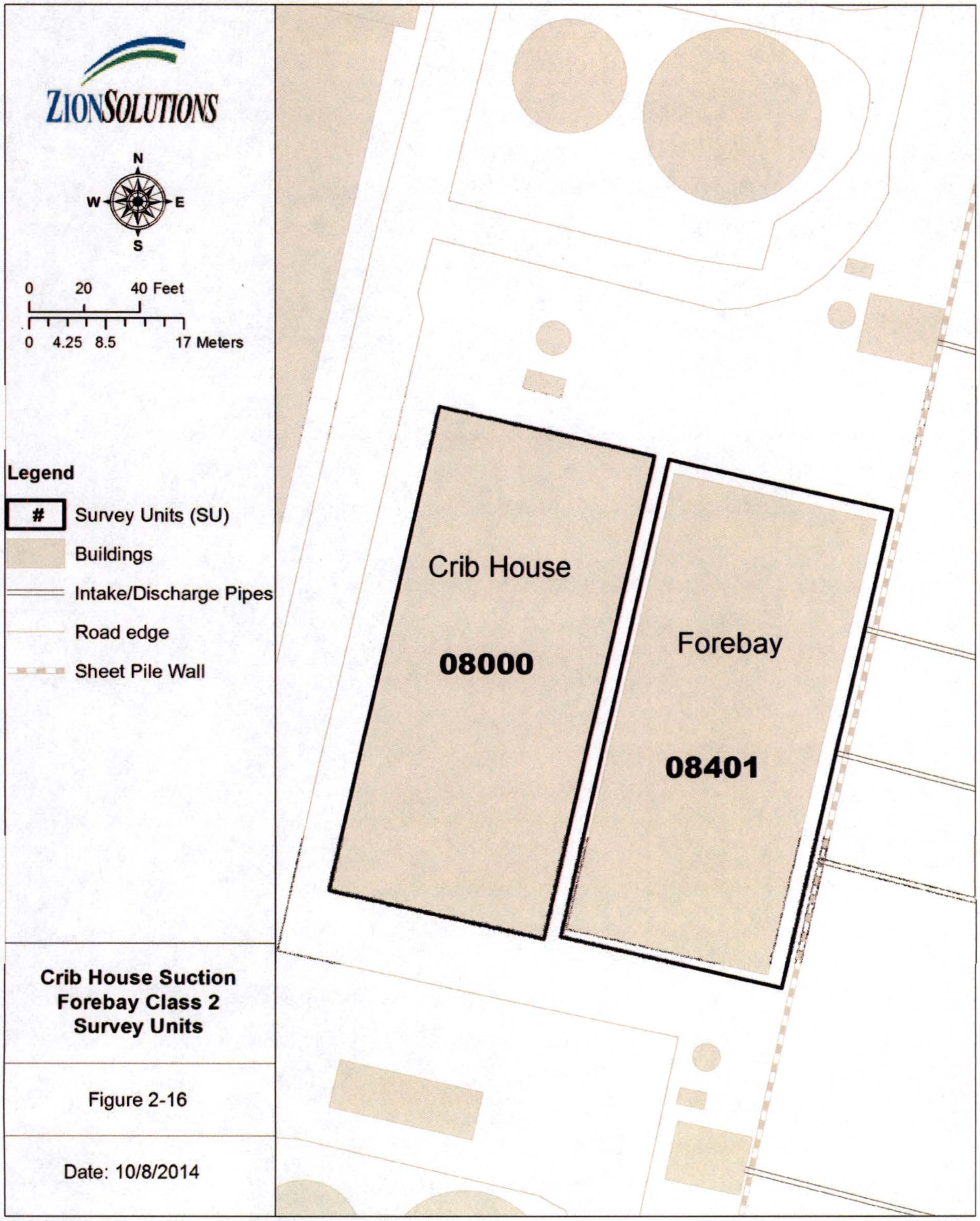


Figure 2-17 Independent Spent Fuel Storage Installation

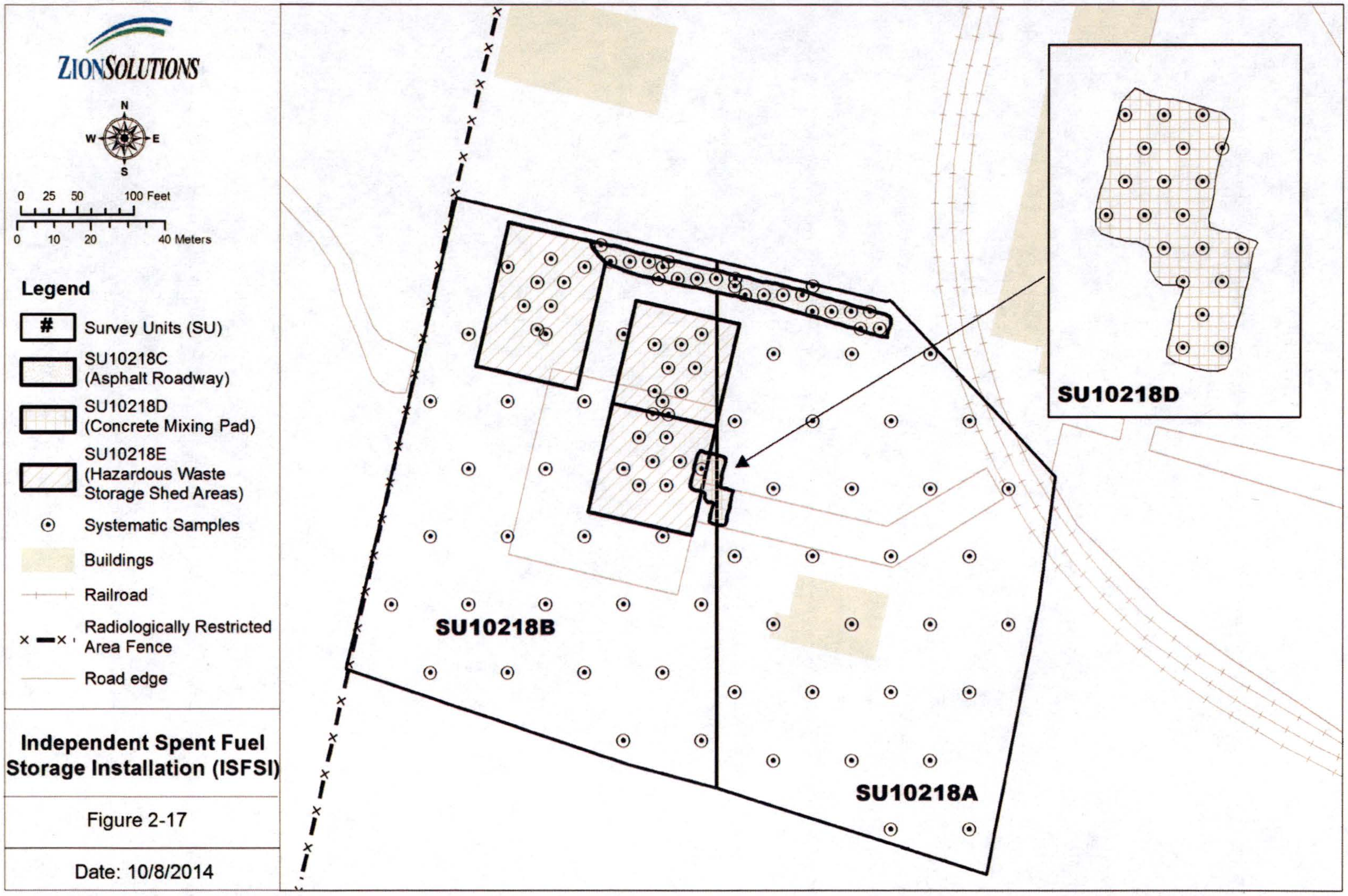


Figure 2-18 Vertical Concrete Cask Construction Area

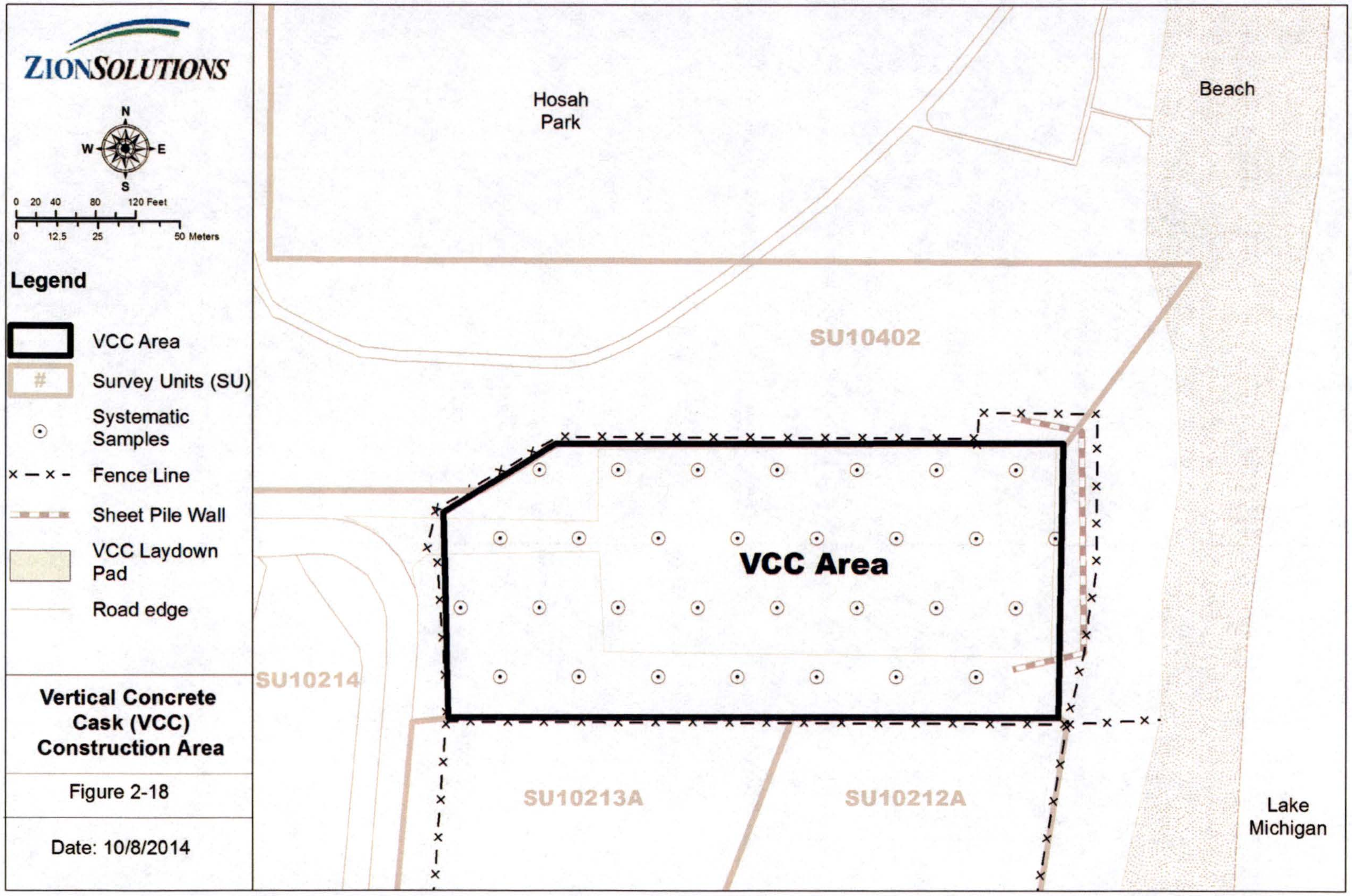


Figure 2-19 Energy Solutions Background Study Locations

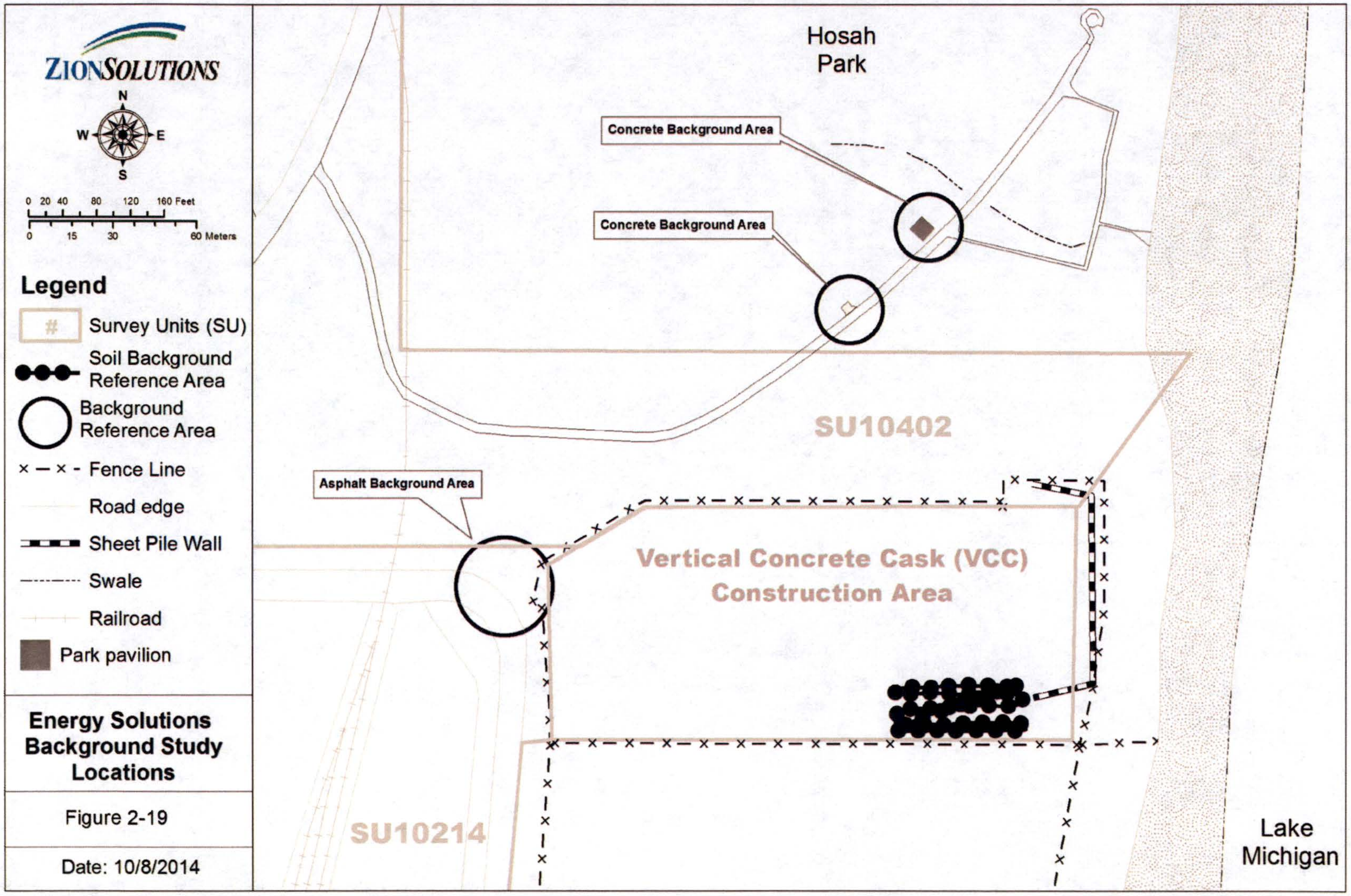


Figure 2-20 Zion City Park District's "Hosah Park" Background Study Location

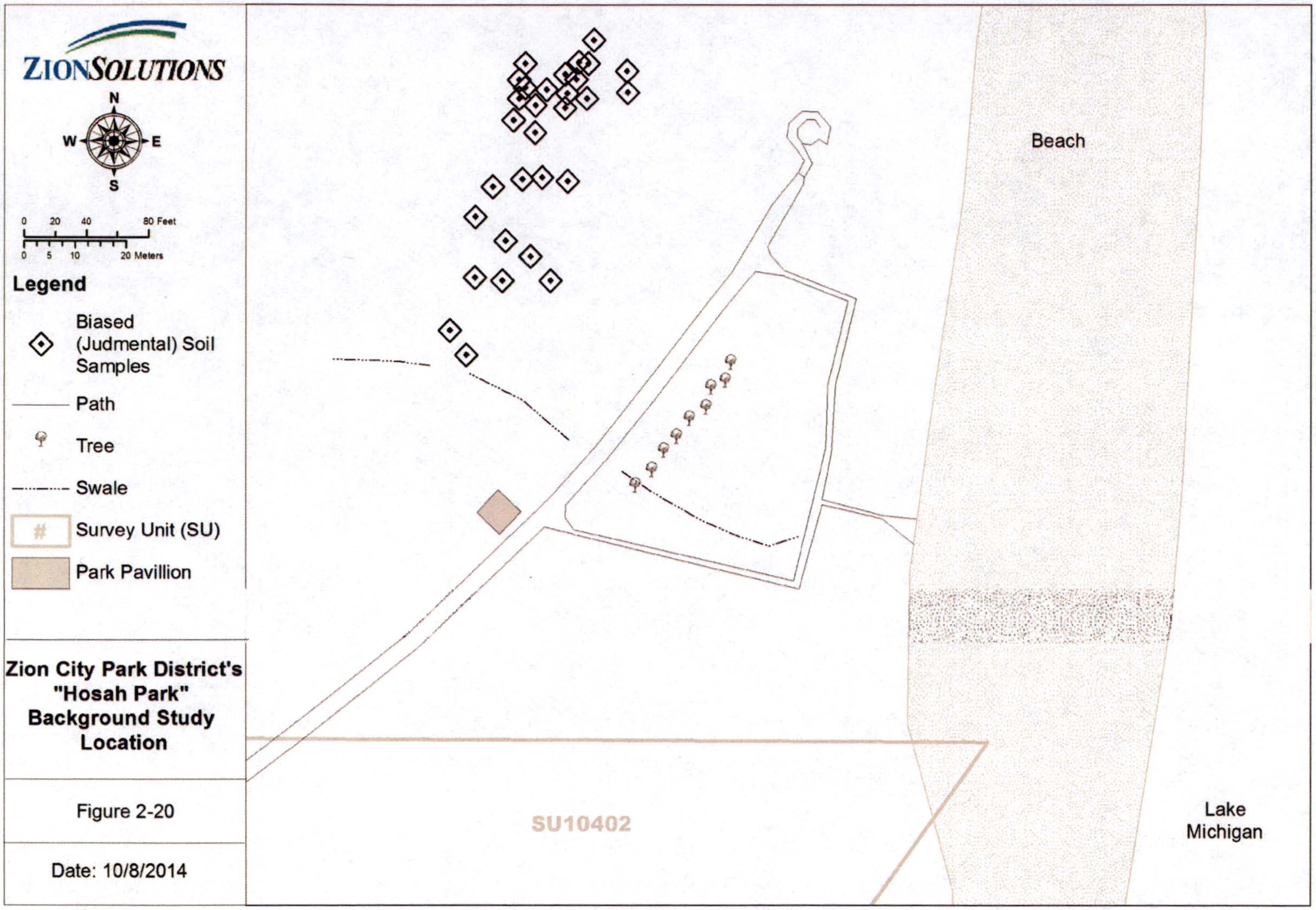


Figure 2-21 Unit 1 Containment 568 foot el. Concrete Core Sample Locations

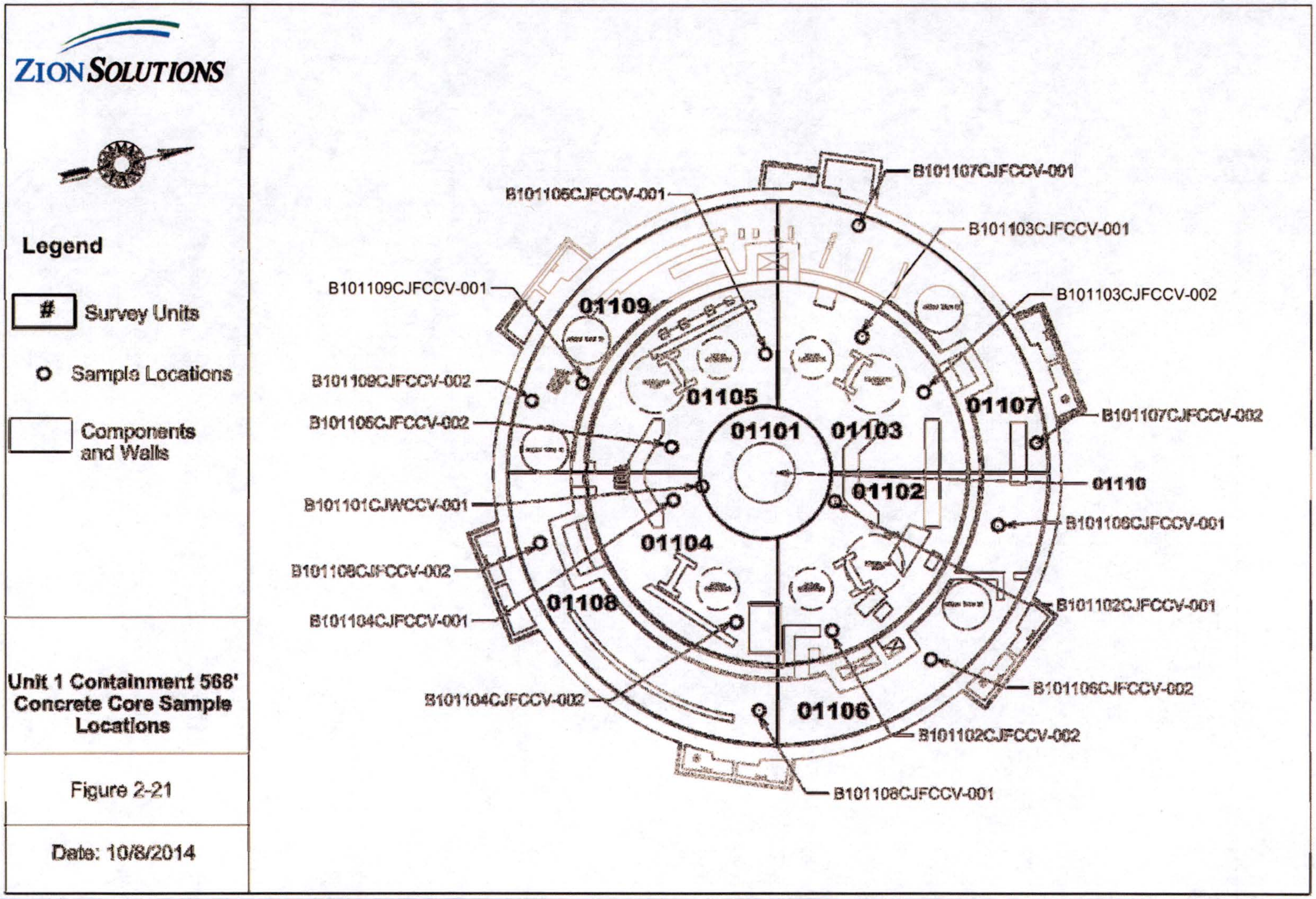


Figure 2-22 Unit 1 Containment 541 foot el. Concrete Core Sample Locations

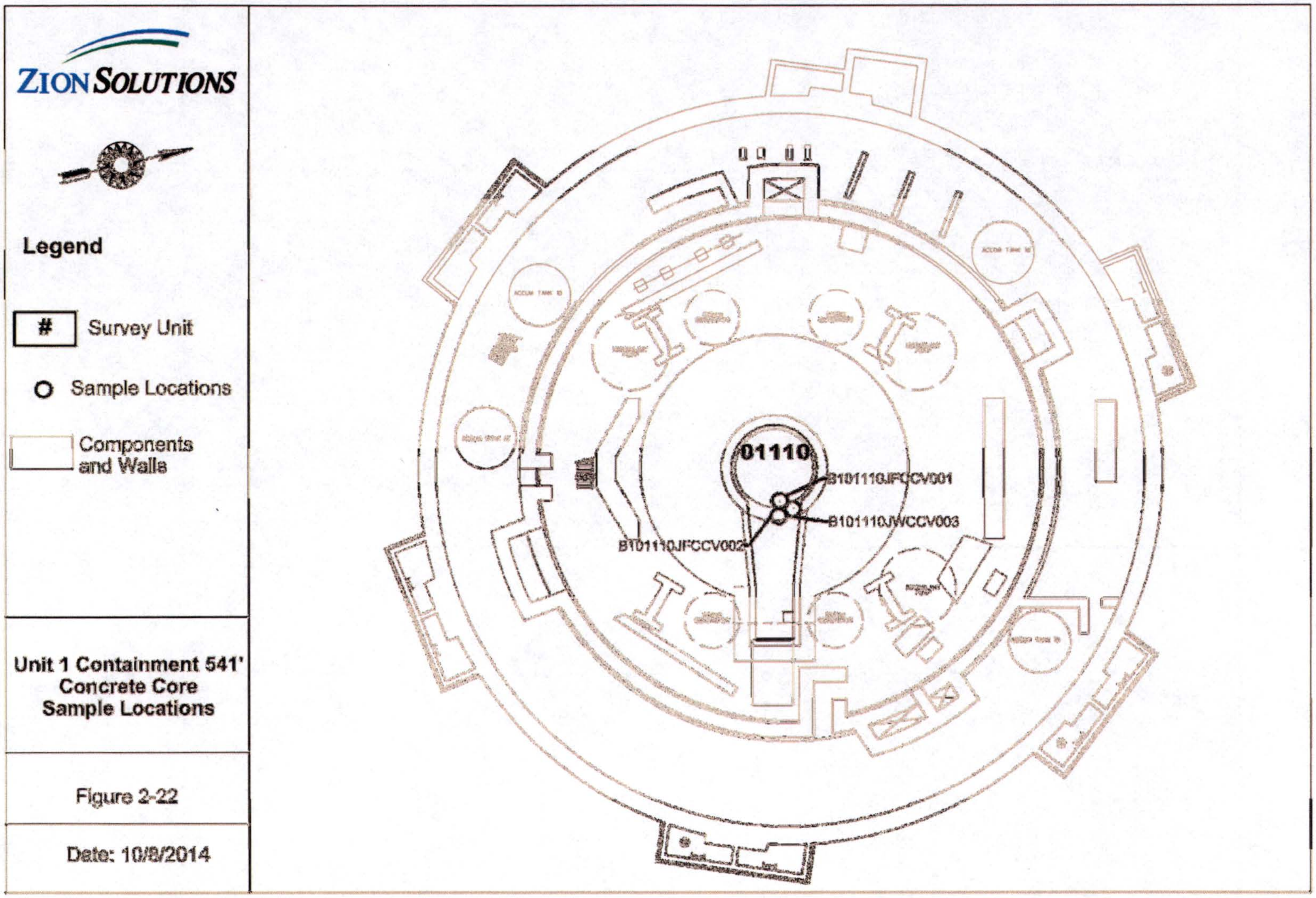


Figure 2-23 Unit 2 Containment 568 foot el. Concrete Core Sample Locations

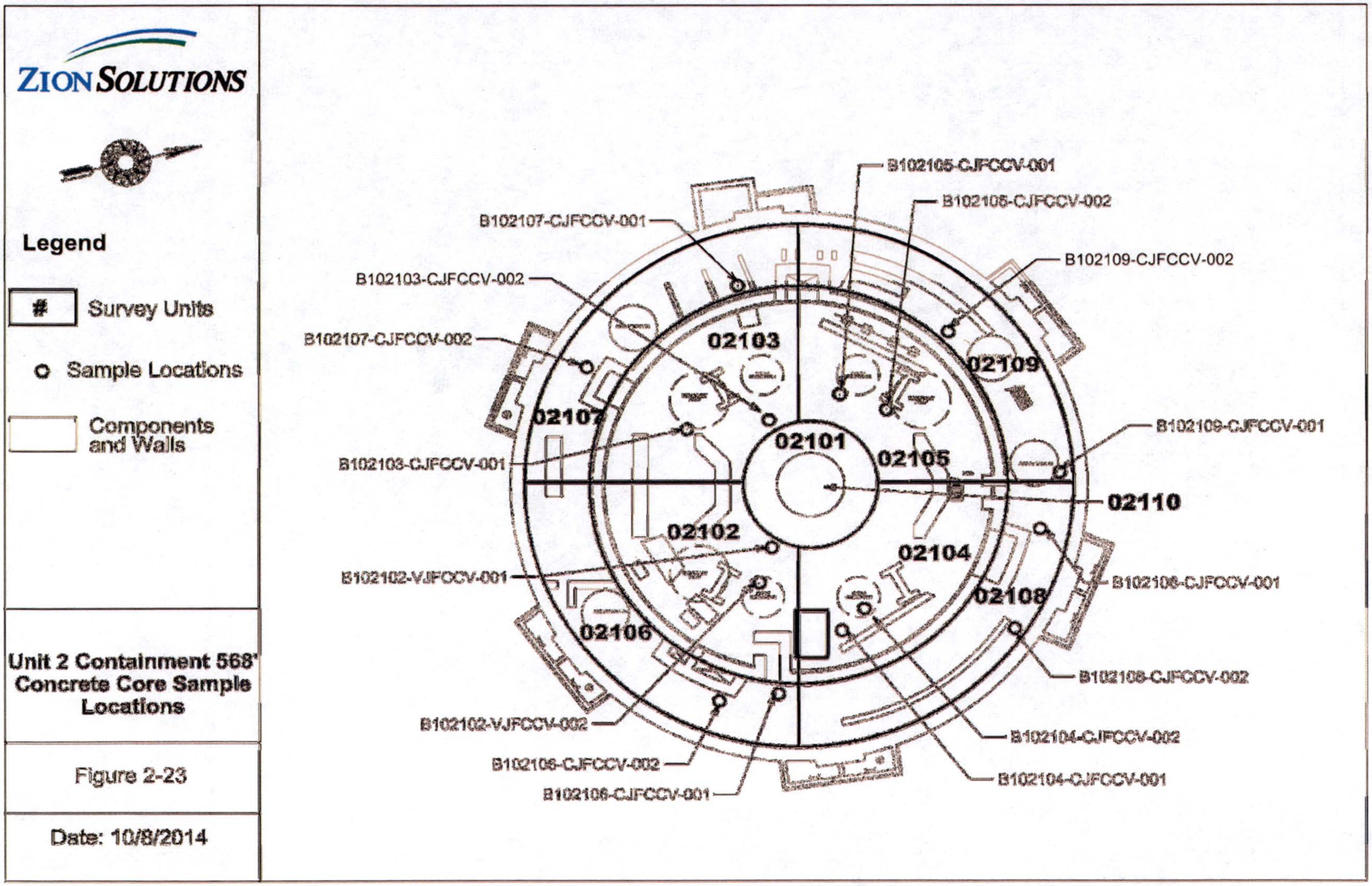


Figure 2-24 Unit 2 Containment 541 foot el. Concrete Core Sample Locations

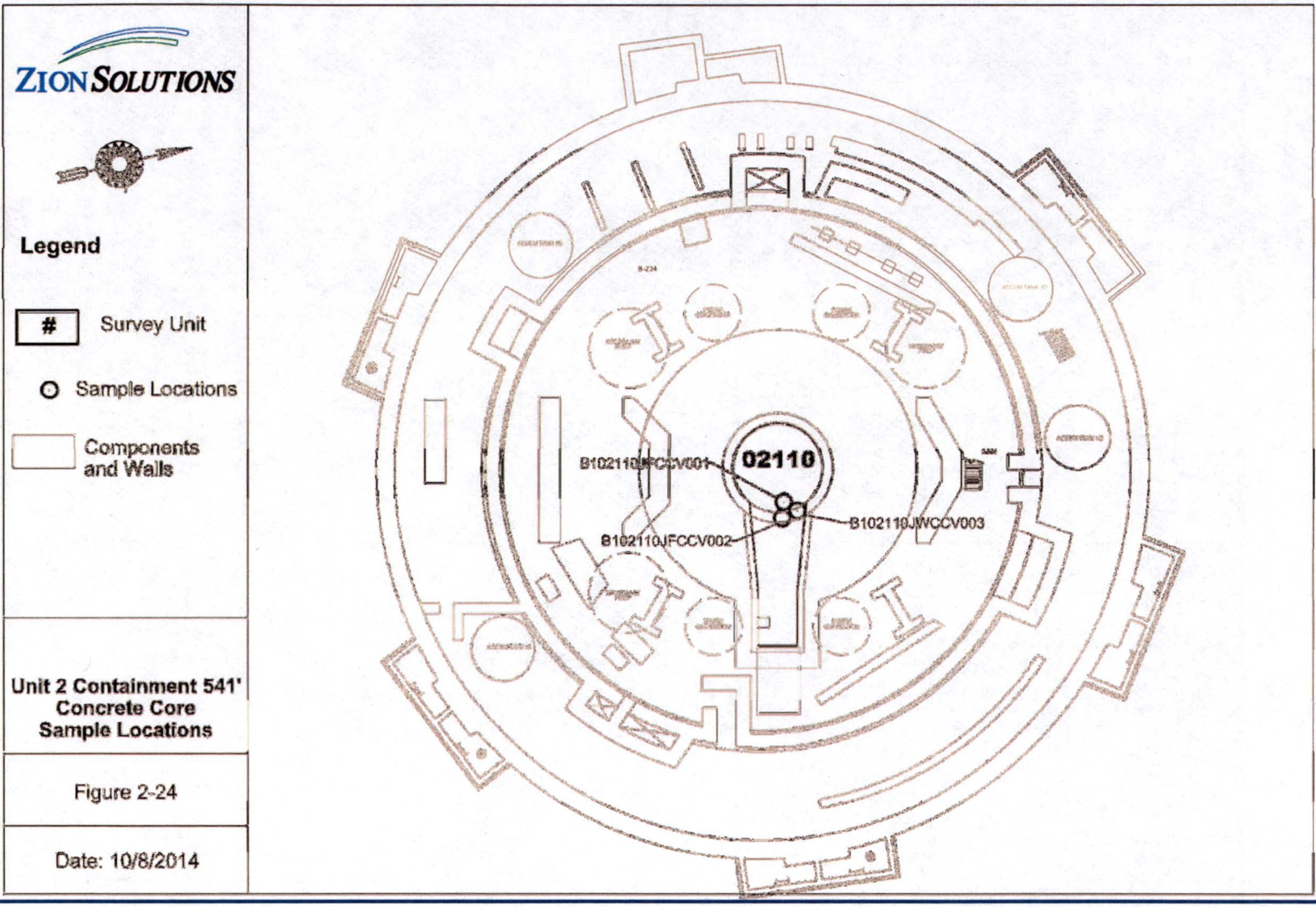


Figure 2-25 Auxiliary Building 542 foot el. Concrete Core Sample Locations

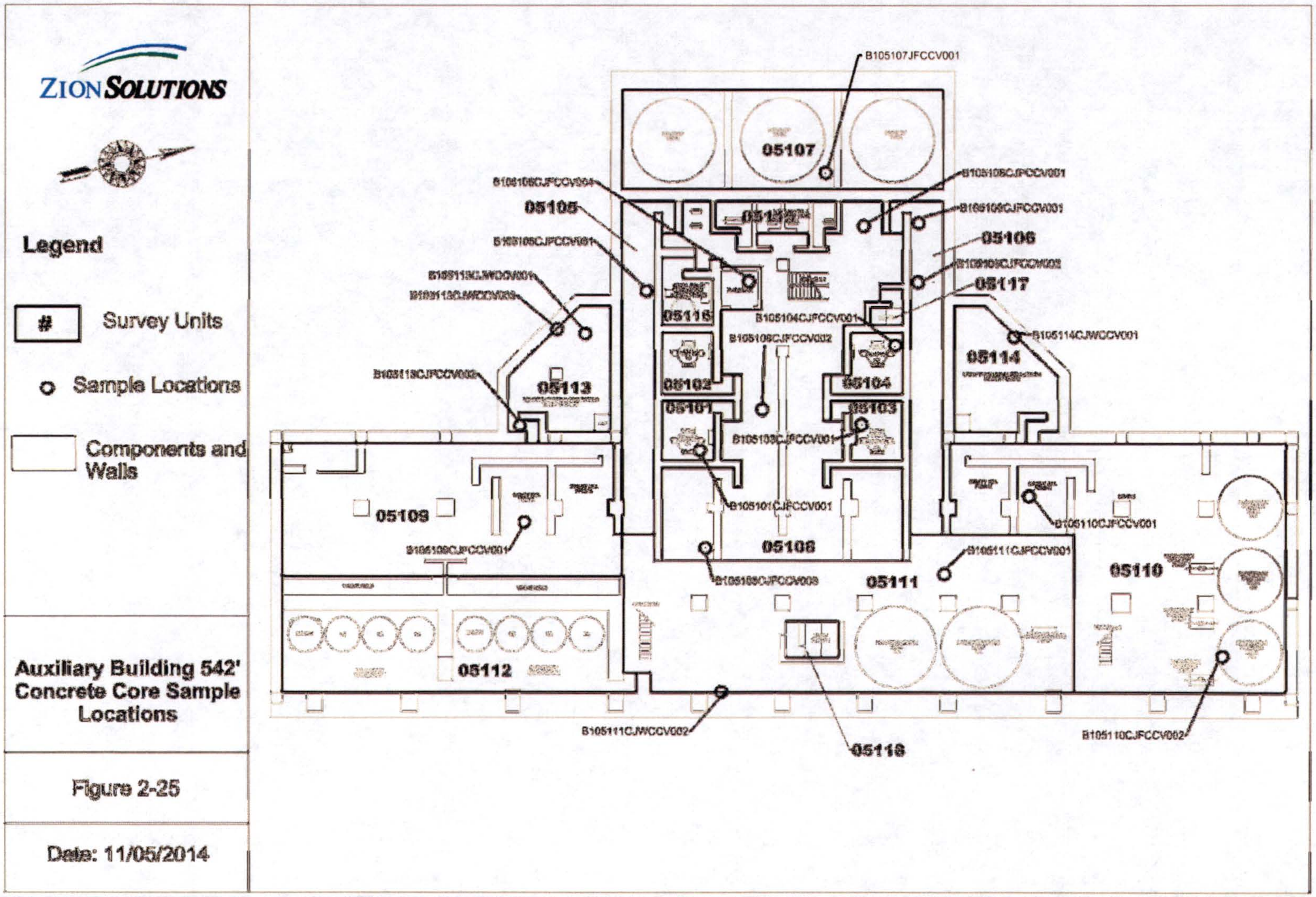


Figure 2-26 Turbine Building 560 foot el. and Steam Tunnel 570 foot el. Concrete Core Sample Locations

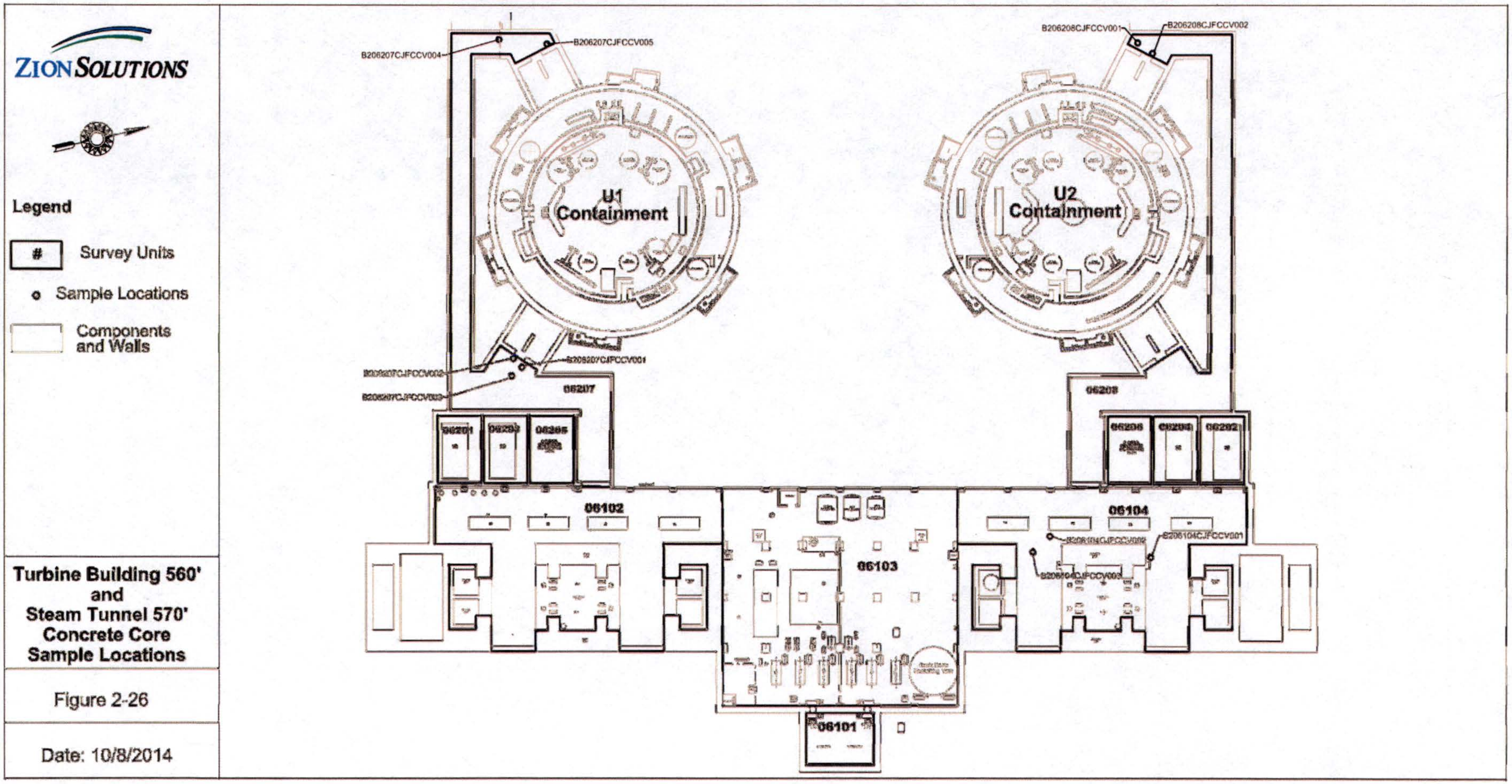


Figure 2-27 Turbine Building 560 foot el. Floor Drain System Sediment Sample Locations

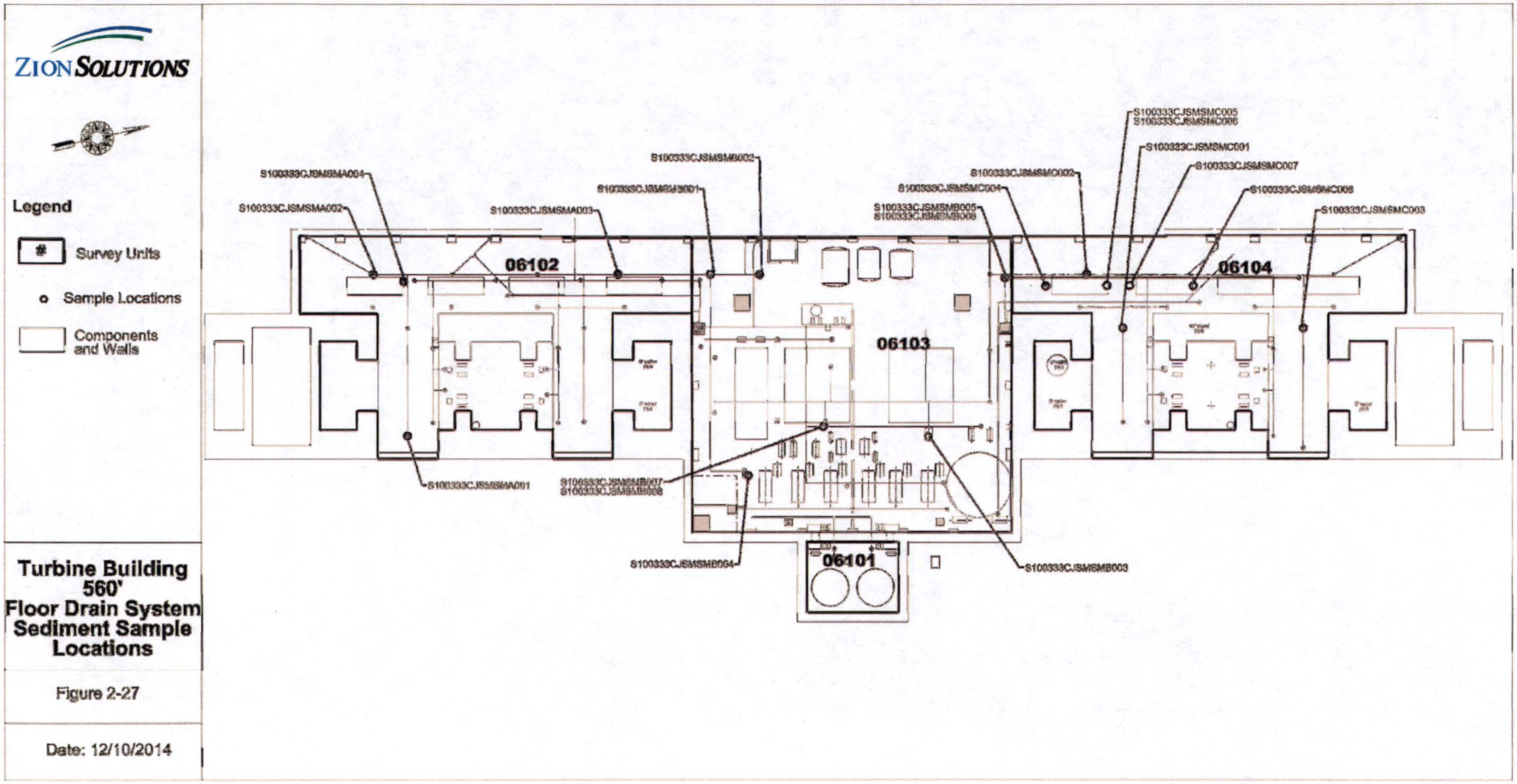


Figure 2-28 Non-Impacted Open Land Survey Units Measurement and Sample Locations

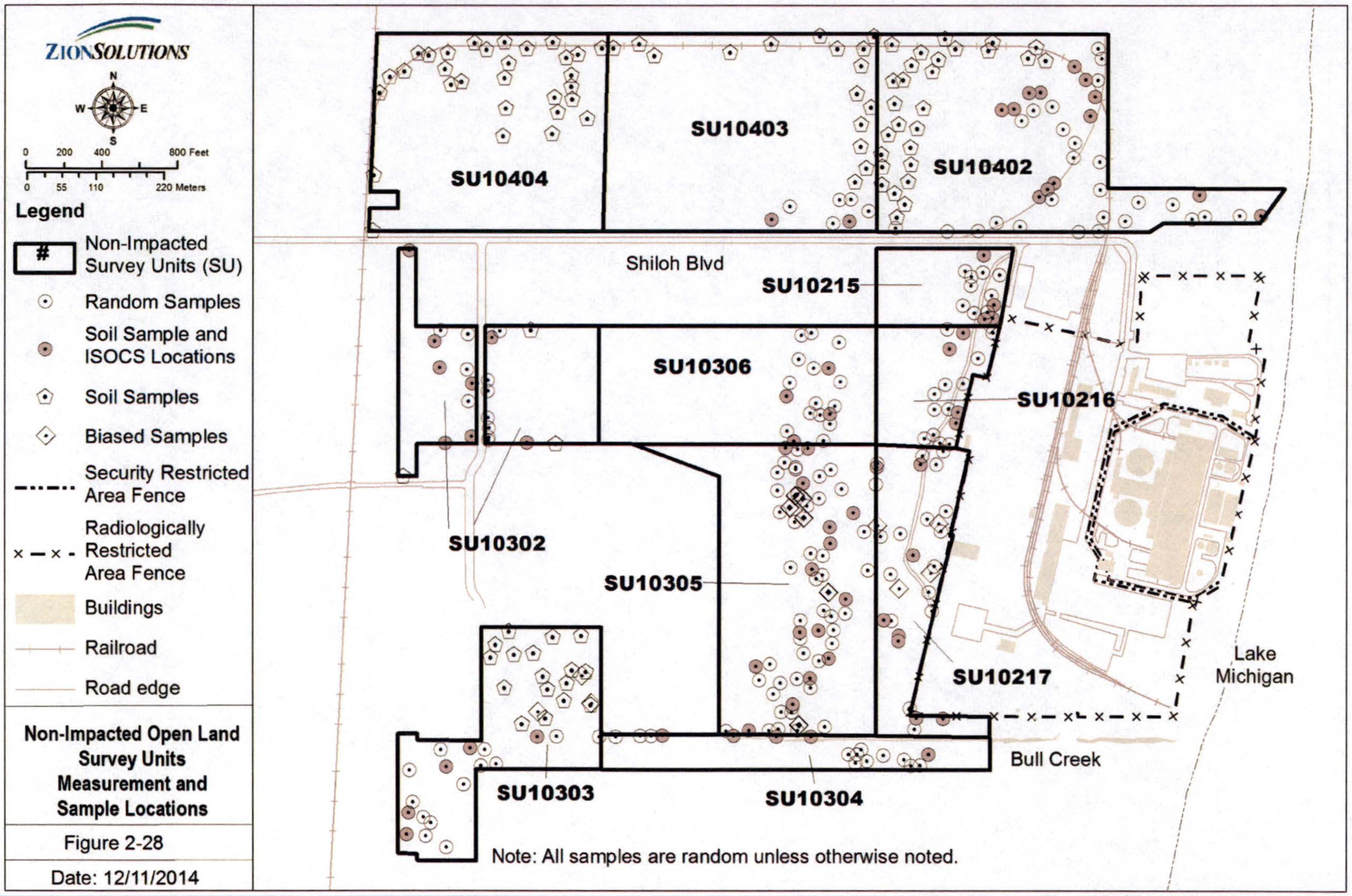


Figure 2-29 Class 3 Open Land Survey Units Located Outside of the "Radiologically-Restricted Area" Measurement and Sample Locations

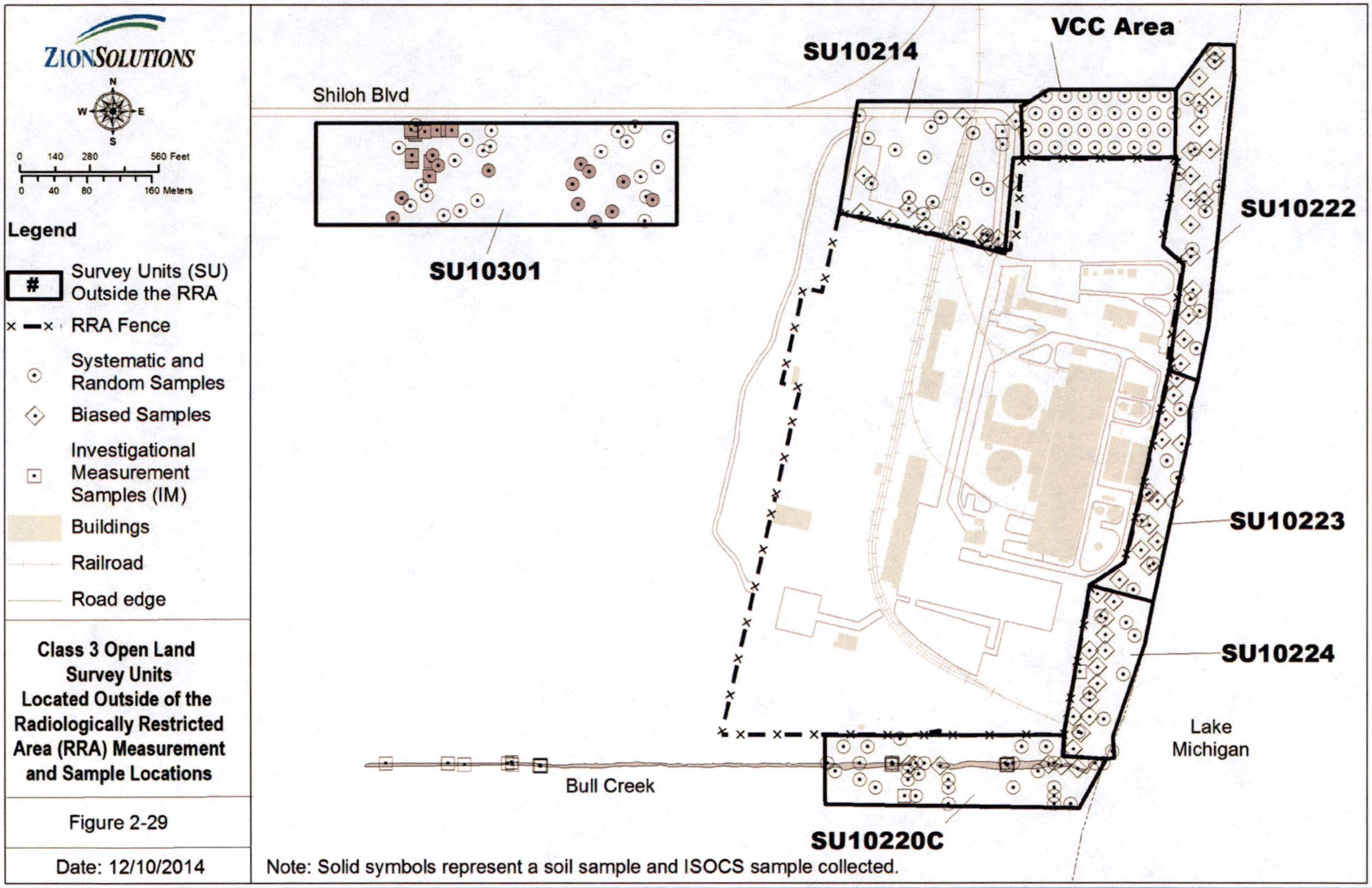


Figure 2-30 Class 3 Open Land Survey Units Located Inside of the "Radiologically-Restricted Area" Measurement and Sample Locations

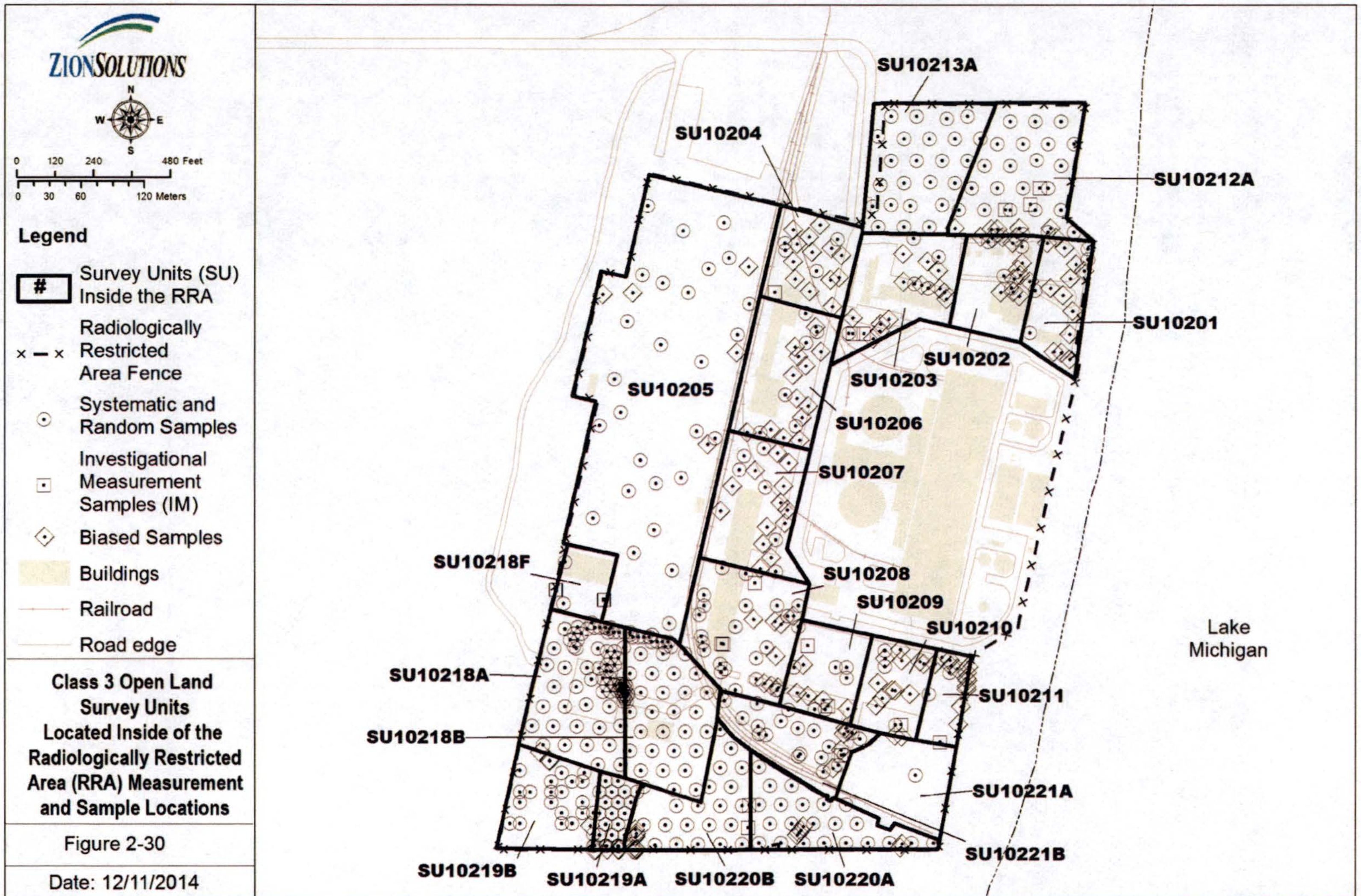


Figure 2-31 Class 2 Open Land Survey Units Measurement and Sample Locations

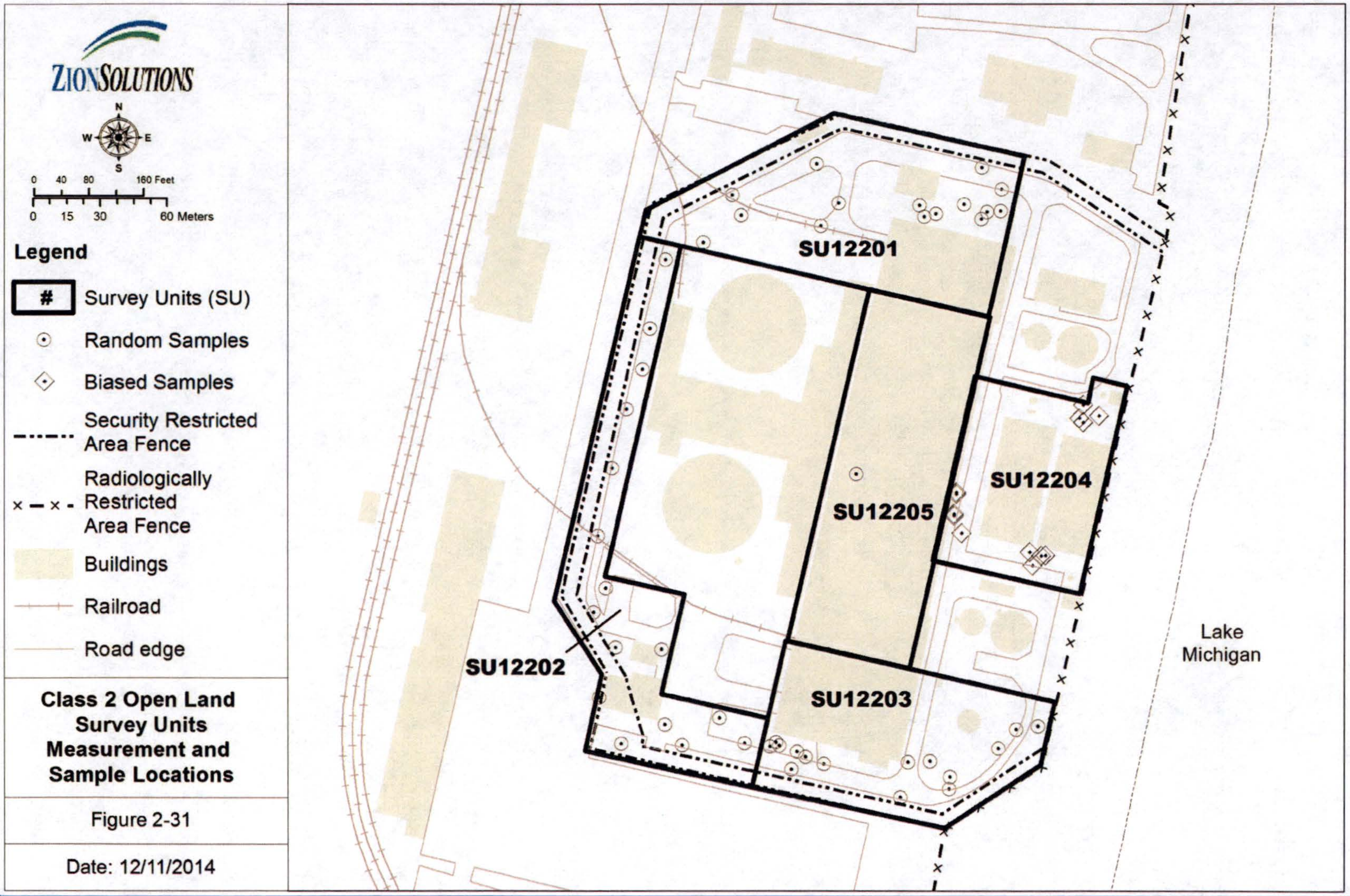


Figure 2-32 Class 1 Open Land Survey Units Measurement and Sample Locations

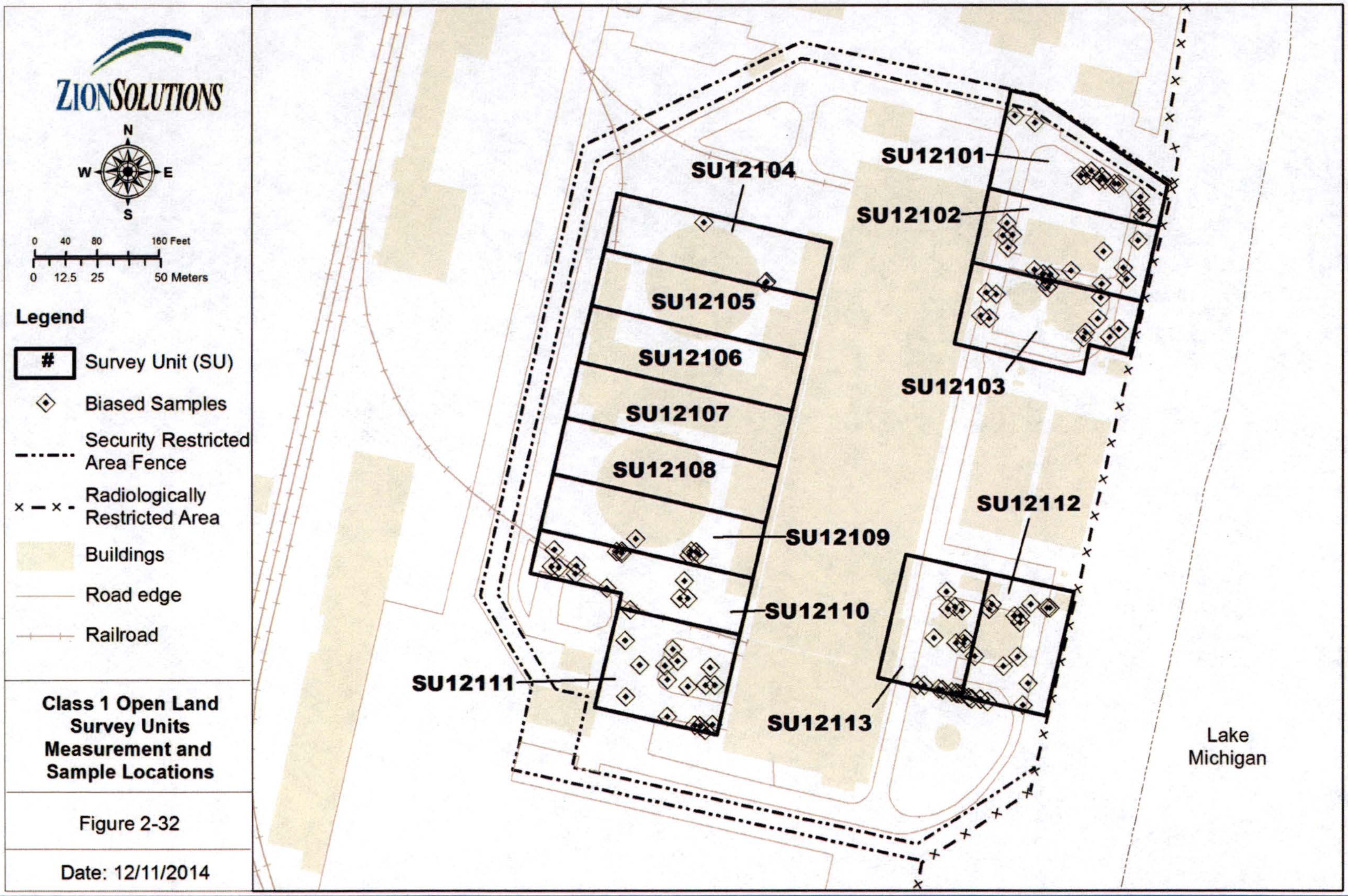
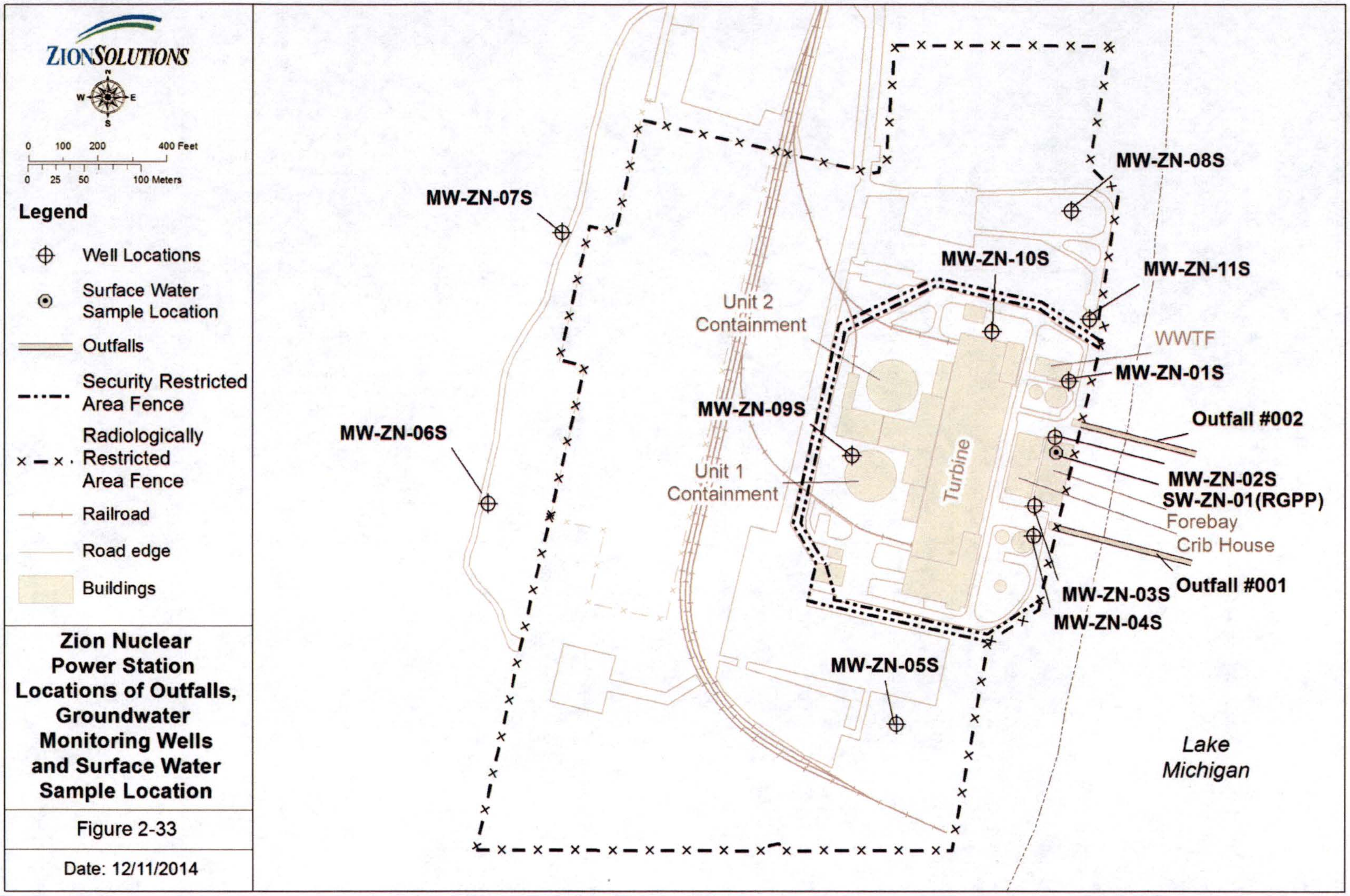


Figure 2-33 ZNPS Locations of Outfalls, Groundwater Monitoring Wells and Surface Water Sampling Locations



ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 3, REVISION 2
IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

TABLE OF CONTENTS

LIST OF TABLES

LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	As Low As Reasonably Achievable
ACM	Asbestos Containing Material
CST	Condensate Storage Tank
DSAR	Defueled Safety Analysis Report
EPA	Environmental Protection Agency
FOT	Fuel Oil Tank
FSS	Final Status Survey
GTCC	Greater than Class C
HVAC	Heating Ventilation Air Conditioning
ISFSI	Independent Spent Fuel Storage Installation
IRSF	Interim Radioactive Waste Storage Facility
LTP	License Termination Plan
MDC	Minimum Detectable Concentration
MMTC	Mechanical Maintenance Training Center
NESHAP	National Emissions Standards for Hazardous Air Pollutants
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
ODCM	Dose Calculation Manual
PSDAR	Post Shutdown Decommissioning Activity Report
PWST	Primary Water Storage Tanks
RA	Radiological Assessment
RCRA	Resource Conservation and Recovery Act
RWP	Radiation Work Permit
SFP	Spent Fuel Pool
SFPI	Spent Fuel Pool Island
SRP	Site Remediation Program
TSCA	Toxic Substances Control Act
TSD	Technical Support Document
URS	Unconditional Release Survey
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

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3. IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1. Introduction

In accordance with 10CFR50.82 (a)(9)(ii)(B), the License Termination Plan (LTP) must identify the remaining major dismantlement and decontamination activities for the decommissioning at the time of submittal. The information includes those areas and equipment that need further remediation and an assessment of the potential radiological conditions that may be encountered. Estimates of the occupational radiation dose for completion of the scheduled task and the projected volumes of radioactive waste that will be generated are also included. These activities will be undertaken pursuant to the current 10 CFR 50 license, are consistent with the Zion Nuclear Station "*Post Shutdown Decommissioning Activity Report*" (PSDAR) (Reference 3-1), and do not depend upon LTP approval to proceed.

Zion*Solutions* primary goals are to decommission the Zion Nuclear Power Station (ZNPS) safely and to maintain the continued safe storage of spent fuel. Zion*Solutions* will decontaminate and dismantle the ZNPS in accordance with the DECON alternative, as described in NUREG-0586 *"Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, Supplement 1, Volume 1"* (Reference 3-2). Completion of the DECON option is contingent upon continued access to one or more low level waste disposal sites. Currently, Zion*Solutions* has access to the low-level waste disposal facility located in Clive, Utah.

Chapter 3 reflects site conditions as of January 6th 2018. Essentially the two Containment domes, Waste Water Treatment Facility, and upper Forebay surfaces are the only structures remaining to be demolished. Description of work completed prior to January 6th 2018 is referred to in past tense.

Zion*Solutions* is currently conducting active decontamination and dismantlement activities at ZNPS in accordance with the PSDAR. Decommissioning activities are being coordinated with the applicable Federal and State regulatory agencies in accordance with plant administrative procedures. Applicable Federal agencies include the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Environmental Protection Agency (EPA). Coordination with applicable State and local regulatory agencies are addressed in section 8.7.2 of Chapter 8. In order to minimize the impact of ongoing decommissioning activities, a Spent Fuel Pool Island (SFPI) was established to separate spent fuel storage functions from other plant functions and other decommissioning activities.

Decommissioning activities at the Zion Station Restoration Project (ZSRP) will be conducted in accordance with the Zion Station *"Defueled Safety Analysis Report"* (DSAR) (Reference 3-3), the NRC Docket Number 50-295, *"Facility Operating License Number DPR-39" (for Unit One)"* (Reference 3-4), NRC Docket Number 50-304, *"Facility Operating License Number DPR-48 (for Unit Two)"* (Reference 3-5), all associated Technical Specifications, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). Currently the remaining activities do not involve any un-reviewed safety questions or changes in the Technical Specifications for ZNPS. If an activity requires prior NRC approval under 10 CFR 50.59(c)(2), or a change to the technical specifications or license, a submittal will be made to the NRC for review and approval before implementing the activity in question. Decommissioning activities are conducted under the

scrutiny of the existing Zion*Solutions* Radiation Protection Program, Industrial Safety Program, and Waste Management Program. Such activities will be conducted in accordance with these programs, which are well established and frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during operations, especially those during major maintenance and outage evolutions.

The remaining decontamination and dismantlement activities that will be performed are described in section 3.3. The specific system considerations that will be taken into account are discussed in sections 3.3.2 through 3.3.7. These sections provide an overview and describe the major remaining components of contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations. Table 3-1 contains a list of major systems and components that have been or are to be removed.

On January 25, 2008, Exelon and Zion*Solutions* submitted an *Application for License Transfers and Conforming Administrative License Amendments* (Reference 3-6) requesting that the NRC approve the transfer of Exelon Corporation's Facility Operating Licenses for ZNPS to Zion*Solutions*. On September 1, 2010, the licenses were transferred from Exelon to Zion*Solutions* (*"Issuance of Conforming Amendments Relating to Transfer of Licenses for Zion Nuclear Power Station, Units 1 and 2"* [Reference 3-7]). Integral to the transfer of the licenses, Zion*Solutions* entered into an agreement with Exelon Corporation titled *"Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement"* (Reference 3-8). This document presents the terms and conditions under which Zion*Solutions* would decommission ZNPS, construct an Independent Spent Fuel Storage Installation (ISFSI), place the spent nuclear fuel in dry cask storage and transfer the loaded fuel casks to the ISFSI, and remediate the site to the unrestricted release criteria as specified in 10 CFR 20.1402. Once the balance of the site is remediated and the as-left radiological conditions are demonstrated to be below the unrestricted release criteria, the 10 CFR Part 50 license will be reduced to the area around the ISFSI and the site will be transferred back to Exelon under the 10 CFR Part 50 license. Zion*Solutions* commenced the active decommissioning of ZNPS on October 13, 2010. Spent fuel and decommissioning activities completed to date are provided in section 3.2.

3.2. Completed and Ongoing Decommissioning Activities and Tasks

3.2.1. Overview

Zion*Solutions* and its subcontractors have completed all demolition activities with the exception of the demolition of the two Containment domes, Waste Water Treatment Facility, and upper surfaces of the Forebay.

Other completed Decommissioning activities include:

- Safely shipped and disposed of the Reactor Vessel Internals and Reactor Vessels.
- Safely shipped and disposed of all large components including the 8 steam generators.
- Safely completed preparation for and the implementation of open air demolition of the Auxiliary Building and Fuel Handling Building.

- Successfully completed all required remediation and sampling for non-Radiological/Environmental site closure under the Illinois Environmental Protection Agency's (IEPA) Site Remediation Program (SRP). This resulted in the issue of a 'No Further Remediation Letter' by IEPA to Zion*Solutions* on November 7, 2017
- Invested significant financial and management resources in ALARA measures/ALARA culture that allowed reduction in the project person-rem goal on 3 occasions from an original estimate of 1100 person-rem to the current estimate of 425 person-rem.

3.2.2. Spent Fuel Island and ISFSI Activities

A priority task for Zion*Solutions* was the construction of the ISFSI and the necessary licensing, training and infrastructure modifications required to transfer spent fuel from the SFP to the ISFSI. As part of this process, the Fuel Handling Building was upgraded with a new single-failure-proof crane. The ISFSI was constructed and became operational in late 2013. All (61) dry cask storage canisters (2226 spent fuel assemblies) and 4 Greater-Than-Class C (GTCC) waste canisters were safely and successfully moved to the ISFSI over a 366 day period.

3.2.3. Demolition and Dismantlement of Initial Structures

To date, with the exception of the two Containment domes, the Waste Water Treatment Facility, the discharge valve houses, and the upper surfaces of the Forebay, all on-site above grade structures have been safely demolished. The demolition of all remaining above grade structures is scheduled for completion by May of 2018.

The Interim Radioactive Waste Storage Facility (IRSF), the Mechanical Maintenance Training Center (MMTC) & Warehouse, and the Fire Maze complex were selected as initial test cases to demonstrate that the plans, programs, and procedures put in place by Zion*Solutions* for the demolition of buildings and structures on the site were ready for implementation. Zion*Solutions* instituted a "Cold, Dark and Dry" methodology that consisted of the following basic activities:

- Identification of any operable systems that may have to be replaced or relocated. For these initial structures, none were identified.
- Isolation of all electrical and mechanical systems servicing the structures.
- Issuance of a contract to a subcontractor designated as the Demolition Contractor to complete demolition activities. For the test case buildings and structures, contracted demolition activities included the installation of any required environmental controls and the demolition and removal of all above grade structures, systems and components. All foundations were either completely removed or demolished and removed to a depth of 3 feet below grade. None of the initial test case structures had sub-grade basements. All buried piping (service air, service water and sewer lines) were removed with the structures. Electrical services (conduits and cables)

were removed to a depth of 3 feet below grade. The remaining excavation void was radiologically surveyed and then backfilled using clean fill to the existing grade.

- Inspection of each structure for all universal, Resource Conservation and Recovery Act (RCRA) or Toxic Substances Control Act (TSCA) wastes that would require removal prior to demolition. These materials included mercury switches, light lamps, electrical ballasts and Asbestos Containing Material (ACM). For the initial test case structures, most of the wastes were directly removed and dispositioned by Zion*Solutions*. The exceptions were mineral oil in a de-energized transformer and the oil and brake shoes in the overhead crane in the IRSF. These wastes were identified to the selected Demolition Contractor and subsequently abated as part of the contracted work scope.

- Completion of unconditional release surveys (URSs) of each structure to ensure the structures can be demolished and free-released. Surveys were performed in accordance with the site procedure for the unconditional release of materials to verify that the material was free of plant-derived radioactive material. Zion*Solutions* Technical Support Document (TSD) 17-010, *"Final Report – Unconditional Release Surveys at the Zion Station Restoration Project"* (Reference 3-9), has been provided to the NRC with an analysis of all unconditional release survey data to ensure that all statistical data requirements were met. For compliance, dose from fill is based the Minimum Detectable Concentration (MDC) of the instrumentation.

- Surveyed and verified concrete debris resulting from the building demolition that was designated for reuse as clean hard fill as radiologically clean. This concrete debris was then processed to remove all exposed rebar and to ensure that individual debris pieces were smaller than 10 inches in diameter. The processed concrete debris was then transported to a designated storage area where it was stockpiled for use as potential backfill material. These stockpile areas are isolated and controlled to prevent the inadvertent introduction of potentially contaminated materials and periodic surveillances are performed.

- All other construction demolition debris that was not stockpiled as potential backfill material was packaged and transported to an appropriate landfill for disposal or to an off-site recycling center. At the ZSRP, all bulk material not leaving site as radwaste, regardless of origin or destination, passes through a radiological truck monitor.

3.2.4. Dismantlement of East Yard Tanks

The next structures to be demolished were the set of tanks located in the east yard of the "Security-Restricted Area". These tanks included the Primary Water Storage Tanks (PWST) and the Condensate Storage Tanks (CST) for both units, the Fuel Oil Tank (FOT), the De-Chlorination and Chlorination tanks for both units and concrete pads and "shacks" constructed to service and house tanks, systems and components. The logic for the removal of the tanks as the next structures in the demolition sequence included;

- The tanks were located within the "Security-Restricted Area" and were considered to be radiologically impacted. Removal of the tanks would allow for the initial use and assessment of the plans, procedures and processes for open-air demolition on radiologically contaminated structures and systems and allow the Demolition Contractor to become acclimated to working in a radiologically controlled environment.
- The radiological contamination of the tanks required contamination mitigation and engineering controls to be implemented as part of the work scope.
- The removal of the tanks provided needed space for the eventual planned demolition of the Crib House and the Turbine Building.

Prior to commencing the dismantlement and demolition of the yard tanks, the tanks and systems were prepared in accordance with the "Cold, Dark and Dry" approach implemented at ZSRP. In order to retire the CSTs as part of this process, it was necessary to design and install a new Demineralized Water Processing system.

The interior and exterior of both PWSTs and CSTs were radiologically surveyed prior to commencing physical dismantlement activities. The survey results indicated that the interiors of the tanks were radiologically contaminated. As a contamination control measure, a fixative was applied to the interior surfaces of the tanks. Following the application of the fixative, a survey was performed to verify that the radiological conditions of the structures met the criteria for open-air demolition as presented in TSD 10-002, *"Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"* (Reference 3-10). Compliance with these criteria minimized the implementation of additional contamination controls that would be required for open-air demolition and allows for the use of heavy equipment to perform the demolition. The intent is to perform this type of survey to verify the radiological conditions in all radiologically-impacted structures, components and systems prior to demolition.

The soils surrounding the tanks were radiologically surveyed as part of the site characterization. No soils were identified during the characterization that would necessitate excavation and removal as radioactive waste. During the course of the tank dismantlement, additional surveys and soil samples were taken of the soil surrounding the tanks. No soil was identified with residual radioactivity in excess of the release criteria. The PWSTs and the CSTs for both units were dismantled, properly packaged and dispositioned as low-level radioactive waste. All radioactive waste was loaded and transported under the direction of Zion*Solutions* Waste Department personnel to the licensed Energy *Solutions* radioactive waste disposal facility in Clive, Utah. Sampling of surface and subsurface soil and groundwater in this area has been performed. Preliminary results demonstrate that some soil will require remediation and disposal off site due to the presence of polynuclear aromatic hydrocarbons above the Illinois EPA limits.

3.2.5. Demolition and Dismantlement of Crib House Structure

The Crib House contained the Circulating Water pumps, the Service Water Pumps and the Fire Protection pumps. In order to implement the "Cold, Dark and Dry" approach in the Crib House,

a number of design modifications were necessary to replace the functioning systems in the Crib House that were required for the operation of operating systems, such as component cooling and fire protection. These included:

- The retirement of the station fire pumps and the integration of a modified fire water ring header that is connected to the city water system for the Town of Zion as a replacement. This modification also resulted in changing the pressurized fire suppression system at ZNPS to a dry system that would be supplied from a new exterior connection for the Town of Zion Fire Department.
- The retirement of the Service Water pumps required the installation of a new pump to supply circulating water to certain systems. This new pump was installed in the Forebay in a manner that would allow for the demolition of the Crib House to proceed. This pump system provided dilution flow for liquid waste releases into the Forebay and also served as a source of emergency make up water to the SFP. The previous Heating Ventilation Air Conditioning (HVAC) functions performed by Service Water were replaced by temporary, local heating and cooling installations and the relocation of certain functions (Hot and Cold Laboratories and Counting Room) to other areas.

Once the system modifications were in place and the Crib House had been successfully made "Cold and Dark", the Crib House was surveyed for unconditional release. Surveys were performed in accordance with the site procedure for the unconditional release of materials to verify that the material was free of plant-derived radioactive material. In addition, all electrical and mechanical systems were isolated and removed as commodities. Once the structure was successfully surveyed and system removal was completed, the above grade portions were demolished by the Demolition Contractor. In order to perform system removal and perform unrestricted release surveys of the deep pump well areas, stop logs and dewatering pumps were installed to isolate the sub-grade areas from the Forebay and Lake Michigan. These measures were implemented by the Demolition Contractor, who also supplied temporary power, lighting and all other support required for the survey performance. Due to the depth of the Circulating Water (centerline elevation 33 feet below grade) and Service Water (centerline elevation 12 feet below grade) headers, they were sealed off at the west wall of the Crib House. These pipe headers were also surveyed for compliance with the unrestricted release criteria prior to being isolated, abandoned in place and filled with grout or fill as appropriate. All concrete structures were removed to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. All remaining structures below 3 feet below grade were surveyed to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402.

3.2.6. Additional Activities

Additional activities that have been completed or are ongoing include, but are not limited to the following:

- Continued assessment of the functional requirements for plant systems, structures, and components.

- Identification of plant systems, structures, and components needed to support safe storage of the spent fuel, support SFP cooling, and facilitate ongoing plant activities.
- Design, installation and operation of a new Liquid Radioactive Waste Processing system.
- Detailed planning and project scheduling.

The Liquid Radioactive Waste system at ZNPS had become degraded since the shutdown of the units and was not capable of successfully processing liquid radwaste for effluent discharge. Consequently, Zion*Solutions* elected to design and install a new Liquid Radioactive Waste Processing system. This system was used to process liquid radwaste at ZNPS and to process the water from each of the reactor cavities (approximately 500,000 gallons each) and the SFP (approximately 700,000 gallons including the transfer canal), once all spent fuel had been moved to the ISFSI. The Turbine Building and Auxiliary Building could not be completely placed in a "Cold, Dark and Dry" status until this liquid waste processing was completed in 2015.

3.3. Future Decommissioning Activities and Tasks

3.3.1. Overview

Spent fuel movement to the ISFSI was completed on January 10th 2015. Once the movement of the spent fuel was complete, other significant dismantlement and decommissioning tasks took place. The removal of the spent fuel from the Fuel Handling Building allowed Zion*Solutions* to implement a license amendment to the 10 CFR 50 license for each unit to remove operational requirements and technical specifications specifically required for the maintenance of spent nuclear fuel in wet pool storage. These license amendments allowed the remaining structures to be placed into a "Cold, Dark and Dry" state, to complete the processing of the remaining liquid radioactive waste, to allow for the complete removal of all remaining commodities and to enhance the ability to freely move material and personnel around the site.

3.3.2. Turbine Building (Unit 1 and Unit 2)

Large component removal in the Turbine Building was completed in 2015. Initial component removal included the dismantlement and removal of most of the large components in Unit 1, including the turbines, generator, moisture separator reheaters, feedwater heaters and coolers, and several feedwater heaters and coolers in Unit 2. In parallel with this effort, the Zion*Solutions* Characterization/License Termination personnel performed surveys for the unconditional release of materials, equipment and structural surfaces throughout the building. In addition, inspections were completed to identify any remaining waste streams. Surveys were performed in accordance with the National Emissions Standards for Hazardous Air Pollutants (NESHAP) to identify any potential ACM, and all accessible friable ACM was removed. Any remaining identified potential ACM that was not accessible, including but not limited to gaskets in piping systems, caulking around windows, floor and wall barrier seals, was appropriately handled and abated by the Demolition Contractor as part of the contracted work scope.

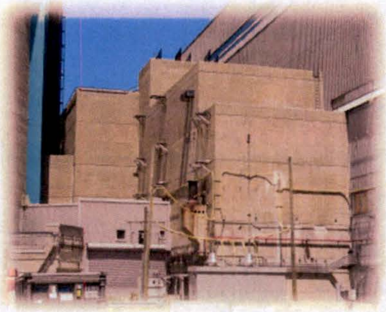
All systems and materials that were identified by radiological survey as contaminated with detectable plant-derived radioactive material were removed by Zion*Solutions* personnel and dispositioned and properly disposed of as radioactive waste. The remaining structure and materials in the Turbine Building were demonstrated to meet the unconditional release criteria. Surveys were performed in accordance with the site procedure for the unconditional release of materials to verify that the material was free of plant-derived radioactive material. The remaining structure were then made "Cold, Dark and Dry" and turned over to a Demolition Contractor as a non-radiologically controlled structure for demolition as a contracted work scope.

The selected Demolition Contractor removed and dispositioned all remaining commodities, and demolished the structure to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. All remaining structures below 3 feet below grade were subjected to a Final Status Survey (FSS) in accordance with the requirements of Chapter 5 to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402. All equipment and components were removed from the structure with the exception of the underground circulating water headers, discharge tunnels and buried service water piping running between the Crib House location and the Auxiliary Building. All remaining buried and piping embedded in concrete has been surveyed for compliance with the unrestricted release criteria prior to being isolated, abandoned in place and filled with grout or fill as appropriate.

Concrete debris from the building demolition was designated for beneficial reuse as clean hard fill. Only concrete which met the non-radiological definition of clean concrete demolition debris and where a URS/FSS demonstrated that the concrete was free of plant derived radionuclides above background was used. This concrete debris was then processed to remove all exposed rebar and to ensure that individual debris pieces were smaller than 10 inches in diameter. The processed concrete debris was then transported to a designated on-site storage area where it was stockpiled for use as potential backfill material. All other construction demolition debris that was not appropriate for reuse as potential backfill material was packaged and transported to an appropriate landfill for disposal or to an off-site recycling center, following final assessment for the presence of any residual radioactive contamination by passing through a radiological truck monitor. Compliance with the unrestricted release criteria was demonstrated, including the completion of confirmatory surveys. The Turbine Building void was backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the original site grade and contours, with at least the top 3 feet as soil only. The top 3 feet of fill will be soil only (i.e. concrete clean hard fill was only utilized as fill up to the 588 foot elevation).

3.3.3. Auxiliary Building

Component and system removal was completed in the Auxiliary Building in 2016. Radiological surveys were performed to verify that as-left contamination levels were below the criteria established as suitable for open-air demolition. Any material identified with radiological contamination in excess of the open-air demolition limits were removed prior to commencing structural demolition. All radioactive waste was loaded and transported under the direction of Zion*Solutions* Waste Department personnel to the licensed Energy*Solutions* radioactive waste disposal facility in Clive, Utah. All structural decontamination activities were performed in



accordance with an approved Radiation Work Permit (RWP) and under the oversight of Zion*Solutions* Radiation Protection personnel.

Structural surfaces were decontaminated to the open-air demolition limits in accordance with TSD 10-002, and the Auxiliary Building was placed in a "Cold, Dark and Dry" configuration. The selected Decommissioning Contractor demolished the structure to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. All below grade interior floors (560 and 579 foot elevations) and walls were removed. Contamination control methods (vacuuming, wiping, etc.) were used to mitigate loose surface contamination on the remaining exposed structural surfaces. All construction debris resulting from the demolition of the Auxiliary Building structure was treated as low level radioactive waste and was shipped to the licensed Energy *Solutions* radioactive waste disposal facility in Clive, Utah by gondola railcar. All remaining structures below 3 feet below grade will be surveyed to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402.

Five 24-inch diameter sleeves that are buried in soil between each of the two containments and the Auxiliary Building 542 foot elevation will be surveyed for compliance with the unrestricted release criteria prior to being isolated, abandoned in place and filled with grout or fill as appropriate. Two of the five sleeves have been capped and were never used. The remaining three sleeves housed 20-inch diameter Recirculating Sump Suction lines, which have been removed.

Several other sections of piping systems associated with the Auxiliary Building may also remain. The bounding list of buried piping, penetrations and embedded piping to remain is provided in Zion*Solutions* TSD 14-016, "*Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State*" (Reference 3-11). The decision to remove or abandon in place will be made based on the results of a cost-benefit analysis that will be performed once access to the pipe sections become possible. In all cases, any buried or embedded piping that will remain will be surveyed for compliance with the unrestricted release criteria prior to being isolated, abandoned in place and filled with grout or fill as appropriate.

Once the remaining concrete structure located 3 feet below grade (extending between the 542 foot and 588 foot elevation) has been satisfactorily surveyed and compliance with the unrestricted release criteria has been demonstrated and, contingent upon the completion of confirmatory surveys, the Auxiliary Building void will be backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the original site grade and contours. The top 3 feet of fill will be soil only (i.e. concrete clean hard fill will only be utilized as fill up to 588 foot elevation).

3.3.4. Unit 1 and Unit 2 Containments

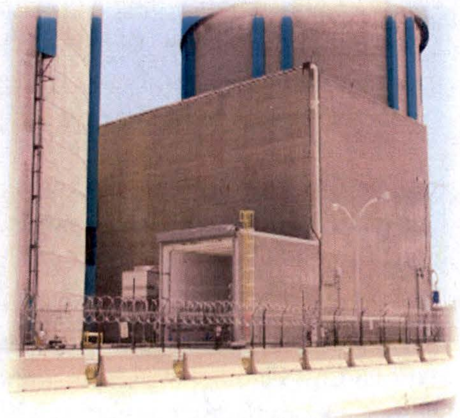
Component and system removal was completed in the Unit 1 and Unit 2 Containment Buildings in 2016. Radiological surveys were performed to verify as-left contamination levels were below the criteria established as suitable for open-air demolition in accordance with TSD 10-002. Any material identified with radiological contamination in excess of the open-air demolition limits was removed. All radioactive waste was loaded and transported under the direction of Zion*Solutions* Waste Department personnel to the licensed Energy *Solutions* radioactive waste disposal facility in Clive, Utah. All structural decontamination activities were performed in accordance with an approved RWP(s) and under the oversight of Zion*Solutions* Radiation Protection personnel.

The Containment Buildings were then placed in a "Cold, Dark and Dry" configuration. In both Containment basements, concrete was removed from the interior side of the steel liner above the 565 foot elevation, leaving only the remaining exposed liner below the 588 foot elevation, 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. Contamination control methods (vacuuming, wiping, etc.) were used to mitigate loose surface contamination on the remaining exposed structural surfaces. All construction debris resulting from the demolition of each of the Containment Building internal structures was treated as low level radioactive waste and was shipped to the licensed Energy*Solutions* radioactive waste disposal facility in Clive, Utah by gondola railcar. The exposed steel liner below the 588 foot elevation and Under-Vessel concrete will be surveyed for compliance with the unrestricted release criteria as specified in 10 CFR 20.1402, including any required confirmatory surveys. Once the Containment structural surfaces located 3 feet below grade (588 foot elevation) have been satisfactorily surveyed and compliance with the unrestricted release criteria has been demonstrated and, contingent upon the completion of confirmatory surveys and regulatory concurrence the Containment Building basements will be filled to above the 588 foot elevation using clean fill or concrete debris suitable for reuse as clean hard fill. The top 3 feet of fill will be soil only.

After backfilling each Containment basement with clean fill, both Containment shells will be demolished to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. Demolition will include the removal of the pre-stressing tendons and the gradual demolition of the containment shells from grade, using ram-hoes to chip away the concrete along the bottom circumference of the shell and allowing the weight of the remaining structure to slowly demolish the structure to grade. This is a similar approach to that used to demolish the Containment structure during the decommissioning of the Connecticut Yankee Atomic Power Company's Haddam Neck Nuclear Power Plant. The process will also allow for the removal of all exposed rebar and to ensure that individual debris pieces are smaller than 10 inches in diameter. Concrete debris resulting from the building demolition may be designated for beneficial reuse as clean hard fill, if it meets the non-radiological definition of clean concrete demolition debris and if URS/FSS demonstrates that the concrete is free of plant derived

radionuclides above background. The processed concrete debris will then be transported to a designated storage area where it will be stockpiled for use as potential backfill material. If the material is not used as clean hard fill, then it will be packaged and transported to an appropriate landfill for disposal or to an off-site recycling center, following final assessment for the presence of any residual radioactive contamination by passing through a radiological truck monitor.

3.3.5. Fuel Handling Building



The dismantlement and decommissioning of the Fuel Handling Building was completed in early 2017 following the placement of all spent nuclear fuel located in the SFP into dry cask storage and transfer of the packaged fuel to the ISFSI facility. The SFP water was processed by the Liquid Radioactive Waste system, sampled and discharged through the normal effluent release pathway into Lake Michigan upon meeting the radiological release criteria. The 23 empty spent fuel storage racks were removed from the pool, packaged and shipped for disposal as radioactive waste.

With the pool empty and dry, all known radioactively contaminated systems and components were removed by a subcontracted Demolition Contractor and properly disposed of as radioactive waste. This included the steel liner of the SFP. In parallel, Zion*Solutions* Radiation Protection personnel performed surveys of the exposed SFP concrete to verify as-left contamination levels were below those established for open-air demolition. All radioactive waste was loaded and transported under the direction of Zion*Solutions* Waste Department personnel to the licensed Energy*Solutions* radioactive waste disposal facility in Clive, Utah. All structural decontamination activities were performed in accordance with an approved RWP(s) and under the oversight of Zion*Solutions* Radiation Protection personnel.

Following commodity removal and structural decontamination, the Fuel Handling Building was turned over to a subcontracted Decommissioning Contractor. The Decommissioning Contractor demolished all of the remaining interior systems, structures, and components down to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. All construction debris resulting from the demolition of the Fuel Handling Building was treated as low level radioactive waste and was shipped to the licensed Energy *Solutions* radioactive waste disposal facility in Clive, Utah by gondola railcar.

Once any remaining concrete structures located below the 588 ft. elevation, which include the concrete sub-slab for the SFP, have been satisfactorily surveyed and demonstrated to be in compliance with the unrestricted release criteria as specified in 10 CFR 20.1402 and, contingent upon the completion of confirmatory surveys, the void where the Fuel Handling Building once stood will be backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the original site grade and contours. The top 3 feet of fill will be soil only (i.e. concrete clean hard fill will only be utilized as fill up to 588 foot elevation).

3.3.6. Waste Water Treatment Facility (WWTF)

The Wastewater Treatment Facility (WWTF) was designed to treat non-radioactive and low-level radioactive liquid from ZNPS sources including building roof run-off and the Turbine Building Fire Sump, which received liquid waste from the Turbine Building Equipment and Floor Drains, and the Fuel Pool Cooling Tower Blowdown. The WWTF was designed to remove suspended solids and oil to ensure compliance with the facility National Pollutant Discharge Elimination System (NPDES) permit. Since the wastewater discharge rates were variable, an equalization tank was installed. The WWTF also includes other equipment such as mixing tanks, mixers, oil skimmers, flocculators, oil coalescers, clarifiers, sludge drying beds and filters. Discharge from the WWTF was by gravity to the Forebay. During ZNPS operations, liquid waste with detectable low-level radioactive contamination was processed by the WWTF. Consequently, the internal surfaces of the WWTF systems are considered as potentially contaminated.

All systems, component and materials associated with the WWTF that are or will be identified by radiological survey as contaminated with detectable plant-derived radioactive material will be removed by Zion*Solutions* personnel and dispositioned and properly disposed of as radioactive waste. The remaining structure will then be made "Cold, Dark and Dry". Once this is complete, all remaining commodities and all structural surfaces will be demolished to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. All remaining structural surfaces from the WWTF below the 588 ft. elevation will undergo a survey to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402.

Once the remaining concrete structure located below 3 feet below grade (588 foot elevation) has been satisfactorily surveyed and compliance with the unrestricted release criteria has been demonstrated and, contingent upon the completion of confirmatory surveys, the WWTF void will be backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the original site grade and contours. The top 3 feet of fill will be soil only (i.e. concrete clean hard fill will only be utilized as fill up to the 588 foot elevation).

3.3.7. Miscellaneous Structures

3.3.7.1. East and West Service Buildings

These two structures were demolished in 2016 and were located to the south of the Unit 1 Turbine Building and were utilized primarily as office space and a machine shop. Both were steel framed structures with no sub-grade basement. The remaining structures and materials in the East and West Service Buildings were surveyed to demonstrate meeting the unconditional release criteria. The remaining structures were then made "Cold, Dark and Dry" and turned over to a Demolition Contractor as a non-radiologically controlled structures for demolition as a contracted work scope. The concrete base slabs and edge beams were removed to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement, and all associated buried piping systems associated

with these buildings are targeted for complete removal and disposal as waste. Any voids created by the demolition of these structures were surveyed by the Zion*Solutions* Characterization/License Termination group and documented as a Radiological Assessment (RA) as part of Continuing Characterization. Upon completion of the RA and acceptance of the survey results, the building footprints were returned to the original site grade and contours.

3.3.7.2. Forebay, Forebay Valve Houses and Intake and Outflow Structures Located in Lake Michigan

The Circulating Water Intake Piping and Discharge Tunnels located at the bottom of Lake Michigan will remain and be abandoned in place. These structures were surveyed in place to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402.

The accessible Forebay surfaces above the 588 foot elevation and the Valve Houses will be radiologically surveyed to demonstrate that the structural surfaces and materials meet the unconditional release criteria. Surveys will be performed in accordance with the site procedure for the unconditional release of materials to verify that the material is free of plant-derived radioactive material. The Valve Houses and the Forebay will then be made "Cold, Dark and Dry" and turned over to a Demolition Contractor as a non-radiologically controlled structure for demolition as a contracted work scope. The structural surfaces of the Forebay located below the 588 foot elevation have been surveyed to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402.

Contingent upon regulatory concurrence, the selected Demolition Contractor will remove and disposition all remaining commodities and completely demolish the Valve Houses and demolish the Forebay structure to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. The Contractor will backfill the Forebay using clean fill or concrete debris suitable for reuse as clean hard fill to the original site grade and contours. The top 3 feet of fill will be soil only (i.e. concrete clean hard fill will only be utilized as fill up to 588 foot elevation). All other construction demolition debris that is not appropriate for beneficial reuse as potential backfill material will be packaged and transported to an appropriate landfill for disposal or, to an off-site recycling center following final assessment for the presence of any residual radioactive contamination by passing through a radiological truck monitor.

3.3.7.3. NGET, ENC, South Warehouse, North Security Access Gatehouse

The NGET, ENC, South Warehouse, North Security Access Gatehouse structures were demolished in 2017. They were located in the "Radiologically-Restricted Area" and had been utilized primarily as office and storage space. All were steel framed structures with no sub-grade basement. Radiological surveys were performed in accordance with the site procedure for the unconditional release of materials to verify that the structural surfaces and materials in each of these buildings were free of plant-derived radioactive material. No recycled materials will

remain on site. The structures were then made "Cold, Dark and Dry" and turned over to a Demolition Contractor as non-radiologically controlled structures for demolition as a contracted work scope. The concrete base slabs and wall foundations were removed to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement and all associated buried piping systems associated with these buildings are targeted for complete removal and disposal as waste. All electrical services were de-energized and removed to a depth of 3 feet below grade. Any void created by the demolition of these structures was surveyed by the Zion*Solutions* Characterization/License Termination group and documented as a RA. Upon completion of the RA and acceptance of the survey results, any voids were backfilled to the original site grade and contours.

3.3.7.4. Laundry Building, North Security Restricted Area Gatehouse, South Security Restricted Area Access, Restricted Area Fence and Vehicle Barrier System

These structures were demolished in 2016-2017. They were located in the "Security-Restricted Area" and were primarily used for security. These structures were no longer required once all the spent nuclear fuel was moved to the ISFSI. The structures were a mix of steel frame, slab on grade and reinforced concrete construction. Radiological surveys were performed in accordance with the site procedure for the unconditional release of materials to verify that the structural surfaces, and materials in each of these buildings were free of plant-derived radioactive material. The structures were then made "Cold, Dark and Dry" and turned over to a Demolition Contractor as non-radiologically controlled structures for demolition as a contracted work scope. The concrete base slabs and wall foundations were removed to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement and all associated buried piping systems associated with these buildings are targeted for complete removal and disposal as waste. All electrical services were de-energized and removed to a depth of 3 feet below grade. Any void created by the demolition of these structures was surveyed by the Zion*Solutions* Characterization/License Termination group and documented as a RA. Upon completion of the RA and acceptance of the survey results, any voids were backfilled to the original site grade and contours.

3.3.7.5. Steam Tunnels and Waste Handling Area

The Steam Tunnels are buried structures that connect the Unit 1 and Unit 2 Containment Building with the Turbine Building at the 570 foot elevation. The roofs of the Steam Tunnels were demolished and removed in 2016. They were constructed of reinforced concrete. The Waste Handling Area was a steel frame building built on an on-grade concrete slab that was also demolished in 2016. Both of these structures underwent radiological surveys to verify that as-left contamination levels were below those established for open-air demolition prior to commencing decommissioning. Based upon the results of these surveys, the remaining systems, components and structural surfaces required to be removed or decontaminated prior to permitting "open air demolition" in accordance with TSD 10-002 were removed or successfully decontaminated prior to structural demolition. All radioactive waste was loaded and transported

under the direction of Zion*Solutions* Waste Department personnel to the licensed Energy*Solutions* radioactive waste disposal facility in Clive, Utah.

Once all identified radioactive systems were removed and all required remediation was completed, the structures were then be made "Cold, Dark and Dry" and turned over to a Decommissioning Contractor for demolition as a contracted work scope. The Waste Handling Building was completely removed, including the concrete slab. The roof of the Steam Tunnels was exposed by excavation. The concrete roof slabs were demolished and removed. All remaining commodities in the Steam Tunnels were removed through the opening created by removing the roof. Concrete debris resulting from the building demolition was designated for beneficial reuse as clean hard fill. Only concrete that met the non-radiological definition of clean concrete demolition debris and where URS/FSS demonstrated that the concrete is free of plant derived radionuclides above background was used. All clean construction debris that was not appropriate for reuse as potential backfill material was packaged and transported to an appropriate landfill for disposal or, to an off-site recycling center following final assessment for the presence of any residual radioactive contamination by passing through a radiological truck monitor.

The remaining structural surfaces of the Steam Tunnels located below the 588 foot elevation were surveyed to demonstrate compliance with the unrestricted release criteria as specified in 10 CFR 20.1402. Once the compliance survey was completed , and contingent upon the completion of confirmatory surveys, the selected Decommissioning Contractor demolished the Steam Tunnel structures to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement.

Following completion of FSS, the Steam Tunnels' voids were backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the original site grade and contours. The top 3 feet of fill consisted of soil only (i.e. concrete clean hard fill will only be utilized as fill up to 588 foot elevation).

3.3.7.6. <u>Old Sewage Treatment Facility and Meteorological Tower</u>

The Old Sewage Treatment Facility and the Meteorological Tower were demonstrated as meeting the unconditional release criteria and demolished in 2016. Surveys were performed in accordance with the site procedure for the unconditional release of materials to verify that the material was free of plant-derived radioactive material. The structures were then made "Cold, Dark and Dry" and turned over to a Decommissioning Contractor as non-radiologically controlled structures for demolition as a contracted work scope. The structures, concrete slabs all associated buried piping systems associated with these buildings were completely removed and disposed of as waste. All electrical services were de-energized and removed to a depth of 3 feet below grade. Any void created by the demolition of these structures was surveyed by the Zion*Solutions* Characterization/License Termination group and documented as Continuing Characterization. Upon completion of the Continuing Characterization and acceptance of the survey results, any voids were backfilled to the original site grade and contours.

3.3.7.7. Storm Drain System, Manholes and Fire Protection Buried Piping

The existing fire protection piping (including hydrants and valves) and a majority of the storm drain system (including the oil separators) were removed and disposed of as clean waste in 2017. Several sections of storm drain pipe are still located in the west yard of the "Security-Restricted Area". This remaining pipe is not radiologically contaminated and is scheduled to be removed completely in 2018.

If a situation occurs where difficulty is encountered with ground water infiltration, some of the piping and/or catch basins located greater than a depth of 5 feet below grade may be abandoned in place. In these cases, any piping or catch basins that remain will be surveyed for compliance with the unrestricted release criteria as specified in 10 CFR 20.1402 prior to being isolated and abandoned in place. Any voids created by excavation to support the removal of these systems will be surveyed by the Zion*Solutions* Characterization/License Termination group and documented as a RA. Upon completion of the RA and acceptance of the survey results, any voids will be backfilled to the original site grade and contours.

3.3.7.8. Surface Soils, Subsurface Soil and Groundwater

Characterization survey results and historical survey data indicate that there is minimal residual radioactivity in soil and no groundwater contamination identified to date. As needed, additional investigations will be performed to ensure that any changing soil radiological contamination profile during decommissioning is adequately identified and addressed. Chapter 5 discusses soil sampling and survey methods.

The release criteria (Base Case Derived Concentration Guideline Levels) that will be used to demonstrate compliance with the 25 mrem/yr dose criterion are provided in Tables 5-5 and 5-6 of Chapter 5. Throughout the course of the decommissioning and through site closure, ZSRP will continue to survey and characterize soils as they are exposed by excavation during building demolition or made accessible by the removal of structures or components. If residual radioactivity is discovered in surface or subsurface soils, ZSRP will excavate, package and dispose of any soil contaminated with residual radioactivity at concentrations greater that the unrestricted release criteria.

3.4. Radiological Impacts of Decommissioning Activities

The decommissioning activities described are and will be conducted under the provisions of the Zion*Solutions* Radiation Protection Program and Radioactive Waste Management Program. These programs are and will continue to be implemented as described in the DSAR. The Zion*Solutions* Radiation Protection Program and written site procedures are intended to provide sufficient information to demonstrate that decommissioning activities will be performed in accordance with 10 CFR 19, *"Notices, Instructions And Reports To Workers"*, 10 CFR 20, *"Standards For Protection Against Radiation"* and to maintain radiation exposures As Low As Reasonably Achievable (ALARA). The Zion*Solutions* Radioactive Waste Management Program controls the generation, characterization, processing, handling, shipping, and disposal of radioactive waste in accordance with the approved Zion*Solutions* Radiation Protection Program, Process Control Program, and written plant procedures.

The current Radiation Protection Program, Waste Management Program, and "*Radiological Effluent Monitoring and Offsite Dose Calculation Manual* (ODCM)" (Reference 3-12) will be used to protect the workers and the public during the various decontamination and decommissioning activities. These well-established programs are routinely inspected by the NRC to ensure that workers, the public, and the environment are protected during facility decommissioning activities. It is also important to note that decommissioning activities involve the same radiation protection and waste management considerations as those encountered during plant operations, maintenance and outages. As described in the PSDAR, the decommissioning will be accomplished with no significant adverse environmental impacts in that:

- No site-specific factors pertaining to the decommissioning of the ZNPS would alter the conclusions presented in NUREG-0586 (see LTP Chapter 8).
- Radiation dose to the public will be minimal.
- Decommissioning is not an imminent health or safety concern and will generally have a positive environmental impact.

Continued application of the current and future Radiation Protection and Radiological Effluent Monitoring Programs at ZNPS ensures public protection in accordance with 10 CFR 20 and 10 CFR 50, Appendix I. ODCM reports for ZNPS to date conclude that the public exposure as a result of decommissioning activities is bounded by the evaluation in NUREG-0586, which concludes the impact is minimal.

3.4.1. Control Mechanisms to Mitigate the Recontamination of Remediated Areas

Due to the scope of remaining structures and systems that will be decontaminated and dismantled, some FSS of areas may be performed in parallel with decommissioning activities. Consequently, a systematic approach will be employed to ensure that areas are adequately remediated prior to performing FSS and ongoing decommissioning activities do not impact the radiological condition of areas where compliance with the unrestricted release criteria as specified in 10 CFR 20.1402 has been demonstrated. These measures and mechanisms are described in Chapter 5, sections 5.6.3 and 5.12.

3.4.2. Occupational Exposure

Table 3-2 provides the "to go" cumulative site dose and estimates for the decommissioning of ZNPS. These estimates were developed to provide site management ALARA goals. The goals are verified by summation of actual site dose, as determined by appropriate dosimetry. ALARA estimates are a compilation of RWP estimates for the period. This information is in addition to information gathered for reporting of yearly site dose. The annual report of occupational dose meets the guidance of NRC Regulatory Guide 1.16, *"Reporting of Operating Information, Appendix A, Technical Specifications"* (Reference 3-13). The total radiation exposure impact for decommissioning and spent fuel management was originally estimated to be 1100 person-rem, with a Project Goal of 900 person-rem. Due to significant financial and management investment in ALARA measures and ALARA culture by Energy*Solutions* (e.g. remotely operated diamond wire saws, remote wireless dose, visual and audio monitoring), this estimate has been reduced 3

times and is currently at 425 person-rem. Table 3-2 includes dose for estimate and goal for the remaining project work of 0.622 and 0.600 person-rem respectively.

3.4.3. Exposure to the Public

Continued application of Zion*Solutions* Radiation Protection, Radioactive Waste, Radiological Effluent Technical Specification and Radiological Environmental Monitoring Programs assures public protection in accordance with 10 CFR 20 and 10 CFR 50, Appendix I.

3.4.4. Radioactive Waste Projections

The Radioactive Waste Management Program is used to control the characterization, generation, processing, handling, shipping, and disposal of radioactive waste during decommissioning. Activated and contaminated systems, structures, and components represent the largest volume of low level radioactive waste expected to be generated during decommissioning. Other forms of waste generated during decommissioning include:

- Contaminated water;
- Used disposable protective clothing;
- Expended abrasive and absorbent materials;
- Expended resins and filters;
- Contamination control materials (e.g., strippable coatings, plastic enclosures); and
- Contaminated equipment used in the decommissioning process.

Table 3-3 provides projections of waste classifications and quantities that will be generated by the decommissioning of ZNPS. These waste quantities are consistent with the waste quantities projected in the PSDAR. As Zion*Solutions* has elected to institute an approach commonly referred to as "rip & ship" verses performing significant on-site decontamination activities, the total volume of low-level radioactive waste for disposal has been estimated at approximately 6,000,000 cubic feet. Actual waste volumes and classifications may vary. The vast majority of this waste will be shipped to the licensed Energy*Solutions* radioactive waste disposal facility in Clive, Utah by gondola railcar.

3.5. References

3-1 Letter from Zion*Solutions* to the U.S. Nuclear Regulatory Commission, "Notification of "Amended Post-Shutdown Decommissioning Activities Report (PSDAR) for Zion Nuclear Power Station, Units 1 and 2" – March 18, 2008.

3-2 U.S. Nuclear Regulatory Commission NUREG-0586 "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities", Supplement 1, Volume 1" – November 2002

3-3 Zion Station, "Defueled Safety Analysis Report" (DSAR) – September 2014

3-4 U.S. Nuclear Regulatory Commission Docket Number 50-295, "Facility Operating License Number DPR-39 (for Unit One)"

3-5 U.S. Nuclear Regulatory Commission Docket Number 50-304, "Facility Operating License Number DPR-48 (for Unit Two)"

3-6 Letter from Exelon and Zion*Solutions* - Application for License Transfers and Conforming Administrative License Amendments-January 25, 2008

3-7 Letter from J.B. Hickman (U.S. Nuclear Regulatory Commission) to J. Christian (Zion*Solutions*), "Issuance of Conforming Amendments Relating to Transfer of Licenses for Zion Nuclear Power Station, Units 1 and 2" – September 2010

3-8 "Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement" – December 2007

3-9 Zion*Solutions* Technical Support Document 17-010, Revision 1, "Final Report – Unconditional Release Surveys at the Zion Station Restoration Project"

3-10 ZionSolutions Technical Support Document 10-002, Revision 1, "Technical Basis for Radiological Limits for Structure Building Open Air Demolition

3-11 Zion*Solutions* Technical Support Document 14-016, Revision 0, "Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State"

3-12 Exelon Nuclear "Radiological Effluent Monitoring and Offsite Dose Calculation Manual (ODCM)" – January 2001

3-13 U.S. Nuclear Regulatory Commission Regulatory Guide 1.16, "Reporting of Operating Information, Appendix A, Technical Specifications" – August 1975

Table 3-1 Status of Major ZNPS Systems, Structures, and Components as of December 2018

System or Component	Required for SFP	Status
Reactor Coolant System	No	Removed-Disposed
Reactor Vessel Internals	No	Removed-Disposed
Reactor Vessels	No	Removed-Disposed
Steam Generators	No	Removed-Disposed
Reactor Coolant Pumps	No	Removed-Disposed
Pressurizer	No	Removed-Disposed
Chemical & Volume Control System	No	Removed-Disposed
Safety Injection System	No	Removed-Disposed
Residual Heat Removal System	No	Removed-Disposed
Containment Spray System	No	Removed-Disposed
Component Cooling Water System	No	Removed-Disposed
Service Water System	No	Removed-Disposed
Spent Fuel Pool (SFP)	Yes	Removed-Disposed
Fuel Handling Equipment	No	Removed-Disposed
Spent Fuel Pool Cooling and Demineralizer System (SFPI systems)	Yes	Removed-Disposed
Condensate System	No	Removed-Disposed
Feedwater System	No	Removed-Disposed
Steam Generator Blowdown System	No	Removed-Disposed
Primary Makeup Water System	No	Removed-Disposed
Refueling Water Storage Tank	No	Removed-Disposed
Plant Effluent Monitoring System	No	In place
Containment Ventilation System	No	Removed-Disposed
Fuel Building Ventilation System	Yes	Removed-Disposed
Aux Building Ventilation System	No	Removed-Disposed
Auxiliary Boiler	No	Removed-Disposed

Table 3-1 (continued)

System or Component	Required for SFP	Status
Instrument and Service Air System	No	Removed-Disposed
Gaseous Radioactive Waste System	No	Removed-Disposed
Solid Radioactive Waste System	No	Removed-Disposed
Liquid Radioactive Waste System	No	Removed-Disposed
Makeup Water Systems	Yes	Removed-Disposed
Radiation Monitoring System	Yes	Removed-Disposed
Process Sampling System	No	Removed-Disposed
Fire Protection System	Yes	Removed-Disposed
Electrical Systems	Yes	Removed-Disposed
Containment Building	No	Domes/basements in place
Auxiliary Building	No	Basement in place
Fuel Handling Building	Yes	Basement in place
Turbine Building	No	Basement in place
Service Buildings	No	Removed-Disposed

Table 3-2 Radiation Exposure Projections for Decommissioning After 1/1/2018

Activity	Exposure (person-rem)
Remaining Activities	
Walkdowns/ Tours/ Security	**0.040**
RP routine Activities	**0.030**
Misc. D&D Work Activities	**0.045**
NRC Activities	**0.005**
Sluicing/ WWTF/ Resin and HIC activities	**0.010**
Asbestos Removal Activities	**0.005**
ISFSI Activities	**0.025**
Waste Operations Activities	**0.150**
FSS Activities	**0.095**
Electrical Activities	**0.005**
Decon of equipment/components/material activities	**0.020**
Per/OAD of Aux/FHB/CTMT's/Steam Tunnels/Valve Houses	**0.190**
Visitor Activities	**0.002**
TOTAL Estimate/Goal (remaing after 01/01/18)	**0.622/0.600**
Zion Project Radiation Exposure (thru 12/31/17)	**421 person-rem (vs original estimate of 1100 person-rem) and current estimate of 425 person-rem.**

Table 3-3 Projected Waste Quantities

WASTE TYPE	WASTE CLASS	WASTE WEIGHT (lbs.)	PACKING DENSITY (lbs./cubic feet)	WASTE VOLUME (cubic feet)
Bulk Concrete	A	307,164,310	84.00	3,642,803
Soils	A	98,865,000	89	1,112,358
Metal Debris	A	67,426,312	47-66	1,020,230
Large Components	A	18,200,000	68 – 388	69,700
HazMat (containerized)	A	1,450,000	59	24,700
Highly Radioactive	B or C	305,000	80	3,800
Very Highly Radioactive	>C	71,600	112	640
Clean Concrete (on-site fill)	-	345,900,000	71	4,870,000
Clean Asbestos	-	1,008,000	8.3	121,400
Clean Debris (local landfill)	-	28,000,000	100	280,000
Clean Scrap Metal (recycler)	-	44,570,000	129	346,000
Totals	-	**912,960,222**		**11,491,631**

Table 3-4 General Project Milestones

Date	Milestone
Q3/2014	Unit 2 Reactor Internals Segmentation Complete
Q4/2014	Unit 1 Reactor Internals Segmentation Complete
Q4/2014	License Termination Plan Submittal to NRC
Q1/2015	Complete Transfer of Spent Nuclear Fuel to ISFSI
Q1/2015	Cold and Dark Complete (Electrical)
Q2/2015	Complete Demolition of Crib House
Q2/2015	Unit 2 Reactor Vessel Segmentation Complete
Q3/2015	Complete Demolition of Service Building (East/West)
Q4/2015	Complete Demolition of Turbine Building
Q4/2015	Unit 1 Reactor Vessel Segmentation Complete
Q4/2016	Complete Interior Dismantlement of Auxiliary Building
Q1/2017	Complete Interior Dismantlement of Unit 2 Containment
Q4/2017	Complete Interior Dismantlement of Unit 1 Containment
Q4/2018	Complete All Major Demolition
Q4/2018	Complete FSS and Site Restoration
Q4/2018	Complete Zion Station Restoration Project

Note; Circumstances can change during decommissioning. If Zion*Solutions* determines that the decommissioning cannot be completed as outlined in this schedule, Zion*Solutions* will provide an updated schedule to the NRC.

ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 4, REVISION 2
REMEDIATION PLAN

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LIST OF ACRONYMS AND ABBREVIATIONS

AF	Area Factor
ALARA	As Low As Reasonably Achievable
AMCG	Average Member of the Critical Group
BFM	Basement Fill Model
CFR	Code of Federal Regulations
CVS	Contamination Verification Survey
DCGL	Derived Concentration Guideline Levels
DSAR	Defueled Safety Analysis Report
EMC	Elevated Measurement Comparison
FSS	Final Status Survey
HEPA	High Efficiency Particulate Air
ISOCS	*In Situ* Object Counting System
LLRW	Low Level Radioactive Waste
LSA	Limited Specific Activity
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
NRC	Nuclear Regulatory Commission
ODCM	Off Site Dose Calculation Manual
ROC	Radionuclides of Concern
RPT	Radiation Protection Technician
SAFSTOR	SAFeSTORage
SFP	Spent Fuel Pool
TEDE	Total Effective Dose Equivalent
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

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4. SITE REMEDIATION PLAN

In accordance with 10 CFR 50.82(a)(9)(ii)(C), the License Termination Plan (LTP) must provide the "plans for site remediation." These plans must include the provisions to meet the criteria from Subpart E of 10 CFR 20 before the site may be released for unrestricted use. The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 are: (1) the Total Effective Dose Equivalent (TEDE) from residual radioactivity that is distinguishable from background radiation must not be greater than 25 mrem/yr to the Average Member of the Critical Group (AMCG) and (2) residual radioactivity levels must be As-Low-As-Reasonably-Achievable (ALARA).

Decontamination and dismantlement activities will be conducted in accordance with established Radiation Protection, Safety and Waste Management programs which include approved written procedures. These programs and procedures are frequently audited for technical content and compliance. Revisions have, and will continue to be made to these programs and procedures to accommodate the changing work environment inherent to reactor decommissioning and, documented, processed and approved in accordance with existing administrative procedures using 10 CFR 50.59 and Regulatory Guide 1.187, *"Guidance for Implementation of 10 CFR 50.59 Changes, Tests and Experiments"* (Reference 4-1) as guidance. Consistent with Regulatory Guide 1.179, *"Standard Format and Contents for License Termination Plans for Nuclear Power Reactors"* (Reference 4-2), details regarding changes to the Radiation Protection Program to address remediation and decommissioning activities are not provided in this LTP, but periodic updates to the Zion Station *"Defueled Safety Analysis Report"* (DSAR) (Reference 4-3) will provide such details.

This chapter describes the methods that may be used to remediate contaminated systems, components and structures. The methods for demonstrating compliance with the ALARA criterion in 10 CFR 20.1402 is also described. Note that Chapter 6 provides the methods for demonstrating compliance with the 25 mrem/yr dose criterion. Also, note that Chapter 3 describes in detail the remaining site remediation and dismantlement activities and the order in which they will occur for each structure, system and/or component.

This chapter also provides a summary of the radiation protection methods and control procedures that will be employed during site dismantlement and remediation.

4.1. Remediation Actions and ALARA Evaluations

When dismantlement and decontamination actions are completed, residual radioactivity may remain on building surfaces and in site soils at concentrations that correspond to the maximum annual dose criterion of 25 mrem/yr. The remaining residual radioactivity must also satisfy the ALARA criterion, 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).

The ALARA evaluation calculates the concentration at which the averted collective radiation dose, converted into dollars, is equal to the costs of continued remediation (e.g., risk of transportation accidents converted into dollars, worker and public doses associated with the remediation action converted into dollars, and the actual costs to perform the remediation

activity). If this concentration is below the concentrations that correspond to the maximum annual dose criterion, then further reduction of residual radioactivity is justified by ALARA.

Regardless of the outcome of the quantified cost/benefit calculation provided in this chapter, the final dose from residual radioactivity is expected to be well below the dose criterion. The majority of the basement surfaces to be backfilled have minimal contamination. In addition, any areas that are identified as potentially containing activity at levels that could exceeding the Derived Concentration Guideline Level (DCGL), as measured during Final Status Survey (FSS) by the *In Situ* Object Counting System (ISOCS), will be remediated. Industry standard remediation methods have been shown to remove contamination to levels significantly below the target levels, in this case the DCGL, and this result is expected for any remediation. The combination of low contamination levels over the majority of the basement surfaces combined with remediated areas likely containing activity well below the DCGL, ensures that the final dose from residual radioactivity at license termination will be well below the 25 mrem/yr dose criterion. Based on characterization results, there is limited contamination expected in soil, buried pipe or end-state structures with a corresponding dose that is also expected to be well below 25 mrem/yr.

4.2. Remediation Actions

Remediation actions are performed throughout the decommissioning process and the techniques, methods and technologies are standard to the commercial nuclear industry. All of the remediation actions described may not necessarily be required, but are listed as possible actions that may be taken during the decommissioning of Zion Nuclear Power Station (ZNPS). The appropriate remediation technique(s), method(s) and/or technologies that will be employed is dependent on the physical composition and configuration of the contaminated media requiring remediation. At ZNPS, the principal media that will be subjected to remediation are concrete structural surfaces. Characterization survey results and historical survey data indicate that there is minimal soil contamination and no groundwater contamination identified to date.

4.2.1. Structures

The general approach to structure remediation at Zion Station Restoration Project (ZSRP) is driven by section 8.5 of Exhibit C, Lease Agreement, "Removal of Improvements; Site Restoration" integral to the *"Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement"* (Reference 4-4) which requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least three feet below grade. Consequently, the only structures that will remain at license termination are the concrete walls and floors below 588 foot elevation in the Unit 1 Containment Building, Unit 2 Containment Building, Auxiliary Building, Turbine Building, Fuel Handling Building, Crib House/Forebay, Waste Water Treatment Facility (WWTF), Circulating Water Intake Piping and Circulating Water Discharge Tunnels. All impacted systems, components and structures above the 588 foot elevation will be removed during the decommissioning process and disposed of as a waste stream. The current decommissioning approach for ZSRP also calls for the beneficial reuse of clean concrete from building demolition as clean fill. The only concrete structures that will be considered are those where the probability of the presence of residual contamination is minimal and surveys demonstrate that the concrete is free of plant derived radionuclides and hazardous paint coatings.

The remaining structural surfaces that will remain at ZNPS following the termination of the license are solid concrete structures which will be covered by at least three 3 feet of soil and physically altered to a condition which would not allow the remaining structural surfaces, if excavated, to be realistically occupied.

Scan measurements, static measurements and/or the analysis of volumetric sample(s) will be used to calculate the remaining total concentration of residual activity. The concrete walls and floors of the basements will be remediated to levels that will provide high confidence that FSS measurements with ISOCS will not exceed radionuclide-specific DCGLs that represent the annual dose criterion for unrestricted use specified in 10 CFR 20.1402.

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. Cost estimates for these techniques also include the amount of water generated and the cost to process, package and ship this waste. Concrete removal may include using machines with hydraulic-assisted, remote-operated, articulating tools. These machines have the ability to exchange scabbling, shear, chisel and other tool heads.

4.2.1.1. Scabbling and Shaving

The principal remediation method expected to be used for removing contaminants from concrete surfaces is scabbling and shaving. Scabbling entails the removal of concrete from a surface by the high-velocity impact of a tool with the concrete surface which transforms the solid surface to a volumetric particulate which can be removed. One method of scabbling is a surface removal process that uses pneumatically operated air pistons with tungsten-carbide tips that fracture the concrete surface to a nominal depth of 0.125 inches at a nominal rate of about 130 ft^2 or 12.07 m^2 per hour. The scabbling pistons (feet) are contained in a close-capture enclosure that is connected by hoses to a sealed vacuum and collector system. Shaving uses a series of diamond cutting wheels on a spindle, and performs at similar rates to scabbling. The wheels are also contained in a close capture enclosure similar to scabbling equipment. The fractured media and dusts from both methods are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute High Efficiency Particulate Air (HEPA) filtration devices. Dust and debris generated through these remediation processes is collected and controlled during the operation.

4.2.1.2. Needle Guns

A second method of scabbling is accomplished using needle guns. The needle gun is a pneumatic air-operated tool containing a series of tungsten-carbide or hardened steel rods enclosed in housing. The rods are connected to an air-driven piston to abrade and fracture the media surface. The media removal depth is a function of the residence time of the rods over the surface. Typically, one to two millimeters are removed per pass. Generated debris collection, transport and dust control are accomplished in the same manner as other scabbling methods. Use of needle guns for removal and chipping of media is usually reserved for areas not accessible to normal scabbling operations. These include, but are not limited to, inside corners, cracks, joints and crevices. Needle gunning techniques can also be applied to painted and oxidized surfaces.

4.2.1.3. Chipping

Chipping includes the use of pneumatically operated chisels and similar tools coupled to vacuum-assisted collection devices. Chipping activities are usually reserved for cracks and crevices. This action is also a form of scabbling.

4.2.1.4. Sponge and Abrasive Blasting

Sponge and abrasive blasting are similar techniques that use media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide, and walnut hulls. Sponge blasting is less aggressive, incorporating a foam media that, upon impact and compression, absorbs contaminants. The medium is collected by vacuum and the contaminants are washed from the medium so the medium may be reused. Abrasive blasting is more aggressive than sponge blasting but less aggressive than scabbling. Both operations use intermediate air pressures. Sponge and abrasive blasting are intended for the removal of surface films and paints.

4.2.1.5. Pressure Washing

Pressure washing uses a nozzle of intermediate water pressure to direct a jet of pressurized water that removes superficial materials from the suspect surface. A header may be used to minimize over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing.

4.2.1.6. Washing and Wiping

Washing and wiping decontamination techniques are actions that are typically performed during the course of remediation activities for housekeeping and to minimize the spread of loose surface contamination. ZSRP will implement good housekeeping throughout decommissioning to ensure ALARA, to reduce the residual activity in structural surfaces to comply with the open air demolition criteria in Zion*Solutions* TSD 10-002, *"Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"* (Reference 4-5) and, to ensure that loose surface contamination is removed prior to evaluating the surface for acceptable concentrations of residual activity.

Washing and wiping techniques are actions that are normally performed during the course of remediation activities and will not always be evaluated as a separate ALARA action. When washing and wiping techniques are used as the sole means to reduce residual contamination below DCGL levels, ALARA evaluations will be performed. Washing and wiping techniques used as housekeeping or good practice measures will not be evaluated.

4.2.1.7. High-Pressure Water Blasting

Most contaminated piping will be removed and disposed of as radioactive waste. Any pipe systems or sections of pipe systems that reside below the 588 foot elevation that will be abandoned in place will be inspected and surveyed as described in Chapter 5. If radiological conditions inside the pipe are in excess of the release criteria, then *in situ* remediation will be performed. One method that may be used to remediate the pipe interior surfaces is high pressure water blasting. A High-Pressure Liquid-Jetting System has a high pressure water pump capable of producing a water pressure of 10,000 psi to 20,000 psi at an actual flow rate that ranges from

44 gallons per minute at 10,000 psi to 23 gallons per minute at 20,000 psi. A rotating jet-mole tip is used for 360 degree coverage of pipe interiors. The jet-mole is attached to a lance and high-pressure hose. The lance is manually advanced through the interior of the pipe. As the lance is advanced, the high-pressure water abrades the interior surface of the pipe, removing the corrosive layer, internal debris and radiological contamination. The waste water containing the removed contamination is then collected and stored for processing as liquid radiological waste.

4.2.1.8. Grit Blasting

Another approach that may be used to remediate the surfaces of pipe interior surfaces is grit blasting. Grit blasting uses grit media such as garnet or sand under intermediate air pressure directed through a nozzle that is pulled through the closed piping at a fixed rate. The grit blasting action removes the interior surface layer of the piping. A HEPA vacuum system maintains the sections being cleaned under negative pressure and collects the media for reuse or disposal. The final system pass is performed with clean grit to remove any residual contamination.

4.2.1.9. Removal of Activated/Contaminated Concrete

As previously stated, the principal means of remediating concrete surfaces is scabbling/shaving. If the concrete structure is designated for complete removal, such as interior concrete walls or the Bio-Shield, the primary method that will be used to completely remove the concrete is through large scale demolition using hydraulic-operated crushing shears and jack-hammers fitted to large tracked excavators. Concrete structures will be fractured and crushed by these tools. As the concrete is reduced to rubble, the embedded rebar will be exposed and segregated from the concrete rubble. In situations where a more surgical removal is required, activated and/or contaminated concrete removal may be accomplished using a machine mounted, remote-operated articulating arm with interchangeable tooling heads. As concrete is fractured and rebar exposed, the metal is cut using flame cutting equipment. The concrete rubble and exposed rebar is collected and transferred into containers for later disposal in both techniques. Dusts, fumes and generated debris are locally collected and as necessary, controlled using temporary enclosures coupled with close-capture HEPA systems or controlled water misting systems. Bulk concrete such as floors and walls may be removed as intact sections after sawing with blades, wires or other cutting methods.

4.2.1.10. Additional Remedial Actions

Mechanical abrasive equipment, such as hones, may be used to remove contamination from the surfaces of embedded/buried piping. Chemical removal means may be used, as appropriate, for the removal of certain contaminants.

4.2.2. Soil

The surface and subsurface soil $DCGL_W$ that will be used to demonstrate compliance with the dose-based criteria of 10 CFR 20, Subpart E for the unrestricted release of open land survey units are provided in Tables 5-5 and 5-6 of Chapter 5. Section 2.5.1.1 of NUREG-1575, *"Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM)*"* (Reference 4-6) addresses the concern for the presence of small areas of elevated radioactivity. A simple

comparison to an investigation level is used to assess the dose impact of potential elevated areas. This is referred to as the Elevated Measurement Comparison (EMC). The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_{w}$ modified by an Area Factor (AF) to account for the small area of the elevated radioactivity. Any radiological contamination in soils identified in concentrations greater than the $DCGL_{EMC}$ will be removed and disposed of as radioactive waste.

The site characterization process has established the location and extent of soil contamination at ZNPS. Characterization survey results and historical survey data indicate that there is minimal residual radioactivity in soil and no groundwater contamination identified to date. As needed, additional investigations will be performed to ensure that any changing soil radiological contamination profile during the remediation actions is adequately identified and addressed. Chapter 5 discusses soil sampling and survey methods.

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. This simple methodology minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket.

Remediation of soils will be performed using established excavation safety and environmental control procedures. Operational constraints and dust control will be addressed in site excavation and soil control procedures. In addition, work package instructions for remediation of soil may include additional constraints and mitigation or control methods to ensure adequate erosion, sediment, and air emission controls during soil remediation.

4.3. Remediation Activities Impact on the Radiation Protection Program

The Radiation Protection Program approved for decommissioning at ZSRP is similar to the regulatory approved program that was implemented during commercial power operation and the subsequent SAFSTOR period. During these periods, in a manner similar to remediation activities during decommissioning, contaminated structures, systems and components were decontaminated in order to perform maintenance or repair actions.

The current approved Radiation Protection Program at ZSRP is adequate to comply with all federal and state regulatory requirements for the protection of occupational personnel from radiological hazards encountered or expected to be encountered during the decommissioning of a two unit commercial reactor facility. In addition, the program ensures the protection of the public from radiological hazards and ensures occupational, effluent and environmental dose from exposure to radioactive materials is, and remains ALARA. To ensure that adequate and proper engineering controls and hazard mitigation techniques are employed, work control programs and procedural requirements allow radiation protection personnel to integrate radiation protection and radiological hazard mitigation measures directly into the work planning and scheduling process. Consequently, the necessary radiological controls are correctly implemented to accommodate each remediation technology as appropriate.

The spread of loose surface contamination is mitigated by the routine remediation of work areas by washing and wiping. Water washing with a detergent is effective in reducing low levels of

loose surface contamination over large surface areas. Wiping with detergent soaked or oil-impregnated media is an effective technique to reduce loose surface contamination on small items, overhead spaces and small hand tools. These same techniques are also effective in reducing low levels of surface contamination on structural surfaces.

For intermediate levels of surface contamination, more aggressive methods such as pressure washing, high-pressure water blasting and grit blasting may be more appropriate. Pipes, surfaces and drain lines can be cleaned and hot spots removed using these techniques and technologies. Small tools, hoses and cables can also be pressure washed in a containment to reduce contamination levels. A paint coating may be applied after surface cleaning to prevent surface contamination from drying out and becoming airborne.

To mitigate high levels of fixed surface contamination embedded in concrete, scabbling or other surface removal techniques may be appropriate. A combination of mechanical and flame cutting will be used to section the reactor vessel and its internals.

The Radiation Protection Program approved for decommissioning is similar to the program in place during commercial power operation. 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 ZSRP for the decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas. Concrete cutting or surface scabbling, mechanical cutting, abrasive water jet cutting, hydrolazing and grit blasting has been used at ZNPS in the past during operations. The current Radiation Protection Program provides adequate controls for these actions.

Decommissioning does not present any new challenge to the Radiation Protection Program above those encountered during normal plant operation and refueling. Decommissioning planning allows radiation protection personnel to focus on each area of the site and plan each activity well before execution of the remediation technique.

The decommissioning organization is experienced in and capable of applying these remediation techniques on contaminated systems, structures or components during decommissioning. The Radiation Protection Program is adequate to safely control the radiological aspects of this work. Because the activities expected during decommissioning are the same or similar to those encountered during operations, as described above, the approval of any changes to the existing approved Radiation Protection Program as described in the Nuclear Regulatory Commission (NRC) Docket Number 50-295, *"Facility Operating License Number DPR-39 (for Unit One)"* (Reference 4-7), NRC Docket Number 50-304, *"Facility Operating License Number DPR-48 (for Unit Two)"* (Reference 4-8) is not requested in this LTP.

4.4. ALARA Evaluation

Guidance for conducting ALARA analyses is provided in Appendix N of NUREG-1757, Volume 2, Revision 1, *"Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report"* (Reference 4-9), which describes acceptable methods for determining when further reduction of residual radioactivity is required to concentrations below the levels necessary to satisfy the 25 mrem/yr dose criterion.

The surface and subsurface soil $DCGL_W$ that will be used to demonstrate compliance with the 25 mrem/yr dose criterion are provided in Tables 5-4 and 5-5 of Chapter 5. Characterization survey results and historical survey data indicate that there is minimal residual radioactivity in soil at ZNPS. Throughout the course of the decommissioning and through to site closure, ZSRP will continue to survey and characterize soils as they are exposed by excavation during building demolition or made accessible by the removal of structures or components. If residual radioactivity is discovered at concentrations greater than the $DCGL_{EMC}$ in surface or subsurface soils, ZSRP will excavate, package and dispose of the soil as Low-Level Radioactive Waste (LLRW).

Section N.1.5 of NUREG-1757 states that *"For residual radioactivity in soil at sites that may have unrestricted release, generic analyses show that shipping soil to a low-level waste disposal facility is unlikely to be cost effective for unrestricted release, largely because of the high costs of waste disposal. Therefore shipping soil to a low-level waste disposal facility generally does not have to be evaluated for unrestricted release."* To illustrate that this is a reasonable approach and applicable to ZSRP, a simple ALARA analysis for the excavation and disposal of soils as low-level radioactive waste is provided in section 4.4.1.

For the subsurface structures that will remain at license termination, the ALARA analysis will determine whether further concrete remediation is necessary by comparing the desired beneficial effects to the undesired costs. Benefits are the averted collective radiation dose (converted into dollars) following the removal of radioactivity. The costs of remediation include transportation accidents, worker and public dose associated with remedial action, and the actual costs to perform the remediation (converted into dollars). If the costs exceed the benefits, then the dose reduction achieved by further remediation is not ALARA.

The ALARA criterion specified in 10 CFR 20.1402 is not met by solely performing remediation. The ALARA analysis is a planning tool to justify that further remediation is not necessary. When remediation is performed, there is no need to analyze whether the action was necessary to meet the ALARA requirement. The remediation required to meet the open air demolition criteria specified in TSD 10-002, including cleaning loose surface contamination to concentrations below 1,000 dpm/100cm^2 and the remediation of concrete surfaces to meet the 2 mR/h exposure rate criteria, will be performed regardless of the outcome of the ALARA evaluation. Consequently, this is an example of when a remediation action is not required to be evaluated for ALARA.

The methods and results of the ALARA evaluation for concrete remediation in structures below 588 foot elevation is provided in section 4.4.2.

4.4.1. ALARA Analysis of Soil Remediation

In order to determine if additional remedial action is warranted by ALARA analysis, the desired beneficial effects (benefits) and the undesirable effects (costs) must be calculated. If the benefits from remedial action will be greater than the costs, then the remedial action is warranted and should be performed. However, if the costs exceed the benefit, then the remedial action is considered to be not ALARA and should not be performed.

Based upon a simple ALARA analysis, the only benefit of reducing residual radioactivity in soil is the monetary value of the collective averted dose to future occupants of the site. For soils, the averted dose is based upon the "resident farmer" scenario.

4.4.1.1. Calculation of Benefits

The benefit from collective averted dose (B_{AD}) is calculated by determining the present worth of future collective averted dose and multiplying by a factor to convert the dose to a monetary value. In accordance with Appendix N of NUREG-1757, the equation is as follows;

Equation 4-1

$$B_{AD} = \$2{,}000 \times PW(AD_{Collective})$$

where;

B_{AD}	=	benefit from an averted dose for a remediation action, in US dollars,
$2,000	=	value in dollars of a person-rem averted and,
$PW(AD_{Collective})$	=	present worth of a future collective averted dose.

The present worth of future collective averted dose $PW(AD_{Collective})$ is then expressed in accordance with the following equation;

Equation 4-2

$$PW(AD_{Collective}) = (P_D)(A)(0.025)(F)\left(\frac{Conc}{DCGL_w}\right)\left(\frac{1 - e^{-(r+\lambda)N}}{r+\lambda}\right)$$

where;

P_D	=	population density for the critical group scenario in people/m^2,
A	=	area being evaluated in square meters (m^2),
0.025	=	annual dose to an AMCG from residual radioactivity at the $DCGL_w$ concentration in rem/yr,
F	=	effectiveness, or fraction of the residual radioactivity removed by the remediation action,
$Conc$	=	average concentration of residual radioactivity in the area being evaluated in units of activity per unit volume (pCi/g),
$DCGL_w$	=	derived concentration equivalent to the average concentration of residual radioactivity that would give a dose of 25 mrem/yr to the AMCG (pCi/g),
r	=	monetary discount rate in units per year (yr^{-1}),
λ	=	radiological decay constant for the radionuclide in units per year and,

N = number of years over which the collective dose will be calculated.

4.4.1.2. ALARA Analysis Parameters

In accordance with Table N.2 of Appendix N of NUREG-1757, the acceptable and relevant parameters for use in performing ALARA analysis are as follows;

- Dollars per person-rem - \$2,000.00/person-rem (per NUREG/BR-0058, *"Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission"* [Reference 4-10])
- Population density (P_D) for the critical group (persons/m^2) - 0.0004 person/m^2 for land (per NUREG-1496, *"Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities,"* Volume 2, [Reference 4-11] Appendix B, Table A.1)
- Monetary discount rate (r) - 0.00 yr^{-1} for soil

  (Note: This variable was established at 0.03 yr^{-1} for soil in Table N.2 of Appendix N of NUREG-1757. The monetary discount for the ALARA analysis was removed from the equation through Federal Register Notice 72 FR 46102 – August 16, 2007. Consequently, the r variable has been conservatively set at 0.00 yr^{-1} for soil, i.e., no monetary discount for soils as well as basements.)
- Area (A) used to calculate the population density (m^2) – 10,000 m^2 (size of reference area that was evaluated)
- Number of years (N) over which the collective averted dose is calculated (yr) - 1,000 yrs (per NUREG-1496, Volume 2, Appendix B, Table A.1)

4.4.1.3. Calculation of Costs

The total cost, ($Cost_T$) which is balanced against the benefits; has several components and may be evaluated according to Equation N-3 of NUREG-1757, Appendix N below:

Equation 4-3

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose}$$

where:

$Cost_R$ = monetary cost of the remediation action (including mobilization costs);

$Cost_{WD}$ = monetary cost for transport and disposal of the waste generated by the action;

$Cost_{ACC}$ = monetary cost of worker accidents during the remediation action;

$Cost_{TF}$ = monetary cost of traffic fatalities during transportation of the waste;

$Cost_{WDose}$ = monetary cost of traffic fatalities during transportation of the waste;

$Cost_{PDose}$ = monetary cost of dose to the public from excavation, transport and disposal of the waste;

4.4.1.4. Calculation of Total Cost for Soil Remediation by Excavation and Disposal

For the analysis of soil excavation and disposal as low-level radioactive waste, the variables for $Cost_R$, $Cost_{ACC}$, $Cost_{WDose}$ *and* $Cost_{PDose}$ were not calculated for this evaluation based upon their anticipated unlikely impact on the total cost ($Cost_T$). This is consistent with the guidance provided in NUREG-1757 which states that if one or two of the costs can be shown to exceed the benefit, then the remediation cost is shown to be unnecessary without calculating all of the costs.

4.4.1.4.1. Transport and Disposal of the Waste ($Cost_{WD}$)

The cost of waste transport and disposal ($Cost_{WD}$) was calculated using Equation N-4 of NUREG-1757, Appendix N which is expressed as follows:

Equation 4-4

$$Cost_{WD} = V_A \times Cost_V$$

where:

V_A = volume of waste produced, remediated in units of m^3;

$Cost_V$ = cost of waste disposal per unit volume, including transportation cost, in units of $\$/m^3$.

Disposal costs for generated waste were based on an average total disposal cost of $2,500/m³. This average cost includes packaging, transportation and disposal fees. The transportation component of this average cost is based on the average transportation cost of using either rail or highway hauling from the Zion site to Clive, Utah (Energy*Solutions* radioactive waste disposal facility). The details of the average total disposal cost ($Cost_V$) of $\$2,500/m^3$ of waste are considered proprietary values defined by negotiated contract.

The volume of waste produced by remediation (V_A) assumes that the reference area of 10,000 m^2 (A) is remediated to a depth of 0.15 meters. This results in a value for waste volume (V_A) of 1,500 m^3, which produces a value for $Cost_{WD}$ of $3,750,000.00.

4.4.1.4.2. Transportation Risks ($Cost_{TF}$)

The cost of traffic fatalities incurred during the transportation of waste ($Cost_{TF}$) was calculated using Equation N-6 of NUREG-1757, Appendix N which is expressed as follows:

Equation 4-5

$$Cost_{TF} = \$3,000,000 \times \frac{V_A}{V_{SHIP}} \times F_T \times D_T$$

where:

\$3,000,000	=	monetary value of a fatality equivalent to \$2000/person-rem (NUREG-1530 *"Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy"* [Reference 4-12])
V_A	=	volume of waste produced in units of m^3;
V_{SHIP}	=	volume of a truck shipment in m^3;
F_T	=	fatality rate per truck-kilometer traveled in units of fatalities/truck-km;
D_T	=	distance traveled in km.

For this evaluation, the waste volume (V_A) is assumed to be 1,500 m^3 and the haul volume of an overland truck shipment per NUREG-1757 is assumed to be 13.6 m^3 (V_{SHIP}).

In accordance with NUREG-1496, Appendix B, Table A.1, a value of 3.8 E-08/hr was used for F_T.

The Clive, Utah round trip distance from the Zion site by highway is 1,463 miles (2,355 km). The distance for rail shipments is further than that for highway shipments because of the route rail shipments must follow, however the difference as it pertains to the calculation is insignificant. The highway shipment distance of 2,355 km (D_T) was used for the calculation of $Cost_{TF}$. For this evaluation, the value for the $Cost_{TF}$ variable is \$29,610.66.

4.4.1.4.3. Total Cost ($Cost_T$)

The total cost, ($Cost_T$) assumed for this evaluation is \$3,779,610.66.

4.4.1.5. Residual Radioactivity in Soils that are ALARA

Determination of residual radioactivity in soils that are ALARA is the concentration at which benefit equals or exceeds the costs of removal and waste disposal. When the total cost ($Cost_T$) is set equal to the dose averted, the ratio of the concentration to the $DCGL_w$ is calculated as follows;

Equation 4-6

$$\frac{Conc}{DCGL_w} = \frac{(Cost_T)(r+\lambda)}{(\$2{,}000)(P_D)(0.025)(F)(A)(1-e^{-(r+\lambda)N})}$$

Assuming the following values for the remaining variables;

- the default parameter values from section 4.4.1.2,
- a value of one for remediation effectiveness (F), assuming all residual radioactivity is removed during the excavation,
- a surface soil $DCGL_w$ of 14.18 pCi/g for Cs-137 from Table 5-5 of Chapter 5,

Equation 4-7

$$\frac{Conc}{DCGL_w} = \frac{(\$3{,}779{,}610.66)\left(0.00 + \frac{0.693}{30.17}\right)}{(\$2{,}000)(0.0004)(0.025)(1)(10{,}000)\left(1 - e^{-\left(0.00+\frac{0.693}{30.17}\right)1{,}000}\right)}$$

the ratio of the concentration to the $DCGL_w$ when the total cost ($Cost_T$) is equal to the dose averted is 434.08.

Assuming a concentration set at 50% of the $DCGL_w$ (based on the investigation level for a Class 3 area), the present worth of future collective averted dose $PW(AD_{Collective})$ can be calculated as follows;

Equation 4-8

$$PW(AD_{Collective}) = (0.0004)(10{,}000)(0.025)(1)\left(\frac{7.09}{14.2}\right)\left(\frac{1 - e^{-\left(0.00+\frac{0.693}{30.17}\right)(1{,}000)}}{0.00 + \frac{0.693}{30.17}}\right)$$

resulting in a value for $PW(AD_{Collective})$ of 2.18 person rems. The benefit from collective averted dose (B_{AD}) is then calculated as follows;

Equation 4-9

$$B_{AD} = \$2{,}000 \times 2.18 = \$4{,}353.54$$

This simple analysis confirms the statement in section N.1.5 of NUREG-1757 that the cost of disposing excavated soil as low-level radioactive waste is clearly greater than the benefit of removing and disposing of soil with residual radioactivity concentrations less than the dose criterion. Since the cost is greater than the benefit, it is not ALARA to excavate and dispose of soils with residual radioactivity concentrations below the $DCGL_w$.

4.4.2. ALARA Analysis for Remediation of Basement Structures

With the exception of some penetrations, embedded and buried piping, all contaminated and non-contaminated systems will be disassembled, removed, packaged and shipped off-site as a waste stream commodity. The list of penetrations and embedded piping to remain is provided in Zion*Solutions* TSD 14-016, 14-016, "*Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State*" (Reference 4-13). Once commodity removal is complete, structural surfaces will be remediated as necessary to meet the open air demolition criteria specified in TSD 10-002. These criteria provide the removable contamination levels and contact exposure rates that will allow structures to be safely demolished without containment. Prior to demolition, a contamination verification survey (CVS) will be performed to identify areas requiring remediation to meet the open-air demolition limits. The CVS will also be used to identify areas on surfaces to remain at license termination (i.e., at least three feet below grade) that could potentially result in a FSS measurement (using ISOCS) to exceed the Basement DCGLs ($DCGL_B$) listed in LTP Chapter 5, Table 5-4. The dose rate target for this objective will be lower than that required for open-air demolition. Identified areas will be remediated to provide high confidence that no FSS ISOCS measurement will exceed the $DCGL_B$. Once

remediation is complete structural surfaces located above the 588 foot elevation and non-load-bearing interior concrete walls below the 588 foot elevation will be demolished, reduced in size, packaged and shipped off-site to a licensed disposal facility.

All concrete inside the liner above the 565 foot elevation will be removed from the interiors of both Containment Buildings prior to demolition. This includes all activated and contaminated concrete. Only the concrete below the 565 foot elevation in the In-core Instrument Shaft leading to and including the area under vessel (or Under-Vessel area) will remain. The source term in the Containment Basements remaining after demolition will consist of the concrete in the Under-Vessel area(s) and low levels of surface contamination on the exposed liner surfaces. There is currently minimal contamination in the Turbine Building, Crib House/Forebay, and Circulating Water Piping at levels that are expected to be well below the open air demolition criteria and below the $DCGL_B$ listed in LTP Chapter 5, Table 5-4. The only portion of the Fuel Handling Building Basement that will remain following building demolition is the lower 13 foot (~4 m) concrete bottom of the Spent Fuel Pool (SFP) and the Transfer Canal, which is located at the 575 foot elevation. The steel liner will be removed from both the SFP and the Transfer Canal. After the liner is removed and the underlying concrete exposed, continuing characterization surveys will be performed. Continuing characterization will consist of scanning of the exposed concrete surfaces and the acquisition of concrete core sample(s) at the location of highest activity. Contamination is expected below the liner but an estimate of levels cannot be made until characterization is completed.

In summary, the vast majority of residual radioactivity remaining in the structures after the open air demolition criteria is met and after the majority of concrete is removed from the Containment Building basements will be located in the 542 foot elevation floor of the Auxiliary Building. Therefore, the ALARA assessment for the remediation of basement structures will focus on the 542 foot elevation floor of the Auxiliary Building as this is the location were the greatest benefit of concrete remediation could be achieved. An ALARA assessment of the 542 foot elevation floor of the Auxiliary Building will bound ALARA assessments for the other buildings which would use the same methods (and cost estimate) but remove less contamination. If continuing characterization indicates significant concentrations of residual radioactivity remaining in other end-state structures (e.g., Under-Vessel area, SFP/Transfer Canal, Auxiliary Building embedded drains), then ZSRP will perform and document a separate ALARA analysis or provide evidence that the ALARA analysis of the 542 foot floor of the Auxiliary Building is still bounding.

The Auxiliary Building basement concrete at the 542 foot elevation is volumetrically contaminated. A total of twenty (20) concrete core samples were collected in the Auxiliary Building during characterization. The sample analysis of these concrete core samples indicates that the majority of the radionuclide inventory resides within the first ½-inch of concrete. However, several core samples show detectable Cs-137 and Co-60 at depths in excess of six inches.

4.4.2.1. ALARA Analysis Equation for Remediation of Basement Structures

For the ALARA analysis for the remediation of basement structures, the equation from section 4.4.1.5 is modified as follows. The DCGLs for concrete are expressed in units of pCi/m^2. The denominator must be summed and the individual dose contribution normalized to account for the

multiple detectable radionuclides that are present in the radionuclide distribution for the Auxiliary Building. The equation from NUREG-1757 therefore becomes:

Equation 4-10

$$\frac{Conc}{DCGL_w} = \frac{(Cost_T)(r + \lambda_i)}{\sum(\$2{,}000)\,(P_D)(f_i)(DOSE_{AMCG})(F)(A)(1 - e^{-(r+\lambda_i)N})}$$

where:

f_i	=	the normalized radionuclide fraction for the Auxiliary Building for each individual Radionuclides of Concern (ROC) (from Chapter 5, Table 5-2)
$DOSE_{AMCG}$	=	averted dose to the AMCG (rem).

The total cost for the remedial action when divided by the total benefit of averted dose determines the cost effectiveness of the remedial action. Values greater than unity demonstrate that no further remediation is necessary beyond that required to meet the 25 mrem/yr dose criterion and are ALARA. Values less than one provide the fraction of the 25 mrem/yr dose criterion where it is necessary to remediate to achieve ALARA.

4.4.2.2. Remedial Action Costs

The only structures that will remain as potential candidate surfaces for remediation are the concrete walls and floors from the Auxiliary Building, the Under-Vessel area(s), Turbine Building, Crib House/Forebay, WWTF, the lower 13 foot concrete bottom of the SFP, the Circulating Water Intake Piping and Circulating Water Discharge Tunnels. With the exception of some sections of buried and embedded pipe, all impacted systems, components as well as all structures above the 588 foot elevation will be removed during the decommissioning process and disposed of as a waste stream. The current decommissioning approach for ZSRP also calls for the beneficial reuse of concrete from building demolition as clean fill. As discussed above, the vast majority of contamination to remain after removal of containment concrete will be in the 542 foot elevation floor of the Auxiliary Building.

Prior to building demolition, all structures will be remediated to meet the open air demolition limits specified in TSD 10-002 and, to provide high confidence that ISOCS measurements taken during FSS will not exceed the $DCGL_B$ from Table 5-4. All loose surface contamination greater than 1,000 dpm/100cm^2 will be removed. The remediation techniques most likely to be implemented to perform this work are vacuuming, pressure washing and hand-wiping, concrete scabbling or concrete shaving. As these efforts will occur prior to evaluating the remaining structural surfaces for acceptable concentrations of residual activity, this remediation action will not be evaluated for ALARA.

The remediation action evaluated for the ALARA analysis for the remediation of basement structures is scabbling the concrete surface of the 542 foot elevation floor of the Auxiliary Building. Concrete core samples indicate that the majority of the radionuclide source inventory in the 542 foot elevation concrete floor resides within the first ½-inch of concrete. For the purposes of the ALARA evaluation, it is conservatively assumed that 100% of the contamination resides in the first ½ inch. In accordance with the guidance in section G.3.1 of NUREG/CR-

5884, *"Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station"*. Volume 2 (Reference 4-14), one pass of scabbling is assumed to remove 0.125 inches (0.635 cm) of concrete. In accordance with Zion*Solutions* TSD 14-013, *"Zion Auxiliary Building End State Estimated Concrete Volumes, Surface Areas, and Source Terms"* (Reference 4-15), the 542 foot elevation floor of the Auxiliary Building has a surface area of 2,543 m^2. This is the surface area which will be evaluated for the remediation cost determination.

4.4.2.2.1. Remediation Activity Rates

The remediation activity rates that were used for this evaluation were based on previous experience, from published literature, or from groups or vendors currently performing these or similar activities. Current project labor costs and past operational experience were also used in developing these rates.

In accordance with NUREG/CR-5884, an assumed crew size for performing concrete scabbling or shaving activities is three full-time laborers, a supervisor at a ¼-time involvement and a Radiation Protection Technician (RPT), also at a ¼-time involvement. Using the current project labor rates for these positions of $66.78 per hour for a laborer, $90.00 per hour for a supervisor and $55.59 per hour for a RPT, the hourly unit rate that will be used for the evaluation is $236.74.

Using the guidance found in NUREG/CR-5884 it is assumed that the concrete scabbling or shaving activity will remove approximately 0.125 inches of concrete per pass and the effective nominal removal rate is approximately 12.07 m^2 per hour. The ALARA evaluation assumes that 100% of the radioactive contamination resides within the first ½ inch. Consequently, removing ½ inch of concrete over an assumed reference area of 2,543 m^2, scabbling at a nominal rate of 12.07 m^2 per hour to a depth of 0.125 inch per pass, equates to approximately 836.5 man-hours of work.

Also in accordance with NUREG/CR-5884 it is assumed that the actual remediation time in a typical eight-hour shift is 5.33 hours. To account for non-remediation work hours for work preparation, donning and removing protective clothing and work breaks, the total man-hours were increased by a factor of 33% which equates to 1,112 man-hours. In addition, a contingency of 25% was added to the manpower hours. This equates to a total of 1,390.61 man-hours, which is multiplied times the hourly unit rate of $236.74 to equal the labor cost for this evaluation of $329,209.54.

4.4.2.2.2. Equipment Costs

Using the guidance found in NUREG/CR-5884, equipment costs are based on the rental of commercially available scabbing equipment, a compressor, a vacuum unit and consumables such as cutting bits, vacuum filters and waste drums for containing waste debris. At 40-hours per work week, 1,391 man-hours equates to approximately 35 work-weeks. This evaluation assumes that two different commercially available concrete removal units will be procured, the Pentek Squirrel Scabbler & Vacuum System with a nominal rental rate of $685.00 per week and a Pentek Moose Scabbler & Vacuum System with a nominal rental rate of $950.00 per week. The compressor required for pneumatic equipment operation can be rented at a nominal rate of $115.00 per week. The cutting bits for the units are assumed to be replaced every 80 hours of

operation, for an equivalent cost of about $13.00 per hour of operation. Additional costs include filter replacements at about $2.50 per hour of operation and waste drums for the collected debris. A 55-gallon drum holds approximately 7 ft^3 of waste and cost approximately $100.00 per drum. As it is assumed that the scabbing activity will generate approximately 1,132 ft^3 (32 m^3) of concrete waste, this will require the procurement of approximately 162 drums at a total cost of $16,171.49. The mobilization and demobilization costs associated with procuring this equipment would be approximately $2,200.00 per piece of equipment for a total of approximately $6,600.00. The total equipment costs assumed for this evaluation is approximately $98,975.87.

4.4.2.2.3. Total Remediation Action Cost ($Cost_R$)

For the evaluation of the remediation activity of concrete scabbling or shaving, the sum of the labor cost of $329,209.54 plus the equipment cost of $98,975.87 results in a total remediation action cost ($Cost_R$) for this activity of $428,185.41.

4.4.2.3. <u>Transport and Disposal of the Waste ($Cost_{WD}$)</u>

As previously described in section 4.4.1.4.1, the cost of waste transport and disposal ($Cost_{WD}$) is expressed as follows:

Equation 4-11

$$Cost_{WD} = V_A \times Cost_V$$

Disposal costs for generated waste were based on an average total disposal cost of $2,500/$m^3$. This average cost includes packaging, transportation and disposal fees. The transportation component of this average cost is based on the average transportation cost of using either rail or highway hauling from the Zion site to Clive, Utah (Energy*Solutions* radioactive waste disposal facility). Based upon an assumed waste volume of 32 m^3, a value of $80,000.00 is calculated for the $Cost_{WD}$ variable.

4.4.2.4. <u>Non-Radiological Risks ($Cost_{ACC}$)</u>

The cost of non-radiological workplace accidents ($Cost_{ACC}$) was calculated using Equation N-5 of NUREG-1757, Appendix N which is expressed as follows:

Equation 4-12

$$Cost_{ACC} = \$3{,}000{,}000.00 \times F_W \times T_A$$

where:

$3,000,000	=	monetary value of a fatality equivalent to $2000/person-rem (NUREG-1530)
F_W	=	workplace fatality rate in fatalities/hour worked;
T_A	=	worker time required for remediation in units of worker-hours.

In accordance with NUREG-1496, Appendix B, Table A.1, a value of 4.2 E-08/hr was used for F_W. For T_A, in accordance with NUREG-1757 the same hours that was determined for labor cost

(1,391 man-hours) was used for worker accident cost. Subsequently, a value of \$175.27 is calculated for the $Cost_{ACC}$ variable.

4.4.2.5. Transportation Risks ($Cost_{TF}$)

As previously described in section 4.4.1.4.2, the cost of traffic fatalities incurred during the transportation of waste ($Cost_{TF}$) is expressed as follows:

Equation 4-13

$$Cost_{TF}=\$3,000,000.00 \times \frac{V_A}{V_{SHIP}} \times F_T \times D_T$$

For this evaluation, the waste volume (V_A) is assumed to be 32 m^3 and the haul volume of an overland truck shipment per NUREG-1757 is assumed to be 13.6 m^3 (V_{SHIP}).

In accordance with NUREG-1496, Volume 2, Appendix B, Table A.1, a value of 3.8 E-08/hr was used for F_T.

The Clive, Utah round trip distance from the Zion site by highway is 1,463 miles (2,355 km). The distance for rail shipments is further than that for highway shipments because of the route rail shipments must follow, however the difference as it pertains to the calculation is insignificant. The highway shipment distance of 2,355 km (D_T) was used for the calculation of $Cost_{TF}$. For this evaluation, the value for the $Cost_{TF}$ variable is \$631.69.

4.4.2.6. Worker Dose Estimates ($Cost_{WDose}$)

The cost of remediation worker dose ($Cost_{WDose}$) was calculated using Equation N-7 of NUREG-1757, Appendix N which is expressed as follows:

Equation 4-14

$$Cost_{WDose}=\$2,000.00 \times D_R \times T$$

where:

D_R	=	total effective dose equivalent (TEDE) rate to remediation workers in units of rem/hr;
T	=	time worked (site labor) to remediate the area in units of person-hour.

Costs associated with worker dose are a function of the hours worked and the workers' radiation exposure for the task. A value of 3 mrem per man-hour was used for D_R. This assumes that a majority of the source inventory will be removed prior to performing the concrete scabbling or shaving activity. The time worked to remediate the area in units of person-hour calculated for this activity (T) was 1,391 man-hours. For this evaluation, the value for the $Cost_{WDose}$ variable is \$8,346.00.

4.4.2.7. Monetary Cost of Dose to the Public ($Cost_{PDose}$)

The cost of remediation worker dose ($Cost_{PDose}$) was calculated using Equation N-7 of NUREG-1757, Appendix N which is expressed as follows:

Equation 4-15

$$Cost_{PDose}=\$2{,}000.00 \times D_R \times T$$

where:

D_R = total effective dose equivalent (TEDE) rate to public in units of rem/hr;

T = time spent near waste shipments in parking lots in units of person-hour.

For this equation, a "worst-case" value of 0.5 mrem/hr was used for D_R. This assumes that the shipment is classified as Limited Specific Activity (LSA) in accordance with 49 CFR 173.427 and the package meets the Zion specific administrative limit of 0.5 mrem/hr on the exterior of the shipment. The exposure time (T) used for this calculation is based upon a transit time of 23 hours driving from Zion to the disposal site in Clive Utah times three shipments, for a total of 69 hours. For this evaluation, the value for the $Cost_{PDose}$ variable is \$69.00.

4.4.2.8. Total Cost ($Cost_T$)

The total cost, ($Cost_T$) assumed for this evaluation is \$517,407.37

4.4.2.9. Residual Radioactivity in Basement Structures that are ALARA

The following parameters were used for performing the ALARA calculation using the equation from NUREG-1757 and presented in section 4.4.2.1:

- Population density (P_D) for the critical group (persons/m^2) - 0.0004 person/m^2 for soil (per NUREG-1496, Appendix B, Table A.1)
- Fraction of residual radioactivity removed by the remedial action (F) – 1 (Removal of desired concrete volume is assumed 100% effective)
- Area (A) used to calculate the population density (m^2)
  - Groundwater scenario –10,000 m^2 (size of resident farmer reference area)
  - Drilling Spoils scenario – 100 m^2 is assumed in order to allow the calculation to generate a population of 1 person exposed to drilling spoils. The actual surface area of the drilling spoils is much smaller at 0.46 m^2 (see LTP Chapter 6)
- Monetary discount rate (r) - 0.00 yr^{-1} for soil

  (Note; This variable was established at 0.03 yr^{-1} for soil in Table N.2 of Appendix N of NUREG-1757. The monetary discount for the ALARA analysis was removed from the equation through Federal Register Notice 72 FR 46102 – August 16, 2007. Consequently, the r variable has been conservatively set at 0.00 yr^{-1} for soil, i.e., no monetary discount for soils as well as basements.)
- Number of years (N) over which the collective averted dose is calculated (yr) - 1,000 yrs (per NUREG-1496, Appendix B, Table A.1)

4.4.2.9.1. Radionuclides Considered for ALARA Analysis

The radionuclide mixture for contaminated concrete developed in Zion*Solutions* TSD 14-019, *"Radionuclides of Concern for Soil and Basement Fill Model Source Terms"* (Reference 4-16) was used for the ALARA analysis. The $DCGL_B$ for the Auxiliary Building for each individual ROC from Chapter 5, Table 5-3 were used for the calculation of f_i for the Auxiliary Building ROC's identified in Table 6-5. DCGLs, in units of pCi/m^2 of basement surface area, are presented in Chapter 6, section 6.6.8.1 for the Basement Fill Model (BFM) Groundwater and BFM Drilling Spoils scenarios individually and are designated as the $DCGL_{BS}$ (Basement Scenario DCGLs). The $DCGL_{BS}$ for the Auxiliary Building are reproduced in Table 4-1. The values for half-life, radiological decay constants (λ) and the radionuclide mixture fractions are presented in Table 4-2. The mixture fractions are based on the analysis of the concrete core samples taken on the Auxiliary Building 542 foot elevation and presented in TSD 14-019, Table 17.

The ALARA calculation was performed in two parts, the first representing the Groundwater scenario and the second representing the Drilling Spoils scenario. Two dose values were required to accurately calculate the averted dose because the compliance dose is based on the sum of both scenarios. In addition, each scenario is applicable to a different area. The Groundwater dose applies to the full 10,000 m^2 site area, the Drilling Spoils dose applies only to the area of material brought to the surface by the well drilling action.

The actual dose from each scenario, assuming a summation of the dose from both scenarios equaled 25 mrem/yr is presented in Table 4-3. Therefore, the dose values for each ROC from Table 4-3 were used to derive the AMCG ($DOSE_{AMCG}$) variable in Equation 4-10 for each scenario.

Table 4-1 Basement $DCGL_{BS}$ for the Auxiliary Building

Radionuclide	Groundwater Scenario DCGL (pCi/m^2)	Drilling Spoils Scenario DCGL (pCi/m^2)
Co-60	3.28E+10	3.07E+08
Ni-63	1.15E+10	1.02E+14
Sr-90	9.98E+06	5.25E+10
Cs-134	3.55E+08	5.23E+08
Cs-137	1.25E+08	1.02E+09

Table 4-2 Radionuclide Half-Life(s), Decay Constant(s) and Mixture

Radionuclide [a]	Half-Life (yrs)	λ (yr^{-1})	Radionuclide Mixture[b]
Co-60	5.27 E 00	1.31 E-01	0.92%
Ni-63	9.60 E+01	7.22 E-03	23.71%
Sr-90	2.91 E+01	2.38 E-02	0.05%
Cs-134	2.06 E+00	3.36 E-01	0.01%
Cs-137	3.02 E+01	2.30 E-02	75.32%

(a) Dose significant ROC for the Auxiliary Building in accordance with TSD 14-019.

(b) Normalized radionuclide mixture for dose significant ROC for Auxiliary Building from Table 20 of TSD 14-019.

Table 4-3 Dose for Individual Scenarios ($DOSE_{AMCG}$)

	Auxiliary Building	
	Groundwater (mrem/yr)	Drilling Spoils (mrem/yr)
Co-60	0.232	24.768
Ni-63	24.997	0.003
Sr-90	24.995	0.005
Cs-134	14.892	10.108
Cs-137	22.271	2.729

4.4.2.9.2. ALARA Calculation

The ALARA calculations performed to evaluate the concrete scabbling or shaving remediation activity is presented in Table 4-4 for the Auxiliary Building 542 foot elevation. A result for the Conc/DCGL ratio that is less than one would justify remediation whereas a result greater than one would demonstrate that residual radioactivity is ALARA. The Conc/DCGL ratio calculated for the summation of *In-Situ* Scenarios (Groundwater + Drilling Spoils) was 2.87.

4.4.2.10. Conclusion

Concrete structural surfaces below the 588 foot elevation will remain in place after license termination. The site dose contribution from remaining residual radioactivity remaining in these buried plant structures will be accounted for by the BFM. The ALARA analysis based on cost benefit analysis shows that further remediation of concrete beyond that required to demonstrate compliance with the 25 mrem/yr dose criterion is not justified.

Table 4-4 ALARA Analysis for Volumetrically Contaminated Subsurface Structures – Auxiliary Building 542 ft.

Cost (in dollars) of remedial action ($Cost_T$) = \$517,407.37

Summation of *In-Situ* Scenarios (Groundwater + Drilling Spoils)

(Groundwater Scenario)

A = 10,000 m^2, r = 0.00 yr^{-1}, N = 1,000 yr, P_D = 0.01 person/m^2 — Fraction of Activity removed by remedial action (F) = 1

Column A	Column B	Column C	Column D	Column E	Column F	Column G	Column H	Column I	Column J	Column K	Column L	Column M	
Nuclide	Half-Life (yrs)[b]	λ (yr^{-1})[b]	$(r+\lambda)$	$(r+\lambda)N$	$e^{-(r+\lambda)N}$	$1-e^{-(r+\lambda)N}$	$[1-e^{-(r+\lambda)}N]/(r+\lambda)$	Mixture[b]	GW DCGL$_{BS}$[a]	(Columns I*J)	f_i Column K divided by sum	Cost Benefit	
Co-60	5.27E+00	1.31E-01	1.31E-01	1.31E+02	7.77E-58	1.00E+00	7.60E+00	0.92%	3.28E+10	3.02E+08	9.66E-02	\$ 179.22	
Ni-63	9.60E+01	7.22E-03	7.22E-03	7.22E+00	7.33E-04	9.99E-01	1.38E+02	23.71%	1.15E+10	2.73E+09	8.73E-01	\$ 174,620.71	
Sr-90	2.91E+01	2.38E-02	2.38E-02	2.38E+01	4.54E-11	1.00E+00	4.20E+01	0.05%	9.98E+06	4.99E+03	1.60E-06	\$ 0.32	
Cs-134	2.06E+00	3.36E-01	3.36E-01	3.36E+02	7.94E-147	1.00E+00	2.97E+00	0.01%	3.55E+08	3.55E+04	1.14E-05	\$1.35	
Cs-137	3.02E+01	2.29E-02	2.29E-02	2.29E+01	1.08E-10	1.00E+00	4.36E+01	75.31%	1.25E+08	9.41E+07	3.01E-02	\$ 5,371.22	
							Check Sum	100%	Sum	3.12E+09	1.00E+00	\$ 180,172.82	$\sum(Cost_B)$

(Drilling Spoils Scenario)

A = 100.00 m^2 [(c)], r = 0.00 yr^{-1}, N = 1,000 yr, P_D = 0.01 person/m^2 — Fraction of Activity removed by remedial action (F) = 1

Column A	Column B	Column C	Column D	Column E	Column F	Column G	Column H	Column I	Column J	Column K	Column L	Column M	
Nuclide	Half-Life (yrs)[b]	λ (yr^{-1})[b]	$(r+\lambda)$	$(r+\lambda)N$	$e^{-(r+\lambda)N}$	$1-e^{-(r+\lambda)N}$	$[1-e^{-(r+\lambda)}N]/(r+\lambda)$	Mixture[b]	DS DCGL$_{BS}$[a]	(Columns I*J)	f_i Column K divided by sum	Cost Benefit	
Co-60	5.27E+00	1.31E-01	1.31E-01	1.31E+02	7.77E-58	1.00E+00	7.60E+00	0.92%	3.07E+08	2.82E+06	1.17E-07	\$ 0.00	
Ni-63	9.60E+01	7.22E-03	7.22E-03	7.22E+00	7.33E-04	9.99E-01	1.38E+02	23.71%	1.02E+14	2.42E+13	1.00E+00	\$ 0.23	
Sr-90	2.91E+01	2.38E-02	2.38E-02	2.38E+01	4.54E-11	1.00E+00	4.20E+01	0.05%	5.25E+10	2.63E+07	1.09E-06	\$ 0.00	
Cs-134	2.06E+00	3.36E-01	3.36E-01	3.36E+02	7.94E-147	1.00E+00	2.97E+00	0.01%	5.23E+08	5.23E+04	2.16E-09	\$ 0.00	
Cs-137	3.02E+01	2.29E-02	2.29E-02	2.29E+01	1.08E-10	1.00E+00	4.36E+01	75.31%	1.02E+09	7.68E+08	3.18E-05	\$ 0.01	
							Check Sum	100%	Sum	2.42E+13	1.00E+00	\$ 0.23	$\sum(Cost_B)$

Summation of *In-Situ* Cost Benefit (Groundwater + Drilling Spoils)

Conc/DCGL (A result < 1 would justify remediation whereas a result > 1 would demonstrate that residual radioactivity is ALARA) **2.87**

(a) From Table 4-1
(b) From Table 4-2
(c) Actual drilling spoils area 0.457 m2, 100 m2 used in calculation to ensure 1 person exposed

4.5. References

4-1 U.S. Nuclear Regulatory Commission Regulatory Guide 1.187 "Guidance for Implementation of 10 CFR 50.59 Changes, Tests and Experiments" – November 2000

4-2 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" – January 1999

4-3 Zion Station, "Defueled Safety Analysis Report" (DSAR) – September 2014

4-4 "Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement" – December 2007

4-5 Zion*Solutions* Technical Support Document 10-002, Revision 1,"Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"

4-6 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" – August 2000

4-7 U.S. Nuclear Regulatory Commission Docket Number 50-295, "Facility Operating License Number DPR-39 (for Unit One)"

4-8 U.S. Nuclear Regulatory Commission Docket Number 50-304, "Facility Operating License Number DPR-48 (for Unit Two)"

4-9 U.S. Nuclear Regulatory Commission NUREG-1757, Volume 2, Revision 1, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report" – September 2003

4-10 U.S. Nuclear Regulatory Commission, NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" – September 2004

4-11 U.S. Nuclear Regulatory Commission, NUREG-1496, Volume 2, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities" – July 1997

4-12 U.S. Nuclear Regulatory Commission, NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy" – December 1995

4-13 Zion*Solutions* Technical Support Document 14-016, Revision 0, "Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State"

4-14 U.S. Nuclear Regulatory Commission, NUREG/CR-5884, Volume 2, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station" – November, 1995

4-15 Zion*Solutions* Technical Support Document 14-013, Revision 0, "Zion Auxiliary Building End State Estimated Concrete Volumes, Surface Areas, and Source Terms"

4-16 Zion*Solutions* Technical Support Document 14-019, Revision 2, "Radionuclides of Concern for Soil and Basement Fill Model Source Terms"

ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 5, REVISION 2
FINAL STATUS SURVEY PLAN

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LIST OF ACRONYMS AND ABBREVIATIONS

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

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

5. FINAL STATUS SURVEY PLAN

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

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

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

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

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

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

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

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

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

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

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

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

The structural surfaces that will remain at ZNPS following the termination of the license are constructed of solid steel and concrete which will be covered by at least three (3) feet of soil and physically altered

to a condition which would not allow the remaining structural surfaces, if excavated, to be realistically occupied.

The End State will also include a range of buried piping, embedded piping and penetrations. Buried piping is defined as pipe that runs through soil. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of buried piping, penetrations and embedded piping to remain is provided in Zion*Solutions* Technical Support Document (TSD) 14-016, *"Description of Embedded Pipe, Penetrations, and Buried Pipe to Remain in Zion End State"* (Reference 5-8). The list of end-state embedded pipe, buried pipe and penetrations presented in Attachment F to TSD 14-016 is intended to be a bounding end-state condition. No pipe that is not listed in Attachment F will be added to the end-state condition however, pipe can be removed from the list and disposed of as waste.

5.1. Radionuclides of Concern and Mixture Fractions

Zion*Solutions* TSD 11-001, *"Potential Radionuclides of Concern during the Decommissioning of Zion Station"* (Reference 5-9) was prepared and approved in November 2011. The purpose of this document was to establish the basis for an initial suite of potential radionuclides of concern (ROC) for the decommissioning. Industry guidance was reviewed as well as the analytical results from the sampling of various media from past plant operations. Based on the elimination of some of the theoretical neutron activation products, noble gases and radionuclides with a half-life less than two years, an initial suite of potential ROC for the decommissioning of the ZNPS was prepared. The initial suite of potential ROC is provided in Table 5-1.

Table 5-1 Initial Suite of Radionuclides

Radionuclide	Half Life (years)	Radionuclide	Half Life (years)	Radionuclide	Half Life (years)
H-3	1.24 E 01	Tc-99	2.13 E 05	Np-237	2.14 E 06
C-14	5.73 E 03	Ag-108m	1.27 E 02	Pu-238	8.77 E 01
Fe-55	2.70 E 00	Sb-125	2.77 E 00	Pu-239/240	2.41 E 04
Ni-59	7.50 E 04	Cs-134	2.06 E 00	Pu-241	1.44 E 01
Co-60	5.27 E 00	Cs-137	3.00 E 01	Am-241	4.32 E 02
Ni-63	9.60 E 01	Eu-152	1.33 E 01	Am-243	7.38 E 03
Sr-90	2.91 E 01	Eu-154	8.80 E 00	Cm-243/244	1.81 E 01
Nb-94	2.03 E 04	Eu-155	4.96 E 00		

LTP Chapter 2 provides detailed characterization data that describes current contamination levels in the basements and the results of surveys taken of soils. The survey data for basements is based on core

samples obtained at biased locations with elevated contact dose rates and/or evidence of leaks/spills. Surface and subsurface soil samples were taken in each impacted open land survey units and analyzed for the presence of plant-derived radionuclides. Zion*Solutions* TSD 14-019, *"Radionuclides of Concern for Soil and Basement Fill Model Source Terms"* (Reference 5-10) evaluates the results of the concrete core analysis data from the Containments and Auxiliary Building and refines the initial suite of radionuclides potential ROC by evaluating the dose significance of each radionuclide.

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

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

Table 5-2 Dose Significant Radionuclides and Mixture

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

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

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

reasonably conservative mixture to apply to soils and buried piping for FSS planning and implementation.

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

For continuing characterization, 10% of all media samples collected in a survey unit during continuing characterization will be analyzed for HTD radionuclides. In addition, a minimum of one sample beyond the 10% minimum will be selected at random, also for HTD radionuclide analysis. All samples will first be analyzed by the on-site gamma spectroscopy system. The sample(s) selected for HTD analysis to meet the 10% requirement will be from the highest gamma activity of the sample population; however additional samples (above 10%) will be sent if they exhibit sufficient activity such that the HTD ROC's will likely be detectable by the laboratory using the nominal surrogate ratios and MDCs. In the absence of detectable gamma activity, locations will be selected based on the potential for the presence of activity using HSA information or other process knowledge data. All samples selected for HTD analysis during continuing characterization will be analyzed for the full suite of radionuclides from Table 5-1.

The actual IC dose will be calculated for each individual sample result using the DCGLs from TSD 14-019 Table 27 for structures and Table 28 for soils. 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. If the actual IC dose calculated from the sample result is greater than the IC dose assigned for DCGL adjustment, then a minimum of five (5) additional investigation samples will be taken around the original sample location. Each investigation sample will be analyzed by the on-site gamma spectroscopy system and sent for HTD analysis (full suite of radionuclides from Table 5-1). As with the original sample, the actual IC dose will be calculated for each investigation sample. In this case, the actual calculated maximum IC dose from an individual sample observed in the survey unit will be used to readjust the DCGLs in that survey unit. If the maximum IC dose exceeds 10%, then the additional radionuclides that were the cause of the IC dose exceeding 10% will be added as additional ROC for that survey unit. The survey unit-specific DCGLs used for compliance, the ROC for that survey unit and the survey data serving as the basis for the IC dose adjustment will be documented in the release record for the survey unit.

The final ROC for the decommissioning of Zion are Co-60, Cs-134 and Cs-137 (as well as Eu-152 and Eu-154 for Containment), which are gamma emitters and Ni-63, Sr-90 and H-3 (applicable only to Containment), which are HTD radionuclides. For sample(s) analyzed for HTD radionuclides during continuing characterization, if the analysis of the sample indicates positive results (greater than MDC) for both a HTD ROC and the corresponding surrogate radionuclide (Cs-137 or Co-60), then the HTD to surrogate ratio will be derived. If the derived HTD to surrogate ratio is less than the maximum HTD to surrogate ratio from section 5.2.11, Table 5-15, then no further action is required. If the HTD to surrogate ratio exceeds the maximum ratio from section 5.2.11, Table 5-15, then a minimum of five (5)

additional investigation samples will be taken around the original sample location. Each investigation sample will be analyzed by the on-site gamma spectroscopy system and then sent for HTD analysis. As with the original sample, the HTD to surrogate ratio will be calculated for each investigation sample. The actual maximum HTD to surrogate ratio observed in any individual sample will be used to infer HTD radionuclide concentrations in the survey units shown to be impacted by the investigation. The survey unit-specific HTD to surrogate ratio and the survey data serving as the basis for the ratio will be documented in the release record for the survey unit(s).

Survey unit-specific surrogate ratios, in lieu of the 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. In these cases, the survey unit-specific radiological data and the derived surrogate ratios will be submitted to the NRC for approval. If approved, then the survey unit-specific ratios used and the survey data serving as the basis for the surrogate ratios will be documented in the release record for the survey unit.

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

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

5.2. Release Criteria

Before the FSS process can proceed, the DCGLs (referred to in this Chapter as Base Case DCGLs) that are used to demonstrate compliance with the 25 mrem/yr unrestricted release criterion must be established. The Base Case DCGLs are calculated by analysis of various pathways (direct radiation,

inhalation, ingestion, etc.), media (concrete, soils, and groundwater) and scenarios through which exposures could occur. Chapter 6 of this LTP describes in detail the approach, modeling parameters and assumptions used to develop the Base Case DCGLs.

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

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

Multiple ROC are known to be present at ZSRP. The dose contribution from each ROC is accounted for using the SOF to ensure that the total dose from all ROC does not exceed the dose criterion.

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

5.2.1. Base Case Derived Concentration Guideline Levels for Basement Surfaces

The Basement Fill Model (BFM) applies to the steel and concrete walls and floor surfaces, below the 588 foot elevation, of the Unit 1 Containment Building, Unit 2 Containment Building, Auxiliary Building, Turbine Building, WWTF, the lower portion of the SFP, Fuel Transfer Canals, Crib House and Forebay, Unit 1 and Unit 2 Steam Tunnels and the Circulating Water Intake Piping and Circulating Water Discharge Tunnels. The BFM source term also includes the end-state embedded piping and penetrations as specified in TSD 14-016. The DCGLs for embedded pipe and penetrations are provided in sections 5.2.7 and 5.2.9. Basement Surface DCGLs are referred to as "Base Case" DCGLs in this LTP Chapter to ensure a clear distinction from Operational DCGLs. Base Case DCGLs for basement surfaces are in units of pCi/m^2. Additional information pertaining to the calculation of DCGLs for basement surfaces is provided in LTP Chapter 6, section 6.6.8.

The Base Case $DCGL_B$ is directly analogous to the $DCGL_W$ as defined in MARSSIM and is the DCGL used during FSS to demonstrate compliance. The IC dose percentage of 5% for the Auxiliary Basement

and 10% for the Containment Basement was used to adjust the Base Case $DCGL_B$ to account for the dose from the eliminated IC radionuclides. The Base Case $DCGL_B$ values, equivalent to the Basement Surfaces DCGLs from LTP Chapter 6, section 6.6.8.1 are reproduced in Table 5-3.

Table 5-3 Base Case DCGLs ($DCGL_B$) for Basements (pCi/m²)

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

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

5.2.2. Operational Derived Concentration Guideline Levels for Basement Surfaces

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

Table 5-4 Operational DCGLs ($OpDCGL_B$) for Basements (pCi/m²)

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

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

5.2.3. Base Case Derived Concentration Guideline Levels for Soil

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

Surface soil is defined as soil residing in the first 0.15 m layer of soil. A subsurface soil category, which is defined as a layer of soil beginning at the surface but extending to a depth of 1 m is also

assessed to allow for flexibility in compliance demonstration if contamination deeper than 0.15 m is encountered. Site-specific DCGLs for soil were calculated for both the 0.15 m and 1 m thicknesses. Based on characterization data and historical information, there are no expectations of encountering a source term geometry that is comprised of a clean surface layer of soil over a contaminated subsurface soil layer. Zion*Solutions* TSD 14-011, *"Soil Area Factors"* (Reference 5-12) and LTP Chapter 6, section 6.8 provides the exposure scenarios and modeling parameters that were used to calculate the site-specific DCGLs for soils (referred to a Base Case Soil DCGLs in this Chapter). The surface and subsurface soil Base Case DCGLs for the unrestricted release of open land survey units are provided in Tables 5-5 and 5-6, respectively. The IC dose percentage of 10% was used to adjust the DCGLs in Tables 5-5 and 5-6 to account for the dose from the eliminated IC radionuclides.

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

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

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

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

5.2.4. Operational Derived Concentration Guideline Levels for Soil

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

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

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

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

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

5.2.5. Base Case Derived Concentration Guideline Levels for Buried Piping

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

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

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

5.2.6. Operational Derived Concentration Guideline Levels for Buried Piping

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

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

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

5.2.7. Base Case Derived Concentration Guideline Levels for Embedded Pipe

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

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

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

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

5.2.8. Operational Derived Concentration Guideline Levels for Embedded Pipe

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

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

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

5.2.9. Base Case Derived Concentration Guideline Levels for Penetrations

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

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

basements. The residual radioactivity in the penetration is assumed to release to both basements simultaneously.

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

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

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

Radionuclide	Auxiliary Bldg. (pCi/m²)	U1/U2 Containment (pCi/m²)	SFP/ Transfer Canal (pCi/m²)	Turbine Bldg. (pCi/m²)	Crib House/ Forebay (1) (pCi/m²)	WWTF[1] (pCi/m²)
H-3	3.99E+09	3.42E+09	4.84E+16	3.23E+09	N/A	N/A
Co-60	8.82E+07	2.26E+09	4.45E+08	1.76E+09	N/A	N/A
Ni-63	6.79E+10	5.78E+10	1.86E+14	5.48E+10	N/A	N/A
Sr-90	2.41E+07	2.06E+07	9.26E+10	1.94E+07	N/A	N/A
Cs-134	3.28E+08	4.32E+08	7.48E+08	4.00E+08	N/A	N/A
Cs-137	6.17E+08	5.66E+08	1.46E+09	5.29E+08	N/A	N/A
Eu-152	3.29E+08	5.26E+09	9.44E+08	4.06E+09	N/A	N/A
Eu-154	2.33E+08	4.58E+09	8.53E+08	3.58E+09	N/A	N/A

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

5.2.10. Operational Derived Concentration Guideline Levels for Penetrations

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

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

Radionuclide	Auxiliary Bldg. (pCi/m²)	Unit 1/Unit 2 Containment (pCi/m²)	SFP/ Transfer Canal (pCi/m²)	Turbine Bldg. (pCi/m²)	Crib House/ Forebay (pCi/m²)	WWTF (pCi/m²)
H-3	3.14E+08	2.33E+08	1.13E+16	2.58E+08	N/A	N/A
Co-60	6.95E+06	1.54E+08	1.04E+08	1.41E+08	N/A	N/A
Ni-63	5.35E+09	3.93E+09	4.33E+13	4.38E+09	N/A	N/A
Sr-90	1.90E+06	1.40E+06	2.16E+10	1.55E+06	N/A	N/A
Cs-134	2.58E+07	2.94E+07	1.74E+08	3.20E+07	N/A	N/A
Cs-137	4.86E+07	3.85E+07	3.40E+08	4.23E+07	N/A	N/A
Eu-152	2.59E+07	3.58E+08	2.20E+08	3.25E+08	N/A	N/A
Eu-154	1.84E+07	3.11E+08	1.99E+08	2.86E+08	N/A	N/A

5.2.11. Surrogate Radionuclides

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

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

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

Table 5-15 Surrogate Ratios

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

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

Using the appropriate scaling factors, the DCGL of the measured radionuclide is modified to account for the represented radionuclide(s) according to the following equation from section 4.3.2 of MARSSIM:

Equation 5-1

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

where:

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

5.2.12. Sum-of-Fractions

The SOF or "unity rule" is applied to the data used for the survey planning, and data evaluation and statistical tests for soil sample analyses since multiple radionuclide-specific measurements will be performed or the concentrations inferred based on known relationships. The application of the unity rule serves to normalize the data to allow for an accurate comparison of the various data measurements to the release criteria. When the unity rule is applied, the $DCGL_w$ (used for the nonparametric statistical test) becomes one (1). The basement structure DCGLs ($DCGL_B$), embedded pipe DCGLs ($DCGL_{EP}$) and penetration DCGLs ($DCGL_{PN}$) are directly analogous to the $DCGL_W$ as defined in MARSSIM. The use and application of the unity rule will be performed in accordance with section 4.3.3 of MARSSIM.

5.2.13. Dose from Groundwater

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

5.2.14. Demonstrating Compliance with Dose Criterion

The Base Case DCGLs for backfilled basements, surface soil, subsurface soil, buried piping, embedded piping and penetrations for each ROC are presented in Tables 5-3, 5-5, 5-6, 5-9, 5-11 and 5-13, respectively. These values are equivalent to the level of residual radioactivity in the media (above background) that could, when considered independently for each ROC, result in a TEDE of 25 mrem

per year to an AMCG. For all media, the dose from the residual radioactivity from each ROC (radionuclide *i*) can be expressed as shown in the following equation:

Equation 5-2

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

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

Equation 5-3

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

where:

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

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

5.2.15. Soil Area Factors

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

The investigation level for this comparison in soils is the $DCGL_{EMC}$, which is the $DCGL_W$ modified by an AF to account for the small area of the elevated radioactivity. At the ZSRP, $DCGL_{EMC}$ only applies to soils as all other media (structural surfaces, embedded pipe, buried pipe and penetrations) will be remediated at their applicable Base Case DCGL. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. Note that the consideration of small areas of elevated radioactivity applies only to Class 1 survey units, as Class 2 and Class 3 survey units by definition should not have contamination in excess of the $DCGL_W$. The following equation defines the calculation of a $DCGL_{EMC}$.

Equation 5-4

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

AFs are calculated using RESRAD for each ROC and for source area sizes ranging from 0.01 m^2 up to the full source area of 64,500 m^2. The AFs for surface and subsurface soils were calculated in TSD 14-011 are provided in Tables 5-16 and 5-17 and discussed in Section 6.11.

Table 5-16 Area Factors for Surface Soils

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

Table 5-17 Area Factors for Subsurface Soils

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

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

5.3. Summary of Characterization Survey Results

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

5.3.1. Survey of Impacted Media

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

were acquired from subsurface basement surfaces, and samples and measurements were taken inside building drain systems.

5.3.2. Field Instrumentation and Sensitivities

The field instrumentation for characterization was selected to provide both reliable operation and adequate sensitivity to detect the ROC identified for ZSRP at levels sufficiently below the established action levels. For characterization of soils, the interim screening DCGLs presented in NUREG-1757, Appendix H, Table H.1 and NUREG/CR-5512 Volume 3, *"Residual Radioactive Contamination from Decommissioning Parameter Analysis"*, (Reference 5-15), Table 6.91 ($P_{crit} = 0.10$) were used as the action levels to assess the correct classification of impacted open land or soil survey units. For structures, the nuclide-specific screening value of 7,100 dpm/100cm^2 total gross beta-gamma surface activity based on Co-60 from NUREG-1757, Appendix H was used as the action level to evaluate the classification of a structural survey unit. In all cases, the field instruments and detectors selected for static measurements and scanning were capable of detecting the anticipated ROC at a MDC of 50% of the applicable action level.

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

Gamma scans were performed over open land surfaces to identify locations of residual surface activity. Sodium iodide (NaI) gamma scintillation detectors (typically 2" x 2") were typically used for these scans. Zion*Solutions* TSD 11-004, *"Ludlum Model 44-10 Detector Sensitivity"* (Reference 5-16) examines the response and scan Minimum Detectable Concentration (MDC) of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. ISOCS measurements were taken in several open land survey units in lieu of scanning and soil sampling.

5.3.3. Laboratory Instrument Methods and Sensitivities

Gamma spectroscopy was primarily performed by the on-site radiological laboratory. Gas proportional counting and liquid scintillation analysis was performed by an approved vendor laboratory in accordance with approved laboratory procedures. ZSRP ensured that the quality programs of the contracted off-site vendor laboratories that were used for the receipt, preparation and analysis of characterization samples provided the same level of quality as the on-site laboratory under Zion*Solutions* ZS-LT-01, *"Quality Assurance Project Plan (for Characterization and FSS"* (QAPP) (Reference 5-17). In all cases, analytical methods were established to ensure that required MDC values are achieved. The analysis of radiological contaminants used standard approved and generally accepted methodologies or other comparable methodologies.

5.3.4. Summary of Survey Results

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

5.3.4.1. Impacted and Non-Impacted Areas

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

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

5.3.4.2. Justification for Non-Impacted Areas

MARSSIM defines non-impacted areas as those areas where there is no reasonable possibility of residual contamination. A review of the operating history of the facility, historical incidents, and operational radiological surveys as documented in the HSA was conducted. Based on the review, the open land areas included in the "owner-controlled" property outside of the footprint of the 87 acre, fence-enclosed "Radiologically Restricted Area", minus the additional impacted areas cited in the previous section, were deemed not impacted by licensed activities or materials.

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

5.3.4.3. Adequacy of the Characterization

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

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

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

5.3.4.4. Inaccessible or Not Readily Accessible Areas

Characterization has been performed in some of the end-state concrete structures to assess the current residual radioactivity concentration, verify the applied radionuclide mixture and, validate the classification of each FSS unit. ZSRP has also characterized surface and subsurface soils surrounding ZNPS. The results of site characterization are presented in Chapter 2.

Continuing characterization surveys will be performed in accordance with Zion*Solutions* 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. Continuing characterization surveys will be designed to gather the appropriate data using the DQO process as outlined in MARSSIM, Appendix D. Survey design will incorporate a graded approach based upon the DQOs for each survey unit. The number of measurements and/or samples that will be taken in each survey unit will be determined by assessing the sample size necessary to satisfy the DQOs. The selection of sample locations will be determined by the professional judgment of the responsible Radiological Engineer during the survey design process. In the design process, consideration is given to locations that exhibit measurable radiation levels above background (i.e. by scanning), depressions, discolored areas, cracks, low point gravity drain points, actual and potential spill locations, or areas where the ground has been disturbed. Historical information from the HSA is used to aid in the selection of biased locations. Characterization sample plan preparation and survey design will be performed in accordance with Zion*Solutions* procedure ZS-LT-100-001-001, "Characterization Survey Package Development" (Reference 5-19).

As a minimum, additional characterization will be performed at the following areas at Zion. Where feasible, the minimum number of samples and their locations are specified.

- The underlying concrete of the SFP/Transfer Canal below the 588 foot elevation after the steel liner has been removed. The objective of the continuing characterization survey will be to validate the use of the Aux Building mixture as a reasonably conservative mixture that is representative of the SFP/Transfer Canal concrete. Three concrete cores were previously taken in the SFP prior to demolition. Additional concrete core samples will be taken to ensure that the mixture is representative. The continuing characterization will consist of a scan of the exposed concrete surfaces and the acquisition of at least five additional concrete core samples at the locations identified by the scan that exhibits the highest activity. The concrete cores will be taken to a depth of 6-inches and each core will be segmented into ½ inch concrete core pucks. Each puck will then be analyzed by the on-site gamma spectroscopy system for gamma-emitting radionuclides.
- The concrete walls and floor of the Under-Vessel areas in Unit 1 and Unit 2 Containments. The objective of the continuing characterization survey will be to assess the depth of activation in the

concrete in order to guide the remediation necessary to meet $OpDCGL_B$ for Containment and to ensure the correct geometry for the ISOCS. In each unit, the continuing characterization will consist of a scan of the exposed concrete surfaces and the acquisition of at least 13 additional concrete core samples. Four (4) locations will be selected on the upper walls, 3 locations on the lower walls, 3 locations on the floor, 3 locations to include the embedded steel support ring and 3 to include shallow rebar. To the extent possible, the locations selected will be identified by scan that exhibits the highest activity. A concrete core sample will be taken at each location that completely penetrates the concrete to the underlying steel liner. Each core will be segmented into ½ inch concrete core pucks, which will then be analyzed by the on-site gamma spectroscopy system for gamma-emitting radionuclides.

- The floors and walls of the Hold-Up Tank (HUT) cubicle. The objective of the continuing characterization survey will be to assess the contamination profile of the concrete to validate the ISOCS Geometry Template as recommended by Zion*Solutions* TSD 14-022, "*Use of In-Situ Gamma Spectroscopy for Source Term Survey of End State Structures*" (Reference 5-20). In each unit, the continuing characterization will consist of a scan of the exposed concrete surfaces and the acquisition of at least 8 additional concrete core samples at the locations identified by the scan that exhibits the highest activity. The concrete cores will be taken to a depth of 6-inches and each core will be segmented into ½ inch concrete core pucks. Each puck will then be analyzed by the on-site gamma spectroscopy system for gamma-emitting radionuclides.
- The floor of the Auxiliary Building 542 foot elevation Pipe Tunnel floors. The objective of the continuing characterization survey will be to assess the contamination profile of the concrete to validate the ISOCS Geometry Template as recommended by Zion*Solutions* TSD 14-022. In each unit, the continuing characterization will consist of a scan of the exposed concrete surfaces and the acquisition of at least 8 additional concrete core samples at the locations identified by the scan that exhibits the highest activity. The concrete cores will be taken to a depth of 6-inches and each core will be segmented into ½ inch concrete core pucks. Each puck will then be analyzed by the on-site gamma spectroscopy system for gamma-emitting radionuclides.
- The floor and lower walls of the 542 foot elevation of the Auxiliary Building. The objective of the continuing characterization survey will be to augment the existing contamination profile of the concrete from the previous characterization and to validate the radionuclide mixture is consistent. In each unit, the continuing characterization will consist of a scan of the exposed concrete surfaces and the acquisition of at least 8 additional concrete core samples at the locations identified by the scan that exhibits the highest activity. The concrete cores will be taken to a depth of 6-inches and each core will be segmented into ½ inch concrete core pucks. Each puck will then be analyzed by the on-site gamma spectroscopy system for gamma-emitting radionuclides.
- The subsurface soils in the "keyways" between the Containment Buildings and the Turbine Building. This will occur once subsurface utilities and subsurface access-interfering structures (e.g., Waste Annex Building) have been removed. The objective of the continuing characterization survey will be to assess the radiological contamination of subsurface soils in these two areas. Continuing characterization will consist of the scanning of soils exposed by the demolition and building removal, the collection of soil samples of the exposed surface soil and collection of additional subsurface soil samples using Geoprobe sampling. The location of the Geoprobe samples will correspond to at least ten (10) locations that exhibit the highest surface-scan measurements of

the exposed soils. If elevated activity is not identified by the scans, then the sample locations will be biased to locations where elevated activity could accumulate, such as below travel paths, below building access points and former waste loading areas. A surface soil and subsurface soil sample will taken at each location. The subsurface soil sample will be taken to a depth of 3 meters below grade. All samples will be analyzed by the on-site gamma spectroscopy system.

- The soils under the basement concrete of the Containment Buildings, the Auxiliary Building and the SFP/Transfer Canal. This will occur once commodity removal and building demolition have progressed to a point where access can be achieved. The objective of the continuing characterization survey will be to assess the radiological contamination of subsurface soils adjacent to and below these basement slabs. Continuing characterization will consist of GeoProbe soil borings at the nearest locations along the foundation walls that can be feasibly accessed and angled GeoProbe soil bores to access the soils under the concrete. A minimum of 4 subsurface soil samples will be taken around each foundation from grade to the depth of approximately 55 feet or refusal, whichever is less. Attempts shall be made to acquire a minimum of 2 subsurface soil samples from beneath each Containment basement foundation and the Auxiliary Building basement floor slab. Samples from under the SFP foundation slab will be acquired from within the excavation prior to backfill. Additional investigations and sampling will be performed in accordance with a sample plan if activity is positively identified. All samples will be analyzed by the on-site gamma spectroscopy system.
- When the interior surfaces become accessible, several potentially contaminated embedded and buried pipe systems that will be abandoned in place will be characterized. The objective of the continuing characterization survey will be to assess the potential radiological classification in the pipe if the HSA or process knowledge is insufficient. Continuing characterization will consist of direct measurements on pipe openings and the acquisition of sediment and/or debris samples (if available) for analysis. Any sediment or debris samples will be analyzed by the on-site gamma spectroscopy system.
- The Containment basements, after concrete removal. Continuing characterization of the steel liner will consist of beta gamma scans and swipe samples. The objective of the continuing characterization survey will be to assess the radiological condition of the exposed steel liner above the 565 foot elevation after the contaminated concrete has been removed. The liner will be subjected to sufficient smear samples and beta scans of accessible surfaces to ensure that the liner is adequately decontaminated prior to FSS. Locations for taking samples and/or measurements will be biased toward locations with high potential for the presence of loose or fixed contamination.

As stated in section 5.1, 10% of all media samples collected in a survey unit during continuing characterization will be analyzed for HTD radionuclides. All samples will first be analyzed by the on-site gamma spectroscopy system. The sample(s) selected for HTD analysis will exhibit the highest gamma activity from the sample population, however additional samples will be selected for HTD analysis beyond the 10% minimum based on outcome of DQO evaluation. In addition, if the level of residual radioactivity in an individual soil sample exceeds the SOF of 0.1, then that sample(s) will be also be analyzed for HTD radionuclides. All samples selected for HTD analysis during continuing characterization will be analyzed for the full suite of radionuclides from Table 5-1.

We believe that all exposed surface soils at ZNPS have been adequately characterized and that additional characterization of surface soil is not anticipated. However, continuing characterization in

the form of Radiological Assessment (RA) surveys will be performed in currently inaccessible subsurface soil areas that are exposed after removal of asphalt or concrete roadways and parking lots, rail lines, or building foundation pads (slab on grade) provided that the removal is not for the purpose of radiological remediation.

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

5.4. <u>Decommissioning Support Surveys</u>

5.4.1. Radiological Assessment (RA)

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

The RA of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-27. In addition to direct and scan radiation measurements, the RA will include the collection of potentially impacted soil, sediment and/or surface residue for laboratory analysis.

5.4.2. Remedial Action Support (In-Process) Surveys

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

RASS of soil areas will rely principally on direct and scan radiation measurements using gamma sensitive instrumentation described in Table 5-27. In addition to direct and scan radiation measurements, the RASS will include the collection of potentially impacted soil, sediment and/or surface residue for laboratory analysis.

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

5.4.3. Instrumentation for RA and RASS

Table 5-27 shows typical field instruments that will be used for performing FSS. The same or similar instruments will be used during the performance of the RA and RASS. The typical MDCs for field instruments used for scanning are provided in Table 5-28 and are sufficient to measure concentrations at the same action levels used during characterization as specified in section 5.3.2.

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

5.4.4. Field Screening Methods for RA and RASS

A gamma walk-over survey will be performed over the surface area, typically using a 2 inch by 2 inch NaI gamma scintillation detector. Appropriate scanning speed and scanning distance will be implemented to ensure that a MDC of 50% of the Operational DCGLs for surface soil ($OpDCGL_{SS}$ from Table 5-7) is achieved. Locations of elevated count rate will be identified for additional scanning and/or the collection of biased soil samples to determine if the elevated count rate indicates the presence of soil concentration in excess of the $OpDCGL_{SS}$. The information obtained during the RA and RASS (scan results and the analytical data from any associated soil samples) will be used to determine if the remaining exposed soils:

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

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

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

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

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

All remaining structural surfaces will be surveyed to meet the criteria for open air demolition specified in Zion*Solutions* TSD 10-002, *"Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"* (Reference 5-21). These criterions are the acceptable removable contamination and contact exposure rate levels that are allowable for open air demolition. A contamination verification survey (CVS) will be performed to identify areas requiring remediation to meet the open air demolition limits. A CVS will be performed within any structure that contains, or previously contained, radiological controlled areas. The CVS will be performed using hand-held beta-gamma instrumentation as presented in Table 5-27 in typical scanning and measurement modes.

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

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

5.4.6. Post-Demolition Survey

Following demolition, after all debris is removed and the floors cleaned, an additional scan survey will be performed to ensure that any individual ISOCS measurement will not exceed the Operational $DCGL_B$ from Table 5-4 during FSS. The survey will be performed using hand-held beta-gamma instrumentation as presented in Table 5-27 in typical scanning and measurement modes.

5.5. <u>Final Status Survey of Basement Structures</u>

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

5.5.1. Instruments Selected for Performing FSS of Basement Surfaces

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

augmented with core sampling. The ISOCS was selected as the instrument of choice to perform FSS of basement surfaces for the following reasons:

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

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

5.5.2. Basement Surface FSS Units

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

The survey units designated for structures below 588 foot elevation from the HSA that were presented in LTP Chapter 2, Table 2-2 were based on screening values and source term assumptions that are significantly different from the BFM and are therefore not applicable.

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

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

House/Forebay FSS unit results. The entire surface areas will be included in the area-weighted average calculation (see section 5.5.6.1).

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

5.5.2.1. Classification and Areal Coverage for FSS of Basement Surfaces

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

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

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

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

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

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

scan of the Turbine Building 560 foot elevation resulted in a maximum observed count rate of 3,922 cpm/100 cm^2.

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

The FSS units for the basements of the Turbine Building, the Crib House/Forebay and the Circulating Water Discharge Tunnels are designated as Class 3 as defined in MARSSIM section 2.2 in that the FSS units are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGLs, based on site operating history and previous radiation surveys. These three FSS units will be subjected to an areal coverage commensurate with the guidance pertaining to Class 3 scan coverage as presented in MARSSIM Table 5.9.

5.5.2.2. Sample Size Determination for FSS of Basement Surfaces

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

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

Following the guidance in MARSSIM, the Type I decision error that was used for this calculation was set at 0.05 and the Type II decision error was set at 0.05. The upper boundary of the gray region was set at the Operational $DCGL_B$. The Lower Bound of the Gray Region (LBGR) was set at the expected fraction of the Operational $DCGL_B$. The expected fraction of the Operational $DCGL_B$ in the Class 1 and Class 2 FSS units was set at 50% and the expected fraction of the Operational $DCGL_B$ in the Class 3 FSS units was set at 1%. The standard deviation of the concrete core samples taken in the Turbine Building was used for sigma (σ) in the FSS units for the Turbine Building, Crib House/Forebay and WWTF. For the Class 2 FSS units in the Containment basements, the entire concrete source term above the 565 foot elevation will be removed. Consequently, the results of the concrete core samples taken in the Containments are also not representative of the conditions at the time of FSS. As reasonable value for sigma (σ) cannot be determined based on existing survey data, a coefficient of variation of 30% was used in accordance with the guidance in MARSSIM, section 5.5.2.2. For the Class 1 survey unit in the Auxiliary Building, the standard deviation of the concrete cores analyzed for

the full initial suite were used for sigma (σ). For the concrete in the Containment Under-Vessel area the standard deviation of the cores collected during characterization were used for sigma (σ).

The relative shift (Δ/σ) was calculated as discussed in section 5.6.4.1.6 of this Chapter. The Δ/σ calculations will be confirmed and documented as part of the FSS design process including evaluation of any additional data from continuing characterization. With the exception of the Under-Vessel area, the relative shift (Δ/σ) was greater than three in all cases. The relative shift (Δ/σ) for the Under-Vessel area was two. Consequently, a value of three was used as the adjusted relative shift (Δ/σ) for all FSS units other than the Under-Vessel area where a value of two was used. From Table 5-5 of MARSSIM, the required number of measurements (N) for use with the Sign Test, using a value of 0.05 for the Type I and Type II decision errors, is 14 measurements for a Δ/σ value of three and 15 for a Δ/σ value of 2. Consequently, the number of ISOCS measurements in several basement surface FSS units was adjusted to meet the larger sample size. Table 5-19 presents the basement surface FSS units and the adjusted number of ISOCS measurements that will be taken in each for FSS.

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

FSS Unit	Classification	Area (m^2)	Minimum Areal Coverage (% of Area)	Minimum # of ISOCS Measurements (FOV-28 m^2)
Aux Bldg. 542 foot Floor and Walls	Class 1	6,503	100%	233
Unit 1 Containment Basement above 565 foot elevation	Class 1	2,465	100%	88
Unit 1 CTMT Under-Vessel Area	Class 1	294	100%	11
Unit 1 Containment Basement above 565 foot elevation	Class 1	2,465	100%	88
Unit 2 CTMT Under-Vessel Area	Class 1	294	100%	11
SFP/Transfer Canal	Class 1	723	100%	26
Turbine Building Basement	Class 3	14,864	1%	6
Circulating Water Discharge Tunnels	Class 3	4,868	1%	2
Crib House/Forebay	Class 3	13,843	1%	5
WWTF	Class 1	1,124	100%	40

As previously noted, the required areal coverage for a Class 1 basement survey unit is 100%. Sufficient measurements will be taken in the Class 1 FSS unit to ensure that 100% of the surface area is surveyed (ISOCS FOV will be overlapped to ensure that there are no un-surveyed corners and gaps). In the case where the physical configuration or measurement geometry would make the acquisition of a 28 m^2 FOV difficult or prohibitive, then the FOV for the ISOCS measurement may be reduced provided

that the adjusted number of samples remains constant and the minimum areal coverage represented by the FSS unit classification (100% areal coverage for a Class 1 FSS unit or 10% areal coverage for a Class 2 FSS unit) is achieved.

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

FSS Unit	Classification	Required Areal Coverage (m^2)	Adjusted # of ISOCS Measurements (FOV-28 m^2)	Adjusted Areal Coverage (m^2)	Adjusted Areal Coverage (% of Area)
Aux Bldg. 542 foot Floor & Walls	Class 1	6,503	407(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).

In the Class 2 basement surface FSS units (where less than 100% ISOCS coverage is required), measurement spacing will be determined in accordance with section 5.6.4.5.2 of this Chapter. The number of measurements will also be increased in survey units that exceed 1,000 m^2 to correspond with the MARSSIM recommended survey size density for a Class 2 structure (measurements/1,000 m^2). If the grid spacing allows, the location of the center of each ISOCS measurement FOV will be determined at a distance equal to the radius of the ISOCS FOV from the boundaries of the FSS unit and the FOV radius of other measurement locations. If possible, the FOV for individual measurements should not overlap. If FOV overlap cannot be avoided, then adjustments shall be made, including taking additional measurements to ensure that the required areal coverage is achieved. If a selected location is found to be either inaccessible or unsuitable, then the location will be adjusted to the closest adjacent suitable location. In these cases, a notation will be made in the field log and the coordinates of the new location documented. In addition to the prescribed areal coverage, additional judgmental measurements will be

collected at locations with higher potential for containing elevated concentrations of residual radioactivity based on professional judgment.

In the Class 3 basement surface FSS units, each measurement location will be randomly selected using a random number generator. If a selected location is found to be either inaccessible or unsuitable, then the location will be adjusted to the closest adjacent suitable location. In these cases, a notation will be made in the field log and the coordinates of the new location documented. In addition to the prescribed areal coverage, additional judgmental measurements will be collected at locations with higher potential for containing elevated concentrations of residual radioactivity based on professional judgment.

5.5.3. Survey Approach for FSS of Basement Surfaces

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

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

For survey units where physical constraints prevent a FOV of 28 m^2, the detector to source distance can be reduced, thereby reducing the FOV, which will increase the number of measurements to ensure that the required FSS coverage as presented in Table 5-18 is achieved. In most cases, the measurement will be acquired using the ISOCS with a geometry that evaluates residual activity over the activity depth.

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

5.5.4. Basement Surface FSS Data Assessment

After a sufficient number of ISOCS measurements are taken in a FSS unit in accordance with the areal coverage requirements specified in Table 5-19, the data will be summarized, including any judgmental or investigation measurements. The measured activity for each gamma-emitting ROC (and any other gamma emitting radionuclide that is positively detected by ISOCS) will be recorded (in units of pCi/m^2). Background will not be subtracted from any measurement. Using the radionuclide mixture fractions applicable to the survey unit, an inferred activity will be derived for HTD ROC using the surrogate approach specified in section 5.2.11. The surrogate ratios that will be used are presented in Table 5-15. A sum of fractions (SOF) calculation will be performed for each measurement by dividing the reported concentration of each ROC by the Operational $DCGL_B$ for each ROC to calculate an individual ROC fraction. The individual ROC fractions will then be summed to provide a total SOF value for the measurement.

As described in section 5.10.3.2, the Sign Test will be used to evaluate the remaining residual radioactivity against the dose criterion. The SOF for each measurement will be used as the sum value for the Sign Test. If the Sign Test demonstrates that the mean activity for each ROC is less than the Operational $DCGL_B$ at a Type 1 decision error of 0.05, then the mean of all the total SOFs for each measurement in a given survey unit is calculated. If the Sign Test fails, or if the mean of the total SOFs in a basement exceeds one (using Operational DCGLs), then the survey unit will fail FSS. If a survey unit fails FSS, then an investigation will be implemented in accordance with section 5.6.4.6.

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

Equation 5-5

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

where:

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

5.5.5. FSS of Embedded Piping and Penetrations

The end state will include embedded piping and penetrations. An embedded pipe is defined as a pipe that runs vertically through a concrete wall or horizontally through a concrete floor and is contained within a given building. A penetration is defined as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration

could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end. The list of penetrations and embedded piping to remain is provided in TSD 14-016. Embedded pipe and penetrations have separate Operational DCGLs as listed in Tables 5-12 and 5-14. However, the survey methods are the same for both. The survey units for embedded pipe and penetrations are presented in Table 5-20.

Table 5-20 Embedded Pipe and Penetration Survey Units

Basement FSS Unit	Embedded Pipe	Penetrations
Auxiliary Building Basement	• Basement Floor Drains (542 ft. elevation)	• Auxiliary Building Penetrations
Containment Basement	• Unit 1 and Unit 2 In-Core Sump Drains (541 ft. elevation) • Unit 1 and Unit 2 Tendon Tunnel Drains	• Containment Penetrations
SFP/Transfer Canal	N/A	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 hydraulically connected to Steam Tunnel/Turbine Building so include with Turbine Building as well as Containment

The residual radioactivity remaining in each section of embedded piping/penetration applicable to each FSS unit will be assessed and quantified by direct survey. Shallow penetrations or short lengths of embedded pipe that are directly accessible will be surveyed using hand-held portable detectors, such as a gas-flow proportional or scintillation detector. Lengths of embedded pipe or penetrations that cannot be directly accessed by hand-held portable detectors will be surveyed using applicable sized NaI or Cesium Iodide (CsI) detectors that will be inserted and transported through the pipe using flexible fiber-composite rods or attached to a flexible video camera/fiber-optics cable. The ISOCS will not be used to perform FSS in any embedded pipe or penetration with the exception of the Circulating Water Intake Pipe and Circulating Water Discharge Tunnels. The specific types of instruments that can be used for both types of scenarios are presented in section 5.8 and Table 5-27.

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

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

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

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

Equation 5-6

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

where:

$SOF_{EP/PN}$	=	SOF for embedded pipe or penetration survey unit within a Basement using Base Case DCGLs
$Mean\ Conc_{EP/PN\ ROC_i}$	=	Mean concentration for the systematic measurements taken during the FSS of embedded pipe or penetrations in survey unit for each ROC_i

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

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

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

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

- If maximum activity exceeds the Base Case $DCGL_{PN}$ from Table 5-13 (SOF >1), then remediation will be performed.
- If the maximum activity in a penetration exceeds the most limiting Operational $DCGL_B$ from Table 5-4 of the two basements where a penetrations interface (SOF>1), but is below the Base Case $DCGL_{PN}$ from Table 5-13, then the penetration will be remediated or grouted.

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

An alternate drilling spoils scenario was evaluated to determine the maximum hypothetical dose from drilling into penetrations or embedded pipe assuming activity is present at the $DCGL_{EP}/DCGL_{PN}$ concentrations. The alternate scenario drilling spoils dose was less than 25 mrem/yr for all penetrations and embedded pipe with the exception of the Steam Tunnel Floor Drains, which resulted in a dose of 71.16 mrem/yr (see LTP Chapter 6 section 6.7). The DCGLs for the Steam Tunnel Floor Drains will be reduced by a factor of 2.89, i.e., from the $DCGL_{EP}$ to $DCGL_{EP} \div 2.89$ which will reduce the maximum dose to 25 mrem/yr.

5.5.6. Summation of Dose for Basement Structures

The BFM source term for a given basement structure includes the contributions from basement surfaces, (concrete and steel liner for Containment), embedded pipe and penetrations that are contained in, or interface with, the basement. Each dose component (surface, embedded pipe, penetrations) has a unique DCGL. Concrete fill is another dose component applicable to any basement where clean concrete debris is used as fill. This is discussed further in LTP Chapter 6, section 6.16. The total dose attributed to the use of concrete fill for each basement, including all ROC, is presented in Table 5-21, which is reproduced from Table 6-53 from LTP Chapter 6, section 6.16. The dose values in Table 5-21 will be added to any basement where concrete fill is used regardless of the volume of concrete fill used.

Table 5-21 Dose Assigned to Clean Concrete Fill

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

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

After the FSS of all dose components in a given basement is complete and all dose component survey units pass the Sign Test, the SOF for each dose component is calculated using Equations 5-5 or 5-6 as applicable. The SOF for concrete fill is calculated by dividing the basement-specific assigned dose in

Table 5-21 by 25 mrem/yr. The total dose for the Basement is then calculated by summing the SOF from all dose components using Equation 5-7 and multiplying by 25 mrem/yr.

Equation 5-7:

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

where:

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

5.5.6.1. Basement Surface Dose Calculation Including Multiple Survey Units

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

Table 5-22 Surface Survey Units Contributing to Each Basement

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

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

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

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

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

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

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

Equation 5-8

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

where:

$SOF_{B,B}$	=	total surface SOF including all surface survey units in basement (B)
$SA_{SUi,B}$	=	surface area of survey unit (i) in basement (B)
$SA_{Adjust,B}$	=	adjusted surface area for DCGL calculation (Table 5-23) for basement (B)
$SOF_{Bi,B}$	=	SOF_B for survey unit (i) in basement (B)

5.6. Final Status Survey (FSS) Design

FSS design is the process used to generate FSS packages and sample plans that when implemented, are designed to demonstrate compliance with the dose-based unrestricted release criteria at ZSRP. Survey design in this section specifically pertains to open land survey units and buried pipe; however the application of survey planning, survey package development, DQOs, data quality, investigations and data assessment as specified in this section is applicable to all FSS, including basement surfaces as described section 5.5. Buried piping is defined as pipe that runs through soil and is addressed in section 5.7.1.9. The FSS design for basement surfaces, embedded pipe and penetrations is described in section 5.5.

5.6.1. Survey Planning

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

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

The FSS process consists of four principal elements:

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

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

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

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

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

- DQOs are properly defined and derived;
- the plan is correctly implemented as prescribed;
- data and samples are collected by individuals with the proper training using approved procedures;

- instruments are properly calibrated and source checked;
- collected data are validated, recorded, and stored in accordance with approved procedures;
- documents are properly maintained; and,
- corrective actions are prescribed, implemented and followed up, if necessary.

The initial open land survey units and survey unit classifications that will be used for the FSS of Zion are presented in LTP Chapter 2, section 2.1.6 and Table 2-4 and shown on Figures 2-4, 2-5, 2-6, and 2-7. A FSS Package will be prepared for each applicable survey unit. This survey package is a collection of documentation detailing FSS Sample Plan survey design, survey implementation and data evaluation. A FSS Package will contain one or more FSS Sample Plans. FSS Packages shall be controlled in accordance with the record quality requirements of Zion*Solutions* QAPP.

5.6.2. Data Quality Objectives

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

5.6.2.1. State the Problem

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

5.6.2.2. Identify the Decision

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

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

5.6.2.3. Identify Inputs to the Decision

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

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

The following information will be utilized to support the decision:

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

5.6.2.4. Define the Study Boundaries

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

5.6.2.5. Develop a Decision Rule

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

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

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

5.6.2.6. Specify Limits on Decision Errors

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

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

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

Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors are considered acceptable. The upper bound corresponds to the release criteria. The Lower Bound of the Gray Region (LBGR) is determined as another limit on decision error. LBGR is influenced by a parameter known as the relative shift. The relative shift is the $DCGL_W$ minus the LBGR (i.e., the width of the Gray Region) divided by the standard deviation of the data set used to design the survey. In accordance with NUREG-1757, Appendix A, the LBGR should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit. However, if no other information is available regarding the survey unit, the LBGR may be initially set equal to 0.5 times the applicable $DCGL_W$. However, if the relative shift exceeds a value of 3, then the LBGR should be adjusted until the relative shift value is equal to 3. The adjustment of decision errors is discussed in more detail in section 5.6.4.

Sample uncertainty is controlled by collecting a small frequency of additional samples from each survey unit. Analytical uncertainty is controlled by using appropriate instrumentation, methods, techniques, training, and Quality Control (QC). The MDC values for individual radionuclides using specific analytical methods will be established. Uncertainty in the decision to release areas for unrestricted use is controlled by the number of samples and/or measurement points in each survey unit and the uncertainty in the estimate of the mean radionuclide or gross radioactivity concentrations. The specific types of instruments that can be used for the FSS of Zion and their respective MDC values are presented in section 5.8 and Tables 5-27 and 5-28.

Graphing the probability that a survey unit does not meet the release criteria may be used during FSS. This graph, known as a power curve, can be performed retrospectively (i.e., after FSS) using actual

measurement data. This retrospective power curve is a tool that can be used to demonstrate that the DQOs are met when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria).

5.6.2.7. Optimize the Design for Obtaining Data

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

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

5.6.3. Area Preparation: Turnover and Control Measures

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

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

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

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

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

5.6.4. Final Status Survey Design Process

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

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

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

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

5.6.4.1.1. Decision Errors

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

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

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

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

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

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

5.6.4.1.2. Unity Rule

The unity rule or SOF, as discussed in section 5.2.12, will be used for the survey planning and data evaluations for soil sample analyses since multiple radionuclide-specific measurements will be performed. As a result, the evaluation criteria and data must be normalized in order to accurately compare and relate the various data measurements to the release criteria.

5.6.4.1.3. Gray Region

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

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

5.6.4.1.4. Upper Bound of the Gray Region

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

5.6.4.1.5. Lower Bound of the Gray Region

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

5.6.4.1.6. Relative Shift

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

5.6.4.2. Statistical Tests

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

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

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

By reading the relative shift from the left side of the Table 5-5 of MARSSIM and cross referencing to the specified decision errors, the number of sampling and measurement locations can be determined. The specified number within the table includes the recommended 20 percent adjustment or increase to attain the desired power level with the statistical tests and allow for possible lost or unusable data. Equation 5-2 of MARSSIM may alternatively be used to calculate the number of sampling and measurement locations. The result will be increased by 20 percent. The sample size calculations will be performed using a specially designed software package such as COMPASS, hand calculations and/or spreadsheets.

5.6.4.3. Small Areas of Elevated Activity

Section 2.5.1.1 of MARSSIM addresses the concern of small areas of elevated radioactivity in the survey unit. Rather than using statistical methods, a simple comparison to an investigation level is used to assess the impact of potential elevated areas. This is referred to as the EMC. The investigation level for this comparison is the $DCGL_{EMC}$, which is the $DCGL_w$ modified by an AF to account for the small

area of the elevated radioactivity. The area correction is used because the exposure assumptions are the same as those used to develop the $DCGL_W$. Note that at ZSRP, the consideration of small areas of elevated radioactivity will only be applied to Class 1 open land (soil) survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_W$. For all other media (structural surfaces, embedded pipe, penetrations and/or buried pipe), any residual radioactivity identified by a FSS measurement at concentrations in excess of the respective Base Case DCGL will be remediated.

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

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

Equation 5-9

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

where:

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

The AF is selected from Tables 5-16 and 5-17 for soils corresponding to the bounded area (A) calculated. If the calculated bounded area (A) falls between two area categories on Tables 5-16 and 5-17, then the larger of the two areas will be selected along with the corresponding AF. $DCGL_{EMC}$ is then derived by multiplying the selected AF by the applicable $DCGL_W$.

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

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

Equation 5-10

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

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

Equation 5-11

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

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

5.6.4.4. Scan Coverage

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

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

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

5.6.4.5. Reference Grid, Sampling and Measurement Locations

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

Table 5-24 Recommended Survey Coverage

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

5.6.4.5.1. Reference Grid

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

5.6.4.5.2. Systematic Sampling and Measurement Locations

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

Equation 5-12

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

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

where:

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

Once the grid spacing is established, a random starting point will be established for the survey pattern using a random number generator. Starting from this randomly-selected location, a row of points will then be established parallel to one of the survey unit axes at intervals of *L*. Additional rows will then be

added parallel to the first row. For a triangular grid, additional rows will be added at a spacing of 0.866*L* from the first row, with points on alternate rows spaced mid-way between the points from the previous row. For a square grid, points and rows will be spaced at intervals of *L*.

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

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

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

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

5.6.4.6. Investigation Process

During the FSS, any areas of concern will be identified and investigated. This will include any areas as identified by the surveyor in real-time during the scanning, any areas identified during post-processing and reviewing of scan survey data, and any results of soil or bulk material analyses that exceed the Operational DCGL. Based on this review, the suspect areas will be addressed by further biased surveys and sampling as necessary. The applicable investigation levels are provided in Table 5-25.

Table 5-25 Investigation Levels

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

5.6.4.6.1. Remediation, Reclassification and Resurvey

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

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

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

The DQO process will be used to evaluate the remediation, reclassification and/or resurvey actions to be taken if an investigation level is exceeded. Based upon the failure of the statistical test or the results of an investigation, Table 5-26 presents actions that will be required.

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

Table 5-26 Remediation, Reclassification and Resurvey Actions

REMEDIATION			
Remediation Criteria			Proposed Remediation
Class 1 FSS Survey Unit	1) Passes Sign Test and the mean SOF for survey unit is less than or equal to unity (1) (SOF EMC for open land survey units or Equation 5-5 or 5-6 for structural survey units)		None
	2) Passes Sign Test and the mean SOF for survey unit is less than or equal to unity (1) with several elevated areas present that require remediation (>$DCGL_{EMC}$ for soils or Base Case DCGL for other media)		Spot Remediation & Resurvey under Existing Survey Design
	3) Does not pass Sign Test, or the mean SOF is greater than unity		General Remediation and Restart FSS under new Survey Design
Class 1 Basement FSS Unit	1) The mean inventory fraction (total mean dose for the survey unit divided by the dose criterion of 25 mrem/yr) is greater than or equal to one.		General Remediation and Restart FSS under new Survey Design
	2) The sum of the mean inventory fractions for each FSS unit contained within a building basement is greater than or equal to one.		
RECLASSIFICATION			
Reclassification Criteria			Proposed Action
Class 2 Survey Unit	One or several survey measurements (scan, sample or direct measurement) exceed the Operational DCGL or a portion of the survey unit is remediated.	The extent of the elevated area relative to the total area of the survey unit is minimal and the source of the residual radioactivity is known	Reclassify only the bounded discrete area of elevated activity to Class 1.
		The extent of the elevated area relative to the total area of the survey unit is minimal and the source of the residual radioactivity is unknown	Reclassify 2,000 m^2 for soils or 100 m^2 for structures around the area of elevated activity as Class 1.
		The extent of the elevated area relative to the total area of the survey unit is significant.	Reclassify the entire survey unit as Class 1.
Class 3 Survey Unit	One or several survey measurements (scan, sample or direct measurement) exceed 50% of the Operational DCGL or a portion of the survey unit is remediated.	The extent of the elevated area relative to the total area of the survey unit is minimal	Reclassify the area of elevated activity to Class 1 and create a Class 2 buffer zone of appropriate size around the area.
		The extent of the elevated area relative to the total area of the survey unit is significant.	Reclassify the area of elevated activity to Class 1 and create a Class 2 buffer zone of appropriate size around the area.
	One or several survey measurements (scan, sample or direct measurement) exceed 1% of the Operational DCGL	The extent of the elevated area relative to the total area of the survey unit is minimal	Reclassify the area of elevated activity to Class 2.
		The extent of the elevated area relative to the total area of the survey unit is significant.	For soils, reclassify 10,000 m^2 around the area of elevated activity to Class 2. For structures, reclassify 1,000 m^2 around the area of elevated

			activity to Class 2.

Table 5-26 (continued) Remediation, Reclassification and Resurvey Actions

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

5.7. Final Status Survey Implementation

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

5.7.1. Survey Methods

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

5.7.1.1. Scanning

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

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

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

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

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

5.7.1.2. Beta-Gamma Scanning

Scanning is performed in order to locate small, elevated areas of residual activity above the investigation level. Miscellaneous materials will be scanned for beta-gamma radiation with appropriate instruments such as those listed in Table 5-27. The measurements will typically be performed at a distance of 1 cm or less from the surface and at a scan speed of 5 cm/sec for hand-held instruments.

5.7.1.3. Volumetric Sampling

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

Trained and qualified individuals will collect and control samples. All sampling activities will be performed under approved procedures. ZSRP will utilize a chain-of-custody (COC) process to ensure sample integrity.

Quality assurance (QA) requirements for FSS activities that apply to sample collection (e.g., split samples, duplicates, etc.) and onsite and offsite laboratories employed to analyze samples as a part of the FSS process will be controlled by approved procedures, in conformance with the QAPP and is further described in section 5.9. Performance of laboratories will be verified periodically in accordance with the QAPP and ZS-QA-10, *"Quality Assurance Project Plan - Zion Station Restoration Project"* (Reference 5-22).

5.7.1.4. Fixed Measurements

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

5.7.1.5. Surface Soils

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

5.7.1.5.1. Gamma Scans of Surface Soils

Gamma scans will be performed over open land surfaces to identify locations of residual surface activity. NaI gamma scintillation detectors (typically 2" x 2") will be used for these scans. Zion*Solutions* TSD #11-004 presents the response and scan MDC of the Ludlum Model 44-10 NaI detectors to Co-60 and Cs-137 radionuclides when used for scanning surface soils. Where appropriate,

gamma emitters such as Co-60 and Cs-137 will be used as surrogates to infer the amount of any HTD radionuclide(s) that may be present in the distribution in accordance with section 5.2.12.

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

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

5.7.1.5.2. Sampling of Surface Soils

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

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

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

5.7.1.6. Subsurface Soils

Subsurface soil refers to soil that resides at a depth greater than 15 cm below the final configuration of the ground surface or soil that will remain beneath structures such as basement floors/foundations or pavement at the time of license termination.

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

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

radioactivity at the Operational DCGL can be used to scan the exposed soils in an open excavation to identify the presence or absence of soil contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the Operational DCGL, additional confirmatory investigation and analyses of soil samples of the suspect areas will be performed.

5.7.1.6.1. Scanning of Subsurface Soils during FSS

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

5.7.1.6.2. Sampling of Subsurface Soils during FSS

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

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

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

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

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

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

5.7.1.6.3. Sampling of Subsurface Soils below Structure Basement Foundations

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

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

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

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

ZSRP will assess the dose consequences of the subsurface soil contamination or will remediate as necessary.

5.7.1.7. Reuse of Excavated Soils

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

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

5.7.1.8. Pavement Covered Areas

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

5.7.1.9. Buried Piping

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

The survey of buried pipe will be achieved in the same manner as described for the survey of embedded pipe as discussed in section 5.5.5. The radiological survey of pipe system interiors involves the insertion of appropriately sized detectors into the pipe interior by a simple "push-pull" methodology,

whereby the position of the detector in the piping system can be easily determined in a reproducible manner.

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

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

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

5.7.1.10. Groundwater

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

5.7.1.11. Sediments and Surface Water

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

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

5.7.1.12. Survey Considerations for Buildings, Structures and Equipment

All above grade buildings will be removed in the end-state for ZSRP. The survey approach that will be used to radiologically assess the residual radioactivity in below-grade basement surfaces is presented in section 5.5 of this FSS Plan. The FSS of minor solid structures, such as but not limited to the Switchyard, the microwave tower, and the Sewage Lift Station, telephone poles, fencing, culverts, duct

banks and electrical conduit will be included in the open land FSS unit in which they reside. These items will be scanned in accordance with recommended survey coverage in Table 5-24.

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

5.8. Final Status Survey Instrumentation

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

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

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

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

5.8.1. Instrument Selection

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

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

Other measurement instruments or techniques may be utilized. The acceptability of additional or alternate instruments or technologies for use in the FSS will be justified in a technical basis evaluation document prior to use. Technical basis evaluations for alternate final status survey instruments or techniques will be provided for NRC review 30 days prior to use. This evaluation will include the following:

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

Instrumentation currently proposed for use in the FSS is listed in Table 5-27. Instrument MDCs are discussed in section 5.8.4 and nominal MDC values for the proposed instrumentation are presented in Table 5-28.

5.8.2. Calibration and Maintenance

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

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

Table 5-27 Typical FSS Survey Instrumentation

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

Table 5-28 Typical FSS Instrument Detection Sensitivities

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

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

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

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

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

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

5.8.3. Response Checks

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

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

5.8.4. Measurement Sensitivity

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

5.8.4.1. Total Efficiency

Instrument efficiencies (ε_i) for surface measurements are derived from the surface emission rate of the radioactive source(s) used during the instrument calibration. Total efficiency (ε_t) is calculated by multiplying the instrument efficiency (ε_i) by the surface efficiency (ε_s) commensurate with the radionuclide's alpha or beta energy using the guidance provided in ISO 7503-1, *"Evaluation of surface contamination - Part 1: Beta-emitters (maximum beta energy greater than 0.15 MeV) and alpha-emitters"* (Reference 5-23).

5.8.4.2. Static Minimum Detectable Concentration

For static (direct) surface measurements with conventional detectors, such as those listed in Table 5-27, the MDC is calculated using the following equation:

Equation 5-13

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

where:

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

5.8.4.3. Beta-Gamma Scan Measurement Minimum Detectable Concentration

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

Equation 5-14

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

where:

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

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

is the Ludlum Model 43-37 with a window width of 13.35 cm which results in a scan speed of 5.25 inches per second. The source efficiency term (ε_s) will be selected to account for effects such as self-absorption, using the values found in Tables 2 and 3 in ISO 7503-1.

5.8.4.4. Gamma Scan Measurement Minimum Detectable Concentration

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

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

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

This method represents the surface scanning process for land areas defined in NUREG-1507 and is the basis for calculation of the scanning detection sensitivity (scan MDC). The gamma scan MDC is discussed in detail in Zion*Solutions* TSD 11-004, which examines the gamma sensitivity for 5.08 by 5.08 cm (2" x 2") NaI detectors to several radionuclide mixtures of Co-60 and Cs-137 using sand (SiO_2) as the soil base. TSD 11-004 derives the MDC for the radionuclide mixtures at various detector distances and scan speeds. The model in TSD 11-004 uses essentially the same geometry configuration as the model used in MARSSIM. TSD 11-004 provides MDC values for the expected ZSRP soil mixture based on detector background condition, scan speed, soil depth (15 cm), soil density (1.6 g/cm^3) and detector distance to the suspect surface.

5.8.4.5. HPGe Spectrometer Analysis

The onsite Zion*Solutions* laboratory maintains gamma isotopic spectrometers that are calibrated to various sample geometries, including a one-liter marinelli geometry for soil analysis. The geometries are created using the Canberra LABSOCS software. These systems are calibrated using a NIST-traceable mixed gamma source. On-site laboratory counting systems are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil; this is calculated in accordance with the following equation:

Equation 5-15

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

where:

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

5.8.4.6. Pipe Survey Instrumentation

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

5.9. Quality Assurance

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

Zion*Solutions* has developed and is implementing a comprehensive QA Program to assure conformance with established regulatory requirements. The quality requirements and quality concepts are presented in ZS-QA-10 which adequately encompasses all risk-significant decommissioning activities. The participants in the Zion*Solutions* QA Program assure that the design, procurement, construction, testing, operation, maintenance, repair, modification, dismantlement and remediation of nuclear reactor components are performed in a safe and effective manner.

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

The QAPP has been prepared to ensure the adequacy of data being developed and used during FSS. It supplements the quality requirements and quality concepts presented in ZS-QA-10. Compliance with the QAPP will serve to ensure that FSS are performed by trained individuals using approved written procedures and properly calibrated instruments that are sensitive to the suspected ROC. In addition, QC measures will be taken to obtain quantitative information to demonstrate that measurement results have

the required precision and are sufficiently free of errors to accurately represent the area being investigated. QC checks will be performed as prescribed by the QAPP for both field measurements and laboratory analysis. Effective implementation of FSS operations will be verified through periodic audit and surveillance activities, including field walk-downs by Characterization/License Termination group management and program self-assessments. Corrective actions will be prescribed, implemented, and verified in the event any deficiencies are identified. These measures will apply to any applicable services provided by off-site vendors, as well as on-site sub-contractors.

5.9.1. Project Management and Organization

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

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

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

5.9.2. Quality Objectives and Measurement Criteria

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

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

The following describe the basic elements of the QAPP.

5.9.2.1. Written Procedures

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

5.9.2.2. Training and Qualifications

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

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

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

5.9.2.3. Measurement and Data Acquisitions

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

5.9.2.4. Instrument Selection, Calibration and Operation

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

5.9.2.5. Chain of Custody

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

5.9.2.6. Control of Consumables

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

5.9.2.7. Control of Vendor-Supplied Services

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

5.9.2.8. Database Control

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

5.9.2.9. Data Management

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

5.9.3. Measurement/Data Acquisition

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

5.9.3.1. Replicate Measurements and Surveys

Replicate measurements will be performed on 5% of the static and scan locations in each applicable FSS package in locations chosen at random.

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

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

5.9.3.2. Duplicate and Split Samples

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

The NRC Inspection Procedure No. 84750 *"Radioactive Waste Treatment, and Effluent and Environmental Monitoring"* (Reference 5-24) will be used to determine the acceptability of split and

duplicate sample analyses. The sample results will be compared to determine accuracy and precision. Agreement is ultimately determined when the same conclusion is reached for each compared result. If the split sample or duplicate sample results do not agree, then further evaluations will be performed.

5.9.3.3. Field Blanks and Spiked Samples

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

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

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

5.9.3.4. QC Investigations

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be performed in accordance with approved procedures; and if necessary, shall warrant a condition report in accordance with Zion*Solutions* procedure ZS-AD-08, *"Corrective Action Program"* (Reference 5-25). The investigation will include verification that the proper data sets were compared, the relevant instruments were operating properly and the survey/sample points were properly identified and located. Relevant personnel will be interviewed to determine if proper instructions and procedures were followed and proper measurement and handling techniques were used including CoC, where applicable. If the investigation reveals that the data is suspect and may not represent the actual conditions, additional measurements will be taken. Following the investigation, a documented determination is made regarding the usability of the survey data and if the impact of the discrepancy adversely affects the decision on the radiological status of the survey unit.

5.9.4. Assessment and Oversight

5.9.4.1. Assessments

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

5.9.4.2. Independent Review of Survey Results

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

5.9.4.3. Corrective Action Process

The corrective action process, already established as part of the site QA Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. The process will be conducted in accordance with ZS-AD-08, which describes the methods used to identify potential

conditions adverse to quality (CAQ), condition reporting, self-assessment resolution and corrective action issues related to FSS. The CAQ evaluation effort is commensurate with the classification of the CAQ and could include root cause determination, extent of condition reviews, and preventive and remedial actions.

5.9.4.4. Corrective Action Process

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

5.9.5. Data Validation

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

5.9.6. NRC Confirmatory Measurements

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

5.10. Final Status Survey Data Assessment

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

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

- determining if the residual radioactivity levels in the survey unit meet the applicable release criterion, and if any areas of elevated radioactivity exist.

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

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

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

Survey results will be converted to appropriate units of measure (e.g., dpm/100 cm^2, pCi/g, pCi/m^2) and compared to investigation levels to determine if the action levels for investigation have been exceeded. Measurements exceeding investigation action levels will be investigated. If confirmed within a Class 1 soil survey unit, the location of elevated concentration may be evaluated using the EMC, or the location may be remediated and re-surveyed. If measurements exceeding investigation action levels are confirmed within a Class 2 or 3 survey unit, in most cases, the entire survey unit will be reclassified and a re-survey performed consistent with the change in classification.

5.10.1. Review of DQOs and Survey Plan Design

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

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

will be resolved and documented prior to proceeding with data analysis. Data assessment will be performed by trained personnel using the approved procedure.

5.10.2. Preliminary Data Review

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

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

5.10.2.1. <u>Data Validation</u>

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

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

If the data review criteria are not met, the discrepancy(s) will be evaluated and the decision to accept or reject the data will be documented in accordance with approved procedures. A condition report generated in accordance with ZS-AD-08 will be used to document and resolve discrepancies as applicable.

5.10.2.2. Graphical Data Review

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

5.10.2.2.1. Posting Plot

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

5.10.2.2.2. Frequency Plot

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

5.10.3. Applying Statistical Test

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

5.10.3.1. Sum-of-Fractions

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

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

Equation 5-16

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

where:

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

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

5.10.3.2. Sign Test

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

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

5.10.4. Elevated Measurement Comparison Evaluation

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

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

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

The average activity of each identified elevated areas is determined as well as the average activity value for the survey unit. The survey unit average activity value is divided by the Base Case DCGL, the survey unit average value is then subtracted from the average activity value for the elevated area and the result is divided by the appropriate $DCGL_{EMC}$. The net average activity for each identified elevated

area is evaluated against its applicable $DCGL_{EMC}$. The fractions are summed and the result must be less than unity for the survey unit to pass. This is summarized in the equation as follows;

Equation 5-17

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

where:

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

5.10.5. Data Conclusions

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

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

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

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

If the retrospective power analysis indicates insufficient power, then an assessment will be performed to determine whether the observed median concentration and/or observed standard deviation are significantly different from the estimated values used during the DQO process. The assessment will

identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions can include failing the unit and starting the DQO process over, remediating some or all of the survey unit and starting the DQO process over and adjusting the LBGR to increase sample size. For example, the assessment determines that the median residual concentration in the survey unit exceeds the Operational DCGL or is higher than was estimated and planned for during the DQO process. A likely course of action might be to fail the unit or remediate and resurvey using a new sample design.

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

5.11. Final Radiation Survey Reporting

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

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

5.11.1. FSS Unit Release Records

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

- Survey unit description, including unit size, descriptive maps, plots or photographs and reference coordinates;
- Classification basis, including significant HSA and characterization data used to establish the final classification;
- DQOs stating the primary objective of the survey;
- Survey design describing the design process, including methods used to determine the number of samples or measurements required based on statistical design, the number of biased or judgmental samples or measurements selected and the basis, method of sample or measurement locating, and a table providing a synopsis of the survey design;

- Survey implementation describing survey methods and instrumentation used, accessibility restrictions to sample or measurement location, number of actual samples or measurements taken, documentation activities, QC requirements and scan coverage;
- Survey results including types of analyses performed, types of statistical tests performed, surrogate ratios, statement of pass or failure of the statistical test(s);
- QC results to include discussion of split samples and/or QC replicate measurements;
- Results of any investigations;
- Any remediation activities, both historic and resulting from the performance of the FSS;
- Any changes from the FSS survey design including field changes;
- DQA conclusions;
- Any anomalies encountered during performance of the survey or in the sample results; and,
- Conclusion as to whether or not the survey unit satisfied the release criteria and whether or not sufficient power was achieved.

5.11.2. FSS Final Reports

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

- A brief overview discussion of the FSS Program including descriptions regarding survey planning, survey design, survey implementation, survey data assessment, and QA and QC measures;
- A description of the site, the applicable survey area(s) and survey unit(s), a summary of the applicable HSA information, conditions at the time of survey, identification of potential contaminants, and radiological release criteria;
- A discussion regarding the DQOs, survey unit designation and classification, background determination, FSS plans, survey design input values and method for determining sample size, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), ISOCS Efficiency Calibration geometry, survey methodology, QC surveys, and a discussion of any deviations during the performance of the FSS from what was described in this LTP;
- A description of the survey findings including a description of surface conditions, data conversion, survey data verification and validation, evaluation of number of sample/measurement locations, a map or drawing showing the reference system and random start systematic sample locations, and comparison of findings with the appropriate Operational DCGL or Action Level including statistical evaluations.

- Description of any judgmental and miscellaneous sample data collected in addition to those required for performing the statistical evaluation.
- Description of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the Operational DCGL.
- If survey unit fails the statistical test, a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure and the impact that the failure has on the conclusion that the facility is ready for final radiological surveys, and a discussion of the impact of the failure on survey design and result for other survey units.
- Description of how good housekeeping and ALARA practices were employed to achieve final activity levels.

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

5.12. Surveillance Following FSS

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

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

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

A routine surveillance will be performed in each completed FSS unit on a semi-annual basis. In addition, a surveillance will be performed when an activity occurs that may have radiologically impacted the survey unit (e.g., transiting a radioactive material package though an FSS area, etc...). These surveillances will be controlled and documented in accordance with the QAPP and approved

procedures. If a routine surveillance identifies physical observations and/or radiological scan measurements that require further investigation, then FSS will be repeated in the affected survey unit.

5.13. References

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

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

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

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

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

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

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

5-8 Zion*Solutions* Technical Support Document 14-016, Revision 0, "Description of Embedded Piping, Penetrations and Buried Piping to Remain in Zion End State"

5-9 Zion*Solutions* Technical Support Document 11-001, Revision 1, "Potential Radionuclides of Concern during the Decommissioning of Zion Station"

5-10 Zion*Solutions* Technical Support Document 14-019, Revision 2, "Radionuclides of Concern for Soil and Basement Fill Model Source Terms"

5-11 Zion*Solutions* Technical Support Document 17-004, Revision 3, "Operational Derived Concentration Guideline Levels for FSS"

5-12 Zion*Solutions* Technical Support Document 14-011, Revision 0, "Soil Area Factors"

5-13 Zion*Solutions* Technical Support Document 14-015, Revision 3, "Buried Pipe Dose Modeling & DCGLs"

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

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

5-16 Zion*Solutions* Technical Support Document 11-004, Revision 0, "Ludlum Model 44-10 Detector Sensitivity"

5-17 Zion*Solutions* ZS-LT-01, Revision 6, "Quality Assurance Project Plan (for Characterization and FSS)" (QAPP)

5-18 Zion*Solutions* ZS-LT-02, Revision 3,"Characterization Survey Plan" – July 2015

5-19 Zion*Solutions* procedure ZS-LT-100-001-001, Revision 2, "Characterization Survey Package Development"

5-20 Zion*Solutions* TSD 14-022, Revision 1, "Use of In-Situ Gamma Spectroscopy for Source Term Survey of End State Structures"

5-21 Zion*Solutions* Technical Support Document 10-002, Revision 1, "Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"

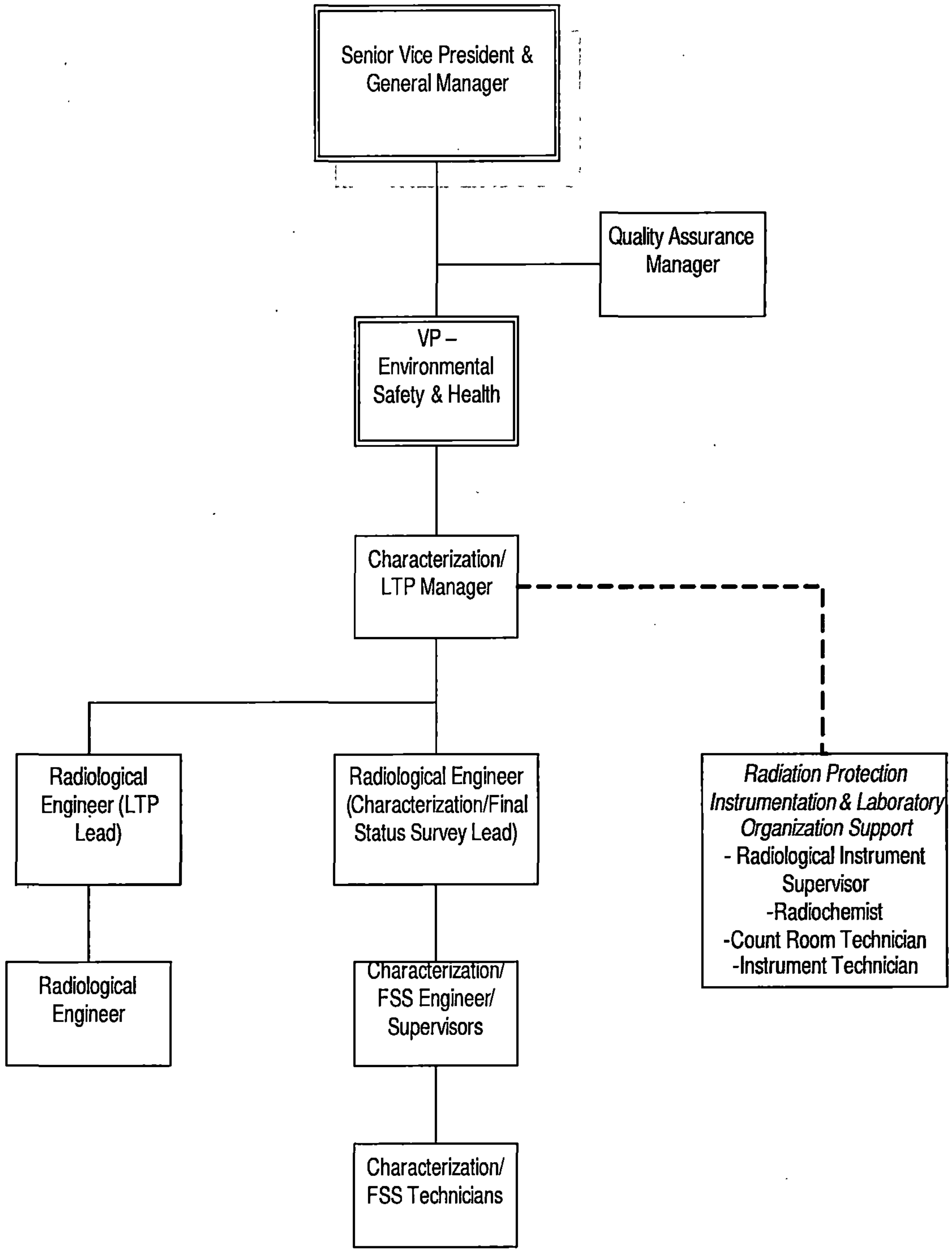
5-22 Zion*Solutions* ZS-QA-10, Revision 9, "Quality Assurance Project Plan - Zion Station Restoration Project"

5-23 International Standard ISO 7503-1, Part 1, "Evaluation of Surface Contamination, Beta-Emitters (maximum beta energy greater than 0.15 MeV) and Alpha-Emitters" – August 1998

5-24 U.S. Nuclear Regulatory Commission Inspection Procedure No. 84750 "Radioactive Waste Treatment, and Effluent and Environmental Monitoring" – March 1994

5-25 ZionSolutions ZS-AD-08, "Corrective Action Program"

Figure 5-1 Characterization/LTP/FSS Organization Chart



ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 6, REVISION 2
COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

:ZIONSOLUTIONS LLC
An EnergySolutions Company

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LIST OF ACRONYMS AND ABBREVIATIONS

AF	Area Factor
ALARA	As Low As (is) Reasonable Achievable
AMSL	Above Mean Sea Level
ANL	Argonne National Laboratory
BFM	Basement Fill Model
CRA	Conestoga Rovers & Associates
DCGL	Derived Concentration Guideline Level
DCF	Dose Conversion Factor
DUST-MS	Disposal Unit Source Term - Multiple Species
EPA	Environmental Protection Agency
FGR	Federal Guidance Report
FOV	Field of View
FSS	Final Status Survey
GW	Groundwater
HSA	Historical Site Assessment
HTD	Hard-to-Detect
IC	Insignificant Contributor
ISFSI	Independent Spent Fuel Storage Installation
ISOCS	In-Situ Object Counting System
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimal Detectable Concentration
NRC	The U.S. Nuclear Regulatory Commission
ODCM	Off-site Dose Calculation Manual
PRCC	Partial Rank Correlation Coefficient
RASS	Remedial Action Support Surveys
REMP	Radiological Environmental Monitoring Program
RESRAD	RESidual RADioactive materials
ROC	Radionuclides of Concern
SFP	Spent Fuel Pool
TEDE	Total Effective Dose Equivalent
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

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6. COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1. Site Release Criteria

The site release criteria for the Zion Station Restoration Project (ZSRP) are the radiological criteria for unrestricted release specified in Title 10, Section 20.1402, of the *Code of Federal Regulations* (10 CFR 20.1402):

- Dose Criterion: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/yr, including that from groundwater sources of drinking water; and
- As Low As (is) Reasonable Achievable (ALARA) Criterion: The residual radioactivity has been reduced to levels that are ALARA.

Chapter 4 describes the methods and results for demonstrating compliance with the ALARA Criterion. This Chapter describes the methods for demonstrating compliance with the Dose Criterion.

6.2. General Site Description

This section provides a general description of the geology and hydrogeology at the Zion Nuclear Power Station (ZNPS) site. Land and groundwater use in the vicinity of site are also summarized. A detailed site description is provided in Zion*Solutions* TSD 14-003, Conestoga Rovers & Associates (CRA) Report: Conestoga Rovers & Associates (CRA) Report, "*Zion Hydrogeologic Investigation Report*" (Reference 6-1).

The ZNPS is located in Northeast Illinois approximately 40 miles north of Chicago, Illinois, and 42 miles south of Milwaukee, Wisconsin. ZNPS is in the extreme eastern portion of the city of Zion, (Lake County) Illinois, on the west shore of Lake Michigan approximately 6 miles NNE of the center of the city of Waukegan, Illinois, and 8 miles south of the center of the city of Kenosha, Wisconsin (see Figure 6-1). The ZNPS owner controlled area is shown in Figure 6-2, with a more detailed view of the "Security-Protected Area" shown in Figure 6-3.

6.2.1. Site Geology

The Site is underlain by overburden deposits and a regionally extensive sequence of consolidated sedimentary deposits. In descending order, the following overburden stratigraphic units have been identified:

- Upper sand unit (also known as the Shallow Aquifer): 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. This unit includes both native and fill sand. Depth ranges from the ground surface to an elevation of approximately 555 feet Above Mean Sea Level (AMSL).

- Upper silty clay unit: Hard silt, silty clay, clayey silt, and sandy silt which contain some sand and gravel and occasional cobbles and boulders. Depth ranges from approximately 525 feet to 555 feet AMSL.
- Lower sand unit: 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. Depth ranges from approximately 480 feet to 525 feet AMSL. This unit is discontinuous.

The lower unconsolidated sand unit layer overlies an upper bedrock layer. This upper bedrock layer is the Niagara Dolomite, a consolidated layer of carbonaceous marine sediments laid down in the Silurian Period. It is about 200 feet thick in the vicinity of ZNPS.

6.2.2. Site Hydrogeology

Two aquifer units are present in the overburden material, the upper sand unit and the lower sand unit. These two units are separated by a silty clay unit and together they comprise the shallow unconsolidated aquifer system. The silty clay unit (found under the upper sand unit) is approximately 30 feet thick and extends approximately 15 feet below the deepest structural feature at ZNPS. The silty clay unit acts as an aquitard and prevents vertical migration of groundwater. Therefore the underlying regional Silurian dolomite bedrock aquifer should not be in hydraulic communication with the upper sand unit at ZNPS.

6.2.3. Area Land Use

The ZNPS Facility is located on the shore of Lake Michigan, in the eastern portion of the City of Zion, and adjacent to the Illinois Beach State Park.

The Illinois Beach State Park is located along the Lake Michigan shoreline and is divided into a northern unit and a southern unit, with ZNPS situated between the two units. The Illinois Beach State Park encompasses 4,160 acres and received approximately 2.75 million visitors in 1998. The Park is considered a natural resource.

The land located to the west of ZNPS is generally undeveloped with a limited number of industrial/commercial facilities present along Deborah Avenue. Residential areas and the City of Zion downtown are located west of the Chicago & Northwestern Railroad, which is west of the Facility. The 2010 census listed the population of Zion as 24,413, with a population density of 2,489 per square mile. Lake Michigan borders the Facility to the east.

6.2.4. 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 mile north of the Site and extending 3,000 feet into the Lake. The City of Zion municipal code requires all improved properties to be connected to the City's water supply. The code states that it is "unlawful for any person to construct, permit or maintain a private well or water supply system within the City which uses groundwater as a potable water supply". There is an exception for some existing wells constructed prior to March 2, 2004. Notwithstanding the fact that current municipal code prohibits construction of residential wells, the conceptual model for dose

assessment of backfilled basements conservatively includes the installation of a water supply well on the site (see 6.5.3).

6.3. Basements and Structures to Remain after License Termination (End State)

The "End State" is defined as the configuration of the remaining below ground buildings, structures, piping and open land areas at the time of license termination.

The Lease Agreement between Zion*Solutions* and Exelon, Section 8.5 of Exhibit C, titled "Removal of Improvements; Site Restoration" integral to the *"Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement"* (Reference 6-2) requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least three feet below grade [designated as an elevation of 588 foot Above Mean Sea Level (AMSL)]. All contaminated systems, components, piping, buildings and structures above 588 foot elevation will be removed during decommissioning and disposed of as waste. The decommissioning approach for ZSRP also calls for the beneficial reuse of concrete from building demolition as clean fill. Concrete that meets the non-radiological definition of Clean Concrete Demolition Debris and where radiological surveys demonstrate that the concrete meets the 10 CFR 20.1402 criteria for unrestricted use will be used.

In both Containment basements (Unit 1 and Unit 2), all concrete will be removed from the inside of the steel liner above 565 foot elevation leaving only the remaining exposed liner below the 588 foot elevation and the concrete in the area under the vessel including the In-Core Instrument Shaft leading to the under vessel area (designated as the "Under-Vessel" area), and the structural concrete outside of the liner. In the Auxiliary and Turbine Building basements, all internal walls and floors will be removed, leaving only the reinforced concrete floors and outer walls of the building structures. For the Fuel Handling Building, the only portion of the structure that will remain is the lower 12 feet of the Spent Fuel Pool (SFP) below the 588 foot elevation and the concrete structure of the Fuel Transfer Canals after the steel liners have been removed. There are five additional below ground structures that will remain, including the lower concrete portions of the Waste Water Treatment Facility (WWTF), Crib House/Forebay, Main Steam Tunnels, Circulating Water Intake Piping and Circulating Water Discharge Tunnels. The basements and structures that will remain at license termination as part of the End State are listed in Table 6-1. Figure 6-4 provides a simple plan view of the End State. A series of four cross-sections showing elevation views of the basements and structures to remain is provided in Figures 6-5 to 6-8.

The End State will also include a range of buried pipe, embedded pipe and penetrations. For the purpose of this License Termination Plan (LTP), buried pipe is defined as pipe that runs through soil, embedded pipe is defined as pipe that runs vertically through a concrete wall or horizontally through a concrete floor, and a penetration is defined as a pipe (or remaining pipe sleeve or concrete if the pipe is removed) that traverses a wall and is cut on both sides of the wall. The list of penetrations and embedded pipe to remain is provided in Zion*Solutions* TSD 14-016, "Description of Embedded Piping, Penetrations and Buried Piping to Remain in Zion End State" (Reference 6-3). The list of end-state embedded pipe, buried pipe and penetrations presented in Attachment F to TSD 14-016 is intended to be a bounding end-state condition. No pipe that is not listed in Attachment F will be added to the end-state condition however, pipe can be removed from the list and disposed of as waste.

Table 6-1 Basements and Below Ground Structures included in the ZNPS End State

Basement/Structure	Material remaining	Lowest Internal Elevation (feet AMSL)
Unit 1 Containment Building	Steel Liner over Concrete	568
Unit 2 Containment Building	Steel Liner over Concrete	568
Auxiliary Building	Concrete	542
Turbine Building	Concrete	560
Crib House and Forebay	Concrete	552
WWTF	Concrete	577
Spent Fuel Pool	Concrete	576
Main Steam Tunnels (Unit 1 and Unit 2)	Concrete	570
Circulating Water Intake Piping	Steel Pipe	(Site) 552/(Lake) 543
Circulating Water Discharge Tunnels	Concrete	(Site) 552/(Lake) 543

There is limited potential for contaminated surface or subsurface soil to be present at ZNPS based on the findings of the "*Zion Station Historical Site Assessment*" (HSA) (Reference 6-5) and the results of extensive characterization performed in 2013. The results of the characterization surveys are summarized in Chapter 2 of this LTP. There has been no groundwater contamination identified by the groundwater monitoring program at ZNPS. The monitoring program and results are described in the TSD 14-003. The groundwater monitoring results are summarized in LTP Chapter 2, section 2.3.6.5.

After all demolition, remediation and backfill is completed, the 10 CFR Part 50 license will be reduced to the area around the Independent Spent Fuel Storage Installation (ISFSI) and the site will be transferred back to Exelon under the 10 CFR Part 50 license.

6.4. Dose Modeling Overview

Dose modeling is performed to demonstrate that remaining residual radioactivity does not result in a dose exceeding the 25 mrem/yr criterion. The Average Member of the Critical Group (AMCG) is assumed to be the Resident Farmer. This section provides a general overview of the dose modeling approach.

There are four potential sources of residual radioactivity that are categorized as follows for the purpose of dose modeling; backfilled basements, buried pipe, soil, and groundwater. As noted above, there is no indication that significant contamination is currently present in surface or

subsurface soil or will be present in the End State. The potential for groundwater contamination is also very low but groundwater dose conversion factors are included as a contingency. The dose from each of the four sources will be summed as applicable.

The backfilled basement dose includes the dose from structure surfaces (wall and floors), embedded pipe, and penetrations in the applicable basement. The dose margin applied to clean concrete fill will also be added to the applicable basement.

An overview of the dose assessment methods for the four sources, and embedded pipe and penetrations, is provided below. Detailed descriptions are provided in subsequent sections.

6.4.1. Backfilled Basement Structure Surfaces

The dose model for backfilled basements and structures to remain below 588 foot elevation at ZNPS (which are generally referred to as "Basements" in this LTP Chapter) is designated as the Basement Fill Model (BFM). The BFM calculates the annual dose to the AMCG from surface and volumetric residual radioactivity remaining in the basement and structures listed in Table 6-1.

The End State Basements will be comprised of steel and/or concrete structures which will be covered by at least three feet of clean soil and physically altered to a condition which would not realistically allow the remaining structures, if excavated, to be occupied. The exposure pathways in the BFM are associated with residual radioactivity in floors and walls that is released through leaching into water contained in the interstitial spaces of the fill material. The BFM assumes that the inventory of residual radioactivity in a given building is released either instantly or over time by diffusion, depending on whether the activity is surficial or volumetric, respectively.

The activity released into the fill water will adsorb onto the clean fill, as a function of the radionuclide-specific distribution coefficients, resulting in equilibrium concentrations between the fill and the water. Consequently, the only potential exposure pathways after backfill, assuming the 'as-left' geometry, are associated with the residual radioactivity in the water contained in the fill.

A water supply well is assumed to be installed within the fill of the Basement. The well water is then used for drinking, garden irrigation, pasture/crop irrigation, and livestock water supply in the Resident Farmer scenario.

The BFM is implemented using two computational models. The Disposal Unit Source Term - Multiple Species (DUST-MS) model is used to calculate the maximum water concentrations in the fill material of each basement for a given inventory of residual radioactivity (pCi/L per mCi). The RESidual RADioactive materials (RESRAD) v7.0 model is used to determine the dose to the Resident Farmer as a function of the water concentration (mrem/yr per pCi/L). BFM Groundwater (GW) Dose Factors are then calculated for each Basement and each Radionuclide of Concern (ROC) by combining the results of the two models with units of mrem/yr per mCi total inventory.

The BFM also includes the dose from drilling spoils that are brought to the surface during the well installation, which is assumed to be at the time of maximum projected future groundwater concentrations. The drilling spoils are assumed to be comprised of fill material containing residual radioactivity at the maximum equilibrium concentrations. Any activity remaining in the

concrete is also included in the drilling spoils source term. BFM Drilling Spoils Dose Factors are also calculated in units of mrem/yr per mCi total inventory.

The final outputs of the BFM are the Basement Derived Concentration Guideline Levels (DCGL), in units of pCi/m^2, which are calculated using the BFM GW and BFM Drilling Spoils Dose Factors. The DCGLs for basement structure surfaces are calculated separately for the GW and Drilling Spoils scenarios and for the summation of both scenarios. The individual Basement Scenario DCGLs for structure surfaces are defined as "$DCGL_{BS}$" and represent a dose of 25 mrem/yr for each scenario individually. The basement summation DCGL for basement structure surfaces includes the dose from both the GW and Drilling Spoils scenarios and represents a dose of 25 mrem/yr from both scenarios combined. The summation DCGL for basement structure surfaces is designated as the "$DCGL_B$" and is used during FSS to demonstrate compliance (equivalent to the $DCGL_W$ as defined in MARSSIM). The DCGLs are radionuclide-specific concentrations that represent the 10 CFR 20.1402 dose criterion of 25 mrem/yr and are calculated for each ROC and each backfilled Basement.

Basement $DCGL_B$ values were calculated for each of the Basements listed in Table 6-1. The Circulating Water Discharge Tunnels were accounted for by adding the surface area (and corresponding source term) to the Turbine Basement during the DCGL calculation (section 6.6.8). The Circulating Water Intake Piping was accounted for by adding the surface area to the Crib House/Forebay Basement during the $DCGL_B$ calculations. Therefore, the $DCGL_B$ values calculated for the Turbine Basement also apply to the Circulating Water Discharge Tunnels and the $DCGL_B$ values for the Crib House/Forebay also apply to the Circulating Water Intake Piping. The Steam Tunnel surface area and volume were included with the Turbine Basement in the calculation of BFM Dose Factors and DCGLs. The Turbine Basement $DCGL_B$ values therefore also apply to the Steam Tunnel. Note that there is expected to be minimal residual radioactivity in the Steam Tunnels, Circulating Water Intake Piping and Circulating Water Discharge Tunnels.

6.4.2. Soil

Derived Concentration Guideline Levels were developed for residual radioactivity in surface and subsurface soil that represent the 10 CFR 20.1402 dose criterion of 25 mrem/yr. A DCGL was calculated for each ROC.

Two soil DCGLs were calculated, surface soil ($DCGL_{SS}$) and subsurface soil ($DCGL_{SB}$) that are defined by the assumed thickness of the soil column from the surface downward. Surface soil is defined as that contained in a 0.15 m depth from the surface. Subsurface soil is defined as that contained in a 1 m depth of soil from the surface. These definitions apply to a continuous soil column from the surface downward. There is no expectation of subsurface contamination in a geometry comprised of a clean soil layer over a contaminated soil layer at depth.

The subsurface soil DCGL, which is based on a 1 m soil depth, can conservatively be applied to any soil depth greater than 0.15 m and less than 1 m. In the unlikely event that geometries are encountered during continuing characterization or during FSS that are not bounded by the 0.15 m and 1 m soil thicknesses, the discovered geometries will be addressed by additional modeling. The U.S. Nuclear Regulatory Commission (NRC) will be notified if additional modeling is required.

Standard methods for RESRAD parameter selection and uncertainty analysis are used in accordance with guidance in NUREG-1757, Volume 2, Revision 1 "*Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria*" (Reference 6-6). The AMCG for soil is the Resident Farmer.

6.4.3. Buried Piping

Buried pipe is defined as pipe that runs through soil. The critical group for the buried piping dose assessment is the Resident Farmer.

The buried pipe DCGLs ($DCGL_{BP}$), in units of $dpm/100cm^2$, are determined for two scenarios; assuming that all pipe is excavated and assuming that all pipe remains *in situ*. Although unrealistic, for the purpose of the bounding modeling approach used, the dose from the two scenarios is summed to determine the $DCGL_{BP}$. RESRAD was used to calculate $DCGL_{BP}$ for both the excavation and *in situ* buried pipe scenarios using the parameters developed for soil modified as necessary for the buried pipe source term geometry. Details on dose assessment methods are provide in section 6.12. A brief overview of scenario assumptions is provided below.

The excavation scenario assumes that all buried pipe is excavated after license termination and all activity on the internal surfaces of the pipes is instantly released and mixed with surface soil. The *in situ* scenario assumes that all of the buried piping remains in the "as-left" condition at the time of license termination and that all activity is instantly released to adjacent soil. Two separate in situ calculations were performed. The first calculation assumes that all pipes are located at 1 m below the ground surface in the unsaturated zone and the second assumes that all pipes are located in the saturated zone. The lowest *in situ* DCGL from either the 1m deep unsaturated or saturated scenario was assigned as the *in situ* $DCGL_{BP}$.

6.4.4. Embedded Piping

Embedded pipe is defined as pipe that runs vertically through concrete walls or horizontally through concrete floors and is contained within a given building. The release pathway for the residual radioactivity in embedded piping is into the basement where the piping is contained. The dose from embedded piping is summed with the dose from the wall and floor surfaces of the basement that contains the embedded pipe (see section 6.12.9). A DCGL, in units of pCi/m^2, was calculated for each embedded pipe survey unit ($DCGL_{EP}$). To eliminate the potential for activity in embedded pipe to result in the release of radioactivity that could potentially result in higher concentrations than predicted by the BFM, remediation and grouting action levels were established (see LTP Chapter 5, section 5.5.5). However, the dose from embedded pipe will be calculated using the $DCGL_{EP}$ values in order to accurately account for the dose.

6.4.5. Penetrations

A penetration is defined as a remaining system pipe (or the metal sleeve if the system pipe is removed, or concrete if the sleeve is removed or no sleeve was present) 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 levels of residual radioactivity in the majority of penetrations is expected to be low.

Penetrations are divided into separate survey units depending on which basements the penetrations interface with. A DCGL, in units of pCi/m^2, was calculated for each penetration survey unit ($DCGL_{PN}$) assuming that the residual radioactivity is released to both basements(s) that the penetrations interface. The $DCGL_{PN}$ calculation conservatively assumes that 100% of the penetration source term is simultaneously released to both basements.

To eliminate the potential for activity in penetrations to result in the release of radioactivity that could potentially result in higher concentrations than predicted by the BFM, remediation and grouting action levels have been established (see LTP Chapter 5, section 5.5.5). However, the dose from penetrations will be assigned based on the calculated $DCGL_{PN}$ values in order to accurately account for the dose. The dose from penetrations is summed with the dose from the wall and floor surfaces of both basements that the penetration interface (see section 6.12.9).

6.4.6. Alternate Scenarios

Several alternate scenarios for land use after backfill were qualitatively considered including industrial use, recreational use (i.e., parkland), and residential use without a water supply well or onsite garden. The Resident Farmer scenario, with onsite well is clearly a very conservative, bounding scenario relative to the alternatives.

The BFM and the alternate scenarios considered above are based on the "as left" geometry of the residual contamination in the backfilled Basements. Two additional low probability alternate scenarios were considered that included changes to the "as left" backfilled geometry. The first entails construction of a basement to the Resident Farmer house within the fill material. Note that the assumed three meter depth of the basement excavation is insufficient to encounter fill material potentially containing residual radioactivity (resulting from leaching of residual radioactivity from surfaces after backfill) assuming the Basement is not constructed within the saturated zone. However, a simple check of direct radiation dose to the resident was conducted to confirm the expectation that the dose would be negligible.

The second alternate scenario that includes disturbance of the as-left geometry considers a very unlikely assumption of a large-scale excavation of the backfilled structures after license termination. The potential doses from large scale excavation were checked by averaging the hypothetical maximum total activity corresponding to 25 mrem/yr in the BFM over the mass of the basement concrete and fill. The average concentrations were compared to the soil concentrations equivalent to 25 mrem/yr based on an industrial use scenario which was assumed to be the only future use that would justify large scale excavation of fill and concrete located deep within the saturated zone.

A third alternate scenario was evaluated which assumed that the drill for a water well encounters penetrations, embedded pipe, and basement surfaces assuming that no activity is released to the fill. The activity in the penetration, embedded pipe or basement surface that is captured by the drill is assumed to be brought to the surface in the drilling spoils. Two receptors were evaluated; the resident farmer and a worker.

6.5. Basement Fill Conceptual Model

This section describes in detail the BFM conceptual model, including the source term, ROC, future land use and exposure scenario, AMCG, and exposure pathways. The BFM is used to calculate dose to the AMCG from residual radioactivity in the backfilled, below ground Basements to remain at the time of license termination. The list of Basements to remain is provided in Table 6-1. The computational model used to implement the conceptual model is described in section 6.6.

6.5.1. Source Term

The source term for the BFM is the residual radioactivity, surface plus volumetric, remaining in each of the Basements at the time of license termination. The source term includes residual radioactivity in wall and floor concrete, or steel liner in the case of the Containment Basements, as well as in embedded piping and penetrations that are contained in or interface with a given basement. Embedded pipe and penetrations are treated as separate survey units within the applicable basement that release activity into the basement fill in the same manner as activity from walls and floors (see sections 6-13 and 6-14). The embedded pipe and penetration source terms are accounted for by adding the dose from the embedded pipe and penetration survey units to the dose from the applicable basement wall and floor survey unit. The total dose from all three sources within a given basement must be less than 25 mrem/yr. See section 6.17.1 for discussion of the process for summing the dose from walls/floors, embedded pipe, and penetrations.

LTP Chapter 2 provides detailed characterization data regarding current contamination levels in the Basements. The data is based on concrete core samples obtained at biased locations with high contact dose rates and/or evidence of leaks/spills. The expected source term configuration and radionuclide distribution expected to remain in each Basement, after remediation is completed, is summarized below.

6.5.1.1. Unit 1 and Unit 2 Containment Building Basements

Both Unit 1 and Unit 2 Containment Buildings are comprised of concrete walls and floors with all interior surfaces of the containment 'shell' covered by a 0.25 inch steel liner. The liner on the 565 foot elevation floor is covered by a 30 inch thick layer of concrete. The floor of the Under-Vessel area is located at the 541 foot elevation. A 30 inch layer of concrete is present above the liner in the Under-Vessel area and a 15 inch layer of concrete is on the walls in the Under-Vessel area. The steel liner on walls above the 568 foot elevation and below the 588 foot elevation has surficial contamination with removable contamination levels ranging from less than 1,000 dpm/100cm^2 to approximately 10,000 dpm/100cm^2 as indicated by operational and routine radiological surveys.

The concrete in the Under Vessel areas is activated. The Bio-shield concrete surrounding the vessel above 568 foot elevation is also activated. Core samples from the Unit 1 Bio-Shield indicate that the concrete was not activated through the entire depth. Core samples from the Under-Vessel areas indicate low concentrations remain in activated concrete at approximately 15 inches deep but activation through the entire depth is not expected. Continuing Characterization of the Under-Vessel concrete is planned (see LTP Rev 1, Chapter 2, section 2.5). Based on the

results of cores to date, activation of the liner, or the concrete outside of the liner, is not expected.

The source term for the Unit 1 and Unit 2 Containment Building Basement End States will be a surface contamination layer distributed over the floor and wall surfaces of the remaining exposed steel liner. All concrete inside of the liner, with the exception of the concrete in the Under-Vessel area, will be removed and disposed of as waste. Any remaining residual radioactivity on the steel liner is anticipated to be the result of the deposition of airborne activity during operations, commodity removal and during the removal of the contaminated interior concrete. Dust suppression measures will be enacted during the removal process and settling of residual radioactivity from airborne dust is expected to be minimal. In addition, operational contamination control measures taken after concrete removal will include removal of loose contamination as required for control of airborne radioactivity. As an illustration of the extent of dust required to deposit on surfaces to be of even trivial consequence, a simple calculation was performed. Based on a nominal estimate of the average radioactivity concentration, before remediation, of 240 pCi/g in the containment concrete (contaminated and activated), approximately 2 g of dust would be required over an area of 100 cm^2 to produce surface contamination levels that exceed 1,000 dpm/100cm^2.

The analyses of concrete core samples from Containment Basements indicate that the majority of the contamination is Cs-137. Ni-63, Co-60, Sr-90, Cs-134, H-3, Eu-152, and Eu-154 were also detected but at significantly lower abundance (see section 6.5.2 for mixture fractions).

The current Containment Basement inventories are not meaningful as a prediction of End State inventories because the vast majority of the contamination is in the concrete which will be completely removed during decommissioning. However, the radionuclide mixtures from the core data are considered reasonably representative of the End State mixture. In accordance with Zion*Solutions* TSD 14-019, *"Radionuclides of Concern for Soil and Basement Fill Model Source Terms"* (Reference 6-7) the nominal estimate of the inventory that will remain in the End State of each Containment Basement is approximately 1.0E-04 Ci, assuming 1,000 dpm/100cm^2 uniformly distributed over the entire interior surface of the remaining exposed liner surface. The activity remaining in the Under Vessel area concrete will be determined through continuing characterization.

6.5.1.2. Auxiliary Building Basement

The source term for the Auxiliary Building Basement End State is contamination in the remaining concrete walls and floors. The Auxiliary Building has no steel liner.

The majority of the remaining End State inventory in the Auxiliary Building Basement will be surface and volumetric contamination in the concrete floor and lower walls of the 542 foot elevation. During the operation of ZNPS, the 542 foot elevation of the Auxiliary Building was routinely flooded with contaminated water, resulting in the contamination of the concrete floor. There are water marks on the lower walls up to approximately one meter high.

The upper walls above 545 foot elevation will also be contaminated but at significantly lower concentrations than the floors. Upper wall contamination is expected to primarily be in the vicinity of floors that will have been removed during demolition. Loose surface contamination will also be present on remaining concrete surfaces due to the deposition of airborne

radioactivity generated during operations, commodity removal and the demolition of interior concrete structures. The inventory attributable to surface contamination on walls has not been estimated but is expected to be a small percentage of the total surface and volumetric inventory in the 542 elevation floor and lower walls.

Characterization results indicate that current levels of loose contamination in the 542 elevation floor range from <1,000 dpm/100 cm^2 to over 250 mrad/swipe.

Fixed contamination is present at the surface and at depth in the concrete primarily at the 542 foot elevation floor. To illustrate the distribution and depth of contamination, a range of core sample results from gamma spectroscopy analysis is provided here (see Chapter 2, section 2.3.3.2 for more details on core sample mean and distribution). Seventeen core samples were collected from the 542 foot elevation floor. The highest concentrations were found in the first 0.5 inch where Co-60 concentrations averaged 46 pCi/g, with a maximum concentration of 456 pCi/g, and Cs-137 concentrations averaged 3,352 pCi/g with a maximum concentration of 25,100 pCi/g. The highest concentrations are expected to be limited to RHR Pump Rooms that total approximately 20 m^2. In some areas, the depth of contamination is greater than 0.5 inches. For example, in the Unit 1 and Unit 2 Pipe Chase Rooms, Cs-137 concentrations of 530 pCi/g and 1,650 pCi/g were observed at depths of 4 and 5 inches, respectively. Additional cores identified a Cs-137 concentration of 57 pCi/g, at a depth of 2 inches in the central common area, a Cs-137 concentration of 31 pCi/g at a depth of 3.5 inches in the east floor area, and a Cs-137 concentration of 63 pCi/g at a depth of 3 inches in the Unit 1 Equipment Drain Collection Tank and Pump room.

The primary radionuclides by mixture percentage in the Auxiliary Building concrete are Cs-137 and Ni-63 (a non-gamma emitting radionuclide) at 75% and 24%, respectively. Cobalt-60, Sr-90, and Cs-134 were also detected but at significantly lower percentages (see section 6.5.2 for discussion of radionuclide mixture). Based on the results of the concrete core samples taken during characterization, which were biased to the "worst-case" radiological conditions, the current total inventory, including all radionuclides, in the Auxiliary Building is estimated to be approximately 0.84 Ci (Reference 6-7).

6.5.1.3. Fuel Handling Building Basement and Transfer Canals

The only portion of the Fuel Handling Building Basement that will remain following building demolition is the lower 12 feet (~4 m) of the SFP and Transfer Canals with floor elevations at 576 foot. The steel liner will be removed from both the SFP and the Transfer Canals. After the liners are removed and the underlying concrete exposed, additional characterization surveys will be performed to assess the radiological condition of the underlying concrete pad and remaining pool walls. Contamination is expected below the liner but estimates of the range, distribution and radionuclide mixture cannot be made until characterization is completed. The mixture is expected to be similar to that found in contaminated concrete in the Auxiliary Building in that the predominant radionuclide is expected to be Cs-137.

6.5.1.4. Turbine Building Basement and Steam Tunnels

Characterization surveys have shown that there is currently minimal residual contamination in the structural surfaces of the Turbine Building. Analyses of concrete cores collected from the

floor of the Turbine Building at 560 foot elevation show the presence of Cs-137 at concentrations greater than Minimal Detectable Concentration (MDC) at two of three sample locations, and only in the first 0.5 inch of concrete. Cs-137 concentrations range from 0.6 pCi/g to 47 pCi/g. In the Steam Tunnels, Cs-137 concentrations in the first 0.5 inch of concrete ranged from 7 pCi/g to 47 pCi/g in Unit 1 and 0.3 pCi/g to 19 pCi/g in Unit 2. At depths greater than 0.5 inch, Cs-137 concentrations were below MDC. No other radionuclides were identified at concentrations exceeding MDC. A nominal inventory estimate assuming 10% of the surface is contaminated at the maximum concentration is 2E-05 Ci.

6.5.1.5. Remaining Basements

Due to access restrictions, characterization was not performed in the remaining Basements, including the Forebay, Circulating Water Intake Piping and Circulating Water Discharge Tunnels. However, based on process knowledge and operational history, minimal or no radioactive contamination is expected in these Basements. Concrete core samples were collected from the Crib House as a part of concrete background studies. Only natural background activity levels were detected.

The Circulating Water Discharge Tunnels were the main authorized effluent release pathway for the discharge of treated and filtered radioactive liquid effluent to Lake Michigan. During plant operations and following shut-down, the liquid effluent release pathway was monitored and the results presented in the annual Radiological Environmental Monitoring Program (REMP) report in accordance with the Off-site Dose Calculation Manual (ODCM). The Unit 2 Circulating Water Discharge Tunnel was used as an authorized effluent release pathway during decommissioning from 6/2013 to 10/2015. The Circulating Water Discharge Tunnels were surveyed as a part of continuing characterization program after effluent release was discontinued (see LTP Chapter 2, section 2.5).

6.5.2. Radionuclides of Concern

NUREG-1757, section 3.3 states that radionuclides contributing no greater than 10% of the dose criterion (i.e., 2.5 mrem/yr) are considered to be "insignificant contributors" (IC). This 10% criterion applies to the sum of the dose contributions from the group of radionuclides considered insignificant.

After the group of IC radionuclides was identified and removed from the initial suite of potential radionuclides, the IC dose was accounted for by adjusting the DCGLs for the remaining radionuclides which are designated as the ROC (see section 6.6.8). The IC radionuclides are then excluded from further detailed evaluations. The ROC are included in the source term for detailed dose modeling.

To identify the IC radionuclides and develop the final ROC list, the first step was to develop the initial suite of radionuclides that have a potential of being present.

6.5.2.1. Potential Radionuclides of Concern and Initial Suite

Zion*Solutions* TSD 11-001, *"Potential Radionuclides of Concern during the Decommissioning of Zion Station"* (Reference 6-9) established the basis for an initial suite of potential ROC prior to characterization. Three industry guidance documents were reviewed including

NUREG/CR-3474, *"Long-Lived Activation Products in Reactor Materials,"* (Reference 6-10), NUREG/CR-4289, *"Residual Radionuclide Concentration Within and Around Commercial Nuclear Power Plants; Origin, Distribution, Inventory, and Decommissioning Assessment"* (Reference 6-11) and WINCO-1191, *"Radionuclides in United States Commercial Nuclear Power Reactors"* (Reference 6-12). Radionuclide half-lives were obtained from ICRP Publication 38, *"Radionuclide Transformations – Energy and Intensity of Emissions"* (Reference 6-13). The review also included the evaluation of 19 post-shutdown waste streams.

Based on the elimination of noble gases, theoretical neutron activation products with an abundance less than 0.01 percent relative to Co-60 and Ni-63 (the prominent activation products identified in ZNPS samples), and radionuclides with half-lives less than two years, an initial suite of radionuclides was selected that were considered to potentially be present during the decommissioning of ZNPS.

After characterization at ZNPS was completed, the results of concrete core sample analyses collected from the Containment Buildings and Auxiliary Building was reviewed in TSD 14-019. Two radionuclides, Ag-108m and Eu-155 were positively identified in one or more characterization cores and were therefore added to the list of potential radionuclides developed in TSD 11-001. The resulting initial suite of potential radionuclides is provided in Table 6-2.

6.5.2.2. Radionuclide Mixture for Initial Suite Radionuclides

The mixture percentages for the initial suite of radionuclides for Containment and Auxiliary Basement concrete were developed in TSD 14-019 using the results of the core sample analyses. Several radionuclides in the initial suite were not positively identified in any of the core sample analyses. The mixture percentages for these radionuclides were conservatively determined using the reported MDC values. The mixture percentages for the initial suite are provided in Table 6-2.

The mixture percentage for the non-gamma emitters, or Hard-to-Detect (HTD) radionuclides, were determined by analyzing selected cores from the Containment and Auxiliary Basements that contained the highest radionuclide concentrations based on gamma spectroscopy. The use of cores with higher concentrations was required to ensure that the percentage assigned to HTD radionuclides were not overly influenced by the MDC values which was the only concentration data available for the majority of the HTD radionuclides in the initial suite.

The radionuclide concentrations identified in core samples from the Turbine Building were very low, which is consistent with expectations based on operational history. Given the very limited data available, the direct determination of mixture percentages, particularly from the HTD radionuclides, was not feasible. No characterization data was collected from the Forebay, WWTF, and Circulating Water Intake Piping but the contamination levels, if any, in these Basements are expected to be minimal. Concrete cores were collected in the Crib House as a part of a background study and only natural background activity levels were identified.

Given the lack of available data and the very low levels of residual radioactivity expected to remain, the radionuclide mixture for the Auxiliary Building was considered to be a reasonably conservative mixture for the Turbine Basement, Crib House/Forebay, WWTF, and Circulating Water Inlet Piping.

The mixtures in the Circulating Water Discharge Tunnels and the SFP/Transfer Canals could be somewhat different than the Auxiliary Building due to the sources of potential contamination, i.e., effluent discharge during decommissioning and fuel pool water leaking into the concrete under the liner, respectively. This will be evaluated as a part of the continuing characterization process (see LTP Chapter 2, section 2.5).

However, the mixture in both the Circulating Water Discharge Tunnel and the SFP/Transfer Canal is expected to be primarily Cs-137 as in the other Basements. Therefore, the Auxiliary Basement mixture is considered reasonable for application to these two structures for planning purposes. The mixtures in these two Basements will be reviewed as continued characterization data is collected from these areas (see LTP Chapter 5, section 5.1).

Note that there is essentially no dose impact from uncertainty in the mixture fractions for beta-gamma emitting radionuclides because the FSS will be performed using gamma spectroscopy and compliance with the 25 mrem/yr dose criterion will be demonstrated using actual measured concentrations. The only potential dose impact of mixture uncertainty is therefore limited to the HTD mixture percentages. The dose impact of HTD radionuclides in the BFM is very low as demonstrated by the very low relative dose contribution of HTD radionuclides as discussed below.

Table 6-2 Initial Suite of Potential Radionuclides for ZNPS and Radionuclide Mixture Based on Auxiliary and Containment Concrete

Nuclide	Containment	Auxiliary
	Percent Activity	Percent Activity
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%	100%

6.5.2.3. Insignificant Dose Contributors and Radionuclides of Concern

The relative and actual dose contributions from each radionuclide in the initial suite was calculated to identify the IC radionuclides and remove them from further detailed consideration. The remaining radionuclides are designated as the ROC. Zion*Solutions* TSD 14-010, '*RESRAD Dose Modeling for Basement Fill Model and Soil DCGL and Calculation of Basement Fill Model Dose Factors and DCGLs*" (Reference 6-14) provides $DCGL_B$ and $DCGL_{BS}$ values for the initial suite. Preliminary analyses indicated that the ROC for the Auxiliary Basement were Cs-137, Co-60, Sr-90, Cs-134, and Ni-63. For Containment, the preliminary ROC were the same five radionuclides with the addition of H-3, Eu-152 and Eu-154.

In TSD 14-019, the $DCGL_B$ and Drilling Spoils $DCGL_{BS}$ values for the initial suite radionuclides were used to calculate the IC dose percentage and corresponding IC dose (i.e., IC dose

percentage times 25 mrem/yr) from the removed radionuclides. Five radionuclide mixtures were assessed;

- mixture for Containment listed in Table 6-2 (which is considered the best estimate),
- mixture for Auxiliary Basement listed in Table 6-2 (which is considered the best estimate),
- mixture using the 11 cores from Unit 1 Containment that were analyzed for the full initial suite,
- mixture using the 10 cores from Unit 2 Containment that were analyzed for the full initial suite,
- mixture using the 6 cores from the Auxiliary Basement that were analyzed for the full initial suite.

The IC dose was also calculated using the actual results (in units of pCi/g) from the individual cores analyzed for the initial suite. The IC dose mean, standard deviation of the mean, and 95% upper confidence level (UCL) were calculated for individual cores from Unit 1 Containment, Unit 2 Containment and the Auxiliary Basement. The total dose and IC dose from the individual cores were calculated assuming that the core concentrations were uniformly distributed over 100% of wall and floor surfaces of a given basement. In many cases, this hypothetical dose exceeded 25 mrem/yr. The high dose was expected given that the cores were collected from areas with the highest pre-remediation gamma activity. A concentration representing a dose greater than 25 mrem/yr would require remediation. Therefore, the total dose, and corresponding IC dose, for cores exceeding 25 mrem/yr were normalized to 25 mrem/yr to provide a value that represents the percentage of the dose criterion to be consistent with the definition in NUREG-1757, section 3.3. The IC dose from a core with a total dose below 25 mrem/yr was reported with no normalization.

The IC dose (normalized as applicable) was calculated for each core individually and the mean, range, and 95% UCL compared to the IC dose calculated from the mixtures (i.e., dose corresponding to the IC dose percentage times 25 mrem/yr). The individual core IC dose was used to assess variability and inform the selection of the IC percentage assigned to adjust the ROC DCGLs and to ensure the assigned percentage is sufficiently conservative. The results of IC dose calculations based on mixtures are provide in Table 6-3. The IC dose from individual cores is provided in Table 6-4.

Table 6-3 IC Dose from Mixtures

Core Data	IC Dose mrem/yr (percent of 25 mrem/yr)	IC Dose Drilling Spoils mrem/yr (percent of 25 mrem/yr)
Table 6-2 Mixture Containment (Unit 1 and 2 Combined) (39 Cores – Initial Suite and Onsite Gamma)	0.13 (0.51%)	0.06 (0.15%)
Table 6-2 Mixture Auxiliary (20 Cores – Initial Suite and Onsite Gamma)	0.33 (1.31%)	0.01 (0.22%)
Unit 1 Containment Mixture (11 Initial Suite Cores)	0.13 (0.51%)	0.08 (0.33%)
Unit 2 Containment Mixture (10 Initial Suite Cores)	0.07 (0.28%)	0.05 (0.22%)
Auxiliary Mixture (6 Initial Suite Cores)	0.33 (1.29%)	0.01 (0.18%)

Table 6-4 IC Dose from Individual Cores (Normalized)

Core Population	Individual Core IC Dose Range mrem/yr (Percentage of 25 mrem/yr)	Individual Core IC Dose Mean mrem/yr	Individual Core IC Dose 95% UCL mrem/yr	Individual Core Total Dose Range(1) mrem/yr
Unit 1 Containment (11 cores)	0.06 to 2.01 (0.22% to 8.06%)	0.37	0.66	2.18 to 2,212
Unit 2 Containment (10 cores)	0.02 to 1.03 (0.08% to 4.10%)	0.30	0.48	0.99 to 3,228
Auxiliary (6 cores)	0.27 to 0.73 (1.06% to 2.91%)	0.53	0.63	6.48 to 76.42

(1) Dose from all radionuclides before normalizing to 25 mrem/yr

As seen in Table 6-3 and 6-4, the highest IC dose from the five mixtures evaluated was 0.33 mrem/yr (1.31%). The maximum individual core dose, was 2.01 mrem/yr (8.06%) and 0.63 mrem/yr (2.91%) for Containment and the Auxiliary Basement, respectively. The maximum mean and 95% UCL for all individual core results were 0.53 mrem/yr and 0.66 mrem/yr, respectively. From the review of the mean and 95% UCL values in Table 6-4, it is clear that the maximum individual core IC dose of 2.01 mrem/yr (8.06%) is an outlier and not representative of widespread conditions. The individual cores represent a range of contamination conditions, with total dose projections (before normalization) from 0.99 mrem/yr to 3,228 mrem/yr, and are considered representative of the range of conditions that that will be encountered during decommissioning.

To account for any additional, unspecified variability and to provide confidence the HTD analyses performed during continuing characterization will not result in an IC dose exceeding that assigned to adjust the ROC DCGLs, a margin was applied to the IC percentage calculated using the Table 6-2 mixture by increasing the percentage to 5% for the Auxiliary Basement and 10% for the Containment Basement (to account for the single core maximum of 8.06%). The resulting IC dose percentage of 5% and 10% (1.25 mrem/yr and 2.5 mrem/yr) will be used to adjust the ROC DCGLs (Basement, Groundwater Scenario and Drilling Spoils Scenario) for the Auxiliary Basement and Containment, respectively, to conservatively account for the IC dose. These values exceed any mixture IC dose, individual core IC dose, or individual core 95% UCL IC dose found in Tables 6-3 and 6-4 and is therefore considered a bounding value.

The final ROC for Containment and the Auxiliary Basement are provided in Table 6-5. As discussed above, the Table 6-2 mixture is considered the most representative. Therefore, the ROC and IC dose percentages in Table 6-5 are considered best estimates and are provided for information and comparison to the selected IC percentage of 5% and 10% that will be used to adjust DCGLs for the Auxiliary Basement and Containment, respectively. As shown in Table 6-5, the IC dose percentages for the Table 6-2 mixture are 0.51% and 1.31% for Containment and Auxiliary Building, respectively. The vast majority of dose is from Cs-137 at 97%. The next highest dose contributor was Co-60 at 1.7%. All radionuclides, except Cs-137 could be included as insignificant contributors and eliminated in accordance with the 10% criterion. However, for conservatism and, in anticipation of potential positive ISOCS results during FSS, the low dose significant gamma emitters Co-60 and Cs-134 are retained as ROC. Sr-90 and Ni-63 are HTD radionuclides that are low dose contributors in the Auxiliary Basement but do have some, albeit low, potential for positive detection during FSS and are also retained as ROC. The Containment ROC includes Eu-152, Eu-154 and H-3 because of their potential for being present in activated concrete, not due to their dose contribution which is less than 0.1% total.

As discussed above, the Auxiliary Basement ROC and selected IC percentage of 5% for adjusting ROC DCGLs will also be applied to all other Basements with the possible exception of the SFP/Transfer Canal depending on the results of continuing characterization.

Table 6-5 Zion Radionuclides of Concern for Containment and Auxiliary Basements.

Radionuclide	Containment		Auxiliary	
	Percent Activity	Percent Annual Dose^{2}	Percent Activity	Percent Annual Dose$^{(2)}$
H-3$^{(1)}$	0.074%	0.017%	NA	NA
Co-60	4.675%	1.669%	0.908%	0.783%
Ni-63	26.275%	0.366%	23.480%	0.270%
Sr-90	0.027%	1.072%	0.051%	0.742%
Cs-134	0.008%	0.015%	0.010%	0.039%
Cs-137	67.582%	96.269%	74.597%	96.959%
Eu-152$^{(1)}$	0.436%	0.067%	NA	NA
Eu-154$^{(1)}$	0.058%	0.010%	NA	NA
IC Dose Percentage (Table 6-2 Mixture)	0.864%	0.512%	0.954%	1.313%
Total	100%	100%	100%	100%

(1) H-3, Eu-152 and Eu-154 are activation products and therefore applicable to Containment Building only
(2) Percent annual dose and IC dose percentage based on best estimate mixture in Table 6-2 for information. IC percentages of 5% and 10% will be used for ROC DCGL adjustment for Auxiliary and Containment Basements, respectively, to provide additional margin.

6.5.2.4. Radionuclide Ratios for Application to Surrogate Approach

The FSS for basement surfaces will be performed using ISOCS gamma spectroscopy. Three radionuclides that are not gamma emitters are included as ROC, i.e., Sr-90 and Ni-63 for the Auxiliary Basement and Sr-90, Ni-63 and H-3 for Containment. As discussed in LTP Chapter 5, the Sr-90, Ni-63 and H-3 concentrations will be accounted for using a surrogate approach during FSS. The ratios of Sr-90/Cs-137, Ni-63/Co-60 and H-3/Cs-137 are required to implement the surrogate approach.

The radionuclide ratios were calculated in TSD 14-019 by calculating the ratios of Sr-90/Cs-137, Ni-63/Co-60 and H-3/Cs-137 within each individual core analyzed for the initial suite. Ratios were calculated separately for Containment and the Auxiliary Basement. The mean, maximum, and 95% UCL of the individual core ratios were calculated. The 95% UCL was conservatively calculated using the standard deviation of the individual results as opposed to the standard deviation of the mean. Table 6-6 provide the results. The maximum individual ratios are all higher than the 95% UCL and will be used in the surrogate calculations during FSS unless different values are justified by the results of continuing characterization or FSS HTD analysis (see LTP Chapter 5 section 5.2.11).

Table 6-6 Radionuclide Ratios from Concrete Cores

Radionuclide Ratio	Containment			Auxiliary Basement		
	Mean	Maximum	95% UCL	Mean	Maximum	95% UCL
Sr-90/Cs-137	0.002	0.021	0.010	0.001	0.002	0.002
Ni-63/Co-60	30.62	442	194	44.14	180.45	154.63
H-3/Cs-137	0.21	1.76	0.96	NA	NA	NA

6.5.3. Critical Group and Exposure Scenario

The critical group for the BFM dose assessment is the Resident Farmer. A well is assumed to be installed onsite (in the center of the Basement with the highest projected future groundwater concentrations), which supplies drinking water, water for livestock and irrigation water for a garden and pasture/crop land. The Resident Farmer is considered a bounding exposure scenario (as defined in NUREG-1757). A simple visualization of the BFM conceptual model is provided in Figure 6-9.

The "Reasonably Foreseeable Scenario", which is defined in NUREG-1757 as a land use scenario that is likely within the next 100 years, could be justified as not including an onsite water well which is prohibited by local municipal code (see section 6.2.4). Municipal water in the vicinity of ZNPS is supplied by Lake Michigan, which is expected to be a viable source for hundreds of years. In addition, Resident Farmer land use, with or without a well, would also be unlikely for a minimum of 100 years after license termination considering current land use and zoning in the area. Any type of residential use is essentially non-credible while the ISFSI is present.

Current zoning at the ZNPS is heavy industrial use. The City of Zion, "*Official Zoning Map City of Zion*" March 2011 (Reference 6-15) contains no agricultural use areas anywhere in the city.

In addition, a 2012 report by the United States Department of Agriculture, "*Custom Soil Resources Report Lake County Illinois*" (Reference 6-16), classified the soil at ZNPS as "Category 3," which is defined as soils with "*severe limitations that reduce the choice of plants or require special conservation practices, or both*". While the zoning and soil classification does not preclude a resident garden, the use of the land for raising livestock such as beef and dairy cattle during the next 100 years could justifiably be categorized as a "less likely but plausible" scenario (as defined in NUREG-1757, Table 5.1). Consistent with this definition, it would be reasonable to not include livestock in the compliance dose assessment.

Using a simple assumption that the Resident Farmer well drilling scenario would not occur on the site for at least the first 100 years after license termination, if at all, the BFM dose would be reduced by about a factor of ten based on the radioactive decay of Cs-137, which is the predominant radionuclide. Assuming that residential occupancy does occur after license termination, eliminating the livestock pathway and retaining the onsite well, resident garden, etc., would reduce the dose by approximately 60%. Notwithstanding all of the above, the BFM applies the Resident Farmer land use to ensure that the critical group and exposure scenario produce a conservative and bounding compliance dose calculation.

6.5.4. Exposure Pathways

The BFM applies to the backfilled Basements which will have a minimum of 3 feet cover and approximately 3 m of clean fill above the potential source term zone as defined by the equilibrium water level in the backfilled Basements. The equilibrium water level is conservatively assumed to be at the natural water table elevation of 579 foot. Therefore, the dose from the water-independent exposure pathways is negligible. Nonetheless, all Resident Farmer exposure pathways, water-dependent and water-independent are included in the model to verify this assumption. The aquatic pathway from an onsite pond is not credible due to engineering and cost issues of construction and proximity to Lake Michigan which negates any foreseeable need (TSD 14-003).

The Resident Farmer Scenario includes the following exposure pathways:

- Direct exposure to external radiation
- Inhalation dose from airborne radioactivity
- Ingestion dose from the following pathways;
  - Plants grown with irrigation water from onsite well,
  - Meat and Milk from livestock consuming fodder from fields irrigated with onsite well water and consuming water from onsite well,
  - Drinking water from onsite well,
  - Soil ingestion.
- Direct exposure, inhalation dose and ingestion dose from contaminated drilling spoils brought to the surface during installation of the onsite well into the fill material.

The last bullet is not a standard Resident Farmer exposure pathway as described in NUREG/CR-5512, Volume 1, "*Residual Radioactive Contamination from Decommissioning Parameter Analysis*" (Reference 6-17) or as contained in RESRAD. However, the BFM Resident Farmer scenario is predicated on the well being installed into the basement fill which will generate drilling spoils. Assuming a well is drilled, exposure to the spoils that are brought to the surface is a potential exposure pathway. The potential dose contribution from this pathway was checked in screening assessments and found to be greater than 10% of the total BFM dose in some cases (see section 6.6.7). Therefore, the pathway was included in the BFM.

The well water dependent BFM exposure pathways are not applicable to the SFP due to the elevation of the SFP floor being at the 576 foot elevation (see Table 6-1), which is only three feet below the water table elevation of 579 foot. Operating water well in an area with only three feet of available water is considered a "land use that because of physical limitations could not occur" and is therefore implausible as defined in Table 5.1 of NUREG-1757. However, this would not preclude a well driller from inadvertently picking a location above the SFP as a potential well location and then rejecting the location based on low water level. Therefore, for the SFP/Transfer Canal Basement, the pathways resulting from well water are not applicable, but the drilling spoils pathway is applicable and will be applied in the BFM assessment. However, the potential contribution of the SFP/ inventory to a well water pathway will be considered by adding the SFP/Transfer Canal surface area to the Containment and Auxiliary Basement surface

areas during the DCGL calculation (see section 6.6.8). Adding the surface area to the DCGL calculation corresponds to adding the inventory. This addition is necessary because the SFP/Transfer Canal will be hydraulically connected to the Containment Basements through the Fuel Transfer Tubes and to the Auxiliary Basement through the opening in the wall between the Transfer Canal and the Auxiliary Basement that was created to facilitate decommissioning.

The same argument regarding implausibility of well operation that was applied to the SFP/Transfer Canal could also be applied to the WWTF, which has a floor that is only two feet below the site groundwater levels and an internal Basement volume of 144 m^3. However, the WWTF is an isolated structure with no connections to other Basements and therefore, the inventory cannot credibly be added to other Basements to conservatively account for the well water exposure pathways. Therefore, the water well pathways are applied in the BFM for the WWTF as a simple, bounding approach.

6.6. <u>Basement Fill Computation Model</u>

6.6.1. DUST-MS Model

The initial environmental transport pathway for the Resident Farmer scenario is the release of radioactivity from Basement concrete (or steel liner surfaces for Containment Basements) to water in the interstitial space of the fill material. The water concentrations in the Basements are calculated using the DUST-MS computer code. The methods and results are summarized here and described in detail in Zion*Solutions* TSD 14-009, *"Brookhaven National Laboratory Report (BNL), "Evaluation of Maximum Radionuclide Groundwater Concentrations for Basement Fill Model, Zion Station Restoration Project*" (Reference 6-18). The water concentrations calculated by DUST-MS were used in conjunction with RESRAD modeling results (see section 6.6.3) to calculate BFM Dose Factors (see section 6.6.8).

The DUST code was originally developed through support from the NRC in 1992. Subsequent development of the code led to the multiple species (MS) version. DUST-MS was used by the NRC to develop guidance on performance assessment for low-level waste disposal and has been accepted for use in LTPs for other power reactor sites.

To calculate the maximum water concentrations, the rate of radionuclide release from concrete the source term to the fill is required. Diffusion controlled release is assumed for Basements with volumetric contamination (i.e., Auxiliary and SFP/Transfer Canal) and instant release is assumed for the remaining Basements where contamination is predominantly on or near the structure surfaces.

After release, the residual radioactivity is assumed to mix instantly with the water the Basements. The concentrations are calculated for each Basement independently. The only mechanism to reduce the water concentration is sorption onto the fill material. The water concentration for this model can be calculated using Equation 6-1.

Equation 6-1

$$C = {}^{I}/{[V \times (\emptyset + \rho K_d)]}$$

where:

C = concentration in water (pCi/L)
I = inventory (pCi)
V = Basement mixing volume (L)
Θ = effective porosity
ρ = bulk density (g/cm^3)
K_d = distribution coefficient (cm^3/g)

Although simple spreadsheet calculations can be performed to determine equilibrium water concentrations for the Basements with instant release, DUST-MS is used to simulate diffusion controlled release for Basements with volumetrically contaminated concrete. In addition, a sensitivity analysis was conducted of the impact of alternate well placement on groundwater concentrations, assuming transport to a well located outside of the Basements (as opposed to the being placed in the Basement fill) which also requires the use of DUST-MS. Therefore, all calculations have been performed with DUST-MS to maintain consistency and for ease of calculation and reporting

The water concentrations are calculated separately for each Basement with no assumption of mixing between buildings. This is conservative given that there will be several open penetrations between the Basements after piping is removed that will provide hydraulic connectivity between the Basements. Zion*Solutions* TSD 14-032, "*Conestoga Rovers & Associates Report, Simulation of the Post-Demoltion Saturation of Foundation Fill Using a Foundation Water Flow Model*" (Reference 6-19) describes the remaining penetrations and the projected equilibrium water levels in the Basements. Mixing and flow of water between Basements will occur, primarily between the Auxiliary, Containment and Turbine Basements. The maximum equilibrium water concentrations are conservatively calculated for the worst case individual Basement (expected to be the Auxiliary Basement) assuming no mixing.

Based on current demolition plans, there will be no connection between the Basements and surrounding groundwater. A number of pipes that penetrate the Turbine Basement walls and enter the outside ground will be removed from both sides of the Basement walls or remain in the ground outside of the Turbine basement. This will leave a number of penetrations open to the outside ground, primarily on the east side of the Turbine Basement. However, none of these open penetrations are below the water table (579 foot elevation). There are two 48 inch diameter Service Water Supply Lines that run from the 549 foot elevation in the Auxiliary Building, to the ground east of the Turbine Basement which will be cut above the 579 foot elevation in the ground and will be filled with grout or sealed in another manner. There are also a number of small diameter buried pipes that penetrate Basements below the 579 foot elevation, primarily in the Auxiliary Basement. These are designated as "Building to Ground Penetrations Buried Pipe" in TSD 14-016. Most of these pipes are currently planned to be cut in the ground above the 579 foot elevation and therefore above the average groundwater elevation. A few are listed as terminating in the ground below 579 foot elevation. To eliminate uncertainty regarding water ingress or egress through these small diameter penetrations that are connected to buried pipe, all

of the penetrations in this category that enter a basement below 579 foot elevation will be grouted regardless of what elevation the buried pipe is cut within the ground. Grouting provides additional assurance that the End State configuration provides no route for groundwater ingress into the Basements, leaving only rainwater infiltration as the source of water in the fill. TSD 14-032 estimated that it will take approximately 28 years to reach an equilibrium water level across all Basements, considering rainwater infiltration rates and existing penetrations between Basements. The DUST-MS model assumes that the Basements are full of water immediately after license termination and capable of supporting a residential well, which is a conservative assumption.

6.6.1.1. Parameter Selection

For DUST-MS modeling, the initial source term in each Basement is nominally assumed to be 1 pCi/m^2 uniform activity over all walls and floor surfaces below 588 foot elevation. The inventory corresponding to this activity (1 pCi/m^2 multiplied by the surface area in a given Basement) is the value used for the equilibrium calculations. However, it is important to note that the value of the assumed inventory in each Basement is immaterial because the DUST-MS modeling results are used to generate unitized Groundwater Concentration Factors in units of pCi/L per mCi. The DUST-MS results will be used in conjunction with the RESRAD results (see section 6.6.3) to calculate BFM Dose Factors in units of mrem/yr per mCi. The BFM Dose Factors are then used to calculated DCGLs, in units pCi/m^2 (see section 6.6.8).

The equilibrium calculation for released activity is simple as shown in Equation 6-1 and includes limited input parameters. The selected model parameters are listed in Tables 6-7 and 6-8.

Table 6-7 General Parameters for DUST-MS Modeling

Parameter	Selected Value
K_d	Table 6-5 (Nuclide Dependent)
Porosity	0.25
Bulk Density	1.5 g/cm^3
Basement Mixing Volume	Table 6-6 (Basement Dependent)

Table 6-8 Distribution Coefficients for DUST-MS Modeling

Radionuclide	Basement Fill K_d (cm^3/g)
Co-60	223
Ni-63	62
Sr-90	2.3
Cs-134	45
Cs-137	45
Eu-152	95
Eu-154	95

The specific composition of the backfill has not yet been determined but is expected to be some combination of sand and debris resulting from building demolition that is designated for beneficial reuse as clean hard fill. The ratios of sand and demolition debris are not known and therefore, the bulk density and porosity not known with certainty. Zion*Solutions* TSD 14-006, a report by Conestoga Rovers & Associates, *"Evaluation of Hydrological Parameters in Support*

of Dose Modeling for the Zion Restoration Project", (Reference 6-20) calculates site-specific values for the porosity and density of local soil. The results were 0.35 and 1.8 g/cm^3, respectively. Inspection of Equation 6-1 shows that calculated water concentrations are inversely proportional to porosity and density. Therefore, a conservative bulk density of 1.5 g/cm^3 and porosity of 0.25 were selected for the DUST-MS parameters. With any of the fill materials, it is unlikely that packing of the material would result in porosity below 0.25.

The distribution coefficients (K_d) are important parameters in the calculation of equilibrium concentrations. As shown in Equation 6-1, water concentration varies inversely with K_d. Consequently, lower K_d values will result in higher projected future water concentrations. Zion*Solutions* TSD 14-004, a report by Brookhaven National Laboratory, *"Recommended Values for the Distribution Coefficient (K_d) to be used in Dose Assessments for Decommissioning the Zion Nuclear Power Plant,"* (Reference 6-21) reviewed K_d values from three sources:

Data Sources for selection of DUST-MS Modeling distribution coefficients:

- literature values,
- site-specific K_d analyses performed by Brookhaven National Laboratory as documented in two reports, Zion*Solutions* TSD 14-017, *"Sorption (Kd) Measurements on Cinder Block and Grout in Support of Dose Assessments for Zion Nuclear Station Decommissioning"* (Reference 6-22), and Zion*Solutions* TSD 14-020, *"Sorption (Kd) measurements in Support of Dose Assessments for Zion Nuclear Station Decommissioning"* (Reference 6-23), and
- the 25th percentile values of the K_d distributions provided in NUREG/CR-6697, *"Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes"* (Reference 6-24).

In selecting values from literature, environmental conditions with high pH (cement sorption data) as well as typical environmental soil sorption data were considered due to the anticipated presence of concrete and cinderblock demolition debris in the fill. For conservatism the minimum values from all of these sources were selected. For nuclides with measured site-specific K_d values, the lowest measured K_d in any potential backfill material or soil was selected.

6.6.1.2. <u>Mixing Volume</u>

The water concentrations calculated by DUST-MS are inversely proportional to the assumed mixing volume which differs for each Basement as a function of the building geometry and distance from the floor to the assumed water elevation in the Basements. Section 6.5.1 describes the source terms and remaining structural configuration of the Basements.

The projected equilibrium water elevation in the Basements was evaluated in TSD 14-032. The water level is driven by the location, elevation and size of existing penetrations between the Basements and between the Basements and outside ground. The current decommissioning approach does not include making additional perforations through Basement walls other than between the SFP and the Transfer Canals. Given these conditions, the equilibrium water level in the Basements was projected to be at the 586 foot elevation. A number of options are presented in TSD 14-032 for perforating the basements to keep water levels at approximately 579 foot elevation. ZSRP has selected Scenario 3 from TSD 14-032 which entails breaching the western

most portion of the north foundation wall of the Unit 2 Steam Tunnel. The breach will be 15-feet wide and extend from the top of the foundation wall after demolition (588' AMSL) to an elevation of 580 feet AMSL (i.e., one foot above the exterior water table). To accommodate any future perforation plans, and ensure conservatism, the mixing volume for the DUST-MS modeling is based on a Basement water elevation equal to the 579 foot elevation of surrounding groundwater. The resulting mixing volumes for each Basement are provided in Table 6-9.

Table 6-9 Basement Mixing Volumes for DUST-MS Modeling

Basement/Structure	Volume (m^3)
Unit 1 Containment Building	6.54E+03
Unit 2 Containment Building	6.54E+03
Auxiliary Building	2.84E+04
Turbine Building	2.61E+04
Crib House and Forebay	3.05E+04
WWTF	1.44E+02
Spent Fuel Pool and Transfer Canal	2.08E+02
Main Steam Tunnels (Unit 1 and Unit 2)	NA - Volume included with Turbine Building volume of 2.61E+04 m^3
Circulating Water Intake Piping	NA - Source term included with Crib House/Forebay and Turbine in DCGL calculation
Circulating Water Discharge Tunnels	NA – Source term included with Turbine Building in DCGL calculation

6.6.1.3. <u>Radionuclide Release Rate</u>

The release rate is a function of the source term geometry. In all of the Basements with the exception of the Auxiliary Basement and possibly the SFP/Transfer Canal, the contamination is expected to be surficial. This surface contamination may be relatively loosely bound. In these Basements, the release is conservatively assumed to occur instantly such that the entire inventory is available immediately after license termination. Activated concrete will remain in the Under-Vessel area of Containment. The assumption of instant release for Containment is very conservative for activated concrete which would actually release radionuclides very slowly.

The contamination in the Auxiliary Basement, and possibly the SFP/Transfer Canal, has diffused into the concrete resulting in volumetric contamination. The Auxiliary Building has been characterized and shown to be contaminated to a depth of at least the first inch of the concrete and deeper in several locations. Leak detection tests have indicated that the steel liner of the SFP does leak, but the extent of the concrete contamination under the liner is not known at this time. After the liner has been removed, the underlying concrete of the SFP/Transfer Canal will be characterized. Due to the volumetric source term, the release of contamination from Auxiliary

and SFP/Transfer Canal concrete, and the resulting maximum water concentrations, will be a driven by time-dependent diffusion controlled release. For these two Basements, a diffusion controlled release model is used. If contamination in the SFP/Transfer Canal is found to be surficial, then the DUST-MS model will be rerun using an instant release rate. Table 6-10 summarizes the release rate assumptions used in DUST-MS modeling for each Basement.

Table 6-10 Summary of DUST-MS Source Term Release Rate Assumptions for the Zion Basements

Basement	Release Rate Assumption
Unit 1 Containment	Instant Release (1) (loose surface contamination on steel liner)
Unit 2 Containment	Instant Release (1) (loose surface contamination on steel liner)
Auxiliary	Diffusion Controlled Release (concrete contamination at depth in concrete)
Turbine	Instant Release (limited contamination present at concrete surface with very limited contamination at depth)
Crib House and Forebay	Instant Release (limited or no surface contamination)
WWTF	Instant Release (limited or no surface contamination)
SFP and Transfer Canals	Diffusion Controlled Release (Concrete contamination at depth expected under liner)

(1) A small volume of activated concrete will remain in the Under-Vessel areas of both Containments. The instant release assumption is very conservative for activated concrete.

Diffusion coefficients for each ROC are required to estimate the rate of release from concrete in addition to the parameters listed in Tables 6-4 and 6-5. The diffusion coefficients from concrete will depend on the water to cement ratio used in forming the concrete and the aggregate. Table 6-11 lists a typical range of diffusion coefficients for concrete and provides reference(s) for the values. The water concentrations are proportional to the diffusion coefficient, so the maximum value in the range was selected for use in the DUST-MS modeling.

The diffusion rate also depends on the contamination depth profile. The majority of the contamination in Auxiliary Basement is found in the first one inch of concrete. However, there are some locations where the contamination is deeper. The diffusion modeling in DUST-MS conservatively assumes that the contamination is 0.5 inch deep. All activity in the concrete, including any activity deeper than 0.5 inch, will be determined during the FSS (see LTP Chapter 5, section 5.5). All activity deeper than 0.5 inch will be assumed to be included in the first 0.5 inch. This is a conservative approach because the deeper contamination would diffuse out more slowly. In addition, assuming that the total inventory is within the first 0.5 inch of concrete will increase the effective concentration in the first 0.5 inch. This assumption will increase the diffusion rate which is driven by the concentration gradient.

Table 6-11 Range of Diffusion Coefficients for Cement and Selected Values for Radionuclides of Concern (Reference 6-21)

Nuclide	Diffusion Coefficient Range (cm^2/s)	Selected Diffusion Coefficient (cm^2/s)
H-3	6.0E-09 – 5.5E-07	5.5E-07
Co-60	5.0E-12 – 4.1E-11	4.1E-11
Ni-63	8.7E-10 – 1.1E-09	1.1E-09
Sr-90	1.0E-11 – 5.2E-10	5.2E-10
Cs-134; Cs-137	4.0E-11 – 3.0E-09	3.0E-09
Eu-152; Eu-154	1.0E-12 – 5.0E-11	5.0E-11

The depth of contamination for DUST-MS diffusion modeling of the concrete in the SFP/Transfer Canal is also assumed to be 0.5 inch. The depth of contamination in the SFP/Transfer Canal concrete is not known at this time and will be characterized after the liner is removed. If contamination is found at depths significantly greater than 0.5 inch, then the model may be re-run using the actual depth profile. This re-run would be at the discretion of ZSRP if it were determined that the 0.5 inch thickness assumption was too conservative. In this case, the results would be made available for NRC review. Any increase in the DCGL as a result of this re-run would require NRC approval. All other DUST-MS parameters would remain the same.

6.6.2. Sensitivity Analysis

Although conservative parameters were selected for DUST-MS as described above, a sensitivity analysis was performed for K_d, porosity, and density to ensure that further parameter review was not necessary. A simple assessment was performed by varying each parameter independently through range of +/- 25% of the selected parameter as shown in Table 6-12.

Table 6-12 Range of DUST-MS Parameters Varied in Sensitivity Analysis

Parameter	Selected Value	Sensitivity Range
K_d	Table 6-5 (Nuclide Dependent)	± 25% of Value in Table 6-5
Porosity	0.25	0.19 – 0.31
Bulk Density	1.5 g/cm^3	1.1 – 1.8 g/cm^3

The results show minimal impact in varying the parameters through the range as listed below. No adjustment of the conservatively selected parameter values is deemed necessary.

- K_d: An increase in K_d caused a decrease in solution concentration and a slight increase in sorbed concentration on fill. Solution concentration is approximately inversely proportional to K_d. The 25% change in K_d had a minimal impact on the amount sorbed or the backfill concentration (pCi/g). Sr-90 showed the largest percentage change in sorbed concentration of all the nuclides but it was less than 2.5%.
- Porosity: Changing porosity had a minor impact on the amount sorbed and solution concentration. The amount of radioactivity in solution was proportional to the porosity (but

the concentration was lower). This reflects the increased volume of water available for mixing in higher porosity media and corresponding higher total amount of activity in the water.

- Density: The solution concentration, sorbed concentration and amount in solution are inversely proportional to density. Increasing density causes a decrease in solution concentration. The change in density has a minor impact (< 2%) on the total amount of radioactivity that is sorbed.

The sensitivity of the depth of contamination in the diffusion release model was also assessed. As expected, the maximum water concentrations decreased with an increased depth of contamination. Therefore, a minimum concrete contamination depth of 0.5 inches was used in the DUST-MS modeling.

6.6.2.1. Sensitivity of Well Placement

The placement of the well inside the Basement(s) is unlikely because it is assumed that the driller will recognize that the spoils are not natural materials and, that the high pH of the water inside the Basement(s) due to the presence of concrete and cinderblock demolition debris will make the water unsuitable for domestic use. In addition, encountering construction debris during drilling and meeting refusal at the Basement floor will further discourage the use of a well drilled into the Basements. However, the BFM conservatively assumes that the well is placed inside of the Basement(s). A simple assessment was performed in TSD 14-009 to determine the potential effect of well placement outside of the walls of the Basement(s) to illustrate that the well placement assumption was conservative.

For the well placement sensitivity assessment, the well was assumed to be located in the shallow sand aquifer at the closest location downstream of the Basement(s) to the east of the Turbine Building. In this assessment, the Auxiliary Building is modeled with contamination released in this building flowing through the Turbine Building similar to the physical layout at the site. The initial inventory in the Turbine Building was reduced by a factor of 0.001 consistent with the much higher measured concentrations in the concrete of the Auxiliary Building. Water flow through the system is assumed to be at the local groundwater velocity (e.g., the Basement walls are assumed to be transparent and allow free flow of groundwater). The results indicate that the water concentration (and corresponding Resident Farmer dose) would be reduced by approximately two orders of magnitude for Cs-137 if the well were located outside of the Basements at the nearest downstream location. The reduction for Co-60 is much greater. The Sr-90 concentrations are only slightly reduced due to the very low assumed K_d of 2.3 cm^3/g. This analysis further supports the conclusion that the BFM conceptual model, which assumes that the well is placed inside a Basement, is bounding.

6.6.2.2. DUST-MS Model Results

Tables 6-13 and 6-14 (Reference 6-18) provide the results of the DUST-MS calculations for each Basement and ROC. The tables report the maximum Groundwater Concentration Factors (pCi/L per mCi) and corresponding Fill Concentration Factors (pCi/g per mCi). Note that both of these values occur at the same point in time.

The maximum concentrations occur at the time of license termination for the Basements with instant source term release. The time of maximum concentrations varies for each ROC in the Auxiliary Building Basement and SFP/Transfer Canal as a function of half-life and diffusion coefficient. For application in the BFM, the maximum concentration factors are used and conservatively assumed to occur at one point in time for all radionuclides.

Table 6-13 Peak Groundwater Concentration Factors (pCi/L per mCi Total Inventory)

Nuclide	Auxiliary (pCi/L/mCi)	Containment (pCi/L/mCi)	Turbine (pCi/L/mCi)	Fuel (pCi/L/mCi)	Crib House /Forebay (pCi/L/mCi)	WWTF (pCi/L/mCi)
Co-60	4.00E-03	4.57E-01	1.14E-01	5.45E-01	9.77E-02	2.08E+01
Cs-134	1.06E-01	2.26E+00	5.65E-01	1.45E+01	4.83E-01	1.03E+02
Cs-137	3.80E-01	2.26E+00	5.65E-01	5.22E+01	4.83E-01	1.03E+02
Eu-152	1.65E-02	1.07E+00	2.68E-01	2.24E+00	2.29E-01	4.83E+01
Eu-154	1.29E-02	1.07E+00	2.68E-01	1.76E+00	2.29E-01	4.83E+01
H-3	1.40E+02	6.13E+02	1.53E+02	1.91E+04	1.31E+02	2.78E+04
Ni-63	2.92E-01	1.64E+00	4.10E-01	4.01E+01	3.52E-01	7.47E+01
Sr-90	3.01E+00	4.13E+01	1.04E+01	4.12E+02	8.85E+00	1.89E+03

Table 6-14 Peak Fill Material Concentration Factors (pCi/g per mCi Total Inventory)

Nuclide	Auxiliary (pCi/g/mCi)	Containment (pCi/g/mCi)	Turbine (pCi/g/mCi)	Fuel (pCi/g/mCi)	Crib House /Forebay (pCi/g/mCi)	WWTF (pCi/g/mCi)
Co-60	8.92E-04	1.02E-01	2.55E-02	1.22E-01	2.18E-02	4.64E+00
Cs-134	4.77E-03	1.01E-01	2.54E-02	6.53E-01	2.18E-02	4.63E+00
Cs-137	1.71E-02	1.01E-01	2.54E-02	2.35E+00	2.18E-02	4.63E+00
Eu-152	1.58E-03	1.02E-01	2.55E-02	2.15E-01	2.18E-02	4.64E+00
Eu-154	1.22E-03	1.02E-01	2.55E-02	1.67E-01	2.18E-02	4.64E+00
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	1.81E-02	1.02E-01	2.54E-02	2.49E+00	2.18E-02	4.64E+00
Sr-90	6.94E-03	9.50E-02	2.38E-02	9.46E-01	2.03E-02	4.33E+00

The Groundwater Concentration Factors are used in conjunction with "Groundwater Exposure Factors" generated by RESRAD to develop the BFM GW Dose Factors which are one of the inputs to the DCGL calculations in section 6.6.8.

6.6.3. RESRAD Model

The RESRADv7.0 computer code was used to calculate the Resident Farmer dose from a unit radionuclide concentration in the well water. A Groundwater Exposure Factor, in units of mrem/y per pCi/L was generated for each ROC. The Groundwater Exposure Factors are combined with the Groundwater Concentration Factors generated using DUST-MS to calculate the BFM GW Dose Factors for each ROC in units of mrem/y per mCi..

6.6.3.1. Parameter Selection

RESRAD parameters are classified as behavioral, metabolic or physical. Some parameters may belong to more than one category. The parameter classification is documented in NUREG/CR-6697. Physical parameters are determined by the source, its location, and geological characteristics of the site (i.e., these parameters are source- and site-specific) including the geohydrologic, geochemical, and meteorologic characteristics of the site. The characteristics of atmospheric and biospheric transport up to, but not including, uptake by, or exposure of, the dose receptor would also be considered physical input parameters.

Behavioral parameters define the receptor's behavior considering the conceptual model selected for the site. For the same group of receptors, a parameter value could change if the scenario changed (e.g., parameters for recreational use could be different from those for residential use). For the ZNPS, the behavioral parameters are based on a Resident Farmer scenario and are the same for both the BFM and soil dose assessments.

Metabolic parameters define certain physiological characteristics of the potential receptor. One set of metabolic parameters applies to both the BFM and soil dose assessments. Physical, behavioral and metabolic parameters are treated as deterministic parameters in the final dose modeling to calculate Groundwater Exposure Factors. The deterministic module of the code uses single values for input parameters and generates a single value for dose. The parameter selection process is described below.

Argonne National Laboratory (ANL) ranked physical parameters by priority as 1, 2, or 3. Priority 1 parameters have the highest potential impact on dose and Priority 3 the least. This ranking is documented in Attachment B to NUREG/CR-6697.

Priority 3 physical parameters were assigned the median values from the parameter distributions defined in NUREG/CR-6697. Priority 1 and 2 parameters were evaluated by uncertainty analysis using the NUREG/CR-6697 parameter distributions. The Partial Rank Correlation Coefficient (PRCC) was used to evaluate the relative sensitivity of the Priority 1 and 2 parameters. A PRCC value less than -0.25 was considered sensitive and negatively correlated to dose. The 25th percentile of the NUREG/CR-6697 distribution was assigned to negatively correlated parameters. A PRCC value greater than 0.25 was considered sensitive and positively correlated to dose. The 75th percentile of the distribution from NUREG/CR-6697 was assigned to positively correlated parameters. Priority 1 and 2 parameters with a |PRCC| less than 0.25 were assigned the median value of the NUREG/CR-6697 parameters.

Consistent with the guidance in NUREG-1757, section I.6.4.2, metabolic and behavioral parameters were assigned the mean values from NUREG/CR-5512 Vol. 3, *"Residual Radioactive Contamination From Decommissioning Parameter Analysis"* Table 6.87 (Reference 6-25).

Figure 6-10 provides a flow chart of the parameter selection process.

The RESRAD code contains several Dose Conversion Factor (DCF) libraries that can be selected by the user. The DCF library selected for the BFM applies inhalation and ingestion DCFs from the Environmental Protection Agency (EPA) Federal Guidance Report (FGR) No. 11, *"Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion"* (Reference 6-26) and direct external exposure dose

conversion factors from FGR No. 12 *"External Exposure to Radionuclides in Air, Water and Soil"* (Reference 6-27).

There are four RESRAD parameters that were assigned values that are specific to the BFM as listed below:

- Time since material placement = 1 year
- Mass Balance Groundwater Model
- 100% of the initial contamination in the water table
- No unsaturated zone (unsaturated zone depth = 0)

The parameter value for "time since material placement" of one year was selected for user convenience to allow RESRAD to calculate an equilibrium well water concentration at run time equal to zero for all radionuclides. The assumption that 100% of the initial contamination is in the water table, no unsaturated zone and Mass Balance Groundwater Model removes the time dependence of travel through an unsaturated zone in the reported well water concentrations as a function of time. All radionuclides achieve maximum well water concentrations at t=0. However, none of these parameters effect the calculation of the Groundwater Exposure Factors (mrem/y per pCi/L), which can be calculated for any year and any well water concentration since they are unitized. The relationship between dose and well water concentration is independent of time or water concentration.

In a similar manner, the saturated zone and contaminated zone hydrogeological parameters have no impact on the calculation of the unitized Groundwater Exposure Factors for the BFM. However, instead of using the default values for these parameters they were selected and justified using the full process shown in Figure 6-10. This was done for two reasons, to allow the same parameter set to be used for the site specific soil DCGL determination in section 6.9 (with slight modification) and to eliminate any potential concerns that the hydrogeological parameters could impact the dose calculations due to unforeseen effects on the RESRAD calculations.

Finally, the RESRAD model was run deterministically, i.e., single values were selected for all parameters. In practice, this only affects the few Priority 1 and 2 physical parameters that are not site-specific or sensitive, which would be run with the distributions from NUREG/CR-6697 in a probabilistic approach as opposed to the mean values.

6.6.4. Uncertainty Analysis

Uncertainty analysis was performed to ensure that conservative values are selected for parameters that have a relatively high correlation to dose. Attachment 1 provides the input parameter set used to perform the uncertainty analysis. The parameter selection process is discussed below.

For the uncertainty analysis, deterministic parameters are selected for behavioral, metabolic and Priority 3 physical parameters in accordance with the process in Figure 6-10. The majority of the Priority 1 and 2 physical parameters are assigned the parameter distributions from NUREG/CR-6697. Three site-specific Priority 1 and 2 physical parameters were assigned deterministic values in the uncertainty analysis including cover depth, precipitation, well

pumping rate (which does not have a recommended distribution in NUREG/CR-6697). In addition, as discussed in section 6.6.1.1, the K_d values were assigned conservative deterministic values based on the review of various literature sources and site-specific data documented in TSD 14-004. The assigned K_d values apply to the basement fill material and are therefore the same as selected for the DUST-MS model (see Table 6-5). There are other site-specific deterministic parameters available, but these are included in the uncertainty analysis by applying the parameter distributions from NUREG/CR-6697 to ensure the appropriate level of justification is provided if one or more of these parameters were determined to be sensitive.

The uncertainty analysis was conservatively run for all ROC individually to maximize the parameter sensitivity. A more realistic approach would be to only apply the radionuclide mixture fractions found at ZNPS. Using the ZNPS fractions could reduce the sensitivity of total dose to some parameters for the low abundance radionuclides. In addition, parameter 'input rank correlations' were not applied in order to maximize variability and corresponding parameter sensitivity. The RESRAD Uncertainty Reports are provided in TSD 14-010. Table 6-15 provides the parameters with |PRCC| values greater than 0.25 and the reported PRCC values.

The PRCC values listed are the highest individual values from the three runs made in the RESRAD Uncertainty Analysis. Table 6-16 and Table 6-17 list the selected 75th or 25th percentile deterministic values from the NUREG/CR-6697 distributions for the sensitive parameters (i.e., those listed in Table 6-15).

The values in Tables 6-16 and 6-17 were used in the RESRAD modeling to determine the Groundwater Exposure Factors. The median of the distributions from NUREG/CR-6697 were assigned to Priority 1 and 2 parameters that were not sensitive (i.e., not listed in Table 6-15).

Table 6-15 BFM Uncertainty Analysis Results for Parameters with |PRCC| > 0.25

Parameter	PRCC Value							
	Co-60	Cs-134	Cs-137	Eu-152	Eu-154	Ni-63	Sr-90	H-3
Depth of Roots	0.33	NS[1]	NS	NS	NS	NS	NS	NS
Weathering Removal Constant of All Vegetation	-0.61	-0.88	-0.87	-0.87	-0.89	-0.78	-0.80	NS
Wet Weight Crop Yield of Fruit Grain and Non-Leafy Vegetables	NS	NS	NS	-0.51	-0.54	NS	NS	NS
Wet Foliar Interception Fraction of Leafy Vegetables	NS	NS	NS	0.56	0.59	NS	NS	NS
Plant Transfer Factor	NS	NS	NS	NS	NS	NS	0.31	NS
Meat Transfer Factor	0.90	0.86	0.85	0.75	0.77	0.32	0.59	NA
Milk Transfer Factor	0.56	0.91	0.90	NS	NS	0.97	0.68	NA
Saturated Zone Hydraulic Conductivity	NS	NS	NS	NS	NS	NS	-0.36	-0.54
Saturated Zone Hydraulic Gradient	NS	NS	NS	NS	NS	NS	-0.59	-0.77
Contaminated Zone Total Porosity	NS	NS	NS	NS	NS	NS	-0.41	-0.77
Density of Contaminated Zone	NS	NS	NS	NS	NS	NS	0.41	0.73

Note 1: NS indicates that the parameter is not sensitive

Table 6-16 BFM Deterministic Values for Sensitive Parameters from Table 6-12 that are Radionuclide Independent

Parameter	Percentile	Parameter Value
Depth of Roots	75th	3.1m
Weathering Removal Constant of All Vegetation	25th	21.5
Wet Weight Crop Yield of Fruit Grain and Non-Leafy Vegetables	25th	1.26 kg/m^2
Wet Foliar Interception Fraction of Leafy Vegetables	75th	0.70
Saturated Zone Hydraulic Conductivity	25th	1695
Saturated Zone Hydraulic Gradient	25th	0.0018
Contaminated Zone Total Porosity	25th	0.37
Density of Contaminated Zone	75th	1.68[1] g/cm^3

Note 1: Site specific density value of 1.8 used in the RESRAD run.

Table 6-17 BFM Deterministic Values for Sensitive Parameters from Table 6-12 that are Radionuclide Dependent

Radionuclide	Plant Transfer Factor 75th Percentile	Meat Transfer Factor 75th Percentile	Milk Transfer Factor 75th Percentile
Co-60	NS[1]	0.058	0.0032
Cs-134	NS	0.065	0.014
Cs-137	NS	0.065	0.014
Eu-152	NS	0.004	NS
Eu-154	NS	0.004	NS
Ni-63	NS	0.0092	0.032
Sr-90	0.59	0.013	0.0028
H-3	NS	NS	NS

Note 1: NS indicates that the parameter is not sensitive

The density of the contaminated zone was identified as sensitive and positively correlated. As noted in Table 6-16, the 75th Percentile of the NUREG/CR-6697 Attachment C distribution is 1.68 g/cm^3. However, the site-specific density value for sand is 1.8 g/cm^3 (TSD 14-006). Because the fill will be a combination of concrete and sand, and concrete density is 2.4 g/cm^3, the 1.8 g/cm^3 for sand is the minimum site-specific value and was therefore applied.

6.6.5. BFM RESRAD Parameter Set and Groundwater Exposure Factor Calculation

The final RESRAD parameter set used to calculate the Groundwater (GW) Exposure Factors is provided in Attachment 2. The RESRAD BFM Summary Report and Concentration Report are provided in TSD 14-010.

A few of the parameters required simple calculations, which are described in the Attachment 2 parameter table. A calculation was also performed to develop a nominal value of 2250 m^3 for the well pumping rate parameter including drinking water, livestock consumption and irrigation in the Resident Farmer Scenario. This calculation is provided at the end of Attachment 2.

The GW Exposure Factors are calculated by dividing the maximum dose, which occurs at t=0 in the RESRAD simulations for all ROC, by the well water concentration at t=0 as shown in Equation 6-2. The RESRAD results for each ROC and the calculated GW Exposure Factors are provided in Table 6-18.

Equation 6-2

$$\textit{GW Exposure Factor}(i) = \textit{Total Dose }(i)/\textit{GW Concentration }(i)$$

where:

GW Exposure Factor (i) = Dose from unitized groundwater concentration (mrem/y per pCi/L)
Total Dose (i) = Total dose from radionuclide (i) calculated by RESRAD (mrem/yr)
GW Concentration (i) = Groundwater concentration for radionuclide (i) calculated by RESRAD (pCi/L)

Table 6-18 RESRAD Results and GW Exposure Factors for BFM model

Radionuclide	Dose (mrem/y)			Groundwater Concentration (pCi/L)	GW Exposure Factor (mrem/y per pCi/L)
	Drinking Water	Plant/Meat/ Milk	Total		
Co-60	5.40E-02	5.82E-02	1.12E-01	4.48E+00	2.50E-02
Cs-134	6.58E-01	1.28E+00	1.94E+00	2.21E+01	8.75E-02
Cs-137	5.23E-01	1.01E+00	1.54E+00	2.21E+01	6.94E-02
Eu-152	3.17E-02	6.30E-03	3.80E-02	1.05E+01	3.62E-03
Eu-154	4.61E-02	9.14E-03	5.52E-02	1.05E+01	5.26E-03
H-3	1.38E-01	7.88E-02	2.17E-01	4.89E+03	4.43E-05
Ni-63	4.42E-03	1.13E-02	1.57E-02	1.61E+01	9.78E-04
Sr-90	2.87E+01	1.49E+01	4.36E+01	3.99E+02	1.09E-01

6.6.6. BFM Groundwater Dose Factors

BFM GW Dose Factors are dose conversion factors in units of mrem/yr per mCi total inventory. The Resident Farmer dose includes the exposure pathways listed in section 6.5.4. The BFM GW Dose Factor accounts for all of the exposure pathways, except the drilling spoils pathway which is addressed in section 6.6.7. The BFM GW Dose Factor is calculated using Equation 6-3. BFM GW Dose Factors were calculated for each Basement and each ROC in TSD 14-010 and are provided in Table 6-19.

Equation 6-3

$$\boldsymbol{BFM\ GW\ Dose\ Factor(i,b)} \\ = \boldsymbol{GW\ Concentration\ Factor(i,b)} \times \boldsymbol{GW\ Exposure\ Factor(i)}$$

where:

BFM GW Dose Factor (i,b) = BFM GW Dose Factor for radionuclide (i) and Basement (b) (mrem/y per mCi)

GW Concentration Factor (i,b) = Groundwater Concentration Factor for Radionuclide (i) and Basement (b) (pCi/L per mCi)

GW Exposure Factor (i,b) = Groundwater Exposure Factor for Radionuclide (i) and Basement (b) (mrem/yr per pCi/L)

Table 6-19 BFM GW Dose Factors (mrem/yr per mCi Total Inventory)

Nuclide	Auxiliary (mrem/yr per mCi)	Containment (mrem/yr per mCi)	Fuel[(1)] (mrem/yr per mCi)	Turbine (mrem/yr per mCi)	Crib House /Forebay (mrem/yr per mCi)	WWTF (mrem/yr per mCi)
Co-60	1.00E-04	1.14E-02	NA	2.87E-03	2.85E-03	5.21E-01
Cs-134	9.27E-03	1.98E-01	NA	4.94E-02	4.91E-02	9.03E+00
Cs-137	2.64E-02	1.57E-01	NA	3.92E-02	3.90E-02	7.17E+00
Eu-152	5.96E-05	3.87E-03	NA	9.69E-04	9.64E-04	1.75E-01
Eu-154	6.77E-05	5.62E-03	NA	1.41E-03	1.40E-03	2.56E-01
H-3	6.21E-03	2.72E-02	NA	6.80E-03	6.75E-03	1.23E+00
Ni-63	2.86E-04	1.61E-03	NA	4.01E-04	4.00E-04	7.31E-02
Sr-90	3.29E-01	4.51E+00	NA	1.13E+00	1.12E+00	2.06E+02

(1) As discussed in section 6.5.4, the BFM GW Dose Factors are not applicable to the SFP/Transfer Canal.

Table 6-19 includes an adjustment to the Table 6-18 peak groundwater concentration factors for the Crib House/Forebay. A revision to the demolition plan for the Crib House/Forebay was made that entailed leaving interior walls as opposed to removing them. This resulted in a decrease in the basement mixing volume as compared to that assumed in the DUST-MS modeling provided in TSD-14-009 and a corresponding increase in the fill and groundwater

concentrations calculated in TSD 14-009. The BFM GW DFs are directly proportional to the groundwater concentrations which are inversely proportional to the ratio of revised/original mixing volumes. The ratio of the revised/original mixing volumes for the Crib House/Forebay was calculated in TSD 14-014, Revision 1 and determined to be 0.86. The Crib House/Forebay BFM GW Dose Factors were therefore adjusted higher by the inverse of 0.86 or a factor of 1.16. Note that the Crib House/Forebay surface area was also adjusted to account for the additional remaining walls but the change in surface area does not affect the calculation of the BFM Dose Factors because the unit inventory approach used was independent of surface area.

6.6.7. BFM Drilling Spoils Dose Factors

The BFM Drilling Spoils scenario addresses one of the BFM exposure pathways listed in section 6.5.4 by calculating the dose from residual radioactivity in fill material (resulting from release from surfaces to clean fill after backfill) which is brought to the surface during the installation of a well in the basement. The activity remaining in the concrete surfaces, if any, is also included in the drilling spoils source term. The drilling spoils exposure pathway was included after initial screening in Zion*Solutions* TSD 14-021 "Basement Fill Model (BFM) Drilling Spoils and Alternate Exposure Scenarios" (Reference 6-28) indicated that the pathway could potentially contribute greater than 10% of the total BFM dose. TSD 14-021 also provides the BFM Drilling Spoils Dose Factor calculations. BFM Drilling Spoils Dose Factors are calculated in units of mrem/yr per mCi total inventory.

The source term for the BFM Drilling Spoils scenario is the average concentration in fill, and remaining in concrete, at the time of maximum groundwater concentration which is the time used to assess exposure for all other BFM pathways. As discussed previously, the fill is clean at the time of license termination but is assumed to adsorb activity after release from the concrete surfaces (or steel liner for Containment).

For Basements with instant release assumptions, the maximum groundwater concentrations occur at t=0 for all radionuclides. The remaining fraction in concrete is assumed to be zero since all activity is released to the water. For Basements with diffusion controlled release (the Auxiliary Basement and the SFP/Transfer Canal), the time of maximum groundwater (and fill) concentrations is a function of half-life and diffusion coefficient, and therefore radionuclide-specific. The corresponding fractions of inventory remaining in concrete at the time of maximum groundwater concentration are also radionuclide-specific. To ensure conservatism and consistency in the BFM source term, the maximum fill concentrations (which occur at the time of maximum groundwater concentrations) are applied for each radionuclide regardless of when the maximum occurs.

There are a number of ways that installers handle and dispose of drilling spoils, including the use of slurry pits, tanks, and dumping the drilling spoils on the existing surface soils. The use of pits would likely involve additional dilution by refilling the pit with the material excavated during its construction. As a conservative assumption, no dilution of the spoil material is assumed after being brought to the surface.

The borehole diameter is assumed to be 8 inches to accommodate the installation of a 4 inch diameter casing. The well is assumed to be drilled into the basement fill down to the concrete floor where refusal is met and drilling stopped. The extent of drilling into concrete is

conservatively assumed to be sufficient to capture 100 percent of the remaining residual radioactivity in concrete. The volume of spoil material brought to the surface is calculated based on the borehole diameter and depth of drilling which is defined as the distance from the ground surface to the bottom of the Basement. All material, including the concrete, fill, and clean overburden is brought to the surface where it is uniformly mixed and spread over a circular area to a depth of 0.15 m.

The dose from the circular area at the surface is calculated using the surface soil DCGLs and Area Factors (AF) (see section 6.9 for soil DCGL calculations). As described in TSD 14-021, the size of the area over which the drilling spoils are spread ranges from 0.92 m^2 to 3.56 m^2, depending on the Basement. The BFM Drilling Spoils Dose Factors are calculated in TSD 14-021 for each Basement and each ROC and are provided Table 6-20. Note that the Drilling Spoils Dose Factors for H-3 are all zero because the distribution coefficient is zero and there is no adsorption onto fill or remaining in concrete.

Table 6-20 BFM Drilling Spoils Dose Factors (mrem/yr per mCi Total Inventory)

Nuclide	Auxiliary (mrem/yr per mCi)	Containment (mrem/yr per mCi)	Fuel (mrem/yr per mCi)	Turbine (mrem/yr per mCi)	Crib House /Forebay (mrem/yr per mCi)	WWTF (mrem/yr per mCi)
Co-60	1.07E-02	2.97E-02	1.58E-01	9.58E-03	2.07E-02	2.26E-01
Cs-134	6.29E-03	1.72E-02	9.41E-02	5.54E-03	1.19E-02	1.31E-01
Cs-137	3.22E-03	7.27E-03	4.83E-02	2.35E-03	5.05E-03	5.57E-02
Eu-152	5.02E-03	1.38E-02	7.46E-02	4.45E-03	9.58E-03	1.05E-01
Eu-154	5.57E-03	1.46E-02	8.25E-02	4.73E-03	1.02E-02	1.12E-01
H-3	0.00E+00	0.00E+00	1.45E-09	0.00E+00	0.00E+00	0.00E+00
Ni-63	3.23E-08	5.61E-08	3.78E-07	1.86E-08	4.81E-08	4.16E-07
Sr-90	6.26E-05	1.39E-04	7.60E-04	4.61E-05	1.16E-04	1.049E-04

Table 6-20 includes an adjustment to the Table 6-18 peak groundwater concentration factors for the Crib House/Forebay which are directly proportional to the peak fill concentrations used in the Drilling Spoils scenario. A revision to the demolition plan for the Crib House/Forebay was made that entailed leaving interior walls as opposed to removing them. This resulted in a decrease in the basement mixing volume as compared to that assumed in the DUST-MS modeling provided in TSD-14-009 and a corresponding increase in the fill and groundwater concentrations calculated in TSD 14-009. The Drilling Spoils Dose Factors are directly proportional to the fill concentrations which are inversely proportional to the ratio of revised/original mixing volumes. The ratio of the revised/original mixing volumes for the Crib House/Forebay was calculated in TSD 14-014, Revision 1 and determined to be 0.86. The Crib House/Forebay BFM Drilling Spoils DFs were therefore adjusted higher by the inverse of 0.86 or a factor of 1.16. Note that the Crib House/Forebay surface area was also adjusted to account for the additional remaining

walls but the change in surface area does not affect the calculation of the BFM Dose Factors because the unit inventory approach used was independent of surface area.

6.6.8. Basement Surface DCGLs

Derived Concentration Guideline Levels, in units of pCi/m^2 of basement surface area, were calculated in Reference 6-13, section 2.5 for the BFM Groundwater and BFM Drilling Spoils scenarios individually and are designated as the $DCGL_{BS}$ (Basement Scenario DCGLs). The Groundwater $DCGL_{BS}$ and Drilling Spoils $DCGL_{BS}$ are combined to generate the Basement DCGL ($DCGL_B$) which represent the combined dose from both the groundwater and drilling spoils scenarios. The $DCGL_B$ is directly analogous to the $DCGL_W$ as defined in MARSSIM and is the DCGL used during FSS to demonstrate compliance.

A DCGL was calculated for each basement surface survey unit. For the purpose of this calculation, a "surface" includes all concrete walls and floors of the basements or the steel liner on the walls and floors of Containment. The areal extent of the walls and floors is the "surface area" and includes all contamination, volumetric at depth and on the surface, within the defined area. Note that embedded pipe and penetrations will remain in the basements. Embedded pipe and penetrations within or interfacing each basement structure were designated as separate survey units, with separate DCGL calculations for each, as described in sections 6.13 and 6.14.

After the basement surface areas were adjusted as described in section 6.6.8.1, the $DCGL_{BS}$ values were calculated for each basement using Equation 6-4. Adjustment factors of 0.90 for Containment and 0.95 for all other basements are included in Equation 6-4 to account for the dose from insignificant contributors (see section 6.5.2.3). The $DCGL_B$ values were calculated by combining the Groundwater and Drilling Spoils $DCGL_{BS}$ values using Equation 6-5.

Equation 6-4

$$DCGL_{BS,i} = \frac{25}{BFM\ Scenario\ DF_i} * \frac{1}{SA_{b(adjusted)}} * 1E+09 * IC\ Dose\ Adjusment$$

Where:

$DCGL_{BS,\,i}$	= Groundwater or Drilling Spoils scenario DCGL for radionuclide (i) (pCi/m^2)
BFM Scenario DF_i	= Basement Fill Model Dose Factor for radionuclide (i) (mrem/yr per mC)
1E+09	= Conversion factor (pCi/mCi)
25	= 25 mrem/yr dose criterion
$SA_{b\ (adjusted)}$	= 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 - see section 6.5.2.3).

Equation 6-5

$$DCGL_{Bi} = \frac{1}{\left(\frac{1}{GW\ DCGL_{BSi}} + \frac{1}{DS\ DCGL_{BSi}}\right)}$$

Where:

$DCGL_{Bi}$	= Basement Surface DCGL for radionuclide (i) (pCi/m^2)
GW $DCGL_{BSi}$	= Groundwater scenario DCGL for radionuclide (i) (mrem/yr per mCi)
DS $DCGL_{BSi}$	= Drilling Spoil scenario DCGL for radionuclide (i) (mrem/yr per mCi)

6.6.8.1. Basement Surface Area Adjustments

Basement surface area adjustments were required to ensure that the DCGLs account for the contribution of residual radioactivity from basements/structures that cannot, on their own, support a water supply well but are hydraulically connected to a basement that can support a well. These include the Circulating Water Intake Pipes, Circulating Water Discharge Tunnels (and associated piping) , Buttress Pits/Tendon Tunnels, and the SFP/Transfer Canal. The surface area adjustments result in lowering the DCGL concentrations (pCi/m^2) in the affected basements and structures, from that which would be calculated for each individually, by requiring the allowable total activity to be uniformly distributed over the larger, combined surface areas.

The first area adjustment is to the Turbine Basement and Crib House/Forebay. As stated in Table 6-9, the activity in the Circulating Water Intake Pipes is included in both the Crib House/Forebay and the Turbine Basement. The activity in the Circulating Water Discharge Tunnels is included with the Turbine Basement. The Intake Pipe has been grouted essentially eliminating the hydraulic connections. The major hydraulic connections between the Discharge Tunnels and the Turbine basement will be isolated as a part of the decommissioning process but two 48 inch diameter service water pipes that run between the Turbine Basement and the Discharge Tunnels will remain open and maintain the hydraulic connection, at least to some extent. For the purpose of the DCGL calculation, the hydraulic connections to the Intake Pipe and Discharge Tunnels are assumed to be fully regained in the future after degradation of the isolation barriers and grout.

The surface DCGL calculations account for the activity in the Intake Pipes and Discharge Tunnels by summing the surface areas of the connected structures and using the summed areas for the DCGL calculation. The Intake Pipe surface area is added to the Crib House/Forebay. The Intake Pipe is also connected to the Turbine basement and therefore, the Intake Pipe surface area is also added to the Turbine Basement. The activity in the Intake Pipe is conservatively assumed to be in both basements simultaneously. The Discharge Tunnel surface area is added to the Turbine Basement. There is also a group of pipes that are within the Turbine building and connected to the Discharge Tunnels including the remaining portions of the 12 foot diameter downcomer pipes, the 36 inch and 48 inch diameter standpipes, and the 48 inch diameter service water return pipes. There are also large diameter pipes on the east side of the Discharge Tunnel Valve House. The internal surface areas of these "Circulating Water Discharge Pipes "are also added to the summed area used for the Turbine Basement DCGL calculation.

The summed areas were then used as the "$SA_{b\ (adjusted)}$" term in Equation 6-4 to calculate the DCGLs for the Crib House/Forebay and Turbine Basement. As seen in Equation 6-4, increasing the surface area decreases the DCGLs. The lower DCGLs calculated for the Crib House/Forebay and Turbine Basement, based on the summed areas, were then also applied to the Intake Pipes and Discharge Tunnels, respectively. The lower DCGL for either the Crib House/Forebay or the Turbine Basement will be applied to the Intake Pipe. However, this is a minor distinction given that FSS measurements in the Intake Pipe have all been below detection limits which are orders of magnitude below the DCGLs. The Discharge Tunnel FSS results will be included in the dose assessment for the Turbine Basement. The Intake Pipe FSS results will be included with both the Crib House/Forebay and Turbine Basement dose assessments (see LTP Chapter 5, section 5.5.6. for discussion of basement surface dose assessment)

A second surface area adjustment was required to account for the contribution of residual radioactivity in the SFP/Transfer Canal to the groundwater pathway. As discussed in section 6.5.4, the SFP/Transfer Canal geometry could not support a water well and therefore the BFM Groundwater Dose Factor was set to zero (see Table 6-19). However, the potential for the residual radioactivity in the SFP/Transfer Canal to contribute to the groundwater pathway is accounted for by adding the SFP/Transfer Canal surface area to the Containment Basement and Auxiliary Basement surface areas in the DCGL calculation. The activity could mix with the Containment Basement through the Fuel Transfer Tube. Activity could mix with the Auxiliary Basement through an opening created by removing the wall between the Transfer Canal and Auxiliary Basement during demolition. The surface area adjustment, and corresponding DCGL calculations, conservatively assume that the activity in the SFP/Transfer Canal is in both the Containment and Auxiliary Basement simultaneously.

The Buttress Pits and Tendon Tunnels are hydraulically connected to the Steam Tunnels. The surface areas of these structures are therefore added to the Turbine Basement.

The inputs to the calculation of adjusted surface areas are provided in Tables 6-21 to 6-23. The SFP/Transfer Canal and WWTF do not require adjustment. The surface areas in Table 6-21 were used in the DCGL calculations for the SFP/Transfer Canal and WWTF. The DCGL calculations, using Equations 6-4 and 6-5, are performed and documented in Reference 13, section 2.5.

Table 6-21 Basement Surface Areas (Walls and Floors)

Basement	Wall and Floor Surface Area (1) m^2
Auxiliary Basement	6503
Containment Basement	2759
Turbine Building Basement	14864
SFP/Transfer Canal	723
Crib House/Forebay	13842
WWTF	1124

(1) Reference: TSD 14-014 Revision 1, Table 64

Table 6-22 Surface Areas for Circulating Water Intake Pipe, Circulating Water Discharge Tunnel, Circulating Water Discharge Pipes and Buttress Pits/Tendon Tunnels

Structure	Surface Area ft^2	Surface Area m^2
Circulating Water Discharge Tunnels (1)	52400	4868
Circulating Water Intake Pipes (2)	47491	4412
Circulating Water Discharge Pipes (3)	11570	1075
Buttress Pits/Tendon Tunnels (4)	20626	1916

(1) Reference TSD 14-014, Table 64.
(2) TSD 14-016, Table 46
(3) TSD 14-016, Table 50.
(4) TSD 14-014, Rev 3, Tables 60 & 63 and TSD 13-005 Rev 1 Table 15

Table 6-23 Adjusted Basement Surface Areas for DCGL Calculation

Basement	Structures Included in Total SA/V Calculation	Total SA m^2
Containment	Containment + SFP/Transfer Canal	3482
Auxiliary	Auxiliary + SFP/Transfer Canal	7226
Turbine	Turbine + Circulating Water Discharge Tunnel + Circulating Water Intake Pipe + Circulating Water Discharge Pipes + Buttress Pits/Tendon Tunnels	27135
Crib House/Forebay	Crib House/Forebay + Circulating Water Intake Pipe	18254
SFP/Transfer Canal (1)	SFP/Transfer Canal	723
WWTF[1]	WWTF	1124

(1) No area adjustment required. The basement surface areas in Table 6-21 are used in the DCGL calculation.

The Groundwater and Drilling Spoils $DCGL_{BS}$ values were calculated using Equation 6-4 with inputs from the BFM Dose Factors in Tables 6-19 and 6-20, respectively, and the surface areas in Table 6-23 for the "$SA_{b\ (adjusted)}$" term. The results are provided in Tables 6-24 and 6-25. The $DCGL_B$ values were calculated using Equation 6-5 with results provided in Table 6-26.

Table 6-24 Adjusted BFM Groundwater Scenario $DCGL_{BS}$ (Adjusted for IC Dose)

Nuclide	Auxiliary (pCi/m^2)	Containment (pCi/m^2)	SFP/ Transfer Canal ((pCi/m^2)	Turbine (pCi/m^2)	Crib House/ Forebay (pCi/m^2)	WWTF (pCi/m^2)
Co-60	3.28E+10	5.65E+08	NA	3.05E+08	4.57E+08	4.05E+07
Cs-134	3.55E+08	3.27E+07	NA	1.77E+07	2.65E+07	2.34E+06
Cs-137	1.25E+08	4.12E+07	NA	2.23E+07	3.34E+07	2.95E+06
Eu-152	5.52E+10	1.67E+09	NA	9.03E+08	1.35E+09	1.21E+08
Eu-154	4.85E+10	1.15E+09	NA	6.22E+08	9.29E+08	8.24E+07
H-3	5.30E+08	2.38E+08	NA	1.29E+08	1.93E+08	1.71E+07
Ni-63	1.15E+10	4.02E+09	NA	2.18E+09	3.25E+09	2.89E+08
Sr-90	9.98E+06	1.43E+06	NA	7.74E+05	1.16E+06	1.03E+05

Table 6-25 Adjusted BFM Drilling Spoils Scenario $DCGL_{BS}$ (Adjusted for IC Dose)

Nuclide	Auxiliary (pCi/m^2)	Containment (pCi/m^2)	SFP/ Transfer Canal (pCi/m^2)	Turbine (pCi/m^2)	Crib House/ Forebay (pCi/m^2)	WWTF (pCi/m^2)
Co-60	3.07E+08	2.18E+08	2.08E+08	9.13E+07	6.28E+07	9.34E+07
Cs-134	5.23E+08	3.77E+08	3.49E+08	1.58E+08	1.09E+08	1.61E+08
Cs-137	1.02E+09	8.89E+08	6.80E+08	3.73E+08	2.58E+08	3.80E+08
Eu-152	6.54E+08	4.69E+08	4.41E+08	1.97E+08	1.36E+08	2.01E+08
Eu-154	5.91E+08	4.41E+08	3.98E+08	1.85E+08	1.28E+08	1.89E+08
H-3	2.26E+15	4.45E+15	2.26E+16	6.02E+14	8.95E+14	1.45E+16
Ni-63	1.02E+14	1.15E+14	8.69E+13	4.71E+13	2.70E+13	5.08E+13
Sr-90	5.25E+10	4.63E+10	4.32E+10	1.90E+10	1.12E+10	2.03E+10

Table 6-26 Adjusted Basement $DCGL_B$ (Adjusted for IC Dose)

Nuclide	Auxiliary (pCi/m^2)	Containment (pCi/m^2)	SFP/ Transfer Canal (1) (pCi/m^2)	Turbine (pCi/m^2)	Crib House/ Forebay (pCi/m^2)	WWTF (pCi/m^2)
Co-60	3.04E+08	1.57E+08	1.57E+08	7.03E+07	5.52E+07	2.83E+07
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
H-3	5.30E+08	2.38E+08	2.38E+08	1.29E+08	1.93E+08	1.71E+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

(1) DCGL for SFP/Transfer Canal set equal to the lower of either the Auxiliary or Containment DCGL Containment DCGL was lower for all ROC therefore SFP/Transfer Canal DCGL set equal to Containment

6.6.9. Basement Surface Elevated Areas

Class 1 survey units that pass the Sign test but have small areas with concentrations exceeding the $DCGL_B$ would be tested to demonstrate that these small areas meet the dose criterion using the Elevated Measurement Comparison (EMC). There are currently seven Class 1 areas at Zion, the Auxiliary Basement, the SFP/Transfer Canal, the Unit 1 and Unit 2 Containment basements (including the Under-Vessel area and the exposed steel liner above the 565 foot elevation) and the WWTF (see LTP Chapter 5 Table 5-18 for survey unit designations in all basements).

Area Factors (AF) would be required to perform the EMC test. The AF is defined as the magnitude by which the concentration within the small area of elevated activity can exceed the $DCGL_B$ while maintaining compliance with the dose criterion.

The BFM is a mixing model that is independent of the distribution of the residual radioactivity. The calculation of the $DCGL_B$ values support the mixing assumption by assuming uniform contamination over all basement walls and floors. An individual FSS ISOCS measurement that exceeds the $DCGL_B$ could conceptually be acceptable if it satisfies an EMC test. For example, assuming full mixing, the AF for an Auxiliary Basement FSS ISOCS measurement could be as high as the total surface area divided by the ISOCS FOV (7226/28 = 258). However, consistent with the bounding approach used to develop the conceptual model, and to support the assumption of uniform mixing, no AF will be assigned to the results of FSS ISOCS measurements. Any FSS ISOCS measurement that exceeds the $DCGL_B$ (or an SOF of one considering all ROC) will result in remediation. Note that as discussed in LTP Chapter 5, lower, Operational DCGLs, will be used as investigation levels as opposed to the $DCGL_B$.

6.7. Alternate Exposure Scenarios for Backfilled Basements

Two alternate scenarios were evaluated in TSD 14-021 that involve a change to the "as left" backfilled geometry in the Resident Farmer scenario. A third alternate scenario was evaluated in TSD 14-010 related to the drilling spoils pathway.

The first alternate scenario entails construction of a house basement within the fill material. Note that the assumed three meter depth of the basement excavation is insufficient to encounter fill material potentially containing residual radioactivity (resulting from leaching of residual radioactivity from surfaces after backfill) assuming the Basement is not constructed within the saturated zone. However, a simple check of direct radiation dose, assuming a residual radioactivity inventory at the hypothetical maximum levels based on the Basement Dose Factors, was conducted to confirm the expectation that the dose would be negligible. The dose calculation is provided in TSD 14-021 with a result of 0.03 mrem/yr for the Auxiliary Basement and 0.5 mrem/yr for the SFP/Transfer Canal. The remaining Basements do not contain significant inventories and were not assessed.

The second alternate scenario assumes large scale excavation of parts or all of the backfilled structural concrete and fill after the ISFSI is decommissioned (assumed to be 10 years after license termination). A simple calculation was performed to estimate the average concentrations in the excavated concrete and fill assuming a residual radioactivity inventory at the hypothetical maximum levels and the ZNPS radionuclide mixture provided in Table 6-3. The assessment was performed for all basements although only the Auxiliary Basement is expected to contain significant levels of residual radioactivity at license termination. (assuming all concrete is removed from Containment).

If a large-scale excavation of the basements were to occur, it would not be for residential use but to develop the property for industrial use. The cost and technical challenges of the excavation required for the deep basements, that are all below the water table, would only be justified for a large scale industrial project that would be present on the site for decades. Therefore, for the assessment of the large scale industrial excavation scenario an Industrial Use soil "DCGL" ($DCGL_I$) was developed assuming an industrial use scenario (see Reference 6-13, section 6). A period of 10 years was assumed before excavation begins. The $DCGL_I$ was used only for the evaluation of the "less likely but plausible" alternate excavation scenario and is not proposed for any compliance demonstration.

The average activity in the excavated concrete and fill was compared to the soil $DCGL_I$ values provided in Reference 6-13, Table 18, as a simple screening assessment for this low probability scenario. Applying the summation rule, and conservatively performing the calculation for each Basement separately, the excavation dose was calculated. The dose results from TSD 14-021, Revision 1, Tables 22 and 26 are reproduced in Table 6-27. In addition, the dose from excavated fill was also evaluated in TSD 14-010 at the Operational DCGL dose fraction of 0.448 assigned in Zion*Solutions* TSD 17-004 "Operational Derived Concentration Guideline Levels for Final Status Surveys" (Reference 6-31). The results are provided in the last row of Table 6-27.

Table 6-27 Large Scale Industrial Excavation Alternate Scenario Dose

	Auxiliary (mrem/yr)	Containment (mrem/yr)	SFP/ Transfer Canal (mrem/yr)	Turbine (mrem/yr)	Crib House/ Forebay (mrem/yr)	WWTF (mrem/yr)
Concrete AF	6.66	2.65	10.72	2.37	7.51	0.64
Concrete No AF	8.57	3.90	20.02	2.83	9.96	1.43
Fill AF	4.24	2.52	16.77	2.71	3.17	0.40
Fill No AF	4.89	2.97	31.40	3.05	3.63	0.74
Fill No AF at Operational DCGL[1]	2.19	1.33	14.07	1.37	1.63	0.33

(1) Dose values in "mrem/yr No AF" row multiplied by Operational DCGL dose fraction for basements of 0.448.

The dose from the "less likely but plausible" industrial excavation scenario was calculated applying the AFs (interpolated) from Table 6-40 to the surface area covered by the excavated material assuming the material is spread over a one meter depth on the ground surface. However, the Table 6-40 AFs were calculated for the Resident Farmer scenario resulting in higher values than would be applicable to for the Industrial Scenario due to elimination of the plant pathway. Therefore, the dose was also calculated without AFs to provide a maximum value.

NUREG-1757 recommends that "greater assurance" be provided to demonstrate that a less likely but plausible land use is "unlikely" if the dose from the scenario is "significant". The maximum dose from all basements, for excavated fill and concrete, was 31.40 mrem/yr for the SFP/Transfer Canal fill assuming no AF adjustment and activity at the $DCGL_B$ concentrations. The dose for the SFP/Transfer Canal including an AF adjustment was 16.77 mrem/yr. If AFs were calculated for the Industrial Use scenario it would likely result in a dose less than 25 mrem/yr at the $DCGL_B$ concentrations. However, the actual source term for the SFP/Transfer canal will be reduced by a factor of 0.448 in order to comply with Operational DCGLs listed in LTP Chapter 5, Table 5-4. The maximum Large Scale Industrial Excavation Dose from fill at Operational DCGL concentrations is 14.07 mrem/yr (0.448*31.40). The low dose values reported in Table 6-29 for the "less likely but plausible" large-scale excavation land use are considered not "significant".

The third alternate scenario assumes that the drill for a water well encounters penetrations, embedded pipe, and basement surfaces with no activity released to the fill. The activity in the penetration, embedded pipe or basement surface that is captured by the drill is assumed to be brought to the surface in the drilling spoils. The source term was the maximum ROC mixture hypothetically allowable given the DCGLs and radionuclide mixture percentages. Two receptors were evaluated; the resident farmer and a worker. The calculation details are provided in TSD 14-010, Revision 6, section 12. The maximum hypothetical dose for the Resident Farmer are provided in Tables 6-28 to 6-30. The maximum hypothetical worker dose from all sources (penetrations, embedded pipe, basement surfaces) was 4.59 mrem/yr. The drilling spoils worker dose was also calculated for each ROC individually, for all penetrations, embedded pipe, and

basement surfaces, assuming residual radioactivity at the DCGL concentrations. The maximum worker dose from any individual ROC assuming DCGL concentrations was 7.61 mrem/yr.

Table 6-28 The maximum hypothetical resident farmer doses from penetrations for the Alternate Drilling Spoils Scenario (assuming well drilled 30 years after license termination)

Auxiliary (mrem/yr)	Containment (mrem/yr)	Fuel (mrem/yr)	Turbine (mrem/yr)	Crib House/ Forebay (1) (mrem/yr)	WWTF (1) (mrem/yr)
6.97	6.96	23.54	6.15	NA	NA

(1) No penetrations in Crib House/Forebay or WWTF

Table 6-29 The maximum hypothetical resident farmer doses from embedded pipe for the Alternate Drilling Spoils Scenario (assuming well drilled 30 years after license termination)

Auxiliary Floor Drain (mrem/yr)	Containment IC Sump Drain (mrem/yr)	Steam Tunnel Floor Drain (mrem/yr)	Tendon Tunnel Floor Drain (mrem/yr)	Turbine Floor Drain (mrem/yr)
13.1	3.39	71.16	16.51	20.16

Table 6-30 The maximum hypothetical resident farmer dose from basement surfaces for the Alternate Drilling Spoils Scenario (assuming well drilled 30 years after license termination)

Auxiliary (mrem/yr)	Containment (mrem/yr)	Fuel (mrem/yr)	Turbine (mrem/yr)	Crib House/ Forebay (mrem/yr)	WWTF (mrem/yr)
0.34	0.12	0.16	0.06	0.08	0.01

The alternate drilling spoils scenario resident farmer dose for the Steam Tunnel Floor Drains is calculated to be 71.16 mrem/yr using the hypothetical maximum activity that could be allowed to remain. However, the actual levels of activity in these drains is expected to be orders of magnitude lower than the hypothetical maximum. The alternate drilling spoils scenario dose for all other embedded pipe, penetrations and basement surfaces are below 25 mrem/yr. The DCGLs for the Steam Tunnel Floor Drains will be reduced by a factor of 2.89 (71.16/25) which will reduce the maximum dose to 25 mrem/yr. The commitment to reduce the Steam Tunnel Floor Drain DCGLs is provided in LTP Chapter 5, section 5.5.5.

6.8. Soil Dose Assessment and DCGL

Site-specific DCGLs were developed for residual radioactivity in surface and subsurface soil that represent the 10 CFR 20.1402 dose criterion of 25 mrem/yr. A DCGL was calculated for each ROC.

Surface soil is defined as contamination contained in the first 0.15 m layer of soil. Subsurface soil is defined as a layer of soil beginning at the surface that extends beyond 0.15 m. The

subsurface soil thickness is arbitrarily set to a 1 m depth. DCGLs are calculated for both the 0.15 m and 1 m thicknesses. Both the surface and subsurface DCGLs assume a continuous source term layer from the ground surface downward. There are no expectations of encountering soil contamination in a geometry consisting of a clean surface layer of soil over a contaminated subsurface soil layer.

6.8.1. Soil Source Term

During the initial characterization of impacted soils at ZNPS, 888 surface soil samples and 723 subsurface soil samples were taken and analyzed for plant-derived radionuclides. Cs-137 was detected at concentrations greater than MDC in 212 samples and Co-60 was detected at concentrations greater than MDC in 42 samples. The majority of the positive Cs-137 samples were in the range of background concentrations and unlikely to be plant-derived activity. The highest concentration of Cs-137 detected was 3.4 pCi/g in surface soils in a Class 1 open land survey unit located next to Unit 1 Containment. The highest level of Cs-137 detected in a surface soil sample taken from a Class 2 or Class 3 open land survey unit was 1.1 pCi/g. The highest concentration of Co-60 detected in any surface soil sample taken was 0.7 pCi/g.

For subsurface soil samples, Cs-137 was detected at concentrations greater than MDC in 15 samples and Co-60 was detected at concentrations greater than MDC in one sample. The highest level of Cs-137 detected in a subsurface soil sample was 1.0 pCi/g and the one sample where Co-60 was positively detected had a concentration of 0.1 pCi/g. In addition, nine surface soil samples and one subsurface soil samples where gamma spectroscopy indicated the presence of Co-60 and/or Cs-137 were analyzed for all ROC, including HTD radionuclides. No other plant-derived radionuclides were positively identified by the HTD analyses.

The results of surface and subsurface soil characterization in the impacted area of ZNPS indicate that there is minimal residual radioactivity in soil above background. However, the assessment of potential subsurface soil contamination is not complete at the time of this LTP submittal (Revision 0). Soil sampling in difficult to access areas such as under building foundations and surrounding buried structures has been deferred until access is more readily available. Based on the characterization survey results to date, ZSRP does not anticipate the presence of significant soil contamination in the areas remaining to be characterized.

6.8.2. Soil Radionuclides of Concern, Insignificant Contributor Dose and Surrogate Ratio

The radionuclides of concern for soil were determined in TSD 14-019 using the same process described in section 6.5.2 but replacing Basement DCGLs with soil DCGLs. 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. Note that the dose contribution from HTD radionuclides at ZNPS has been shown to be trivial based on characterization to date and is expected to be trivial at license termination. Gamma emitters are directly measured during the FSS.

The IC dose percentage for soil was calculated using the Table 6-2 mixture which is considered the most representative available. As a cross-check of the Table 6-2 mixture, the IC dose was

also calculated using a mixture comprised of the data from the 10 soil samples analyzed for the initial suite. The dose from individual samples was calculated in two ways; using the mean of the MDC values and using the mean of the actual net results. Due to the fact that essentially all of the soil characterization results were non-detect, with the exception of Cs-137 at very low levels and generally in the range of background, a significant and unrealistic bias in the IC dose calculation results was introduced by the use of MDC values. To provide a more realistic evaluation of the IC dose, a separate calculation was performed using the mean of actual net results. The analysis of individual soil samples was not considered meaningful given that all of the results were less than MDC. Other than low level Cs-137, the only positive result in soil samples was Co-60 in one sample at a concentration of 0.24 pCi/g.

The IC dose percentage for soil using the Table 6-2 mixture is provided in Table 6-30. The mixture and dose percentages for the ROC are also shown in Table 6-31. The mean IC dose and IC dose percentage from the two evaluations of the soil sample mixture are listed in Table 6-31.

The IC dose percentage assuming the "best estimate" Table 6-2 mixture was 0.171% (Table 6-31). The IC dose calculated using the results of the 10 soil samples analyzed for the initial suite and applying MDC values for all non-detect radionuclides (i.e., essentially all radionuclides) was 9.9% (Table 6-32). The more realistic calculation of IC dose using the actual reported results from the 10 soil samples, as opposed to MDCs, resulted in an IC dose percentage of 1.96% (Table 6-32). The actual results better correspond to the fact that the underlying assumption for the MDC calculation is that the mean net result is zero when no activity is present.

Table 6-31 Soil ROC Mixture and IC Dose Percentage Using the Table 6-2 "Best Estimate" Mixture.

Radionuclide	Mixture Percent	Percent Annual Dose
Co-60	0.91%	3.878%
Ni-63	23.48%	0.119%
Sr-90	0.05%	0.072%
Cs-134	0.01%	0.028%
Cs-137	74.60%	95.733%
Insignificant Contributor Percent	0.95%	0.171%
Total	100%	100%

Table 6-32 Soil IC Dose and Dose Percentage using Soil Sample Results

Data Used for Non-Detect	IC Dose mrem/yr	IC Dose Percentage (of 25 mrem/yr)
MDC Values	2.47	9.9%
Actual Reported Results	0.49	1.96%

The IC dose percentage for soil is considered to be between 0.17% and 1.96%. The 9.9% IC dose percentage is not considered representative of the actual site mixture due to MDC biasing issues. For example, it is very likely that the same MDC values would have been reported in the analysis of a soil sample from an offsite location, with the same calculation results. However, to ensure conservatism, an IC dose percentage of 10% will be used to adjust the ROC DCGLs for soil to conservatively account for the IC dose. The 10% (2.5 mrem/yr) value significantly exceeds the IC dose percentage calculated using the best estimate Table 6-2 mixture or the actual soil analytical results and provides a significant margin to account for uncertainty.

None of the 10 soil samples analyzed for the initial suite contained positive results for a HTD ROC, or any HTD radionuclide. Therefore, it is not technically feasible to develop radionuclide ratios for use with the surrogate approach during FSS. The maximum radionuclide ratios for Sr-90/Cs-137 and Ni-63/Co-60 calculated for the Auxiliary Basement in section 6.5.2.4 will be used in the surrogate evaluations for soil unless different values are justified by the results of continuing characterization or FSS HTD analysis results (see LTP Chapter 5 section 5.2.11).

6.8.3. Soil Exposure Scenario and Critical Group

The Resident Farmer exposure scenario and critical group as described in section 6.5.3 for the BFM also applies to the soil dose assessment. The Resident Farmer Scenario includes the following exposure pathways:

- Direct exposure to external radiation
- Inhalation dose from airborne radioactivity
- Ingestion dose from the following pathways:
  - Plants grown with irrigation water from onsite well
  - Meat and milk from livestock consuming fodder from fields irrigated with onsite well water and consuming water from onsite well
  - Drinking water from onsite well
  - Soil ingestion

6.9. Soil Computation Model – RESRAD v7.0

RESRAD version 7.0 was used to calculate DCGLs for surface and subsurface soil.

6.9.1. Parameter Selection

The parameters selection process described in section 6.6.3.1 and summarized in Figure 6-10 was used to select the RESRAD input parameters for soil. The vast majority of the behavioral, metabolic and physical parameters are the same as those developed for the BFM RESRAD modeling. However, the conceptual model for soil required changes to the following parameters:

- K_d values for site soil (sand) were selected based on the review provided by Brookhaven National Laboratory in TSD 14-004 (see Table 6-33),
- Cover depth = 0,
- "Time Since Material Placement" parameter set to zero,
- No initial contamination penetrates the saturated zone,

- An unsaturated zone is assumed to be present, and
- Non-dispersion groundwater model used.

Table 6-33 Distribution Coefficients for Surface and Subsurface Soil RESRAD Analysis

Radionuclide	Soil K_d (cm^3/g)
Co-60	1161
Ni-63	62
Sr-90	2.3
Cs-134	615
Cs-137	615

6.9.2. Uncertainty Analysis

Parameter uncertainty analysis was performed following the process described in section 6.6.3.1. The parameters used for the uncertainty analysis of the surface and subsurface soil dose modeling are the same as were used for the BFM RESRAD uncertainty analysis, with the exception of contaminated zone thickness. A 0.15 m thickness is used for surface soil and 1.0 m thickness for subsurface soil. The unsaturated zone depth was also adjusted to ensure that the depth to the water table remains constant for both the 0.15 m and 1.0 m contaminated zone thicknesses.

The RESRAD input parameters used for the uncertainty analysis of both surface and subsurface soil are provided in Attachment 3. Deterministic parameters were selected for behavioral, metabolic and Priority 3 physical parameters in accordance with the process in Figure 6-10. The majority of the Priority 1 and 2 physical parameters are assigned the parameter distributions from NUREG/CR-6697. Three site-specific Priority 1 and 2 physical parameters are assigned deterministic values in the uncertainty analysis including cover depth, precipitation, and well pumping rate (which does not have a recommended distribution in NUREG/CR-6697). The distribution coefficients were assigned either deterministic site-specific values based on the most conservative laboratory analysis of site soil as documented in TSD 14-004 or the distribution from NUREG/CR-6697 if site-specific data were not available. There are other site-specific parameters available, but these are included in the uncertainty analysis. The distributions from NUREG/CR-6697 were used to ensure that the appropriate level of justification is provided if one or more of these site-specific parameters are determined to be sensitive.

The uncertainty analysis was conservatively run for all ROC individually to maximize the number of parameters deemed sensitive. A more realistic approach would apply the radionuclide mixture fractions for ZNPS which could reduce the sensitivity of total dose to some parameters for the low abundance radionuclides. In addition, parameter 'input rank correlations' were not applied because this also maximizes variability and corresponding parameter sensitivity. Surface soil parameters that exhibited sensitivity to dose (i.e., with a |PRCC| result greater than 0.25) are listed in Table 6-34. The PRCC values listed are the highest individual values from the three runs made in the RESRAD Uncertainty Analysis. Tables 6-35 and 6-36 provide the selected 75th

or 25th percentile deterministic values for surface soil from the NUREG/CR-6697 distributions for the positively and negatively correlated parameters, respectively.

Table 6-34 Surface Soil DCGL Uncertainty Analysis Results for Parameters with |PRCC| >0.25

Parameter	PRCC Value				
	Co-60	Cs-134	Cs-137	Ni-63	Sr-90
Depth of Soil Mixing Layer	NS	-0.30	-0.36	-0.56	NS
Depth of Roots	-0.30	-0.47	-0.53	-0.78	-0.90
External Gamma Shielding Factor	0.99	0.96	0.93	NS	NS
Density of Contaminated Zone	0.59	0.32	NS	NS	NS
Plant Transfer Factor	0.34	0.57	0.63	0.86	0.96
Meat Transfer Factor	0.26	0.25	0.31	NS	NS
Milk Transfer Factor	NS	0.25	0.31	0.88	0.36

Table 6-35 Selected Deterministic Values for Surface Soil DCGL Sensitive Parameters from Table 6-21 That Are Radionuclide Independent

Parameter	Percentile	Parameter Value
Depth of Soil Mixing Layer	25th	0.15
Depth of Roots	25th	1.22m
External Gamma Shielding Factor	75th	0.40
Density of Contaminated Zone	75th	1.68 g/cm^3 (site-specific value of 1.8 g/cm^3 used)

Table 6-36 Deterministic Values for Surface Soil DCGL Sensitive Parameters from Table 6-21 that are Radionuclide Dependent

Radionuclide	Plant Transfer Factor 75th Percentile	Meat Transfer Factor 75th Percentile	Milk Transfer Factor 75th Percentile
Co-60	0.15	0.058	NS
Cs-134	0.078	0.065	0.014
Cs-137	0.078	0.065	0.014
Ni-63	0.092	NS	0.032
Sr-90	0.59	NS	0.0027

Subsurface soil parameters with a |PRCC| result greater than 0.25 are listed in and Table 6-37. Tables 6-38 and 6-39 provide the selected 75th or 25th percentile deterministic values for

subsurface soil. The median of the NUREG/CR-6697 distributions was assigned to the Priority 1 and 2 parameters that were not sensitive (i.e., not listed in Tables 6-34 and 6-37). The RESRAD Uncertainty Reports for each ROC are provided in TSD 14-010.

Table 6-37 Subsurface Soil DCGL Uncertainty Analysis Results for Parameters with |PRCC| > 0.25

Parameter	PRCC Value				
	Co-60	Cs-134	Cs-137	Ni-63	Sr-90
Depth of Roots	-0.45	-0.60	-0.69	-0.86	-0.93
External Gamma Shielding Factor	0.97	0.90	0.84	NS	NS
Plant Transfer Factor	0.67	0.83	0.88	0.96	0.98
Meat Transfer Factor	0.40	0.29	0.37	NS	NS
Milk Transfer Factor	NS	0.35	0.45	0.91	0.44

Table 6-38 Selected Deterministic Values for Subsurface Soil DCGL Sensitive Parameters from Table 6-28 that are Radionuclide Independent

Parameter	Percentile	Parameter Value
Depth of Roots	25th	1.22m
External Gamma Shielding Factor	75th	0.40

Table 6-39 Deterministic Values for Subsurface Soil DCGL Sensitive Parameters from Table 6-28 that are Radionuclide Dependent

Radionuclide	Plant Transfer Factor 75th Percentile	Meat Transfer Factor 75th Percentile	Milk Transfer Factor 75th Percentile
Co-60	0.15	0.058	NS
Cs-134	0.078	0.065	0.014
Cs-137	0.078	0.065	0.014
Ni-63	0.092	NS	0.032
Sr-90	0.59	NS	0.0027

6.10. RESRAD Results and Soil DCGLs

The surface and subsurface soil DCGLs were calculated using the deterministic parameter set provided in Attachment 4. The RESRAD Summary Reports are provided in TSD 14-010. The surface and subsurface soil DCGLs are provided in Table 6-40. Note that the values reported in

Table 6-40 also include adjustment to account for the 10% dose contribution from removed insignificant contributors (see section 6.8.2)

Table 6-40 Adjusted Surface Soil and Subsurface Soil DCGLs (Adjusted for IC Dose)

Radionuclide	Surface Soil DCGL (pCi/g)	Subsurface Soil 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

6.11. Soil Area Factors

The RESRAD modeling for soil assumes a large source term area of 64,500 m^2. Isolated areas of contamination that are smaller than 64,500 m^2 will have a lower dose for a given concentration. The ratio of the dose from the full source term area to the dose from a smaller area is defined as the AF.

Zion*Solutions* TSD 14-011, "*Soil Area Factors*" (Reference 6-30), calculates Area Factors (AF) for each ROC using RESRAD. The source area sizes ranged from 0.01 m^2 up to the full source area of 64,500 m^2. The AFs are relatively insignificant for areas greater than 100 m^2 and in practice are very unlikely to be required for greater areas. The RESRAD parameter set in Attachment 4 was used in TSD 14-011 to generate the AFs by varying the source term areas in each run. The RESRAD Summary Reports are provided in TSD 14-011. The surface soil and subsurface soil AFs for areas up to 100 m^2 are listed in Tables 6-41 and 6-42. A comprehensive list of AFs is provided in LTP Chapter 5, Table 5-16 and 5-17.

6.12. Buried Piping Dose Assessment and DCGL

Buried piping is defined as pipe that runs through soil. The dose assessment methods and resulting DCGLs for buried piping are described in detail in Zion*Solutions* TSD 14-015, "*Buried Pipe Dose Modeling & DCGLs*" (Reference 6-4). This section summarizes the methods and provides the resulting DCGLs for buried pipe.

As discussed in section 6.14, the maximum dose from buried piping will be added to the maximum dose from the open land survey unit(s). The rationale for this approach is identical to the standard process presented in MARSSIM for accounting for dose from elevated areas of residual radioactivity within an open land survey unit.

Table 6-41 Surface Soil Area Factors

Area (m^2)	Area Factors for Radionuclides of Concern				
	Cs-137	Co-60	Cs-134	Ni-63	Sr-90
1	1.50E+01	1.23E+01	1.33E+01	8.06E+03	8.90E+02
3	6.46E+00	5.24E+00	5.73E+00	2.73E+03	3.13E+02
10	3.06E+00	2.47E+00	2.72E+00	8.23E+02	1.03E+02
30	2.10E+00	1.68E+00	1.86E+00	2.75E+02	4.02E+01
100	1.62E+00	1.29E+00	1.44E+00	8.26E+01	1.64E+01

Table 6-42 Subsurface Soil Area Factors

Area (m^2)	Area Factors for Radionuclides of Concern				
	Cs-137	Co-60	Cs-134	Ni-63	Sr-90
1	2.04E+01	1.10E+01	1.52E+01	6.49E+03	1.50E+03
3	9.26E+00	4.91E+00	6.92E+00	2.17E+03	5.23E+02
10	4.48E+00	2.36E+00	3.35E+00	6.51E+02	1.64E+02
30	3.23E+00	1.70E+00	2.42E+00	2.17E+02	5.72E+01
100	2.59E+00	1.37E+00	1.95E+00	6.51E+01	1.76E+01

6.12.1. Buried Pipe Source Term and Radionuclides of Concern

Buried piping, with internal diameters ranging from one inch to 48 inches is expected to remain at the time of license termination. The Circulating Water Intake Pipes and Circulating Water Discharge Tunnels (and associated "Discharge Tunnel Pipe" located in the Turbine Building) are not considered buried pipe. The dose from residual radioactivity that is assumed to remain in the Intake Pipe and Discharge Tunnel is accounted for by adding the surface area (representing source term) to the applicable Basement in the DCGL calculation. The list of buried piping expected to remain is provided in TSD 14-016 (Reference 6-3).

The list of end-state buried pipe presented in TSD 14-016 is meant as a bounding condition. No pipe that is not listed in TSD 14-016 will be added to the end-state condition however, pipe can be removed from the list and disposed of as waste. As discussed below, the Buried Pipe DCGL is based on the summation of the surface area of all pipe to ensure conservatism regardless of the pipe location. Decreasing the amount of Buried Pipe to remain, i.e., removing more pipe than currently planned, would decrease the source term and corresponding dose. The DCGL becomes more conservative if less than 2,153 m^2 of pipe surface area remains and therefore no DCGL revision is necessary if additional pipe is removed.

None of the listed buried piping was associated with systems involving reactor coolant. Based on process knowledge, the majority of piping is expected to contain minimal residual radioactivity at levels well below the DCGLs.

To date, samples from piping systems have not been collected. Therefore, ZSRP is currently using the results of Auxiliary Basement concrete cores to represent the ROC and mixture for buried piping (see Table 6-3). Buried piping will be characterized as part of the continuing characterization program in accordance with LTP Chapter 2 section 2.5.

6.12.2. Buried Pipe Exposure Scenario and Critical Group

The critical group for the buried piping dose assessment is the Resident Farmer.

The buried pipe DCGL is determined for two scenarios; assuming that all pipe is excavated and assuming that all pipe remains *in situ*. Although unrealistic, for the purpose of the bounding modeling approach used, the dose from the two scenarios is summed to determine the Buried Pipe DCGL.

The excavation scenario assumes that all buried pipe is excavated and all activity on the internal surfaces of the pipes instantly released and mixed with surface soil. The *in situ* scenario assumes that all of the buried piping remains in the "as-left" condition at the time of license termination and that all activity is instantly released to adjacent soil. Two separate *in situ* calculations were performed. The first assumes that all pipes are located at 1 m below the ground surface and the second assumes that all pipes are located in the saturated zone.

6.12.3. Buried Pipe RESRAD Model for Excavation Scenario

The Excavation scenario assumes that all of the buried piping is excavated, brought to the surface and spread over a contiguous area equal to the internal surface area of the pipe. After being brought to the surface all of the activity on the internal surfaces of the pipe is assumed to instantly release and mix in a 0.15 m depth of surface soil.

RESRAD modeling is used to determine the dose from excavated buried pipe in units of mrem/yr per pCi/g. The RESRAD parameters used are the same as those used for surface soil DCGLs (see Attachment 4) with the following exceptions:

- Area of Contaminated Zone 2153 m^2
- Length Parallel to Flow SFP/Transfer Canal 46 m
- Cover Depth 0 m
- Unsaturated Zone Thickness 3.45 m

The Area of Contaminated Zone parameter is equal to the total internal surface are of all buried pipe. The complete list of buried pipe and total surface internal surface area is provided in Reference 6-4, Attachment 1. The length parallel to flow is the square root of the contaminated area under a nominal assumption that the shape of the contaminated area is square. The bases for the remaining parameters are self-explanatory. Note that the buried pipe list was revised after the RESRAD runs were made (the de-icing lines were initially listed twice). The total internal surface area was reduced from 2153 m^2 to 1539 m^2. The reduced area results in lower dose to source ratios (DSRs) (mrem/yr per pCi/g) and therefore the DSRs using 2153 m^2 were retained and used to calculate the Buried Pipe DCGLs which is conservative. Using the larger surface area also provides margin to account for the potential for additional buried pipe to be identified and added to the Reference 6-4, Attachment 1 list as decommissioning proceeds. Although not expected, if additional buried pipe is identified and added to the list, and the total surface area is

increased but remains below the 2153 m^2 assumed in the RESRAD model, the calculated buried pipe DCGLs would remain conservative. The area revision (and associated conservatism) also applies to the Insitu Saturated and Insitu Unsaturated scenarios RESRAD runs described in section 6.12.4.

6.12.4. Buried Pipe RESRAD Model for Insitu Scenarios

The Buried Pipe Insitu scenarios assume that the pipe remains in place. Two *in situ* geometries are evaluated. One scenario assumes that the buried pipe is in the unsaturated zone and a second scenario assumes that the pipe is in the saturated zone.

For the Insitu Unsaturated Zone scenario, the pipes are assumed to be located 1 m below the ground surface. The ZSRP decommissioning approach calls for removal of all material, including piping, to 3 feet below grade. Note that portions of the storm drain system that will remain in place and functional after license termination are closer to the surface than 1 m but this minor exception is considered insignificant. Assuming that the pipe is within 1m of the surface allows the roots to penetrate the 0.15 m thick *in situ* source which maximizes dose.

The RESRAD parameters used for the Buried Pipe Unsaturated Zone Insitu scenario are the same as those used for surface soil DCGLs (see Attachment 4) with the following exceptions:

- Area of Contaminated Zone — 2153 m^2
- Length Parallel to Flow — 46 m
- Cover Depth — 1 m
- Unsaturated Zone Thickness — 2.45

The second *in situ* scenario evaluated assumed that all buried pipe is in the saturated zone. This scenario is intended to conservatively address the possibility that GW could possibly enter some portions of the buried piping.

The RESRAD parameters used for the Buried Pipe Saturated Zone Insitu scenario are the same as those used for surface soil DCGLs (see Attachment 4) with the following exceptions:

- Area of Contaminated Zone — 2153 m^2
- Length Parallel to Flow — 46 m
- Cover Depth — 3.6 m
- Unsaturated Zone Thickness — 0 m
- Contaminated Fraction Below the Water Table — 1
- All Kds set to minimum site-specific value since dose is 100% from water pathways

6.12.5. Buried Pipe Uncertainty Analysis

An uncertainty analysis was performed for the three Buried Pipe dose scenarios to identify parameters that are sensitive in the Buried Pipe scenarios that were not identified as sensitive in the soil dose modeling uncertainty analysis. The process and criteria used to identify sensitive parameters and select conservative deterministic parameters were the same as that describe in Figure 6-10.

The RESRAD parameters assigned for the uncertainty analysis are the same as those used for the soil uncertainty analysis listed in Attachment 7 with a few exceptions:

- The Buried Pipe scenario parameters listed in section 6.12.4 were used as opposed to the corresponding soil parameters.
- Kd distributions were included to represent the range of site-specific sand Kd values determined by laboratory analysis
- To allow the dose from plant ingestion to vary with contaminated zone area, the two plant ingestion rate parameters were doubled to account for the fact that RESRAD automatically divides the entered ingestion rates by a factor of 2 when a value of -1 is used for the "Contaminated Fraction of Plant Food" parameter. The modified parameters are:
    - Fruits, non-leafy vegetables, grain consumption (kg/y) = 224
    - Leafy vegetable consumption (kg/y) = 42.8

The only parameters that required change as a result of the uncertainty analysis were the Saturated Zone Hydraulic Gradient for the Insitu Saturated scenario and the Depth of Roots for the Insitu Unsaturated scenario. All of the remaining parameters identified as sensitive in Reference 6-4, Table 1 were already identified as sensitive, with the same correlation, in the soil DCGL sensitivity analyses. The corresponding parameters, either 25th or 75th percentile, were included in the baseline surface soil DCGL deterministic parameter sets used for the Buried Pipe RESRAD runs.

The sensitivity of the assumed source term thickness required a separate analysis. The buried pipe scenarios assume that residual radioactivity is released from the pipes into adjacent soil. The thickness of soil into which the released activity was assumed to mix was 0.15 m which is considered the minimum reasonable mixing depth, particularly for the excavation scenario. As the "Thickness of Contaminated Zone" parameter is increased, assuming a unit concentration for all radionuclides, the dose increases. However, as the contaminated zone thickness increases the source term concentration decreases as an inverse linear function of the mixing depth. To determine the effect of these conflicting effects of increasing the Thickness of Contaminated Zone a separate sensitivity analysis was performed that accounts for both effects for source term thicknesses of 0.15 m and 1.0 m.

Reference 6-4, Attachment 3 provides the results of the sensitivity analysis. Note that for all scenarios and all radionuclides except Sr-90 increasing the Thickness of Contaminated Zone either has no effect on dose (indicated by a value of 1 in the column labeled "DSR Ratio*Source Term Decrease" in the Reference 6-4, Attachment 3 Tables) or causes the dose to decrease (indicated by a fraction in the column labeled "DSR Ratio*Source Term Decrease" in Reference 6-4, Attachment 3 Tables). The one exception, i.e., 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.

For the Insitu Saturated Scenario, increasing the source term thickness had no effect on dose for any radionuclides other than Sr-90. Note that the actual dose impact from the slightly increased Sr-90 dose for a 1 m thick source, as opposed to 0.15 m, is much lower than the values calculated individually for Sr-90 when the mixture percentages are considered. As shown in LTP Chapter 5, Table 5-2, the Auxiliary Basement mixture fraction (which is assumed to apply to buried pipe) for Cs-137 is 75.32% while the mixture fraction for Sr-90 is 0.05%. Therefore, the actual fractional dose attributable to the 8% and 20% increased values can be approximated as the ratio

of percentages times the percentage increase, i.e., 1.08*0.05/75.32 and 1.13 *0.05/75.32, or 0.07% and 0.08% of the final compliance dose which is insignificant.

For the excavation scenario, there are conflicting results for Sr-90 and the gamma emitters. While the Sr-90 dose shows an increase of 13% for the 1 m depth the Cs-137 dose decreases by 79%. When the mixture fractions are considered it is clear that the decrease in Cs-137 dose at 1 m source term depth would be orders of magnitude greater than the slight Sr-90 increase which would result in a non-conservative dose calculation.

In conclusion, the Thickness of Contaminated Zone parameter was set to 0.15 for all scenarios. However, to account for the indicated dose increase for Sr-90 at 1 m depth DSRs for Sr-90 were increased by factors of 1.08 and 1.13 for the Insitu Saturated and Excavation scenarios, respectively.

6.12.6. Buried Pipe RESRAD Results

Three RESRAD runs were performed for Buried Pipe; Excavation Scenario, Insitu Unsaturated Scenario, and Insitu Saturated Scenario (Reference 6-4). The RESRAD DSR results are summarized in Table 6-43.

Table 6-43 RESRAD DSR Results for Buried Pipe Dose Assessment to Support DCGL Development

Radionuclide	Excavation (mrem/yr per pCi/g)	Insitu Unsaturated (mrem/yr per pCi/g)	Insitu Saturated (mrem/yr per pCi/g)
Co-60	4.975E+00	7.298E-02	5.710E-04
Cs-134	2.836E+00	1.070E-01	2.881E-03
Cs-137	1.238E+00	8.491E-02	2.287E-03
Ni-63	1.445E-03	1.285E-03	2.745E-04
Sr-90 (1)	1.489E+00	1.384E+00	1.480E+00

(1) The Sr-90 DSRs for Excavation and Insitu Saturated were multiplied by factors of 1.13 and 1.08, respectively, to adjust for potentially higher dose from thicker source terms (Reference 6-3)

6.12.7. Buried Piping DCGL

The Buried Pipe DCGL is determined by first calculating the pCi/g concentration in the 0.15 m soil mixing layer that corresponds to a unit concentration, 1 dpm/100 cm2, on the pipe surface. The second input to the DCGL calculation is the sum of the DSR for Excavation and the maximum DSR for the Insitu Scenarios. As seen in Table 6-43, the maximum Insitu DSR is from the Unsaturated Scenario for all radionuclides except Sr-90. Therefore, the DSR summation used in the Buried Pipe DCGL calculation is comprised of the Excavation and Insitu Unsaturated Scenario DSRs for all radionuclides except Sr-90 which is based on the summation of the Excavation and Insitu Saturated Scenario DSRs. The summed DSRs are shown in Table 6-44.

Table 6-44 Maximum Summed RESRAD DSRs from Excavation and Insitu Scenarios

Radionuclide	Maximum Summed DSR Excavation + Insitu (mrem/yr per pCi/g)
Co-60	5.048E+00
Cs-134	2.943E+00
Cs-137	1.323E+00
Ni-63	2.730E-03
Sr-90	2.969E+00

The dpm/100 cm^2 per pCi/g conversion factor is used with the maximum summation DSR in Table 6-43 to calculate the Buried Pipe DCGL as shown in Equation 6-8.

Equation 6-8

$$BP\ DCGL = \frac{1}{Max\ Summed\ DSR} * \frac{dpm/100\ cm^2}{pCi/g} * 25\ mrem/yr$$

where:
BP DCGL = Buried Pipe DCGL (dpm/100 cm^2)
Max Summed DSR = Maximum Summed DSR values from Table 6-43 (pCi/g per mrem/yr)
(dpm/100 cm^2)/pCi/g = dpm/100 cm^2 in pipe per pCi/g in soil

The calculation of Buried Pipe DCGLs is provided in Reference 6-4, Attachment 2. Table 6-45 provides the resulting Buried Pipe DCGLs.

Table 6-45 Buried Piping DCGLs (Not Adjusted for IC Dose)

Radionuclide	Buried Pipe DCGL (dpm per 100 cm^2)
Co-60	2.94E+04
Cs-134	5.04E+04
Cs-137	1.12E+05
Ni-63	5.44E+07
Sr-90	5.00E+04

6.12.8. Adjustment for Dose from Insignificant Contributors

The buried pipe DCGLs must be adjusted to account for the radionuclides in the initial suite that were removed due to insignificant dose contribution. The Excavation scenario is closely related to the soil DCGL scenario. The Buried Pipe Insitu scenarios, particularly the Insitu Saturated,

have a greater potential groundwater dose contribution than the soil DCGL scenario and are more closely related to the BFM scenarios. The activity in buried pipes originate in one of the basements and the activity is assumed to mix with basements as well as mix with soil. Therefore, the insignificant dose contribution percentage assigned for the Buried Pipe DCGL adjustment was the maximum for either soil or the BFM. The maximum IC dose percentage was 10% for both soil and the BFM (Containment) and was the value used for Buried Pipe DCGL adjustment.

The Adjusted Buried Pipe DCGLs are provided in Table 6-46.

Table 6-46 Adjusted Buried Pipe DCGLs (Adjusted for IC Dose)

Radionuclide	Adjusted Buried Pipe 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

6.13. Embedded Piping DCGL

Embedded piping is defined as piping that runs vertically in a concrete wall or horizontally in a concrete floor. The residual radioactivity in embedded piping to remain has no release pathway other than into the Basement(s) where the piping terminates. Each embedded pipe run is treated as a separate survey unit within the basement that the embedded pipe is located and the DCGL calculated accordingly.

The embedded pipe to remain in the End State is identified and quantified in TSD 14-016. The embedded pipe survey units are listed in Table 6-47 along with the total internal survey area of the pipes in the survey unit. The IC-sump embedded pipe is very limited with a total surface area of 1.05 m2 each for Unit 1 and Unit 2. To provide a reasonable maximum value for the DCGL a nominal area of 100 m2 was assumed for the surface area of IC sump embedded pipe survey unit. The U2 Steam Tunnel surface area was slightly lower than the U1 area (46.88 m2 versus 46.39 m2). For simplicity, the higher, more conservative, area was applied to both Steam Tunnel Floor Drain DCGL calculations.

Table 6-47 Embedded Pipe Survey Unit Surface Areas

Embedded Pipe	EP SU Surface Area (m^2)
Auxiliary Floor Drains	299.41
Turbine Floor Drains	302.43
U1 Containment IC-Sump Drain	1.05 (100) (1)
U2 Containment IC-Sump Drain	1.05 (100) (1)
U1 Steam Tunnel Floor Drain	46.88(2)
U2 Steam Tunnel Floor Drain	46.88(2)
U1 Tendon Tunnel Floor Drain	51.41
U2 Tendon Tunnel Floor Drain	51.41

(1) The total surface area of unit 1 and unit 2 IC Sump Drains are 1.05 m^2 each. To provide a reasonable maximum value for the DCGL a nominal area of 100 m^2 was assumed for the DCGL calculation.
(2) Higher surface area applied to both U1 and U2 Steam Tunnel Floor Drains. U2 area is 46.39 m^2.

DCGLs were calculated for each of the embedded pipe survey units using Equation 6-9.

Equation 6-9

$$DCGL_{EP}(b,i) = \frac{25}{BFM\ DF_{b,i}} * \frac{1}{EP\ SU\ Area} * 1E09 * IC\ Dose\ Factor$$

Where:

$DCGL_{EP}$ (b,i)	= Embedded Pipe DCGL for radionuclide (i) in basement (b) (pCi/m^2)
BFM DF (b,i)	= Summation of Basement Fill Model Dose Factors for Groundwater and Drilling Spoils scenarios for radionuclide (i) in basement (b) (mrem/yr per mCi)
25	= 25 mrem/yr release criterion
EP SU Area	= Total internal surface area of all embedded pipe in the survey unit (m^2)
1E+09	= conversion factor of 1E+09 pCi/mCi
IC Dose Factor	= Insignificant contributor dose adjustment factor equal to 0.90 for Tendon Tunnel and IC-Sump embedded pipe and 0.95 for remaining embedded pipe (see section 6.5.2.3)

The embedded pipe DCGL calculations are provided in Reference 13. Note that the Tendon Tunnel Floor drains are included in both the Containment and Turbine Basement compliance demonstrations (see LTP Rev 1, Chapter 5, Table 5-20). The embedded pipe DCGL was therefore calculated for both basements. The DCGLs calculated using the Containment Basement Dose Factors in Equation 6-9 were lower than using the Turbine Basement Dose Factors and

were therefore assigned as the Tendon Tunnel floor drain DCGLs. The embedded pipe DCGLs are provided in Table 6-48.

Table 6-48 Embedded Pipe $DCGL_{EP}$ (Adjusted for Insignificant Contributor Dose)

Radionuclide	Auxiliary Floor Drain (pCi/m^2)	Turbine Floor Drain (pCi/m^2)	IC-Sump Drain U1 and U2 (pCi/m^2)	Steam Tunnel Floor Drain U1 and U2 (pCi/m^2)	Tendon Tunnel Floor Drains U1 and U2 (pCi/m^2)
Co-60	7.33E+09	6.31E+09	5.47E+09	4.07E+10	1.06E+10
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
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
H-3	NA	NA	8.28E+09	NA	1.61E+10
Eu-152	NA	NA	1.28E+10	NA	2.48E+10
Eu-154	NA	NA	1.11E+10	NA	2.16E+10

6.13.1. Dose Calculation for Grouted Auxiliary Basement Floor Drains

The FSS of the Auxiliary Basement floor drains is complete and documented in "Zion Station Restoration Project Final Status Survey Release Record Auxiliary Building 542 ft Embedded Floor Drain Pipe Survey Unit 051198A". After NRC review of the Release Record the drains were grouted to refusal.

The Auxiliary Drain FSS applied the DCGLs listed in Table 6-48 which assume that activity is released from the drains at the same rate as from the floor or wall surfaces in the Auxiliary Basement. A dose calculated using the Table 6-48 DCGL values is highly conservative because it does not take credit for the reduction in release due to the presence of grout. Therefore, a more realistic, yet still reasonably conservative dose calculation is performed for the Auxiliary Floor Drains that accounts for the presence of grout and will be used use in the final demonstration of compliance with the dose criterion (see section 6.17).

The reduction in radionuclide release from the Auxiliary Floor Drains due to the presence of grout was calculated in Attachment F of TSD 14-009, Revision 3, "Brookhaven National Laboratory: Evaluation of Maximum Radionuclide Groundwater Concentrations for Basement Fill Model" (Reference 6-18). Reduction in radionuclide release is linearly correlated to reduction in dose. The calculation assumed a one foot length of grout in the pipe and that all the residual radioactivity in the pipe is located directly under the one foot grout layer. This is very conservative given that the length of the floor drain sections range from 18 to 188 feet and the entire drain system was grouted to refusal. The vast majority of the activity in the pipes would have a much longer length of grout to diffuse through than 1 foot. The minimum depth to an identified obstruction in a grouted pipe (which may or may not cause refusal of grout flow) was

six feet. The fractional release of total residual radioactivity remaining in the Auxiliary Floor Drain pipe for each ROC, assuming a one foot length of grout, is provided in Table 6-49.

Table 6-49 Fractional Release of Residual Radioactivity from Auxiliary Floor Drains Due to presence of 1 Foot of Grout

Radionuclide	Fractional Release (1)
Co-60	<1E-30
Ni-63	8.28E-07
Sr-90	1.19E-16
Cs-134	1.38E-25
Cs-137	4.98E-07

(1) From TSD 14-009, Revision 3, Attachment F (Reference 6-18)

The maximum fractional release for all ROC occurs for Ni-63 with a value of 8.28E-07. As a measure of conservatism in the assumption of a one foot grout length, the fractional release was also calculated for grout lengths of two and four feet resulting in much lower Cs-137 fractional release values of 1.24E-13 and 2.31E-29, respectively. The dose from Ni-63 calculated in the Release Record for the Auxiliary Floor Drains, assuming no grout, would therefore be reduced by at least a factor of 8.28E-07 when grout is accounted for. The dose from the other ROC would be reduced further as indicated by the lower fractional releases in Table 6-49.

Based on the FSS results and the DCGLs in Table 6-48, the total dose including all ROC was calculated in the FSS Release Record for the Auxiliary Building Floor Drains to be 4.241 mrem/yr. Using the highest fractional release from Table 6-49 of 8.28E-07, a conservative estimate of dose, including the effect of diffusion through grout, is 4.241 mrem/yr x 8.28E-07 = 3.51E-06 mrem/yr.

The dose calculation of 3.51E-06 mrem/yr accounts for the fractional release of the ROC and includes a general assumption that the fractional release of the HTD radionuclides included in the DCGL adjustment for insignificant contributor dose (see Equation 6-9) would be also be a very low. However, a review of Reference 6-18 shows that the fractional release for H-3 is 0.34 (the value was much lower for Eu-152 and Eu-154 at <1E-30). While the higher fractional release is not unexpected for H-3, which has a diffusion coefficient that is approximately two orders of magnitude higher than Cs-137, it raises a general question as to the actual fractional release for the insignificant contributor radionuclides. To address this question in a simple, highly conservative and bounding manner, the dose from insignificant contributors is calculated without application of a grout fractional release factor, i.e., assuming no grout is present. This calculation of dose from the insignificant contributor radionuclides is in addition to the 5% adjustment for insignificant contributor dose already included in the DCGL calculation (see Equation 6-9) and therefore included in the 3.51E-06 mrem/yr dose calculation accounting for the presence of grout.

The insignificant contributor dose percentage used in the calculation was based on HTD analysis of sediment samples collected from the Auxiliary Floor Drains prior to FSS. As shown in

Attachment 5 of the FSS Release Record for the Auxiliary Building Floor Drains, the dose percentage assigned to the insignificant contributor radionuclides is 3.952%. Therefore, the bounding dose from the insignificant contributors alone, with no credit for grout, would be 3.952% of the 4.241 mrem/yr that was calculated assuming no grout present. This results in a dose of 0.0392*4.241mrem/yr = 0.17 mrem/yr.

In conclusion, the total dose to be assigned to the Auxiliary Building Floor Drains for the compliance calculation in section 6.17 is 3.51E-06 mrem/yr + 0.17 mrem/yr = 0.17 mrem/yr with rounding. ZSRP will add insignificant radionuclide dose in this same manner for any other end-state Class 1 embedded pipe that is grouted.

6.14. Penetration DCGL

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

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

The penetration DCGL (DCGLPN) is calculated in the same manner as embedded pipe using Equation 6-9 but replacing the embedded pipe survey unit surface area with the penetration survey unit surface area. The penetration survey units are defined in TSD 14-016 including the total area of each penetration survey unit as listed in Table 6-50.

Table 6-50 Penetration Survey Unit Surface Areas

Embedded Pipe	Penetration Survey Unit Surface Area (m^2)
Auxiliary Basement	948.75
Containment Basement	242.36
Turbine Basement	1081.14
SFP/Transfer Canal	337.45
Crib House/Forebay	1.14
WWTF	0.89

An additional adjustment is required for the calculation of the DCGLPN for the Auxiliary Basement penetration survey unit. The release of residual radioactivity from the Auxiliary basement concrete assumes diffusion release. In most cases the remaining penetrations will be either the remaining pipe or steel pipe sleeve after a pipe is removed. Because the residual

radioactivity is not contained at depth in concrete, the assumption of diffusion release through concrete is not applicable and instant release is conservatively assumed for the penetrations. As seen in Equation 6-9, the penetration DCGL calculation uses the BFM Dose Factors, which in the case of the Auxiliary basement are based on an assumption of diffusion release.

An adjustment is therefore required to account for the higher maximum release rate under an instant release assumption as compared to diffusion release. The correction factor was calculated in TSD14-009, Revision 3, Attachment G, where the maximum concentration in the Auxiliary Basement under an instant release assumption was compared to the maximum concentration using a diffusion release assumption. The adjustment factor was calculated as the ratio of maximum instant release to maximum diffusion release. The results from TSD 14-009, Revision 3, Attachment G are reproduced in Table 6-51.

Table 6-51 Ratio of Instant Release Maximum to Diffusion Release Maximum for Auxiliary Basement

Radionuclide	Auxiliary Floor Drain
Co-60	26.23
Cs-134	4.91
Cs-137	1.37
Ni-63	14.96
Sr-90	19.1
H-3	1.01
Ni-63	1.29
Sr-90	3.15

The $DCGL_{PN}$ for the Auxiliary Basement penetration survey unit is then calculated using Equation 6-10 which is the same as Equation 6-9 with an additional term, i.e., "$Ratio_{ID}$", to account for instant release from Auxiliary Basement penetration survey unit to the Auxiliary basement.

Equation 6-10

$$DCGL_{EP}(A,i) = \frac{25}{(BFM\ DF_{gw}(i) + BFM\ DF_{ds}(i)) * Ratio_{ID}} * \frac{1}{PEN\ SU\ Area} * 1E09 * IC\ Dose\ Factor$$

Where:

$DCGL_{EP}$ (A,i)	= Embedded Pipe DCGL for radionuclide (i) in Auxiliary basement (A) (pCi/m^2)
BFM DF_{gw} (i)	= Basement Fill Model Groundwater Dose Factor for radionuclide (i) in Auxiliary basement (b) (mrem/yr per mCi)
BFM DF_{ds} (i)	= Basement Fill Model Drilling Spoils Dose Factor for radionuclide (i) in Auxiliary basement (b) (mrem/yr per mCi)
$Ratio_{ID}$	= ratio of instant release maximum concentration to diffusion release concentration

25	= 25 mrem/yr release criterion
PEN SU Area	= Total internal surface area of Auxiliary Basement penetration survey unit (m^2)
1E+09	= conversion factor of 1E+09 pCi/mCi
IC Dose Factor	= Insignificant contributor dose adjustment factor equal to 0.90 for Containment and 0.95 for all other basements (see LTP Chapter 6 section 6.8.2)

The penetration $DCGL_{PN}$ are calculated in Reference 13. The $DCGL_{PN}$ values are provided in Table 6-52. Note that the $DCGL_{PN}$ for the Crib House/Forebay and WWTF are listed as not applicable due the very small surface areas of the few penetrations present (1.14 m2 and 0.89 m^2). The Crib House/Forebay and WWTF penetrations DCGLs are set equal to the wall/floor surface DCGL and included with the Crib House/Forebay and WWTF surface survey units.

Table 6-52 Adjusted Penetration $DCGL_{PN}$ (adjusted for insignificant contributor dose)

Nuclide	Auxiliary (pCi/m^2)	Containment (pCi/m^2)	SFP/ Transfer Canal(1) (pCi/m^2)	Turbine (pCi/m^2)	Crib House/ Forebay1 (pCi/m^2)	WWTF (pCi/m^2)
Co-60	8.82E+07	2.26E+09	4.45E+08	1.76E+09	NA	NA
Cs-134	3.28E+08	4.32E+08	7.48E+08	4.00E+08	NA	NA
Cs-137	6.17E+08	5.66E+08	1.46E+09	5.29E+08	NA	NA
Eu-152	3.29E+08	5.26E+09	9.44E+08	4.06E+09	NA	NA
Eu-154	2.33E+08	4.58E+09	8.53E+08	3.58E+09	NA	NA
H-3	3.99E+09	3.42E+09	4.84E+16	3.23E+09	NA	NA
Ni-63	6.79E+10	5.78E+10	1.86E+14	5.48E+10	NA	NA
Sr-90	2.41E+07	2.06E+07	9.26E+10	1.94E+07	NA	NA

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

6.15. Existing Groundwater Dose

As previously stated, no groundwater contamination has been identified by groundwater monitoring performed as of the date of this LTP (Revision 1) and is not expected to be present at the time of license termination. However, if groundwater contamination is identified during decommissioning, the dose will be calculated using the BFM Groundwater Exposure Factors in Table 6-18. Table 6-18 was developed as a part of the BFM, but the BFM Groundwater Exposure Factors presented in Table 6-18 are fully applicable to any groundwater contamination, regardless of the location.

6.16. Clean Concrete Fill

ZSRP will demonstrate that all concrete designated as backfill material in basements is clean through the Unconditional Release Survey (URS) program at Zion presented in ZionSolutions procedure ZS-LT-400-001-001, "Unconditional Release of Materials, Equipment and Secondary

Structures". Materials unconditionally released from Zion, regardless of their point of origin on the site, have been verified to contain no detectable plant-derived radioactivity and are free to be used and relocated anywhere offsite without tracking, controls, or dose considerations.

Although the concrete debris to remain onsite and used as clean fill can be viewed as having a no dose impact, a dose value will be assigned for the purpose of demonstrating compliance with 10 CFR 20.1402 in the same manner as other materials to remain at license termination that are surveyed and found to not contain detectable activity. The "detection limit" used for the dose calculation is conservatively assumed to be the maximum scan MDC of 5,000 dpm/100 cm2 allowed in the URS program. Actual detection limits in the unconditional release program are lower than this value.

The vast majority of clean concrete fill to be used will come from five buildings; Containment, Turbine, Crib House/Forebay, Service Building and Interim Waste Storage Facility. Because the concrete will be from both Containment and other structures the dose calculation was performed using both the Containment and Auxiliary ROC mixtures. The dose was essentially the same for both mixtures but the dose with the Containment mixture was slightly higher (with trivial exception of WWTF). Consistent with the bounding approach used for the clean concrete assessment, the Containment mixture was applied to all concrete. In addition, when applying the ROC mixture, the 5,000 dpm/100 cm2 maximum detection limit was assumed to be 100% Cs-137. The remaining radionuclide concentrations were added to the Cs-137 concentration at their respective ratios to Cs-137.

The dose values are calculated separately for each basement assuming that the entire basement void is filled with concrete only. This conservatively includes the top three feet of fill which will be soil for all basements and not concrete. Details regarding the calculation are provided in Reference 13, section 8. The total dose results for each basement, assuming a scan MDC value of 5,000 dpm/100 cm2 and including all ROC, are provided in Table 6-53.

The dose values in Table 6-53 will be adjusted based on the actual maximum scan MDC after all URS surveys are completed. The adjusted dose will be calculated by multiplying the ratio of the actual maximum scan MDC to 5,000 dpm/100 cm2 by the values in Table 53. The adjusted dose will be added to any basement where concrete fill is used regardless of the volume of concrete fill used. This is a conservative and bounding approach (see section 6-17).

Table 6-53 Dose Assigned to Clean Concrete Fill

	Auxiliary	Containment	SFP/ Transfer Canal	Turbine	Crib House/ Forebay	WWTF
Dose (mrem/yr)	9.94E-01	1.77E+00	1.52E-01	1.58E+00	1.57E+00	6.40E+00

6.17. Demonstrating Compliance with Dose Criterion

There will be four distinct source terms in the ZNPS End State; backfilled basements, soil, buried piping, and groundwater. Demonstrating compliance with the Dose Criterion requires the summation of dose from the four source terms as shown in Equation 6-11. The embedded pipe dose, penetration dose and clean concrete fill dose (see sections 6.13, 6.14, and 6.16

respectively) will be added to the dose from wall and floor surfaces in the applicable basement to calculate the total basement dose. See LTP Chapter 5, Table 5-20 for a list of embedded pipe and penetration survey units and which basement they associated with. The maximum total basement SOF will be used for the "Max $SOF_{BASEMENT}$" term in Equation 6-11.

The dose summation described in Equation 6-11 is conservative because the various source terms are likely not contiguous or simultaneous. For example, the maximum open land soil survey unit dose could be from an area that is not within the footprint of the Basement assigned with the maximum dose. Another example is the buried pipe that delivers the greatest dose may not be under or contiguous with the open land survey unit assigned with the maximum dose.

The final compliance dose will be calculated using Equation 6-11 after FSS has been completed in all survey units. The Release Record for each FSS unit will be reviewed to determine the maximum mean dose from each for each of the four source terms (e.g. basement, soil, buried pipe and existing GW if applicable). The compliance dose must be less than or equal to 25 mrem/yr. The calculation of the compliance dose will be documented in the final FSS Report for the site.

A detailed description of the terms in Equation 6-11 and the process for calculating the Compliance Dose is provided in Zion*Solutions* TSD 17-004,"Operational Derived Concentration Guideline Levels for Final Status Surveys" (Reference 6-31). The Operational DCGLs selected in TSD 17-004 are less than the standard DCGLs calculated in this chapter. The application of the Operational DCGLs provides additional assurance that the compliance dose will be less than or equal to 25 mrem/yr after FSS is completed for all four source terms. See LTP Chapter 5 for additional information on the application of Operational DCGLs during FSS.

Equation 6-11

$$\mathbf{Compliance\ Dose} = (\mathbf{Max\ SOF_{BASEMENT}} + \mathbf{Max\ SOF_{SOIL}} + \mathbf{Max\ SOF_{BURIED\ PIPE}} + \mathbf{Max\ SOF_{GROUNDWWATER}}) * \mathbf{25\ mrem/yr}$$

where:

Compliance Dose	=	must be less than or equal to 25 mrem/yr,
Max $SOF_{BASEMENT}$	=	Maximum Sum of Fractions (SOF) (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,
Max $SOF_{GROUNDWATER}$	=	Maximum SOF for from existing groundwater

6.18. References

6-1 Zion*Solutions* Technical Support Document 14-003, Revision 3, Conestoga Rovers & Associates (CRA) Report, "Zion Hydrogeologic Investigation Report"

6-2 "Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement" – December 2007

6-3 Zion*Solutions* TSD 14-016, Revision 0, "Description of Embedded Piping, Penetrations and Buried Piping to Remain in Zion End State"

6-4 Zion*Solutions* Technical Support Document 14-015, Revision 3, "Buried Pipe Dose Modeling & DCGLs"

6-5 "Zion Station Historical Site Assessment (HSA)" – September 2006

6-6 U.S. Nuclear Regulatory Commission NUREG-1757, Volume 2, Revision 1, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report" – September 2003

6-7 Zion*Solutions* Technical Support Document 14-019, "Radionuclides of Concern for Soil and Basement Fill Model Source Terms"

6-8 Zion*Solutions* Technical Support Document 10-002, Revision 1, "Technical Basis for Radiological Limits for Structure/Building Open Air Demolition"

6-9 Zion*Solutions* Technical Support Document 11-001, Revision 1, "Potential Radionuclides of Concern during the Decommissioning of Zion Station"

6-10 Pacific Northwest Laboratory, NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials, Pacific Northwest Laboratory" – 1984

6-11 Pacific Northwest Laboratory, NUREG/CR-4289, "Residual Radionuclide Concentration Within and Around Commercial Nuclear Power Plants; Origin, Distribution, Inventory, and Decommissioning Assessment" – 1985

6-12 Westinghouse Idaho Nuclear Company, Inc., WINCO-1191, "Radionuclides in United States Commercial Nuclear Power Reactors" – 1994

6-13 International Commission on Radiological Protection, ICRP Publication 38, "Radiological Transformations - Energy and Intensity of Emissions" – 1983

6-14 Zion*Solutions* Technical Support Document 14-010, Revision 6, 'RESRAD Dose Modeling for Basement Fill Model and Soil DCGL and Calculation of Basement Fill Model Dose Factors and DCGLs"

6-15 The City of Zion, "Official Zoning Map City of Zion" – March 2011

6-16 United States Department of Agriculture, "Custom Soil Resources Report Lake County Illinois" – August 2013

6-17 Pacific Northwest Laboratory, NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning" – October 1992

6-18 Zion*Solutions* TSD 14-009, Revision 3, "Brookhaven National Laboratory Report (BNL),

"Evaluation of Maximum Radionuclide Groundwater Concentrations for Basement Fill Model, Zion Station Restoration Project"

6-19 Zion*Solutions* Technical Support Document 14-032, Revision 0, "Conestoga Rovers & Associates Report, Simulation of the Post-Demoltion Saturation of Foundation Fill Using a Foundation Water Flow Model"

6-20 Zion*Solutions* Technical Support Document 14-006, Revision 5, Conestoga Rovers & Associates (CRA) Report, "Evaluation of Hydrological Parameters in Support of Dose Modeling for the Zion Restoration Project"

6-21 Zion*Solutions* Technical Support Document 14-004, Revision 1, Brookhaven National Laboratory (BNL), "Recommended Values for the Distribution Coefficient (K_d) to be used in Dose Assessments for Decommissioning the Zion Nuclear Power Plant"

6-22 Zion*Solutions* Technical Support Document 14-017, Revision 0, Brookhaven National Laboratory (BNL), "Sorption (Kd) Measurements on Cinder Block and Grout in Support of Dose Assessments for Zion Nuclear Station Decommissioning"

6-23 Zion*Solutions* Technical Support Document 14-020, Revision 0, Brookhaven National Laboratory (BNL), "Sorption (Kd) measurements in Support of Dose Assessments for Zion Nuclear Station Decommissioning"

6-24 Argonne National Laboratory, NUREG/CR-6697 "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes" – December 2000

6-25 Sandia National Laboratory, NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination From Decommissioning Parameter Analysis" – October 1999

6-26 Environmental Protection Agency, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion" – September 1988

6-27 Environmental Protection Agency, Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water and Soil" – September 1993

6-28 ZionSolutions Technical Support Document 14-021 Revision 1, "Basement Fill Model (BFM) Drilling Spoils and Alternate Exposure Scenarios"

6-29 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" – August 2000

6-30 Zion*Solutions* Technical Support Document 14-011, Revision 0, "Soil Area Factors"

6-31 Zion*Solutions* TSD 17-004, Revision 3, "Operational Derived Concentration Guideline Levels for Final Status Surveys"

Figure 6-1 Zion Nuclear Power Station Geographical Location

Figure 6-2 Zion Nuclear Power Station Owner Controlled Area

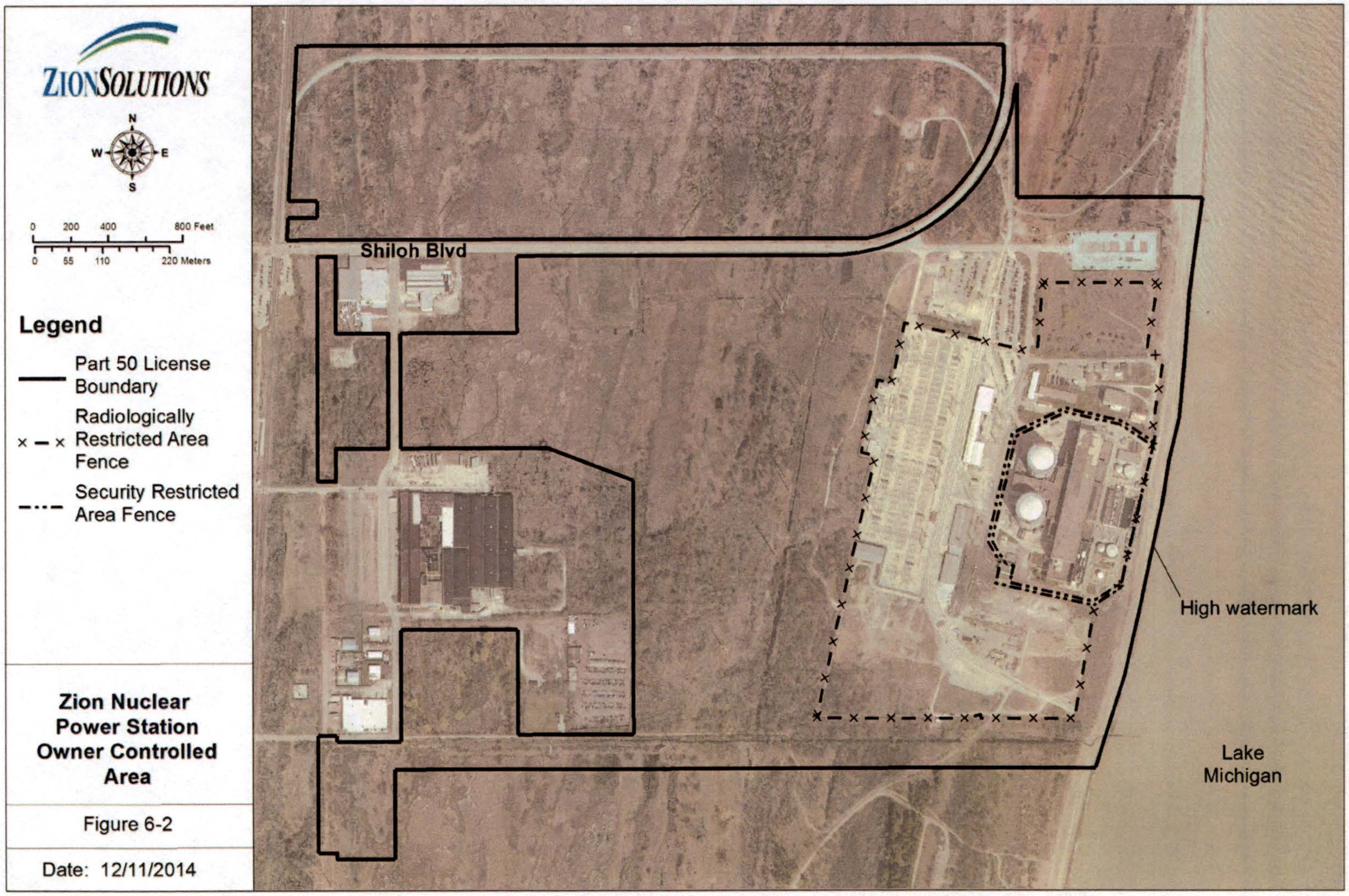


Figure 6-3 Zion Nuclear Power Station Security Restricted Area

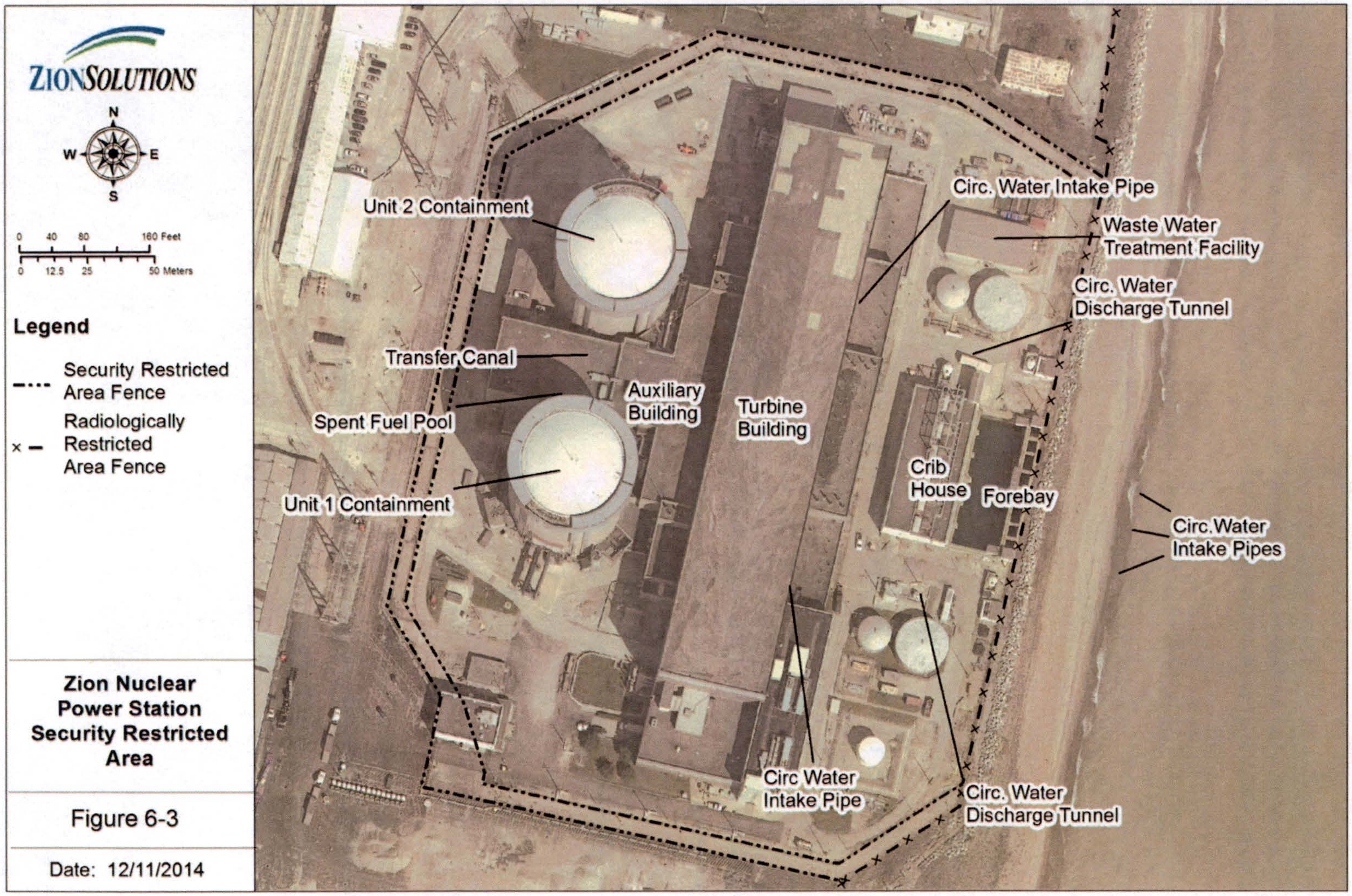


Figure 6-4 Backfilled Basement and Structures to Remain Below 588' Elevation

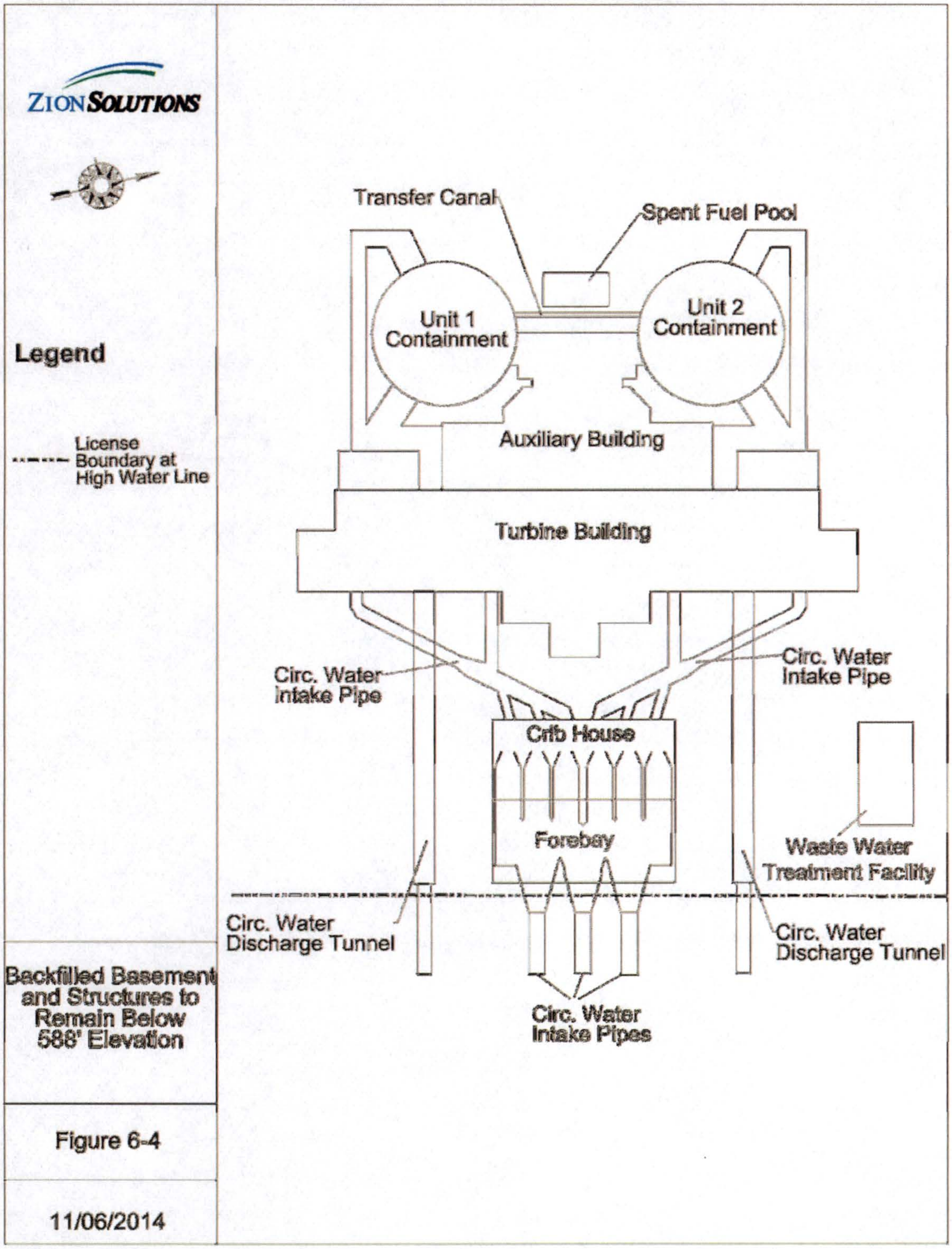


Figure 6-5 Cross Section A-A of Basements/Structures Below

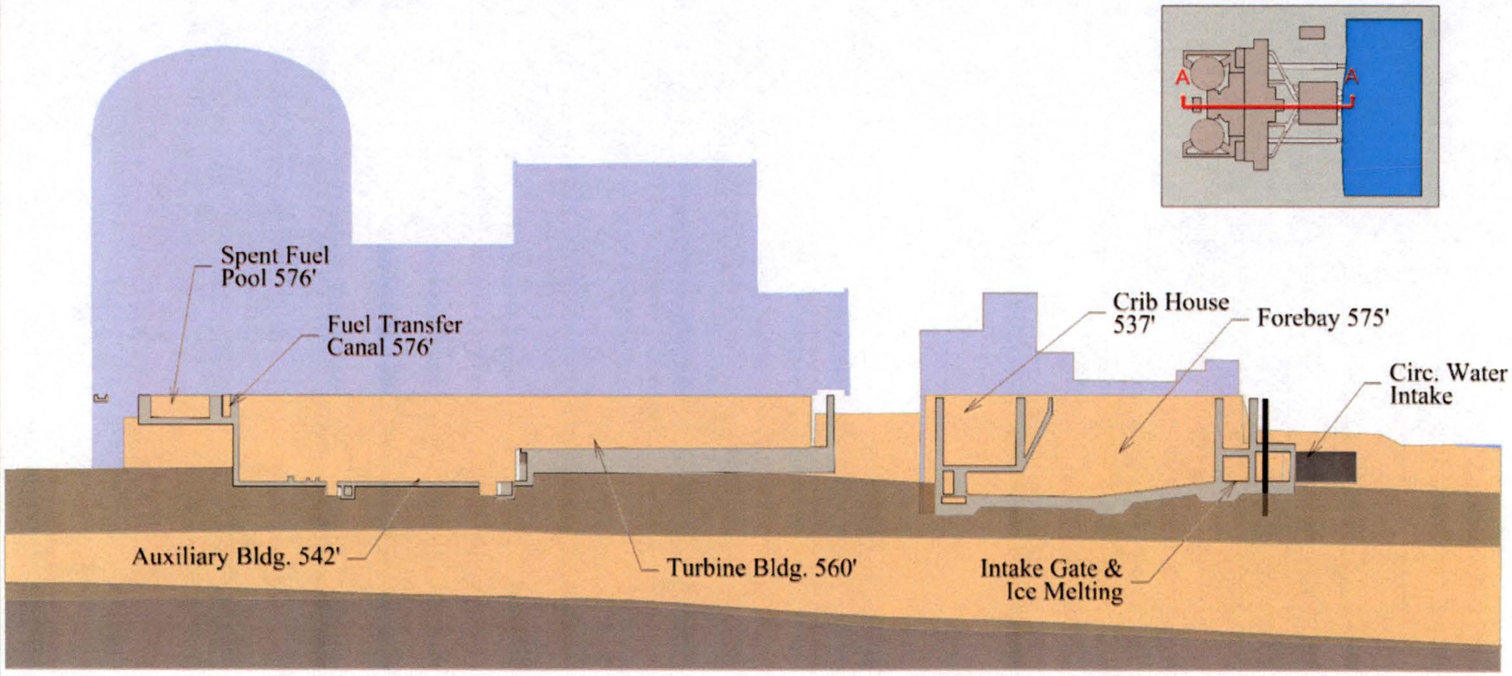


Figure 6-5 Cross Section A-A of Basements/Structures Below 588' Elevation to Remain at License Termination

Figure 6-6 Cross Section B-B of Basements/Structures Below 588' Elevation to Remain at License Termination

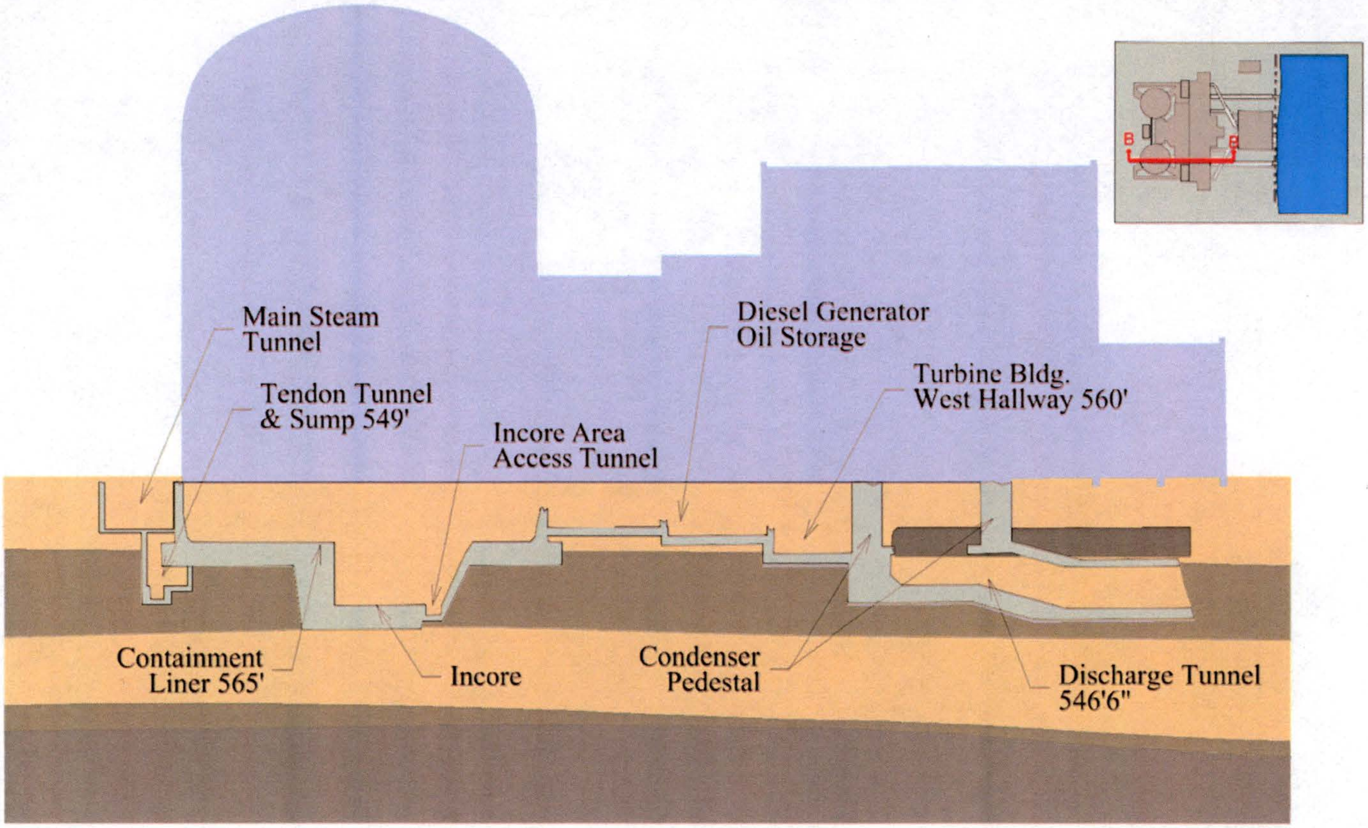


Figure 6-6 Cross Section B-B of Basements/Structures Below 588' Elevation to Remain at License Termination

Figure 6-7 Cross Section C-C of Basements/Structures Below 588' Elevation to Remain at License Termination

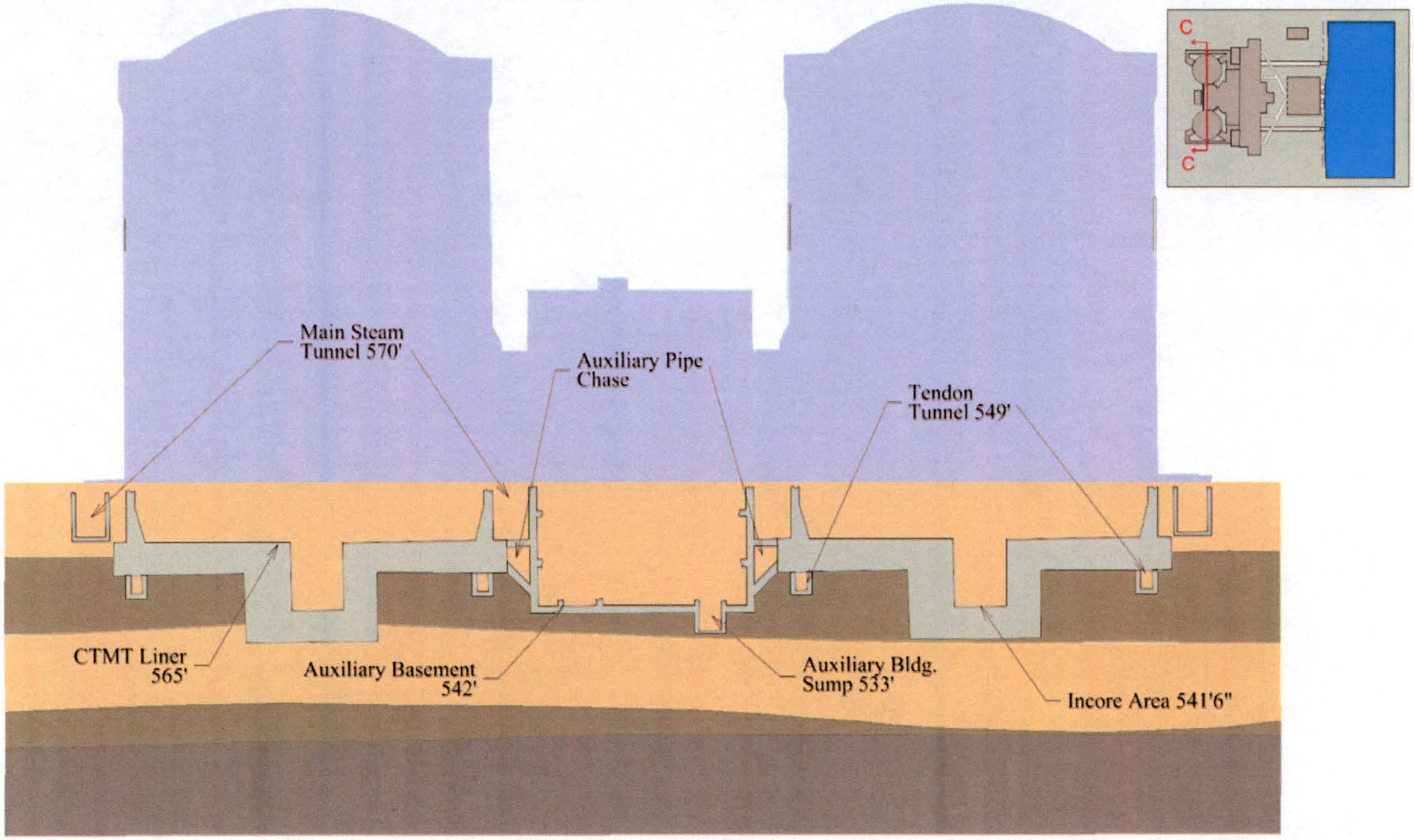


Figure 6-7 Cross Section C-C of Basements/Structures Below 588' Elevation to Remain at License Termination

Figure 6-8 Cross Section D-D of Basements/Structures Below 588'

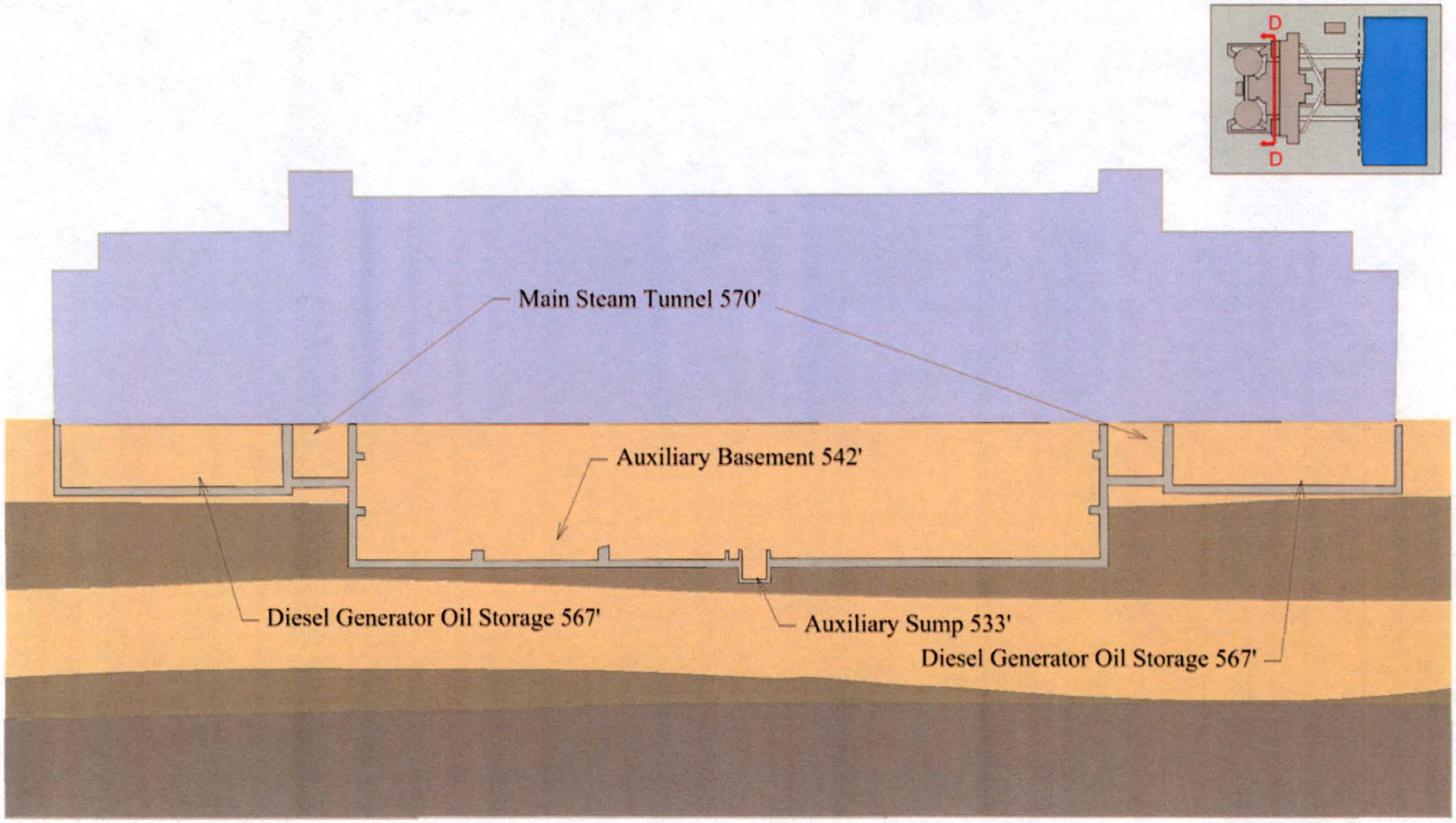


Figure 6-8 Cross Section D-D of Basements/Structures Below 588' Elevation to Remain at License Termination

Figure 6-9 Visualization of BFM Conceptual Model

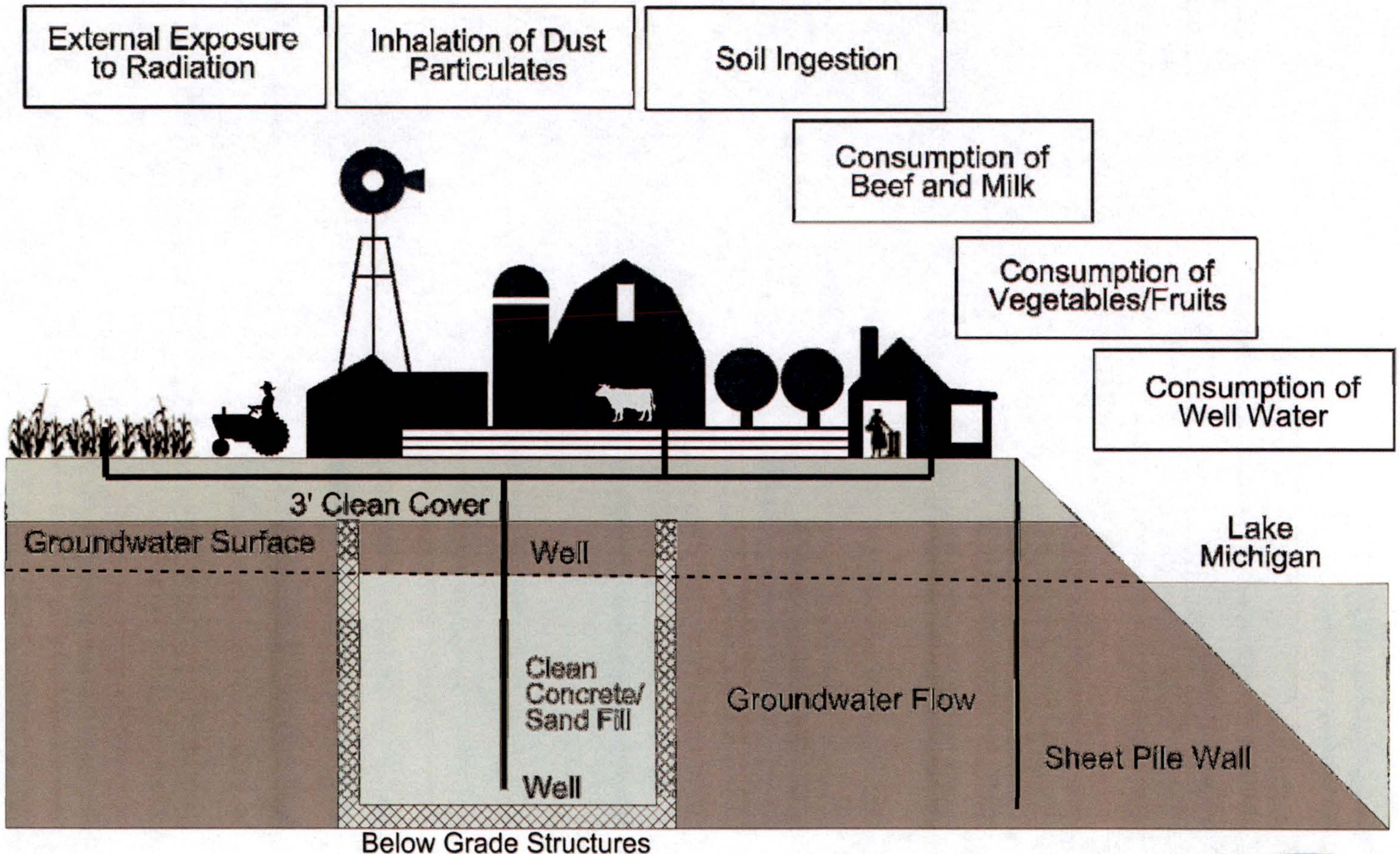
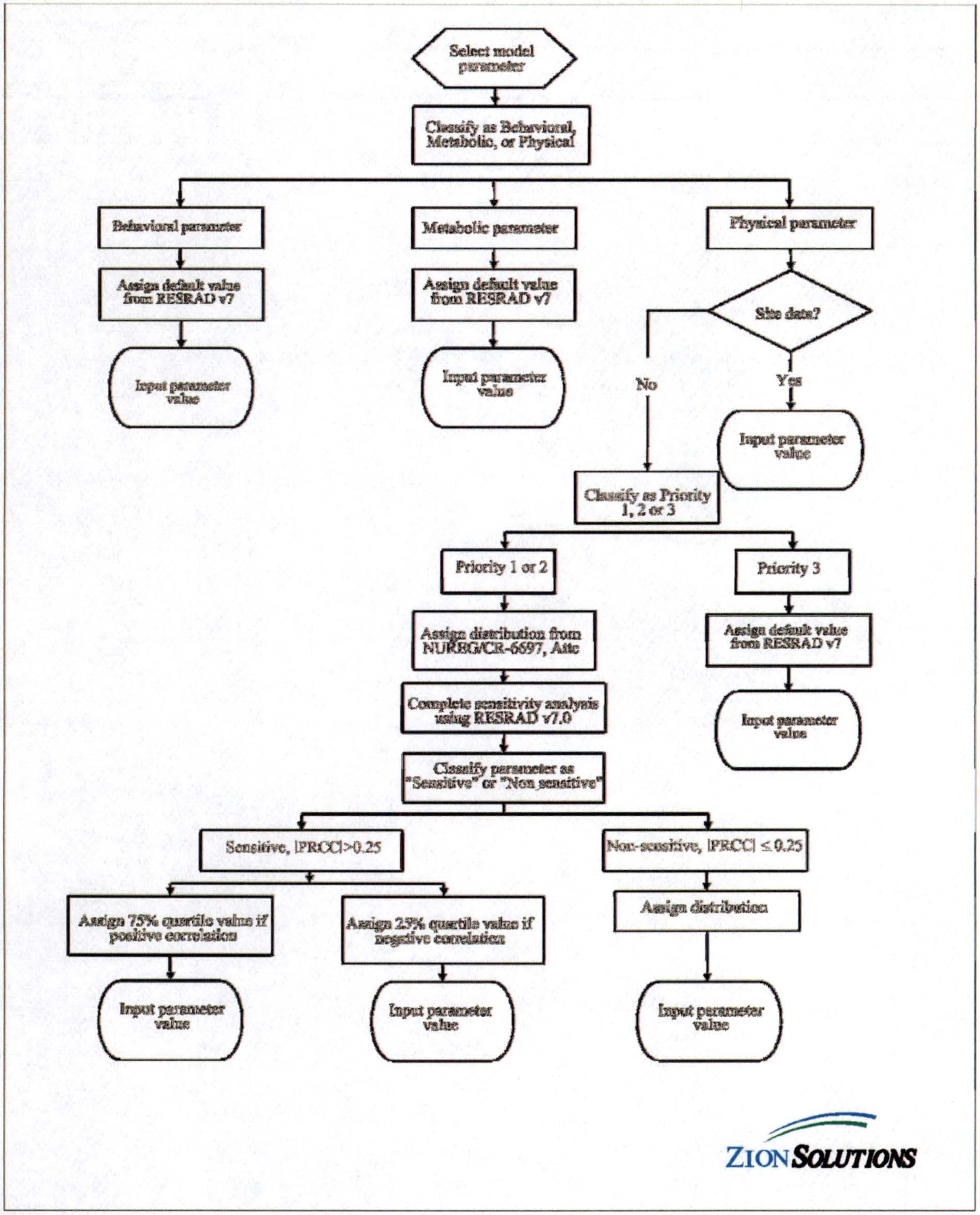


Figure 6-9 Visualization of BFM Conceptual Model

Figure 6-10 RESRAD Parameter Selection Flow Chart



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ATTACHMENT 1

RESRAD Input Parameters for ZSRP BFM Uncertainty Analysis

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Soil Concentrations										
Basic radiation dose limit (mrem/y)		3	D	25	10 CFR 20.1402	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR	
Distribution coefficients (contaminated, unsaturated, and saturated zones) (cm^3/g)										
Co-60	P	1	D	223	TSD 14-004	5.46	2.53	0.001	0.999	235
Cs-134	P	1	D	45	TSD 14-004	6.1	2.33	0.001	0.999	446
Cs-137	P	1	D	45	TSD 14-004	6.1	2.33	0.001	0.999	446
Eu-152	P	1	D	95	TSD 14-004	6.72	3.22	0.001	0.999	825
Eu-154	P	1	D	95	TSD 14-004	6.72	3.22	0.001	0.999	825
Gd-152 (daughter for Eu-152)	P	1	D	825	Median Value NUREG/CR-6697, Att. C	6.72	3.22	0.001	0.999	825
H-3	P	1	D	0	TSD 14-004	-2.81	0.5	0.001	0.999	0.06
Nd-144 (daughter for Eu-152)	P	1	D	158	RESRADv.7.0 Default Neodymium (Nd) not listed in NUREG/CR-6697	NA	NA	NA	NA	NA
Ni-63	P	1	D	62	TSD 14-004	6.05	1.46	0.001	0.999	424
Sm-148 (daughter Eu-152)	P	1	D	825	Median Value NUREG/CR-6697, Att. C	6.72	3.22	0.001	0.999	825
Sr-90	P	1	D	2.3	TSD 14-004	3.45	2.12	0.001	0.999	32
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	No existing groundwater contamination	NR	NR	NR	NR	
Calculation Times										
Time since placement of material (y)	P	3	D	1	For user convenience: Allows use of t=0 in dose and concentration output reports to calculate unitized Exposure Factors	NR	NR	NR	NR	
Time for calculations (y)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD Default	NR	NR	NR	NR	
Contaminated Zone										
Area of contaminated zone (m^2)	P	2	D	64,500	Area of the 'Security Protected Area' on Zion Site	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Thickness of contaminated zone (m)	P	2	D	11.2	Contaminated Zone is the Basement fill depth where mixing occurs. Depth of fill mixing zone depends on Basement floor elevation. 11.2 m is used as nominal value based on difference between elevations of the water table (579') and Auxiliary Basement floor (542') which equals 11.2m. Note: this parameter has no effect on the calculated values for unitized Exposure Factors.	NR	NR	NR	NR	
Length parallel to aquifer flow (m)	P	2	D	287	Diameter of 64,500 m2 contaminated area. Note: not applicable to Basement Fill Model because Mass Balance groundwater model used.	NR	NR	NR	NR	
Does the initial contamination penetrate the water table?	NA	NA	NA	Yes	100% of the contamination assumed to be in the basement fill water mixing zone	NA	NA	NA	NA	
Contaminated fraction below water table	P[e]	3[e]	D	1	100% of the contamination assumed to be in the basement fill water mixing zone	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	3.6m	Difference between ground level elevation at 591' (179.6m) and equilibrium water level in basements at 579' (176m)	NR	NR	NR	NR	NA
Density of cover material	P	2	D	1.8	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.5.	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Cover erosion rate	P,B	2	D	Continuous Logarithmic	NUREG/CR-6697 Att. C	5E-08	0,0007	0,005	.2	0.0015
Density of contaminated zone (g/cm^3)	P	1	S	Truncated Normal	NUREG/CR-6697 Att. C Fill to be comprised of undetermined combination of clean concrete and native sand. NUREG/CR-6697 distribution used as placeholder – parameter has no effect on calculation of unitized Exposure Factors	1.52	0.23	0.001	0.999	1.52
Contaminated zone erosion rate (m/y)	P,B	2	S	Continuous Logarithmic	NUREG/CR-6697 Att. C	5E-08	0.0007	0,005	0.2	0.0015
Contaminated zone total porosity	P	2	S	Truncated Normal	NUREG/CR-6697 Att. C Fill to be comprised of undetermined combination of clean concrete and native sand. NUREG/CR-6697 distribution used as placeholder – parameter has no effect on calculation of unitized Exposure Factors	0.425	0.0867	0.001	0.999	0.42
Contaminated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Contaminated zone hydraulic conductivity (m/y)	P	2	S	Loguniform	Site-specific distribution from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Contaminated zone b parameter	P	2	S	Bounded Lognormal - N	NUREG/CR-6697, Att. C Fill to be comprised of undetermined combination of clean concrete and native sand. NUREG/CR-6697 distribution used as placeholder – parameter has no effect on calculation of unitized Exposure Factors	1.06	0.66	0.5	30	2.89
Humidity in air (g/m^3)	P	3	D	7.2	Median NUREG/CR-6697 Att. C	1.98	0.334	0.001	0.999	7.2
Evapotranspiration coefficient	P	2	S	Uniform	NUREG/CR-6697 Att. C	0.5	0.75	NR	NR	0.625
Average annual wind speed (m/s)	P	2	S	Bounded Lognormal n	NUREG/CR-6697 Att. C	1.445	0.2419	1.4	13	4.2

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Precipitation (m/y)	P	2	D	0.83	Site-specific value from Reference 6-21, Table 5.12	NR	NR	NR	NR	
Irrigation (m/y)	B	3	D	0.19	NUREG-5512, Vol. 3, Table 6-18 (Illinois Average) Converted 0.52 L/m2/d to m/y.	NR	NR	NR	NR	
Irrigation mode	B	3	D	Overhead	Overhead irrigation is common practice in U. S.	NR	NR	NR	NR	
Runoff coefficient	P	2	S	Uniform	NUREG/CR-6697 Att. C	0.1	0.8	NR	NR	0.45
Watershed area for nearby stream or pond (m^2)	P	3	D	1.0E+06	RESRAD Default	NR	NR	NR	NR	
Accuracy for water/soil computations	-	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Saturated Zone Hydrological Data										
Density of saturated zone (g/cm^3)	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	1.51	0.16	0.001	0.999	1.51
Saturated zone total porosity	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.43	0.06	0.001	0.999	0.43
Saturated zone effective porosity	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.383	0.0610	0.001	0.999	0.383
Saturated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Saturated zone hydraulic conductivity (m/y)	P	1	S	Loguniform	Site-specific distribution from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Saturated zone hydraulic gradient	P	2	S	Bounded Lognormal - N	NUREG/CR-6697 Att. C	-5.11	1.77	0.00007	0.5	0.006
Saturated zone b parameter	P	2	D	NA saturated zone b not active because water table drop rate =0	NUREG/CR-6697	NR	NR	NR	NR	NR
Water table drop rate (m/y)	P	3	D	0	Basement fill water assumed to supply well with no water table drop.	NR	NR	NR	NR	
Well pump intake depth (m below water table)	P	2	S	Triangular	NUREG/CR-6697 Att. C	6	10	30		10
Model: Non-dispersion (ND) or Mass-Balance (MB)	P	3	D	MB	MB model most applicable to assumption that well located in center of basement fill.	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Well pumping rate (m^3/y)	B,P	2	D	2250	Calculated according to method described in NUREG/CR-6697, Att. C section 3.10 using Illinois average irrigation rate and NUREG/CR-5512 Vol. 3 livestock water consumption rate. Calculation provided at end of this table as Footnote 1.	NR	NR	NR	NR	NR
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	NA	NA	0	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone thickness (m)	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone soil density (g/cm^3)	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone total porosity	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone effective porosity	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone field capacity	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone hydraulic conductivity (m/y)	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone soil-specific b parameter	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Occupancy										
Inhalation rate (m^3/y)	M,B	3	D	8400	NUREG/CR-5512, Vol. 3 Table 6.29 (= $23m^3$/d x 365d)	NR	NR	NR	NR	
Mass loading for inhalation (g/m^3)	P,B	2	S	Continuous Linear	NUREG/CR-6697, Att. C	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	2.35E-05
Exposure duration	B	3	D	30	RESRAD User's Manual (Parameter not used in dose calculation)	NR	NR	NR	NR	
Indoor dust filtration factor	P,B	2	S	Uniform	NUREG/CR-6697, Att. C	0.15	0.95			0.55
Shielding factor, external gamma	P	2	S	Bounded Lognormal - N	NUREG/CR-6697, Att. C	-1.3	0.59	0.044	1	0.27

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Fraction of time spent indoors	B	3	D	0.649	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Fraction of time spent outdoors (on site)	B	3	D	0.124	NUREG/CR-5512, Vol. 3 Table 6.87 (outdoors + gardening)	NR	NR	NR	NR	
Shape factor flag, external gamma	P	3	D	Circular	Circular contaminated zone assumed for modeling purposes	NR	NR	NR	NR	
Ingestion, Dietary										
Fruits, non-leafy vegetables, grain consumption (kg/y)	M,B	2	D	112	NUREG/CR-5512, Vol. 3 Table 6.87 (other vegetables + fruits + grain)	NR	NR	NR	NR	
Leafy vegetable consumption (kg/y)	M,B	3	D	21.4	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk consumption (L/y)	M,B	2	D	233	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry consumption (kg/y)	M,B	3	D	65.1	NUREG/CR5512, Vol. 3 Table 6.87 (beef + poultry)	NR	NR	NR	NR	
Fish consumption (kg/y)	M,B	3	D	20.6	NUREG/CR-5512, Vol. 3 Table 6.87 Note: Aquatic Pathway inactive in BFM	NR	NR	NR	NR	
Other seafood consumption (kg/y)	M,B	3	D	0.9	RESRAD User's Manual Table D.2 Note: Aquatic Pathway inactive in BFM	NR	NR	NR	NR	
Soil ingestion rate (g/y)	M,B	2	D	18.3	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Drinking water intake (L/y)	M,B	2	D	478	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Contamination fraction of drinking water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of household water (if used)	B,P	3		NA						
Contamination fraction of livestock water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of irrigation water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Contamination fraction of aquatic food	B,P	2	NA	NA	Assumption that pond is constructed that intercepts contaminated water not credible at Zion site	NR	NR	NR	NR	
Contamination fraction of plant food	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Contamination fraction of meat	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Contamination fraction of milk	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Ingestion, Non-Dietary										
Livestock fodder intake for meat (kg/day)	M	3	D	28.3	NUREG/CR5512, Vol. 3 Table 6.87 (forage, grain and hay for beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock fodder intake for milk (kg/day)	M	3	D	65.2	NUREG/CR5512, Vol. 3 Table 6.87 (forage + grain + hay)	NR	NR	NR	NR	
Livestock water intake for meat (L/day)	M	3	D	50.6	NUREG/CR5512, Vol. 3 Table 6.87 (beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Livestock soil intake (kg/day)	M	3	D	0.5	RESRAD User's Manual, Appendix L	NR	NR	NR	NR	
Mass loading for foliar deposition (g/m^3)	P	3	D	4.00E-04	NUREG/CR-5512, Vol. 3 Table 6.87, gardening	NR	NR	NR	NR	
Depth of soil mixing layer (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C	0	0.15	0.6		0.23
Depth of roots (m)	P	1	S	Uniform	NUREG/CR-6697, Att. C	0.3	4.0			2.15
Drinking water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Household water fraction from ground water (if used)	B,P	3		NA	Not used					
Livestock water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Irrigation fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Wet weight crop yield for Non-Leafy (kg/m^2)	P	2	S	Truncated Lognormal - N	NUREG/CR-6697, Att. C	0.56	0.48	0.001	0.999	1.75

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Wet weight crop yield for Leafy (kg/m^2)	P	3	D	2.89	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet weight crop yield for Fodder (kg/m^2)	P	3	D	1.91	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Non-Leafy (y)	P	3	D	0.25	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Leafy (y)	P	3	D	0.12	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Fodder (y)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/y)	P	2	S	Triangular	NUREG/CR-6697, Att. C	5.1	18	84		33
Wet Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	S	Triangular	NUREG/CR-6697, Att. C	0.06	0.67	0.95		0.58
Wet Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Storage times of contaminated foodstuffs (days):										
Fruits, non-leafy vegetables, and grain	B	3	D	14	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef = 20d and poultry =1 day. Lowest value used)	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Fish	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive in BFM	NR	NR	NR	NR	
Crustacea and mollusks	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive in BFM	NR	NR	NR	NR	
Well water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Surface water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Livestock fodder	B	3	D	45	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Special Radionuclides (C-14)										
C-12 concentration in water (g/cm^3)	P	3	NA	NA	NA	NR	NR	NR	NR	
C-12 concentration in contaminated soil (g/g)	P	3	NA	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from soil	P	3	NA	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from air	P	3	NA	NA	NA	NR	NR	NR	NR	
C-14 evasion layer thickness in soil (m)	P	2	NA	NA	NA	NR	NR	NR	NR	
C-14 evasion flux rate from soil (1/sec)	P	3	NA	NA	NA	NR	NR	NR	NR	
C-12 evasion flux rate from soil (1/sec)	P	3	NA	NA	NA	NR	NR	NR	NR	
Fraction of grain in beef cattle feed	B	3	NA	NA	NA	NR	NR	NR	NR	
Fraction of grain in milk cow feed	B	3	NA	NA	NA	NR	NR	NR	NR	
Dose Conversion Factors (Inhalation mrem/pCi)										
Co-60	M	3	D	2.19E-04	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	4.62E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	3.19E-05	FGR11	NR	NR	NR	NR	
Eu-152	M	3	D	2.21E-04	FGR11	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Eu-154	M	3	D	2.86E-04	FGR11	NR	NR	NR	NR	
Gd-152	M	3	D	2.43E-01	FGR11	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	6.29E-06	FGR11	NR	NR	NR	NR	
Nd-144[l]	M	3	D	7.04E-02	ICRP60	NR	NR	NR	NR	
Sm-148[l]	M	3	D	7.34E-02	ICRP60	NR	NR	NR	NR	
Sr-90	M	3	D	1.30E-03	FGR11	NR	NR	NR	NR	
Dose Conversion Factors (Ingestion mrem/pCi)										
Co-60	M	3	D	2.69E-05	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	5.00E-05	FGR11	NR	NR	NR	NR	
Eu-152	M	3	D	6.48E-06	FGR11	NR	NR	NR	NR	
Eu-154	M	3	D	9.55E-06	FGR11	NR	NR	NR	NR	
Gd-152	M	3	D	1.61E-04	FGR11	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR11	NR	NR	NR	NR	
Nd-144[l]	M	3	D	1.51E-04	ICRP60	NR	NR	NR	NR	
Sm-148[l]	M	3	D	1.58E-04	ICRP60	NR	NR	NR	NR	
Sr-90	M	3	D	1.42E-04	FGR11	NR	NR	NR	NR	
Plant Transfer Factors (pCi/g plant)/(pCi/g soil)										
Co-60	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-2.53	0.9			7.9E-02
Cs-134	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Cs-137	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Eu-152	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Eu-154	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Gd-152	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
H-3	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	1.57	1.1			4.8E+00
Nd-144	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Ni-63	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.00	0.9			5.0E-02
Sm-148	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Sr-90	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-1.20	1.0			3.0E-01
Meat Transfer Factors (pCi/kg)/(pCi/d)										
Co-60	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.51	1.0			3.0E-02
Cs-134	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Cs-137	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Eu-152	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
Eu-154	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
Gd-152	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
H-3	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.42	1.0			1.2E-02
Nd-144	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
Ni-63	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-5.30	0.9			5.0E-03
Sm-148	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Sr-90	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C	-4.61	0.4			1.0E-02
Milk Transfer Factors (pCi/L)/(pCi/d)										
Co-60	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	0.7			2.0E-03

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Cs-134	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Cs-137	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Eu-152	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Eu-154	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Gd-152	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
H-3	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.6	0.9			1.0E-02
Nd-144	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Ni-63	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.91	0.7			2.0E-02
Sr-90	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	0.5			2.0E-03
Sm-148	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/L))										
Co-60	P	2	NA	Inactive	NUREG/CR-6697, Att. C	5.7	1.1			3.0E+02
Cs-134	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Cs-137	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Eu-152	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.9	1.1			4.9E+01
Eu-154	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.9	1.1			4.9E+01
Gd-152	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.2	1.1			2.5E+01
H-3	P	2	NA	Inactive	NUREG/CR-6697, Att. C	0	0.1			1.0E+00
Nd-144	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.6	1.1			9.9E-01
Ni-63	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.6	1.1			9.9E+01
Sm-148	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.2	1.1			2.5E+01
Sr-90	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.1	1.1			6.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/L))										
Co-60	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-134	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-137	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP BFM Sensitivity Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Eu-152	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Eu-154	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Gd-152	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
H-3	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Nd-144	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Ni-63	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Sm-148	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Sr-90	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	
Spacing				log	RESRAD Default	NR	NR	NR	NR	
Time integration parameters										
Maximum number of points for dose				17	RESRAD Default	NR	NR	NR	NR	

Notes: a P = physical, B = behavioral, M = metabolic; (see NUREG/CR-6697, Attachment B, Table 4.)
b 1 = high-priority parameter, 2 = medium-priority parameter, 3 = low-priority parameter (see NUREG/CR-6697, Attachment B, Table 4.1)
c D = deterministic, S = stochastic
d Distributions Statistical Parameters:
Lognormal-n: 1= mean, 2 = standard deviation
Bounded lognormal-n: 1= mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Truncated lognormal-n: 1= mean, 2 = standard deviation, 3 = lower quantile, 4 = upper quantile
Bounded normal: 1 = mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Beta: 1 = minimum, 2 = maximum, 3 = P-value, 4 = Q-value
Triangular: 1 = minimum, 2 = mode, 3 = maximum
Uniform: 1 = minimum, 2 = maximum

Footnote 1

Basement Fill Model: RESRAD Well Pumping Rate Parameter Calculation

Input	**Value**			**Reference**
Water Table Elevation	579.00 ft	176.0 m		Reference 6-21, Table 5.1
Ground Surface Elevation	591.00 ft	179.7 m		Reference 6-21, Table 5.2
Auxiliary Bldg. Floor Surface Area	27484.42 ft2	2540 m2		Reference 6-7, TSD 14-019
Auxiliary Bldg. Floor Elevation	542.00 ft	164.8 m		Reference 6-7, TSD 14-019
Demolition Below Grade	3.00 ft	0.9 m		Project Plans
Post-Dem Wall Height Aux Bldg	46.00 ft	14.0 m		Calculation
Water Table Height above Aux Floor	37.00 ft	11.2 m		Calculation
Ground Surface to Water Table	12.00 ft	3.648 m		Calculation
Cont Zone total porosity	0.35			[1]
Precipitation	0.83 m/y			[1]
well pump rate	**2250 m3/y**			NUREG-6697, Table 3.10-1 method
inputs	0.52 L/m2/d	irrigation rate		NUREG-5512, Volume 3, Table 6-18 (Illinios Average)
	0.19 m3/m2/yr	irrigation rate conversion		
	10000.00 m2	contaminated area	(nominal 2 cattle at ~1 per acre + 2000 m2 garden) [1]	
Calculation	328.70 m3	domestic use		Reference 6-25, Table 3.10-1
50.6 and 60 L/d (1 dairy 1 meat)	40.37 m3/y	livestock		Reference 6-25, Table 3.10-1
	189.80 m3/y	vegetable garden irrigation		Reference 6-25, Table 3.10-1
	1689.22 m3/y	pasture irrigation		Reference 6-25, Table 3.10-1
	1.64 m3/y	drinking water		Reference 6-25, Table 3.10-1

Conversion Factors

m/ft	0.304
m2/ft2	0.09

Ref [1]: Pastures for Profit: A guide to Rotational Grazing (A3529), University of Wisconson Extension, 2002.

ATTACHMENT 2

RESRAD Input Parameters for ZSRP BFM

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Soil Concentrations										
Basic radiation dose limit (mrem/y)	NA	3	D	25	10 CFR 20.1402	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR	
Distribution coefficients (contaminated, unsaturated, and saturated zones) (cm^3/g)										
Co-60	P	1	D	223	TSD 14-004	5.46	2.53	0.001	0.999	235
Cs-134	P	1	D	45	TSD 14-004	6.1	2.33	0.001	0.999	446
Cs-137	P	1	D	45	TSD 14-004	6.1	2.33	0.001	0.999	446
Eu-152	P	1	D	95	TSD 14-004	6.72	3.22	0.001	0.999	825
Eu-154	P	1	D	95	TSD 14-004	6.72	3.22	0.001	0.999	825
Gd-152 (daughter for Eu-152)	P	1	D	825	Median Value NUREG/CR-6697, Att. C	6.72	3.22	0.001	0.999	825
H-3	P	1	D	0	TSD 14-004	-2.81	0.5	0.001	0.999	0.06
Nd-144 (daughter for Eu-152)	P	1	D	158	RESRADv.7.0 Default Nd not listed in NUREG/CR-6697	NA	NA	NA	NA	NA
Ni-63	P	1	D	62	TSD 14-004	6.05	1.46	0.001	0.999	424
Sm-148 (daughter Eu-152)	P	1	D	825	Median Value NUREG/CR-6697, Att. C	6.72	3.22	0.001	0.999	825
Sr-90	P	1	D	2.3	TSD 14-004	3.45	2.12	0.001	0.999	32
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	No existing groundwater contamination	NR	NR	NR	NR	
Calculation Times										
Time since placement of material (y)	P	3	D	1	For user convenience: Allows use of t=0 in dose and concentration output reports to calculate unitized Exposure Factors	NR	NR	NR	NR	
Time for calculations (y)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD Default	NR	NR	NR	NR	
Contaminated Zone										
Area of contaminated zone (m^2)	P	2	D	64,500	Area of the 'Radiological Protected Area' on Zion Site	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Thickness of contaminated zone (m)	P	2	D	11.2	Contaminated Zone is the Basement fill depth where mixing occurs. Depth of fill mixing zone depends on Basement floor elevation. 11.2 m is used as nominal value based on difference between elevations of the water table (579') and Auxiliary Basement floor (542') which equals 11.2m. Note: this parameter has no effect on the calculated values for unitized Exposure Factors.	NR	NR	NR	NR	
Length parallel to aquifer flow (m)	P	2	D	287	Diameter of 64,500 m2 contaminated area. Note: not applicable to Basement Fill Model because Mass Balance groundwater model used.	NR	NR	NR	NR	
Does the initial contamination penetrate the water table?	NA	NA	NA	Yes	100% of the contamination assumed to be in the basement fill water mixing zone	NA	NA	NA	NA	
Contaminated fraction below water table	P[e]	3[e]	D	1	100% of the contamination assumed to be in the basement fill water mixing zone	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	3.6m	Difference between ground level elevation at 591' (179.6m) and equilibrium water level in basements at 579' (176m)	NR	NR	NR	NR	NA
Density of cover (g/cm^3)	P	1	D	1.8	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.5.	1.52	0.23	0.001	0.999	1.52
Cover erosion rate (m/y)	P,B	2	D	0.0015	Median NUREG/CR-6697 Att. C	5E-08	0.0007	0,005	0.2	0.0015

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Density of contaminated zone (g/cm^3)	P	1	D	1.8	Density identified as sensitive and positively correlated. The 75th Percentile of the NUREG/CR-6697 Att. C distribution is 1.67 g/cm^3. However, the site-specific value for sand density is 1.8 g/cm^3. Fill to be comprised of undetermined combination of clean concrete and native sand therefore higher value for site-specific sand used.	1.52	0.23	0.001	0.999	1.52
Contaminated zone erosion rate (m/y)	P,B	2	D	0.0015	Median NUREG/CR-6697 Att. C	5E-08	0.0007	0,005	0.2	0.0015
Contaminated zone total porosity	P	2	D	0.37	25th Percentile NUREG/CR-6697 Att. C.	0.425	0.0867	0.001	0.999	0.42
Contaminated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Contaminated zone hydraulic conductivity (m/y)	P	2	D	2880	Site-specific value from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Contaminated zone b parameter	P	2	D	2.89	Median NUREG/CR-6697, Att. C	1.06	0.66	0.5	30	2.89
Humidity in air (g/m^3)	P	3	D	7.2	Median NUREG/CR-6697 Att. C	1.98	0.334	0.001	0.999	7.2
Evapotranspiration coefficient	P	2	D	0.625	Median NUREG/CR-6697 Att. C	0.5	0.75	NR	NR	0.625
Average annual wind speed (m/s)	P	2	D	4.2	Median NUREG/CR-6697 Att. C	1.445	0.2419	1.4	13	4.2
Precipitation (m/y)	P	2	D	0.83	Site-specific value from Reference 6-21, Table 5.12	NR	NR	NR	NR	
Irrigation (m/y)	B	3	D	0.19	NUREG-5512, Vol. 3, Table 6-18 (Illinois Average). Converted 0.52 L/m2/d to m/y.	NR	NR	NR	NR	
Irrigation mode	B	3	D	Overhead	Overhead irrigation is common practice in U. S.	NR	NR	NR	NR	
Runoff coefficient	P	2	D	0.2	Site-specific value from Reference 6-21, Section 5.10	0.1	0.8	NR	NR	0.45
Watershed area for nearby stream or pond (m^2)	P	3	D	1.0E+06	RESRAD Default	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Accuracy for water/soil computations	-	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Saturated Zone Hydrological Data										
Density of saturated zone (g/cm^3)	P	1	D	1.8	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.5.	1.51	0.16	0.001	0.999	1.51
Saturated zone total porosity	P	1	D	0.35	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.6	0.43	0.06	0.001	0.999	0.43
Saturated zone effective porosity	P	1	D	0.29	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.7	0.383	0.0610	0.001	0.999	0.383
Saturated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Saturated zone hydraulic conductivity (m/y)	P	1	D	1695	25th percentile Site-specific distribution from Reference 6-21, Table 5.9.	786	17000	NA	NA	3649
Saturated zone hydraulic gradient	P	2	D	0.0018	25th Percentile NUREG/CR-6697 Att. C distribution Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.10 is greater at 0.0039 but lower value used	-0.511	1.77	0.00007	0.5	0.006
Saturated zone b parameter	P	2	D	NA saturated zone b not active in RESRAD because water table drop rate =0	RESRAD User Manual	NR	NR	NR	NR	NR
Water table drop rate (m/y)	P	3	D	0	Basement fill water assumed to fully supply well.	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Well pump intake depth (m below water table)	P	2	D	5.6	Basement depths vary. 5.2m selected as nominal value based on mid-point of 11.2m contaminated zone for Auxiliary Basement. Note: this parameter has no effect on the calculated values for unitized Exposure Factors.	6	10	30	NA	10
Model: Non-dispersion (ND) or Mass-Balance (MB)	P	3	D	MB	MB model most applicable to assumption that well located in center of basement fill.	NR	NR	NR	NR	
Well pumping rate (m^3/y)	B,P	2	D	2250	Calculated according to method described in NUREG/CR-6697, Att. C Section 3.10 using Illinois specific irrigation rate and NUREG/CR-5512 vol. 3 livestock water consumption rate.	NR	NR	NR	NR	NR
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	NA	NA	0	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone thickness (m)	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone soil density (g/cm^3)	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone total porosity	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone effective porosity	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone field capacity	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone hydraulic conductivity (m/y)	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Unsat. zone soil-specific b parameter	P	NA	NA	NA	No unsaturated zone in Basement Fill Model	NA	NA	NA	NA	
Occupancy										

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Inhalation rate (m^3/y)	M,B	3	D	8400	NUREG/CR-5512, Vol. 3 Table 6.29 (= 23 m^3/d x 365 d/y)	NR	NR	NR	NR	
Mass loading for inhalation (g/m^3)	P,B	2	D	2.35E-05	Median NUREG/CR-6697, Att. C	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	2.35E-05
Exposure duration	B	3	D	30	RESRAD User's Manual (Parameter not used in dose calculation)	NR	NR	NR	NR	
Indoor dust filtration factor	P,B	2	D	0.55	Median NUREG/CR-6697, Att. C	0.15	0.95			0.55
Shielding factor, external gamma	P	2	D	0.27	Median NUREG/CR-6697, Att. C	-1.3	0.59	0.044	1	0.27
Fraction of time spent indoors	B	3	D	0.649	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Fraction of time spent outdoors (on site)	B	3	D	0.124	NUREG/CR-5512, Vol. 3 Table 6.87 (outdoors + gardening)	NR	NR	NR	NR	
Shape factor flag, external gamma	P	3	D	Circular	Circular contaminated zone assumed for modeling purposes	NR	NR	NR	NR	
Ingestion, Dietary										
Fruits, non-leafy vegetables, grain consumption (kg/y)	M,B	2	D	112	NUREG/CR-5512, Vol. 3 Table 6.87 (other vegetables + fruits + grain)	NR	NR	NR	NR	
Leafy vegetable consumption (kg/y)	M,B	3	D	21.4	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk consumption (L/y)	M,B	2	D	233	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry consumption (kg/y)	M,B	3	D	65.1	NUREG/CR5512, Vol. 3 Table 6.87 (beef + poultry)	NR	NR	NR	NR	
Fish consumption (kg/y)	M,B	3	D	20.6	NUREG/CR-5512, Vol. 3 Table 6.87 Note: Aquatic Pathway inactive in BFM	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Other seafood consumption (kg/y)	M,B	3	D	0.9	RESRAD User's Manual Table D.2 Note: Aquatic Pathway inactive in BFM	NR	NR	NR	NR	
Soil ingestion rate (g/y)	M,B	2	D	18.3	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Drinking water intake (L/y)	M,B	2	D	478	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Contamination fraction of drinking water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of household water (if used)	B,P	3		NA						
Contamination fraction of livestock water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of irrigation water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of aquatic food	B,P	2	D	NA	Assumption that pond is constructed that intercepts contaminated water not credible at Zion site	NR	NR	NR	NR	
Contamination fraction of plant food	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Contamination fraction of meat	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Contamination fraction of milk	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Ingestion, Non-Dietary										
Livestock fodder intake for meat (kg/day)	M	3	D	28.3	NUREG/CR5512, Vol. 3 Table 6.87 (forage, grain and hay for beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock fodder intake for milk (kg/day)	M	3	D	65.2	NUREG/CR5512, Vol. 3 Table 6.87 (forage + grain + hay)	NR	NR	NR	NR	
Livestock water intake for meat (L/day)	M	3	D	50.6	NUREG/CR5512, Vol. 3 Table 6.87 (beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Livestock soil intake (kg/day)	M	3	D	0.5	RESRAD User's Manual, Appendix L	NR	NR	NR	NR	
Mass loading for foliar deposition (g/m^3)	P	3	D	4.00E-04	NUREG/CR-5512, Vol. 3 Table 6.87, gardening	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Depth of soil mixing layer (m)	P	2	D	0.23	Median NUREG/CR-6697, Att. C	0	0.15	0.6		0.23
Depth of roots (m)	P	1	D	3.1	75th Percentile NUREG/CR-6697, Att. C	0.3	4.0			2.15
Drinking water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Household water fraction from ground water (if used)	B,P	3		NA						
Livestock water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Irrigation fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Wet weight crop yield for Non-Leafy (kg/m^2)	P	2	D	1.26	25th Percentile NUREG/CR-6697, Att. C	0.56	0.48	0.001	0.999	1.75
Wet weight crop yield for Leafy (kg/m^2)	P	3	D	2.89	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet weight crop yield for Fodder (kg/m^2)	P	3	D	1.91	NUREG/CR-5512, Vol. 3 Table 6.87 (maximum of forage, grain and hay)	NR	NR	NR	NR	
Growing Season for Non-Leafy (y)	P	3	D	0.25	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Leafy (y)	P	3	D	0.12	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Fodder (y)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/y)	P	2	D	21.5	25th Percentile NUREG/CR-6697, Att. C	5.1	18	84		33
Wet Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	D	0.70	75th Percentile NUREG/CR-6697, Att. C	0.06	0.67	0.95		0.58
Wet Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Storage times of contaminated foodstuffs (days):										
Fruits, non-leafy vegetables, and grain	B	3	D	14	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef = 20d and poultry =1 day. Lowest value used)	NR	NR	NR	NR	
Fish	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive in BFM	NR	NR	NR	NR	
Crustacea and mollusks	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive in BFM	NR	NR	NR	NR	
Well water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Surface water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Livestock fodder	B	3	D	45	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Special Radionuclides (C-14)										
C-12 concentration in water (g/cm^3)	P	3	D	NA	NA	NR	NR	NR	NR	
C-12 concentration in contaminated soil (g/g)	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from soil	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from air	P	3	D	NA	NA	NR	NR	NR	NR	
C-14 evasion layer thickness in soil (m)	P	2	D	NA	NA	NR	NR	NR	NR	
C-14 evasion flux rate from soil (1/sec)	P	3	D	NA	NA	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
C-12 evasion flux rate from soil (1/sec)	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of grain in beef cattle feed	B	3	D	NA	NA	NR	NR	NR	NR	
Fraction of grain in milk cow feed	B	3	D	NA	NA	NR	NR	NR	NR	
Dose Conversion Factors (Inhalation mrem/pCi)										
Co-60	M	3	D	2.19E-04	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	4.62E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	3.19E-05	FGR11	NR	NR	NR	NR	
Eu-152	M	3	D	2.21E-04	FGR11	NR	NR	NR	NR	
Eu-154	M	3	D	2.86E-04	FGR11	NR	NR	NR	NR	
Gd-152	M	3	D	2.43E-01	FGR11	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR11	NR	NR	NR	NR	
Nd-144[f]	M	3	D	7.04E-02	ICRP60	NR	NR	NR	NR	
Ni-63	M	3	D	6.29E-06	FGR11	NR	NR	NR	NR	
Sm-148[f]	M	3	D	7.34E-02	ICRP60	NR	NR	NR	NR	
Sr-90	M	3	D	1.30E-03	FGR11	NR	NR	NR	NR	
Dose Conversion Factors (Ingestion mrem/pCi)										
Co-60	M	3	D	2.69E-05	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	5.00E-05	FGR11	NR	NR	NR	NR	
Eu-152	M	3	D	6.48E-06	FGR11	NR	NR	NR	NR	
Eu-154	M	3	D	9.55E-06	FGR11	NR	NR	NR	NR	
Gd-152	M	3	D	1.61E-04	FGR11	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR11	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Nd-144[f]	M	3	D	1.51E-04	ICRP60	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR11	NR	NR	NR	NR	
Sm-148[f]	M	3	D	1.58E-04	ICRP60	NR	NR	NR	NR	
Sr-90	M	3	D	1.42E-04	FGR11	NR	NR	NR	NR	
Plant Transfer Factors (pCi/g plant)/(pCi/g soil)										
Co-60	P	1	D	7.9E-02	Median NUREG/CR-6697, Att. C	-2.53	0.9			7.9E-02
Cs-134	P	1	D	4.0E-02	Median NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Cs-137	P	1	D	4.0E-02	Median NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Eu-152	P	1	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Eu-154	P	1	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Gd-152	P	1	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
H-3	P	1	D	4.8E+00	Median NUREG/CR-6697, Att. C	1.57	1.1			4.8E+00
Nd-144	P	1	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Ni-63	P	1	D	5.0E-02	Median NUREG/CR-6697, Att. C	-3.00	0.9			5.0E-02
Sm-148	P	1	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Sr-90	P	1	D	5.9E-01	75th Percentile NUREG/CR-6697, Att. C	-1.20	1.0			3.0E-01
Meat Transfer Factors (pCi/kg)/(pCi/d)										
Co-60	P	2	D	0.058	75th Percentile NUREG/CR-6697, Att. C	-3.51	1.0			3.0E-02
Cs-134	P	2	D	0.065	75th Percentile NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Cs-137	P	2	D	0.065	75th Percentile NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Eu-152	P	2	D	0.004	75th Percentile NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
Eu-154	P	2	D	0.004	75th Percentile NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Gd-152	P	2	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
H-3	P	2	D	0.012	Median NUREG/CR-6697, Att. C	-4.42	1.0			0.012
Nd-144	P	2	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.0			2.0E-03
Ni-63	P	2	D	0.0092	75[th] Percentile NUREG/CR-6697, Att. C	-5.30	0.9			5.0E-03
Sm-148	P	2	D	2.0E-03	Median NUREG/CR-6697, Att. C	-6.21	1.1			2.0E-03
Sr-90	P	2	D	0.013	75[th] Percentile NUREG/CR-6697, Att. C	-4.61	0.4			1.0E-02
Milk Transfer Factors (pCi/L)/(pCi/d)										
Co-60	P	2	D	0.0032	75[th] Percentile NUREG/CR-6697, Att. C	-6.21	0.7			2.0E-03
Cs-134	P	2	D	1.4E-02	75[th] Percentile NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Cs-137	P	2	D	1.4E-02	75[th] Percentile NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Eu-152	P	2	D	6.0E-05	Median NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Eu-154	P	2	D	6.0E-05	Median NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Gd-152	P	2	D	6.0E-05	Median NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
H-3	P	2	D	0.010	Median NUREG/CR-6697, Att. C	-4.6	0.9			1.0E-02
Nd-144	P	2	D	6.0E-05	Median NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Ni-63	P	2	D	0.032	75[th] Percentile NUREG/CR-6697, Att. C	-3.91	0.7			2.0E-02
Sm-148	P	2	D	6.0E-05	Median NUREG/CR-6697, Att. C	-9.72	0.9			6.0E-05
Sr-90	P	2	D	0.0028	75[th] Percentile NUREG/CR-6697, Att. C	-6.21	0.5			2.0E-03
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/L))										
Co-60	P	2	NA	Inactive	NUREG/CR-6697, Att. C	5.7	1.1			3.0E+02
Cs-134	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Cs-137	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Eu-152	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.9	1.1			4.9E+01
Eu-154	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.9	1.1			4.9E+01
Gd-152	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.2	1.1			2.5E+01
H-3	P	2	NA	Inactive	NUREG/CR-6697, Att. C	0	0.1			1.0E+00
Nd-144	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.6	1.1			9.9E-01
Ni-63	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.6	1.1			1.0E+02
Sm-148	P	2	NA	Inactive	NUREG/CR-6697, Att. C	3.2	1.1			2.5E+01
Sr-90	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.1	1.1			6.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/L))										
Co-60	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-134	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-137	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Eu-152	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Eu-154	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Gd-152	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
H-3	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Nd-144	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Ni-63	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Sm-148	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Sr-90	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Spacing				log	RESRAD Default	NR	NR	NR	NR	
Time integration parameters										
Maximum number of points for dose				17	RESRAD Default	NR	NR	NR	NR	

Notes:
a P = physical, B = behavioral, M = metabolic; (see NUREG/CR-6697, Attachment B, Table 4.)
b 1 = high-priority parameter, 2 = medium-priority parameter, 3 = low-priority parameter (see NUREG/CR-6697, Attachment B, Table 4.1)
c D = deterministic, S = stochastic
d Distributions Statistical Parameters:
Lognormal-n: 1= mean, 2 = standard deviation
Bounded lognormal-n: 1= mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Truncated lognormal-n: 1= mean, 2 = standard deviation, 3 = lower quantile, 4 = upper quantile
Bounded normal: 1 = mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Beta: 1 = minimum, 2 = maximum, 3 = P-value, 4 = Q-value
Triangular: 1 = minimum, 2 = mode, 3 = maximum
Uniform: 1 = minimum, 2 = maximum

ATTACHMENT 3

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil Uncertainty Analysis

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Soil Concentrations										
Basic radiation dose limit (mrem/y)		3	D	25	10 CFR 20.1402	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR	
Distribution coefficients (contaminated, unsaturated, and saturated zones) (cm^3/g)										
Co-60	P	1	D	1161	TSD 14-004	5.46	2.53	0.001	0.999	235
Cs-134	P	1	D	615	TSD 14-004[2]	6.1	2.33	0.001	0.999	446
Cs-137	P	1	D	615	TSD 14-004	6.1	2.33	0.001	0.999	446
Ni-63	P	1	D	62	TSD 14-004	6.05	1.46	0.001	0.999	424
Sr-90	P	1	D	2.3	TSD 14-004	3.45	2.12	0.001	0.999	32
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	No existing groundwater contamination	NR	NR	NR	NR	
Calculation Times										
Time since placement of material (y)	P	3	D	0	RESRAD Default	NR	NR	NR	NR	
Time for calculations (y)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD Default	NR	NR	NR	NR	
Contaminated Zone										
Area of contaminated zone (m^2)	P	2	D	64,500	Area of the 'Security Protected Area' on Zion Site	NR	NR	NR	NR	
Thickness of contaminated zone (m)	P	2	D	0.15 or 1	Surface soil depth 0.15 Subsurface soil depth 1 m	NR	NR	NR	NR	
Length parallel to aquifer flow (m)	P	2	D	287	Diameter of 64,500 m2 contaminated area.	NR	NR	NR	NR	
Does the initial contamination penetrate the water table?	NA	NA	NA	No	No contamination in water table	NA	NA	NA	NA	
Contaminated fraction below water table	P[e]	3[e]	D	0	No contamination in water table	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	0	No Cover	NR	NR	NR	NR	NA
Density of cover (g/cm^3)	P	1	NA	NA	No Cover	NA	NA	NA	NA	NA
Cover erosion rate (m/y)	P,B	2	NA	Continuous Logarithmic	NUREG/CR-6697 Att. C Table 3.8-1	5E-08	0.0007	0.005	0.2	0.0015

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Density of contaminated zone (g/cm^3)	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	1.51	0.16	0.001	0.999	1.51
Contaminated zone erosion rate (m/y)	P,B	2	S	Continuous Logarithmic	NUREG/CR-6697 Att. C Table 3.8-1	5E-08	0.0007	0.005	0.2	0.0015
Contaminated zone total porosity	P	2	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.43	0.06	0.001	0.999	0.43
Contaminated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Contaminated zone hydraulic conductivity (m/y)	P	2	S	Loguniform	Site-specific distribution from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Contaminated zone b parameter	P	2	S	Truncated Lognormal - N	NUREG 6697 distribution for site soil type - sand	-.0253	0.216	0.001	0.999	0.97
Humidity in air (g/m^3)	P	3	D	7.2	Median NUREG/CR-6697 Att. C	1.98	0.334	0.001	0.999	7.2
Evapotranspiration coefficient	P	2	S	Uniform	NUREG/CR-6697 Att. C	0.5	0.75	NR	NR	0.625
Average annual wind speed (m/s)	P	2	S	Bounded Lognormal N	NUREG/CR-6697 Att. C	1.445	0.2419	1.4	13	4.2
Precipitation (m/y)	P	2	D	0.83	Site-specific value from Reference 6-21, Table 5.12	NR	NR	NR	NR	
Irrigation (m/y)	B	3	D	0.19	NUREG-5512, Vol. 3, Table 6-18 (Illinois Average) Converted 0.52 $L/m^2/y$ to m/y	NR	NR	NR	NR	0.56
Irrigation mode	B	3	D	Overhead	Overhead irrigation is common practice in U. S.	NR	NR	NR	NR	
Runoff coefficient	P	2	S	Uniform	NUREG/CR-6697 Att. C	0.1	0.8	NR	NR	0.45
Watershed area for nearby stream or pond (m^2)	P	3	D	1.0E+06	RESRAD Default	NR	NR	NR	NR	
Accuracy for water/soil computations	-	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Saturated Zone Hydrological Data										
Density of saturated zone (g/cm^3)	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	1.51	0.16	0.001	0.999	1.51
Saturated zone total porosity	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.43	0.06	0.001	0.999	0.43
Saturated zone effective porosity	P	1	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.383	0.0610	0.001	0.999	0.383
Saturated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Saturated zone hydraulic conductivity (m/y)	P	1	S	Loguniform	Site-specific distribution from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Saturated zone hydraulic gradient	P	2	S	Bounded Lognormal - N	NUREG/CR-6697 Att. C	-5.11	1.77	0.00007	0.5	0.006
Saturated zone b parameter	P	2	D	NA saturated zone b not active because water table drop rate =0	NUREG/CR-6697 Att. C	NR	NR	NR	NR	NR
Water table drop rate (m/y)	P	3	D	0	Well pumping rate assumed small relative to water table volume.	NR	NR	NR	NR	
Well pump intake depth (m below water table)	P	2	S	Triangular	NUREG/CR-6697	6	10	30		10
Model: Non-dispersion (ND) or Mass-Balance (MB)	P	3	D	ND	Non Dispersion Model used	NR	NR	NR	NR	
Well pumping rate (m^3/y)	B,P	2	S	2250	Calculated according to method described in NUREG/CR-6697, Att. C Section 3.10 using Illinois specific irrigation rate and NUREG/CR-5512 vol. 3 livestock water intake rate	NR	NR	NR	NR	NR
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	3	D	1	One unsaturated zone	NA	NA	NA	NA	
Unsat. zone thickness (m)	P	1	D	3.45 (for 0.15 m contaminated zone thickness) 2.6 (for 1.0 m contaminated zone thickness)	Distance from ground surface (591') to water table (579') = 3.6 Reference 6-21, Tables 5.1 and 5.2 For 0.15 m contaminated zone thickness unsaturated zone = 3.6 – 0.15 = 3.45 m For 1.0 m contaminated zone thickness unsaturated zone = 3.6 – 1.0 = 2.6 m	NA	NA	NA	NA	
Unsat. zone soil density (g/cm^3)	P	2	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	1.51	0.16	0.001	0.999	1.51
Unsat. zone total porosity	P	2	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.43	0.06	0.001	0.999	0.43

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Unsat. zone effective porosity	P	2	S	Truncated Normal	NUREG 6697 distribution for site soil type - sand	0.383	0.0610	0.001	0.999	0.383
Unsat. zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Unsat. zone hydraulic conductivity (m/y)	P	2	S	Loguniform	Site-specific distribution from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Unsat. zone soil-specific b parameter	P	2	S	Truncated Lognormal - N	NUREG 6697 distribution for site soil type - sand	-.0253	0.216	0.001	0.999	0.97
Occupancy										
Inhalation rate (m^3/y)	M,B	3	D	8400	NUREG/CR-5512, Vol. 3 Table 6.29 (23 m^3/d x 365 d)	NR	NR	NR	NR	
Mass loading for inhalation (g/m^3)	P,B	2	S	Continuous Linear	NUREG/CR-6697, Att. C	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	2.35E-05
Exposure duration	B	3	D	30	RESRAD User's Manual (Parameter not used in dose calculation)	NR	NR	NR	NR	
Indoor dust filtration factor	P,B	2	S	Uniform	NUREG/CR-6697, Att. C	0.15	0.95			0.55
Shielding factor, external gamma	P	2	S	Bounded Lognormal - N	NUREG/CR-6697, Att. C	-1.3	0.59	0.044	1	0.2725
Fraction of time spent indoors	B	3	D	0.649	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Fraction of time spent outdoors (on site)	B	3	D	0.124	NUREG/CR-5512, Vol. 3 Table 6.87 (outdoors + gardening)	NR	NR	NR	NR	
Shape factor flag, external gamma	P	3	D	Circular	Circular contaminated zone assumed for modeling purposes	NR	NR	NR	NR	
Ingestion, Dietary										
Fruits, non-leafy vegetables, grain consumption (kg/y)	M,B	2	D	112	NUREG/CR-5512, Vol. 3 Table 6.87 (other vegetables + fruits + grain)	NR	NR	NR	NR	
Leafy vegetable consumption (kg/y)	M,B	3	D	21.4	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Milk consumption (L/y)	M,B	2	D	233	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry consumption (kg/y)	M,B	3	D	65.1	NUREG/CR5512, Vol. 3 Table 6.87 (beef + poultry)	NR	NR	NR	NR	
Fish consumption (kg/y)	M,B	3	D	20.6	NUREG/CR-5512, Vol. 3 Table 6.87 Note: Aquatic Pathway inactive	NR	NR	NR	NR	
Other seafood consumption (kg/y)	M,B	3	D	0.9	RESRAD User's Manual Table D.2 Note: Aquatic Pathway inactive	NR	NR	NR	NR	
Soil ingestion rate (g/y)	M,B	2	D	18.3	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Drinking water intake (L/y)	M,B	2	D	478	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Contamination fraction of drinking water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of household water (if used)	B,P	3		NA						
Contamination fraction of livestock water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of irrigation water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of aquatic food	B,P	2	D	NA	Assumption that pond is constructed that intercepts contaminated water not credible at Zion site	NR	NR	NR	NR	
Contamination fraction of plant food	B,P	3	D	1	100% of food consumption assumed contaminated	NR	NR	NR	NR	
Contamination fraction of meat	B,P	3	D	1	100% of food consumption assumed contaminated	NR	NR	NR	NR	
Contamination fraction of milk	B,P	3	D	1	100% of food consumption assumed contaminated	NR	NR	NR	NR	
Ingestion, Non-Dietary										

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Livestock fodder intake for meat (kg/day)	M	3	D	28.3	NUREG/CR5512, Vol. 3 Table 6.87 (forage, grain and hay for beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock fodder intake for milk (kg/day)	M	3	D	65.2	NUREG/CR5512, Vol. 3 Table 6.87 (forage + grain + hay)	NR	NR	NR	NR	
Livestock water intake for meat (L/day)	M	3	D	50.6	NUREG/CR5512, Vol. 3 Table 6.87 (beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Livestock soil intake (kg/day)	M	3	D	0.5	RESRAD User's Manual, Appendix L	NR	NR	NR	NR	
Mass loading for foliar deposition (g/m^3)	P	3	D	4.00E-04	NUREG/CR-5512, Vol. 3 Table 6.87, gardening	NR	NR	NR	NR	
Depth of soil mixing layer (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C	0	0.15	0.6		0.23
Depth of roots (m)	P	1	S	Uniform	NUREG/CR-6697, Att. C	0.3	4.0			2.15
Drinking water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Household water fraction from ground water (if used)	B,P	3		NA						
Livestock water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Irrigation fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Wet weight crop yield for Non-Leafy (kg/m^2)	P	2	S	Truncated Lognormal - N	NUREG/CR-6697, Att. C	0.56	0.48	0.001	0.999	1.75
Wet weight crop yield for Leafy (kg/m^2)	P	3	D	2.90	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet weight crop yield for Fodder (kg/m^2)	P	3	D	1.90	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Non-Leafy (y)	P	3	D	0.246	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Leafy (y)	P	3	D	0.123	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Fodder (y)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/y)	P	2	S	Triangular	NUREG/CR-6697, Att. C	5.1	18	84		33
Wet Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	D	Triangular	NUREG/CR-6697, Att. C	0.06	0.67	0.95		0.58
Wet Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Storage times of contaminated foodstuffs (days):										
Fruits, non-leafy vegetables, and grain	B	3	D	14	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef = 20d and poultry =1 day. Lowest value used)	NR	NR	NR	NR	
Fish	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive	NR	NR	NR	NR	
Crustacea and mollusks	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive	NR	NR	NR	NR	
Well water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Surface water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Livestock fodder	B	3	D	45	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Special Radionuclides (C-14)										
C-12 concentration in water (g/cm^3)	P	3	D	NA	NA	NR	NR	NR	NR	
C-12 concentration in contaminated soil (g/g)	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from soil	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from air	P	3	D	NA	NA	NR	NR	NR	NR	
C-14 evasion layer thickness in soil (m)	P	2	D	NA	NA	NR	NR	NR	NR	
C-14 evasion flux rate from soil (1/sec)	P	3	D	NA	NA	NR	NR	NR	NR	
C-12 evasion flux rate from soil (1/sec)	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of grain in beef cattle feed	B	3	D	NA	NA	NR	NR	NR	NR	
Fraction of grain in milk cow feed	B	3	D	NA	NA	NR	NR	NR	NR	
Dose Conversion Factors (Inhalation mrem/pCi)										
Co-60	M	3	D	2.19E-04	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	4.62E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	3.19E-05	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	6.29E-06	FGR11	NR	NR	NR	NR	
Sr-90	M	3	D	1.30E-03	FGR11	NR	NR	NR	NR	
Dose Conversion Factors (Ingestion mrem/pCi)										
Co-60	M	3	D	2.69E-05	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	5.00E-05	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR11	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Sr-90	M	3	D	1.42E-04	FGR11	NR	NR	NR	NR	
Plant Transfer Factors (pCi/g plant)/(pCi/g soil)										
Co-60	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-2.53	0.9			7.9E-02
Cs-134	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Cs-137	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Ni-63	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.00	0.9			5.0E-02
Sr-90	P	1	S	Lognormal - N	NUREG/CR-6697, Att. C	-1.20	1.0			3.0E-01
Meat Transfer Factors (pCi/kg)/(pCi/d)										
Co-60	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.51	1.0			3.0E-02
Cs-134	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Cs-137	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Ni-63	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-5.30	0.9			5.0E-03
Sr-90	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.61	0.4			1.0E-02
Milk Transfer Factors (pCi/L)/(pCi/d)										
Co-60	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	0.7			2.0E-03
Cs-134	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Cs-137	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Ni-63	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-3.91	0.7			2.0E-02
Sr-90	P	2	S	Lognormal - N	NUREG/CR-6697, Att. C	-6.21	0.5			2.0E-03
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/L))										
Co-60	P	2	NA	Inactive	NUREG/CR-6697, Att. C	5.7	1.1			3.0E+02

RESRAD Input Parameters for ZSRP Soil Uncertainty Analysis

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Cs-134	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Cs-137	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Ni-63	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.6	1.1			9.9E+01
Sr-90	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.1	1.1			6.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/L))										
Co-60	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-134	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-137	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Ni-63	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Sr-90	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	
Spacing				log	RESRAD Default	NR	NR	NR	NR	
Time integration parameters										
Maximum number of points for dose				17	RESRAD Default	NR	NR	NR	NR	

Notes:
a P = physical, B = behavioral, M = metabolic; (see NUREG/CR-6697, Attachment B, Table 4.)
b 1 = high-priority parameter, 2 = medium-priority parameter, 3 = low-priority parameter (see NUREG/CR-6697, Attachment B, Table 4.1)
c D = deterministic, S = stochastic
d Distributions Statistical Parameters:
Lognormal-n: 1= mean, 2 = standard deviation
Bounded lognormal-n: 1= mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Truncated lognormal-n: 1= mean, 2 = standard deviation, 3 = lower quantile, 4 = upper quantile
Bounded normal: 1 = mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Beta: 1 = minimum, 2 = maximum, 3 = P-value, 4 = Q-value
Triangular: 1 = minimum, 2 = mode, 3 = maximum
Uniform: 1 = minimum, 2 = maximum

ATTACHMENT 4

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Soil Concentrations										
Basic radiation dose limit (mrem/y)		3	D	25	10 CFR 20.1402	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR	
Distribution coefficients (contaminated, unsaturated, and saturated zones) (cm^3/g)										
Co-60	P	1	D	1161	TSD 14-004[2]	5.46	2.53	0.001	0.999	235
Cs-134	P	1	D	615	TSD 14-004[2]	6.1	2.33	0.001	0.999	446
Cs-137	P	1	D	615	TSD 14-004[2]	6.1	2.33	0.001	0.999	446
Ni-63	P	1	D	62	TSD 14-004[2]	6.05	1.46	0.001	0.999	424
Sr-90	P	1	D	2.3	TSD 14-004[2]	3.45	2.12	0.001	0.999	32
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	No existing groundwater contamination	NR	NR	NR	NR	
Calculation Times										
Time since placement of material (y)	P	3	D	0	RESRAD Default	NR	NR	NR	NR	
Time for calculations (y)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD Default	NR	NR	NR	NR	
Contaminated Zone										
Area of contaminated zone (m^2)	P	2	D	64,500	Area of the 'Security Protected Area' on Zion Site	NR	NR	NR	NR	
Thickness of contaminated zone (m)	P	2	D	0.15 or 1.0	Surface Soil Depth = 0.15m Subsurface Soil Depth = 1m	NR	NR	NR	NR	
Length parallel to aquifer flow (m)	P	2	D	287	Diameter of 64,500 m2 contaminated area.	NR	NR	NR	NR	
Does the initial contamination penetrate the water table?	NA	NA	NA	No	No initial contamination in water table	NA	NA	NA	NA	
Contaminated fraction below water table	P[e]	3[e]	D	0	No initial contamination in water table	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	0	No Cover	NR	NR	NR	NR	NA
Density of cover (g/cm^3)	P	1	NA	NA	No Cover	NA	NA	NA	NA	NA
Cover erosion rate (m/y)	P,B	2	NA	NA	No Cover	NA	NA	NA	NA	NA
Density of contaminated zone (g/cm^3)	P	1	D	1.8	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.5.	1.51	0.16	0.001	0.999	1.51

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Contaminated zone erosion rate (m/y)	P,B	2	D	0.0015	Median NUREG/CR-6697 Att. C	5E-08	0.0007	0.005	0.2	0.0015
Contaminated zone total porosity	P	2	D	0.35	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.6	0.43	0.06	0.001	0.999	0.43
Contaminated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Contaminated zone hydraulic conductivity (m/y)	P	2	D	2880	Site-specific value from Reference 6-21, Table 5.9	786	17000	NA	NA	3649
Contaminated zone b parameter	P	2	D	0.97	Median NUREG 6697 distribution for site soil type - sand	-0.0253	0.216	NA	NA	0.97
Humidity in air (g/m^3)	P	3	D	7.2	Median NUREG/CR-6697 Att. C	1.98	0.334	0.001	0.999	7.2
Evapotranspiration coefficient	P	2	D	0.625	Median NUREG/CR-6697 Att. C	0.5	0.75	NR	NR	0.625
Average annual wind speed (m/s)	P	2	D	4.2	Median NUREG/CR-6697 Att. C	1.445	0.2419	1.4	13	4.2
Precipitation (m/y)	P	2	D	0.83	Site-specific value from Reference 6-21, Table 5.12	NR	NR	NR	NR	
Irrigation (m/y)	B	3	D	0.19	NUREG-5512, Vol. 3, Table 6-18 (Illinois Average)	NR	NR	NR	NR	
Irrigation mode	B	3	D	Overhead	Overhead irrigation is common practice in U. S.	NR	NR	NR	NR	
Runoff coefficient	P	2	D	0.2	Site-specific value from Reference 6-21, Section 5.10	0.1	0.8	NR	NR	0.45
Watershed area for nearby stream or pond (m^2)	P	3	D	1.0E+06	RESRAD Default	NR	NR	NR	NR	
Accuracy for water/soil computations	-	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Saturated Zone Hydrological Data										
Density of saturated zone (g/cm^3)	P	2	D	1.8	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.5.	1.51	0.16	0.001	0.999	1.52
Saturated zone total porosity	P	1	D	0.35	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.6	0.43	0.0699	0.214	0.646	0.43

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Saturated zone effective porosity	P	1	D	0.29	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.7	0.43	0.06	0.001	0.999	0.43
Saturated zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Saturated zone hydraulic conductivity (m/y)	P	1	D	2880	Site-specific average from Reference 6-21, Table 5.9.	786	17000	NA	NA	3649
Saturated zone hydraulic gradient	P	2	D	0.0039	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.10	-5.11	1.77	0.00007	0.5	0.006
Saturated zone b parameter	P	2	D	NA saturated zone b not active because water table drop rate =0	NUREG/CR-6697, Att. A, Table 2	NR	NR	NR	NR	NR
Water table drop rate (m/y)	P	3	D	0	Well pumping rate assumed small relative to water table volume.	NR	NR	NR	NR	
Well pump intake depth (m below water table)	P	2	D	3.3	Mid-point of Shallow Aquifer Reference 6-21, Table 5.1	NA	NA	NA		NA
Model: Non-dispersion (ND) or Mass-Balance (MB)	P	3	D	ND	Non-dispersion model used	NR	NR	NR	NR	
Well pumping rate (m^3/y)	P	2	D	2250	Calculated according to method described in NUREG/CR-6697, Att. C Section 3.10using Illinois specific irrigation rate and NUREG/CR-5512 vol. 3 livestock water intake rate	NR	NR	NR	NR	NR
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	3	D	1	One unsaturated zone	NA	NA	NA	NA	

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Unsat. zone thickness (m)	P	1	D	3.45 (for 0.15 m contaminated zone thickness) 2.6 (for 1.0 m contaminated zone thickness)	Distance from ground surface (591') to water table (579') = 3.6 Reference 6-21, Tables 5.1 and 5.2 For 0.15 m contaminated zone thickness unsaturated zone = 3.6 – 0.15 = 3.45 m For 1.0 m contaminated zone thickness unsaturated zone = 3.6 – 1.0 = 2.6 m	NA	NA	NA	NA	
Unsat. zone soil density (g/cm^3)	P	2	D	1.8	Site-specific value from Reference 6-21, Table 5.5	NA	NA	NA	NA	
Unsat. zone total porosity	P	1	D	0.35	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.6	0.43	0.0699	0.214	0.646	0.43
Unsat. zone effective porosity	P	1	D	0.29	Site-specific average native sand and disturbed sand from Reference 6-21, Table 5.7	0.342	0.0705	0.124	0.56	0.342
Unsat. zone field capacity	P	3	D	0.066	Site-specific value from Reference 6-21, Table 5.4	NR	NR	NR	NR	
Unsat. zone hydraulic conductivity (m/y)	P	2	D	2880	Site-specific average from Reference 6-21, Table 5.9.	-0.511	1.77	0.00007	0.5	0.006
Unsat. zone soil-specific b parameter	P	2	D	0.97	Median NUREG/CR-6697 Att. C Sand soil type	-0.0253	0.216	0.501	1.90	0.97
Occupancy										
Inhalation rate (m^3/y)	M,B	3	D	8400	NUREG/CR-5512, Vol. 3 Table 6.29 (=23 m^3/d x 365 d/y)	NR	NR	NR	NR	
Mass loading for inhalation (g/m^3)	P,B	2	D	2.35E-05	Median NUREG/CR-6697, Att. C	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	See NUREG-6697 Table 4.6-1	2.35E-05
Exposure duration	B	3	D	30	RESRAD User's Manual (Parameter not used in dose calculation)	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Indoor dust filtration factor	P,B	2	D	0.55	Median NUREG/CR-6697, Att. C	0.15	0.95			0.55
Shielding factor, external gamma	P	2	D	0.40	75th Percentile NUREG/CR-6697, Att. C	-1.3	0.59	0.044	1	0.272
Fraction of time spent indoors	B	3	D	0.649	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Fraction of time spent outdoors (on site)	B	3	D	0.124	NUREG/CR-5512, Vol. 3 Table 6.87 (outdoors + gardening)	NR	NR	NR	NR	
Shape factor flag, external gamma	P	3	D	Circular	Circular contaminated zone assumed for modeling purposes	NR	NR	NR	NR	
Ingestion, Dietary										
Fruits, non-leafy vegetables, grain consumption (kg/y)	M,B	2	D	112	NUREG/CR-5512, Vol. 3 Table 6.87 (other vegetables + fruits + grain)	NR	NR	NR	NR	
Leafy vegetable consumption (kg/y)	M,B	3	D	21.4	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk consumption (L/y)	M,B	2	D	233	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Meat and poultry consumption (kg/y)	M,B	3	D	65.1	NUREG/CR5512, Vol. 3 Table 6.87 (beef + poultry)	NR	NR	NR	NR	
Fish consumption (kg/y)	M,B	3	D	20.6	NUREG/CR-5512, Vol. 3 Table 6.87 Note: Aquatic Pathway inactive	NR	NR	NR	NR	
Other seafood consumption (kg/y)	M,B	3	D	0.9	RESRAD User's Manual Table D.2 Note: Aquatic Pathway inactive	NR	NR	NR	NR	
Soil ingestion rate (g/y)	M,B	2	D	18.3	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Drinking water intake (L/y)	M,B	2	D	478	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Contamination fraction of drinking water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of household water (if used)	B,P	3		NA						
Contamination fraction of livestock water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Contamination fraction of irrigation water	B,P	3	D	1	All water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of aquatic food	B,P	2	D	NA	Assumption that pond is constructed that intercepts contaminated water not credible at Zion site	NR	NR	NR	NR	
Contamination fraction of plant food	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Contamination fraction of meat	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Contamination fraction of milk	B,P	3	D	1	100% of food consumption rate from onsite source	NR	NR	NR	NR	
Ingestion, Non-Dietary										
Livestock fodder intake for meat (kg/day)	M	3	D	28.3	NUREG/CR5512, Vol. 3 Table 6.87 (forage, grain and hay for beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock fodder intake for milk (kg/day)	M	3	D	65.2	NUREG/CR5512, Vol. 3 Table 6.87 (forage + grain + hay)	NR	NR	NR	NR	
Livestock water intake for meat (L/day)	M	3	D	50.6	NUREG/CR5512, Vol. 3 Table 6.87 (beef cattle + poultry + layer hen)	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Livestock soil intake (kg/day)	M	3	D	0.5	RESRAD User's Manual, Appendix L	NR	NR	NR	NR	
Mass loading for foliar deposition (g/m^3)	P	3	D	4.00E-04	NUREG/CR-5512, Vol. 3 Table 6.87, gardening	NR	NR	NR	NR	
Depth of soil mixing layer (m)	P	2	D	0.15 for Surface Soil 0.23 for Subsurface Soil	25th Percentile NUREG/CR-6697, Att. C Median NUREG/CR-6697, Att. C	0	0.15	0.6		0.23
Depth of roots (m)	P	1	D	1.22	25th Percentile NUREG/CR-6697, Att. C	0.3	4.0			2.15
Drinking water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Household water fraction from ground water (if used)	B,P	3		NA						
Livestock water fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Irrigation fraction from ground water	B,P	3	D	1	All water assumed to be supplied from groundwater	NR	NR	NR	NR	
Wet weight crop yield for Non-Leafy (kg/m^2)	P	2	D	1.75	Median NUREG/CR-6697, Att. C	0.56	0.48	0.001	0.999	1.75
Wet weight crop yield for Leafy (kg/m^2)	P	3	D	2.90	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet weight crop yield for Fodder (kg/m^2)	P	3	D	1.90	NUREG/CR-5512, Vol. 3 Table 6.87 (maximum of forage, grain and hay)	NR	NR	NR	NR	
Growing Season for Non-Leafy (y)	P	3	D	0.246	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Leafy (y)	P	3	D	0.123	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Growing Season for Fodder (y)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/y)	P	2	D	33	Median NUREG/CR-6697, Att. C	5.1	18	84		33
Wet Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	D	0.58	Median NUREG/CR-6697, Att. C	0.06	0.67	0.95		0.58
Wet Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Storage times of contaminated foodstuffs (days):										
Fruits, non-leafy vegetables, and grain	B	3	D	14	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Meat and poultry	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef = 20d and poultry =1 day. Lowest value used)	NR	NR	NR	NR	
Fish	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive in BFM	NR	NR	NR	NR	
Crustacea and mollusks	B	3	D	7	RESRAD User's Manual Table D.6 Note: Aquatic pathway inactive in BFM	NR	NR	NR	NR	
Well water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Surface water	B	3	D	1	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Livestock fodder	B	3	D	45	RESRAD User's Manual Table D.6	NR	NR	NR	NR	
Special Radionuclides (C-14)										
C-12 concentration in water (g/cm^3)	P	3	D	NA	NA	NR	NR	NR	NR	
C-12 concentration in contaminated soil (g/g)	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from soil	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of vegetation carbon from air	P	3	D	NA	NA	NR	NR	NR	NR	
C-14 evasion layer thickness in soil (m)	P	2	D	NA	NA	NR	NR	NR	NR	
C-14 evasion flux rate from soil (1/sec)	P	3	D	NA	NA	NR	NR	NR	NR	
C-12 evasion flux rate from soil (1/sec)	P	3	D	NA	NA	NR	NR	NR	NR	
Fraction of grain in beef cattle feed	B	3	D	NA	NA	NR	NR	NR	NR	
Fraction of grain in milk cow feed	B	3	D	NA	NA	NR	NR	NR	NR	
Dose Conversion Factors (Inhalation mrem/pCi)										
Co-60	M	3	D	2.19E-04	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	4.62E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	3.19E-05	FGR11	NR	NR	NR	NR	

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Ni-63	M	3	D	6.29E-06	FGR11	NR	NR	NR	NR	
Sr-90	M	3	D	1.30E-03	FGR11	NR	NR	NR	NR	
Dose Conversion Factors (Ingestion mrem/pCi)										
Co-60	M	3	D	2.69E-05	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR11	NR	NR	NR	NR	
Cs-137	M	3	D	5.00E-05	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR11	NR	NR	NR	NR	
Sr-90	M	3	D	1.42E-04	FGR11	NR	NR	NR	NR	
Plant Transfer Factors (pCi/g plant)/(pCi/g soil)										
Co-60	P	1	D	1.5E-01	75th Percentile NUREG/CR-6697, Att. C	-2.53	0.9			7.9E-02
Cs-134	P	1	D	7.8E-02	75th Percentile NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Cs-137	P	1	D	7.8E-02	75th Percentile NUREG/CR-6697, Att. C	-3.22	1.0			4.0E-02
Ni-63	P	1	D	9.2E-02	75th Percentile NUREG/CR-6697, Att. C	-3.00	0.9			5.0E-02
Sr-90	P	1	D	5.9E-01	75th Percentile NUREG/CR-6697, Att. C	-1.20	1.0			3.0E-01
Meat Transfer Factors (pCi/kg)/(pCi/d)										
Co-60	P	2	D	5.8E-02	75th Percentile NUREG/CR-6697, Att. C	-3.51	1.0			3.0E-02
Cs-134	P	2	D	6.5E-02	75th Percentile NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Cs-137	P	2	D	6.5E-02	75th Percentile NUREG/CR-6697, Att. C	-3.00	0.4			5.0E-02
Ni-63	P	2	D	5E-03	Median NUREG/CR-6697, Att. C	-5.30	0.9			5.0E-03
Sr-90	P	2	D	8E-03	Median NUREG/CR-6697, Att. C	-4.61	0.4			1.0E-02
Milk Transfer Factors (pCi/L)/(pCi/d)										
Co-60	P	2	D	2E-03	Median NUREG/CR-6697, Att. C	-6.21	0.7			2.0E-03
Cs-134	P	2	D	1.4E-02	75th Percentile NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Cs-137	P	2	D	1.4E-02	75th Percentile NUREG/CR-6697, Att. C	-4.61	0.5			1.0E-02
Ni-63	P	2	D	3.2E-02	75th Percentile NUREG/CR-6697, Att. C	-3.91	0.7			2.0E-02

RESRAD Input Parameters for ZSRP Surface Soil and Subsurface Soil DCGL

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Mean/ Median
Sr-90	P	2	D	2.7E-03	75th Percentile NUREG/CR-6697, Att. C	-6.21	0.5			2.0E-03
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/L))										
Co-60	P	2	NA	Inactive	NUREG/CR-6697, Att. C	5.7	1.1			3.0E+02
Cs-134	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Cs-137	P	2	NA	Inactive	NUREG/CR-6697, Att. C	7.6	0.7			2.0E+03
Ni-63	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.6	1.1			9.9E+01
Sr-90	P	2	NA	Inactive	NUREG/CR-6697, Att. C	4.1	1.1			6.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/L))										
Co-60	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-134	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Cs-137	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Ni-63	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Sr-90	P	3	NA	Inactive	RESRAD User's Manual Appendix D	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	
Spacing				log	RESRAD Default	NR	NR	NR	NR	
Time integration parameters										
Maximum number of points for dose				17	RESRAD Default	NR	NR	NR	NR	

Notes: a P = physical, B = behavioral, M = metabolic; (see NUREG/CR-6697, Attachment B, Table 4.)
b 1 = high-priority parameter, 2 = medium-priority parameter, 3 = low-priority parameter (see NUREG/CR-6697, Attachment B, Table 4.1)
c D = deterministic, S = stochastic
d Distributions Statistical Parameters:
Lognormal-n: 1= mean, 2 = standard deviation
Bounded lognormal-n: 1= mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Truncated lognormal-n: 1= mean, 2 = standard deviation, 3 = lower quantile, 4 = upper quantile
Bounded normal: 1 = mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Beta: 1 = minimum, 2 = maximum, 3 = P-value, 4 = Q-value
Triangular: 1 = minimum, 2 = mode, 3 = maximum
Uniform: 1 = minimum, 2 = maximum

ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
SECTION 7, REVISION 2
UPDATE OF THE SITE-SPECIFIC DECOMMISSIONING COSTS

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LIST OF TABLES

LIST OF ACRONYMS AND ABBREVIATIONS

ComEd	Commonwealth Edison
Exelon	Exelon Corporation
FSS	Final Status Survey
FSAR	Final Safety Analysis Report
GTCC	Greater Than Class C
ISFSI	Independent Spent Fuel Storage Installation
LTP	License Termination Plan
NRC	Nuclear Regulatory Commission
PSDAR	Post-Shutdown Decommissioning Activities Report
TLG	TLG Services, Inc.
WCS	Waste Control Specialists
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

7. UPDATE OF THE SITE-SPECIFIC DECOMMISSIONING COSTS

7.1. Introduction

In accordance with 10 CFR 50.82(a)(9)(ii)(F) (Ref 7-1) and Regulatory Guide 1.179 (Ref 7-2), the updated site specific cost estimates and funding plans for the Zion Station Restoration Project (ZSRP) are provided. Regulatory Guide 1.179 provides guidance on the details of the information to be presented in the License Termination Plan (LTP).

The LTP must provide an estimate of the remaining decommissioning costs at the time of LTP submittal and also compare these estimated costs with the present funds set aside for decommissioning. If it is determined that there is a deficit in the present funding, the LTP must indicate the means for ensuring that adequate funds are available to complete the decommissioning.

The decommissioning cost estimate, at a minimum, needs to include an evaluation of the following cost elements:

- Cost assumptions used, including contingency factor;
- Major decommissioning activities and tasks;
- Unit cost factors;
- Estimated costs of decontamination and removal of equipment and structures;
- Estimated costs of waste disposal, including disposal site surcharges;
- Estimated Final Status Survey (FSS) costs; and
- Estimated total costs.

The cost estimate should focus on the remaining work, detailed activity by activity, including costs of labor, materials, equipment, energy, and services. The cost estimate should include the cost of the planned remediation activities as well as the cost of the transportation and disposal of the waste generated by the remedial work conducted.

Historical Perspective

By 1998, Exelon Corporation (Exelon), formerly Commonwealth Edison Company, or ComEd, had completely shut down the two unit Zion Nuclear Power Station (ZNPS) and made plans to implement a delayed-DECON decommissioning scenario, with decommissioning expected to commence at the original license expiration date (November 14, 2013). As part of its post shutdown planning, Exelon contracted a specialty decommissioning consultant, TLG Services, Inc. (TLG) to develop a decommissioning cost estimate for the ZNPS Units 1 and 2. This cost estimate was provided with the Post-Shutdown Decommissioning Activities Report (PSDAR) (Ref 7-3) Exelon submitted in 2000 to the NRC.

Exelon later entered into discussions with Energy*Solutions* for the possible transfer of the ZNPS licenses and the decommissioning fund to Energy*Solutions* to accelerate the decommissioning of the plant. As part of its application for the license transfers, Zion*Solutions*, LLC, a wholly owned subsidiary of Energy*Solutions*, amended the PSDAR and submitted it to the NRC in 2008. This amended PSDAR provided significant decommissioning cost milestone changes and an estimate of expected decommissioning costs.

The license transfers were completed and Zion*Solutions* started decommissioning operations by September 2010. In compliance with 10 CFR 50.75(f)(1) (Ref 7-4) and 10 CFR 50.82(a)(8)(v)-(viii), Zion*Solutions* continues to demonstrate financial assurance on an annual basis.

7.1.1. Cost Estimates Previously Docketed with the NRC

Exelon submitted its PSDAR to the NRC on February 14, 2000. As previously noted, in accordance with 10 CFR 50.82(a)(8)(iii), a Zion site-specific decommissioning cost estimate was also prepared and submitted in a letter dated February 14, 2000 (Ref 7-5). This submittal, docketed with the NRC, included an Attachment, "Zion Nuclear Power Station Units 1 and 2 Site-Specific Decommissioning Cost Estimate" which was the cost estimate study prepared by TLG.

During the process of Zion*Solutions*, LLC's application to take over the licenses of Zion Units 1 and 2 from Exelon, Zion*Solutions* submitted an Amended PSDAR, including an estimate of expected decommissioning costs, on March 18, 2008. This submittal was also docketed with the NRC.

7.2. Decommissioning Cost Estimate

The decommissioning cost estimate presented herein represents the cost to complete the remaining decommissioning work as of the end of the 3rd quarter 2014. This estimate was prepared based upon the schedule of the remaining work, incorporating the experience that has been gained while performing similar decommissioning tasks over the past four years. To a large extent, this decommissioning cost estimate is based upon an existing and operating decommissioning organization, in which actual contracts for services are already in place. As such, there is a high degree of certainty regarding expected work productivity, the cost of labor and the cost of services required to support the remainder of the project. The decommissioning cost estimate also includes application of contingency, as specific provision for unforeseeable elements of cost within the defined project scope. Contingencies are particularly important where previous experience has shown that unforeseeable events, which may increase costs, are likely to occur. The contingency, as used in this estimate, does not account for price escalation and inflation in the costs of decommissioning over the remaining project duration.

The cost estimate was prepared to include all costs associated with the decommissioning and unrestricted release of the Zion site other than the area bounded by the Independent Spent Fuel Storage Installation (ISFSI), and includes radiological decommissioning (i.e., those costs required to accomplish such unrestricted release), spent fuel management (transfer of the spent fuel to the ISFSI and operation of the ISFSI until the partial site release is achieved, at which time the released portion of the site and the remaining ISFSI will be transferred back to Exelon), and site restoration (i.e., non-radiological remediation aimed at leaving the site in a safe and stable condition). As was reflected in the Application relating to the transfer of the Zion licenses to Zion*Solutions*, Exelon has retained title to the spent fuel and Greater Than Class C (GTCC) waste, as well as the obligation for ultimate disposition of the spent fuel and the GTCC waste in the ISFSI and the decommissioning of the ISFSI.

The site-specific decommissioning cost estimate provided with this LTP has been prepared as a collaborative effort by Zion*Solutions* and TLG and presents a breakdown of the remaining costs

to complete the decommissioning process and release all portions of the site for unrestricted release, with the exception of the area bounded by the ISFSI.

The following subsections present a description of how the cost estimate was prepared and a summary and breakdown of the estimated costs.

7.2.1. Cost Estimate Description and Methodology

During the summer and fall of 2014, the Zion*Solutions* decommissioning project organization undertook an effort to update the baseline schedule, risks and the costs to complete the decommissioning project. This resulted in a revised work breakdown structure that provided a detailed listing of the remaining work activities and support services needed to complete the project. Task durations, crew compositions and material and contracted services needs were derived from the results of detailed process planning carried out by each of the decommissioning and support organizations (e.g., decommissioning operations, engineering, security, radiation protection, radiological engineering, waste management, safety, FSS, etc.).

Additionally, Zion*Solutions* performed a contingency and risk analysis so that the potential additional costs due to expected but undefined risks and uncertainties could be addressed and included in the cost estimate.

The resulting information was then compiled into a decommissioning cost estimate by TLG. The following sections provide a summary of those results.

7.2.2. Summary of the Site-Specific Decommissioning Cost Estimate

The overall remaining decommissioning cost (including scope risk contingency) was estimated to be $389 Million (in year of expenditure dollars), with a base estimated cost of $358 Million (without the scope risk contingency). The cost estimates include provisions for cost escalation based upon the following assumptions:

- Labor costs are assumed to escalate at 1.992% per year, this cost escalation factor being based on the forecast of the Consumer Price Index, Services, CUSASNS as published by Global Insight Company, and applied per the Zion project Asset Sale Agreement.
- Non-Labor costs that are not covered by fixed prices, fixed rates or escalation provisions in contractual agreements, are similarly assumed to escalate at 1.992% per year, this cost escalation factor being based on the Consumer Price Index, Services, CUSASNS as published by Global Insight Company, and applied per the Zion project Asset Sale Agreement.
- For Class A and Class B&C waste costs, Zion*Solutions* has largely mitigated this escalation risk by having a fixed price arrangement with Energy*Solutions* (Class A) and contractually defined costs for B/C waste.

The cost estimate includes the costs for radiological decommissioning, spent fuel management, and site restoration. A breakout of the cost for each part of the decommissioning program is provided in Table 7-1.

Table 7-1 Cost for Radiological Decommissioning, Spent Fuel Management, and Site Restoration

	Radiological Decommissioning	Spent Fuel Management*	Site Restoration*
Base Amount	$284.3 Million	$37.4 Million	$36.2 Million
Contingency	$24.7 Million	$3.3 Million	$3.2 Million
Total	**$309.0 Million**	**$40.7 Million***	**$39.4 Million***

*included for completeness, but not required for license termination funding purposes.

Detailed breakdowns of the estimated costs for radiological decommissioning, spent fuel management and site restoration programs are provided in sections 7.2.3, 7.2.4 and 7.2.5, respectively. Section 7.2.6 presents the estimated contingency costs for each of these programs.

7.2.3. Radiological Decommissioning Costs

Consistent with the NRC definition of decommissioning under 10 CFR 50.2, the radiological decommissioning costs under this category consider only those costs associated with normal decommissioning activities necessary for release of the site (other than the ISFSI) for unrestricted use. It does not include costs associated with the disposal of non-radiological materials or structures beyond those necessary to terminate the Part 50 license or the costs associated with construction or operation of an ISFSI.

As summarized in section 7.2.2 above, the total estimated cost for radiological decommissioning, including contingency is $309 Million. The estimated cost for the anticipated base work scope is $284.3 Million. Application of a contingency of $24.7 Million results in a total estimated cost of $309 Million.

The remaining decommissioning scope of work included in this estimate is described in detail in other chapters of this LTP. Overall, that work scope includes completion of the removal, transportation and disposal of the major components; completion of the removal, transportation and disposal of the remaining equipment; decontamination and/or bulk demolition of radiological impacted structures and transportation and disposal of the resulting radioactive wastes; performance of the FSS and associated license termination activities. The estimated costs include the labor, equipment, materials, services and fees needed to conduct the work. The estimated cost also includes all of the program support activities and services necessary to manage and safely carry out a large scale dismantlement and demolition project. These program support activities include project management, work controls and site administration; technical support services, such as radiation protection, safety, engineering, security, QA/QC, environmental monitoring, waste management and decommissioning subject matter experts needed to support the project.

A high level breakdown of the estimated base radiological decommissioning cost, by major resource category, is provided in Table 7-2.

Table 7-2 Estimated Base Radiological Decommissioning Cost by Resource Category

Labor	[redacted]	$119.8 Million (b)
Equipment, Materials and Supplies		$24.9 Million
Fixed- Price Contracts, Services & Fees		$55.5 Million
Radioactive Waste Packaging, Transportation & Disposal		$84.1 Million
Total (c)		**$284.3 Million**

[redacted]
(b) Includes contracted specialty labor costs
(c) Columns may not add due to rounding

A high level breakdown of the estimated radiological decommissioning cost, alternatively by major project activity, is provided in Table 7-3.

Table 7-3 Estimated Radiological Decommissioning Cost by Major Project Activity

Major Component Removal	$30.8 Million
Equipment and Structure Decontamination / Removal	$63.8 Million
Waste Disposition	$84.1 Million
Program Management and Support Services (excluding Final Status Survey and License Termination Activities)	$75.7 Million
Final Status Survey and License Termination Activities	$8.0 Million
Other Lump-Sum Costs (e.g., regulatory fees, financing)	$21.9 Million
Total (a)	**$284.3 Million**

(a) Columns may not add due to rounding

A more detailed breakdown of the costs by resource requirements (e.g., labor, materials, services, etc.) and by decommissioning activity (e.g., component removal, structural decontamination, program support functions, waste management functions, etc.) are provided in Tables 7-6 and 7-7 respectively.

The total estimated cost for radioactive waste disposition (containers, transportation and disposal) is $84.1 Million. As presented in Table 7-7, these waste management costs are comprised of four distinct categories; Class A Large Components, Class B/C Waste, Class A Containerized Wastes and Class A Bulk Materials. Costs for on-site handling of GTCC waste (i.e., reactor vessel internals) are included in the "Major Component Removal" category shown

on Table 7-7. However, no costs for disposal of this waste is included in the estimate, as it is assumed that disposal of this waste will be included as a part of spent fuel disposition.

The project has in place a unique contracting approach for disposal of the resulting radioactive wastes that eliminates much of the cost uncertainty and waste volume estimation risk that is often associated with decommissioning projects. As such, the reported waste management costs are unlikely to vary due to waste volume uncertainties. The resulting radioactive waste streams and the disposal and transportation contracts that are in place can be categorized by the following:

7.2.3.1. <u>Class A Large Components</u>

This category of waste includes equipment that will be transported and disposed of intact, enclosed in rail cars or prepared to serve as its own waste container. These items have been radiologically and physically characterized. As such, the inventory of these items and their disposal volumes are known. The associated waste management costs are covered by existing fixed-price contracts with Energy*Solutions*. Therefore, the waste management costs for these items are well known and not likely to vary. [redacted]

7.2.3.2. <u>Class A Bulk Materials</u>

This category of waste primarily consists of concrete rubble or similar materials contaminated with very low levels of radioactivity (and large components described above). This material will be transported in covered gondola rail cars to the Energy*Solutions* disposal site in Clive, Utah. The cost for disposal and transportation of this material is covered by a fixed-price contract that covers any and all material of this type from this decommissioning project, without regard to the total mass or volume. Therefore, these costs are known and are unlikely to vary. This category of waste comprises > 95% of the total volume and mass and > 80% of the estimated waste management costs for all radioactive waste expected to be generated by this decommissioning effort. [redacted]

7.2.3.3. <u>Class A Containerized Wastes</u>

This category of waste primarily consists of material that will need to be packaged in strong-tight / Industrial containers, such as intermodals or steel boxes. Typically, this would include small pieces of contaminated equipment, pipe or debris which require containerization to meet DOT regulations or mitigate radiological handling concerns. This material will be transported by rail, for disposal at the Energy*Solutions* disposal site in Clive, Utah. [redacted]

7.2.3.4. <u>Class B/C Waste</u>

This category of waste is primarily composed of segmented pieces of the activated reactor internals and, to a much lesser extent, higher radioactivity level resins, filters, sludge and cutting fines. These materials will require packing in liners or high integrity containers, and transported

in shielded licensed transportation casks by truck to the Waste Control Specialists (WCS) facility in Andrews, Texas. The volume (or mass) of this waste material is well known, characterized, and will be generated from a limited set of reactor components. [redacted] Disposal cost variability for this category of waste has been largely mitigated by established contractual terms in place with WCS.

7.2.4. Spent Fuel Management Costs

Zion*Solutions* acknowledges that the costs to construct and operate an ISFSI (previously defined) and other spent fuel related management costs are not considered by the NRC staff as part of decommissioning costs. Nevertheless, as there is significant interest by many stakeholders in these costs, they are presented herein. As presented in Section 7.2.2 above, the estimated cost for the anticipated base work scope is $37.4 Million. A contingency of $3.3 Million was applied resulting in total spent fuel management costs of $40.7 Million.

Overall, the spent fuel management work scope includes transfer of the remaining spent fuel to the ISFSI and operation of the ISFSI until termination of the reactor license, with the exception of the area bounded by the ISFSI, projected to take place in 2019.

Construction of the ISFSI was completed in April 2013 and spent fuel transfer operations were started by December 2013 with the first spent fuel cask being placed on the ISFSI in early January 2014. As of the end of September 2014, approximately 64% of the spent fuel has been transferred to the ISFSI, contained in 39 dry storage casks on the ISFSI pad. Note that spent fuel transfer was completed on January 10, 2015.

The estimated costs include the labor, equipment, materials, services, fees, and program support activities necessary to safely manage the spent nuclear fuel. ISFSI operational costs are estimated through mid-year 2019, when partial site release and the transfer of the site and ISFSI back to Exelon are expected, and subsequent management of the spent fuel is consistent with the Irradiated Fuel Management Plan for Zion under 10 CFR 50.54 (bb) (Ref 7-6). Exelon has provided a decommissioning funding plan to the NRC for the Zion ISFSI (Ref 7-7).

A high level breakdown of the estimated base spent fuel management cost, by major resource category, is provided in Table 7-4.

Table 7-4 Estimated Base Spent Fuel Management Cost by Major Resource

Labor	[redacted]	$29.9 Million (b)
Equipment, Materials and Supplies		$1.3 Million
Fixed- Price Contracts, Services & Fees		$6.2 Million
Radioactive Waste Packaging, Transportation & Disposal		$0
Total (c)		**$37.4 Million**

[redacted]
(b) Includes contracted specialty labor costs
(c) Columns may not add due to rounding

A more detailed breakdown of the cost by resource requirements (e.g., labor, materials, services, etc.) is provided in Table 7.8.

7.2.5. Site Restoration Costs

Zion*Solutions* acknowledges that the costs to restore the Zion Plant property are not considered by the NRC staff as part of decommissioning costs. Nevertheless, there is significant interest by many stakeholders in these costs and they are presented herein. The estimated cost for the anticipated work scope is $36.2 Million. A contingency of $3.2 Million was estimated, bringing the total to $39.4 Million. Overall, that work scope includes removal of any remaining hazardous materials, demolition of remaining structures, backfilling of any open excavations or void spaces, and final grading and stabilization against erosion.

The estimated costs include the labor, equipment, materials, services and fees needed to conduct the work. In general, most of this work is anticipated to be performed by contractors; however the estimated cost also includes all of the program support activities and services necessary to manage and safely carry out project.

A high level breakdown of the estimated site restoration cost, by major resource category, is provided in Table 7-5.

Table 7-5 Estimated Site Restoration Cost by Major Resource Category

Labor	[redacted]	$58.8 Million (b)
Equipment, Materials and Supplies		$0.71 Million
Fixed- Price Contracts, Services & Fees		$29.7 Million
Radioactive Waste Packaging, Transportation & Disposal		$0
Total (c)		$36.2 Million

(a[redacted]
(b) Includes contracted specialty labor costs
(c) Columns may not add due to rounding

A more detailed breakdown of the cost by resource requirements (e.g., labor, materials, services, etc.) is provided in Table 7.8.

7.2.6. Contingency

Uncertainty associated with the decommissioning cost estimate, and the need to allocate additional funding to cover contingency for this project has been included in this estimate. Accounting for contingency has been evaluated from two standpoints, operational efficiency and scope expansion risk. Within the context of this cost estimate, operational efficiency contingency is defined as the occurrence of events or circumstances that can prolong project duration or make the execution of a given work scope more difficult. Examples of these types of events include weather related delays, equipment or tool breakage or unavailability, and interferences from other work activities. Scope expansion risk within the context of this estimate is defined as the need to perform unplanned work activities or expansion of the work activities that were planned. Examples of this type of project risk would be discovering new or additional contaminated media requiring remediation, or a need to perform work in a different manner due to unforeseen conditions or changes in requirements.

As was initially shown in section 7.2.2, contingency was estimated at $31.1 Million; apportioned as $24.7 Million for radiological decommissioning, $3.3 Million for spent fuel management and $3.2 Million for site restoration. This contingency was estimated using a quantitative Monte Carlo type probability analysis, with the $31.1 Million amount corresponding to the resulting 85 percent confidence level amount.

7.3. Decommissioning Funding Plan

As indicated in section 7.2, the estimated cost to complete the radiological decommissioning of the Zion nuclear station, including contingency, is $309 Million (year of expenditure dollars) as of Sept 30, 2014. These decommissioning costs will be paid for with funds from the station's nuclear decommissioning trust fund (NDT). Discounting those escalated costs at the rate of cost inflation described in section 7.2.2 yields a cost of radiological decommissioning at constant 2014 dollars of [redacted].

The actual cash balance of the NDT, as recorded by the Zion*Solutions* trustee as of Sept 30, 2014, was [redacted]. Recognizing that there were project costs incurred and recorded on Sept 30, 2014 that had not been paid for from the NDT (outstanding disbursements), plus other transactions in the last quarter of 2014 that have a bearing on these outstanding disbursements, the net balance of the NDT available to cover the future costs of radiological decommissioning was $317.1 Million.

Based on a time phased cash flow analysis of the radiological decommissioning costs, and assuming NDT returns at an annual 2% real, after tax rate of return, the required minimum funding assurance amount to fund the future radiological decommissioning costs equals $302.6 Million, which is below the $317.1 Million available balance described above.

This NDT position, together with Energy*Solutions* resources and the $200 Million Letter of Credit backup for the NDT agreed with Exelon in the Zion Nuclear Power Station Unit 1 and 2 Asset Sale Agreement, that are available but are not relied upon here, provides for sufficient funding and financial assurance for completion of radiological decommissioning of the Zion Project.

On or before March 31, 2015, as required by 10 CFR 50.75(f)(1) and 10 CFR 50.82(a)(8)(v)-(viii), Zion*Solutions* will be submitting the annual demonstration of financial assurance for the year ending Dec 31, 2014. That submission will be based upon future project costs of radiological decommissioning and the NDT balance as of that date.

7.4. References

7-1 Code of Federal Regulations, Title 10, Part 50.82, "Termination of License"

7-2 US Nuclear Regulatory Commission Regulatory Guide 1.179, Revision 1, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors", June 2011

7-3 Letter from Zion*Solutions* ,LLC to the U.S. Nuclear Regulatory Commission, "Notification of Amended Post-Shutdown Decommissioning Activities Report (PSDAR) for Zion Nuclear Power Station, Units 1 and 2 in Accordance with 10 CFR 50.82(a)(7)", dated March 18, 2008

7-4 Code of Federal Regulations, Title 10, Part 50.75, "Reporting and Recordkeeping for Decommissioning Planning"

7-5 Letter from Commonwealth Edison to the U.S. Nuclear Regulatory Commission, "Submittal of the Zion Nuclear Power Station Site-Specific Decommissioning Cost Estimate", dated February 14, 2000

7-6 Letter from Commonwealth Edison to the U.S. Nuclear Regulatory Commission, "Submittal of the Zion Nuclear Power Station Irradiated Fuel Management Plan", dated February 14, 2000

7-7 Letter from Exelon Generation to the U.S. Nuclear Regulatory Commission, "Proposed Independent Spent Fuel Storage Installation (ISFSI) Decommissioning Funding Plan for Zion", dated October 17, 2013

Table 7-6 Detailed Breakdown of Radiological Decommissioning Costs by Resource Requirement

Labor:	**TOTAL** [redacted]	**$119.8 Million**
	Craft [redacted]	$38.6 Million
	Technician [redacted]	$14.3 Million
	Professional (Sci. & Eng.) [redacted]	$36.8 Million
	Management [redacted]	$18.1 Million
	Other - contract service labor [redacted]	$11.9 Million
Equipment & Materials:	**TOTAL**	**$24.9 Million**
	Durable Equipment	$5.3 Million
	Consumable Supplies	$16.0 Million
	Utilities and Energy	$3.6 Million
Contracts, Services & Fees:	**TOTAL**	**$55.5 Million**
	Equipment Rental	<$0.1 Million
	Contracted Services	$27.9 Million
	Laboratory & Analytical Services	$1.8 Million
	Travel & Living	$1.5 Million
	Insurance and Finance Fees	$20.1 Million
	Licensee Fees & Permits	$4.2 Million
Radioactive Waste Packaging, Transportation & Disposal:	**TOTAL**	**$84.1 Million**
	Class A Waste	[redacted]
	Class B/C Waste	[redacted]
TOTAL		**$284.3 Million**

Columns may not add due to rounding

Table 7-7 Detailed Breakdown of Radiological Decommissioning Costs by Decommissioning Activity

Major Component Removal	**TOTAL**	**$30.8 Million**
	Reactor Vessels and Internals	$21.1 Million
	Steam Generators	$9.0 Million
	Pressurizers	$0.7 Million
SSC Removal and Decontamination	**TOTAL**	**$27.2 Million**
	Equipment Removal / Structural Decontamination	$18.4 Million
	Bulk Structural Material Removal	$7.9 Million
	In-process Characterization and Remedial Action Support	$0.9 Million
Waste Management	**TOTAL**	**$84.1 Million**
	Class B/C Waste: Packaging, Transportation and Disposal Surveys and Sampling	[redacted]
	Class A Waste: Rail Car Preparation for Large Components	[redacted]
	Class A Bulk Waste: Rail Car Transportation and Disposal	[redacted]
	Class A Packaged Waste: Containers, Transportation and Disposal	[redacted]
Program Management and Support Services	**TOTAL**	**$120.3 Million**
	Program and Project Management and Site Administration	$32.1 Million
	Technical Services and Services- (e.g., Engineering, Rad. Protection, Environmental Monitoring, Site Characterization, Waste Mgmt, QA/QC, Safety, Worker Qualifications)	$47.9 Million
	Security	$7.7 Million
	Site O&M	$4.9 Million
	Special Projects (Cold & Dark, Bld. Mods.)	$8.5 Million
	Equipment, Materials, Consumable Supplies and Utilities	$11.2 Million
	FSS, LT and Material Release Program	$8.0 Million
Other Lump-Sum Costs	**TOTAL**	**$21.9 Million**
	Financing	$9.8 Million
	Regulatory Fees	$12.1 Million
TOTAL		**$284.3 Million**

Columns may not add due to rounding

Table 7-8 Detailed Breakdown of Spent Fuel Management Costs by Resource Requirement

Labor:	**TOTAL** [redacted]	**$29.9 Million**
	Craft [redacted]	$3.7 Million
	Technician [redacted]	$1.5 Million
	Professional (Sci. & Eng.) [redacted]	$3.7 Million
	Management [redacted]	$1.8 Million
	Other (contract service labor, primarily security - exclusive of management) [redacted]	$19.2 Million
Equipment & Materials:	**TOTAL**	**$1.7 Million**
	Durable Equipment	<$0.1 Million
	Consumable Supplies	$1.2 Million
	Utilities and Energy	$0.4 Million
Contracts, Services & Fees:	**TOTAL**	**$5.8 Million**
	Equipment Related Services	$1.3 Million
	Contracted Services (excluding security provided in labor above)	$2.2 Million
	Laboratory & Analytical Services	<$0.1 Million
	Travel & Living	<$ 0.1 Million
	Insurance, Finance, Licensee and Permit fees	$2.3 Million
Radioactive Waste Packaging, Transportation & Disposal:	**TOTAL**	**$0**
	Class A Waste	$0
	Class B/C Waste	$0
TOTAL		**$37.4 Million**

Columns may not add due to rounding

Table 7-9 Detailed Breakdown of Site Restoration Costs by Resource Requirement

Labor:	**TOTAL** [redacted]	**$5.8 Million**
	Craft [redacted]	$2.2 Million
	Technician [redacted]	$0.16 Million
	Management and Professional (Sci. & Eng.) [redacted]	$1.9 Million
	Other- contract service labor [redacted]	$1.6 Million
Equipment & Materials:	**TOTAL**	**$1.2 Million**
	Durable Equipment	<$0.1 Million
	Consumable Supplies	$0.7 Million
	Utilities and Energy	$0.5 Million
Contracts, Services & Fees:	**TOTAL**	**$29.2 Million**
	Equipment Rental	<$0.1 Million
	Contracted Services	$27.6 Million
	Laboratory & Analytical Services	<$0.1 Million
	Travel & Living	<$0.1 Million
	Insurance, Finance, Licensee & Permit fees,	$1.6 Million
Radioactive Waste Packaging, Transportation & Disposal:	**TOTAL**	**$0**
	Class A Waste	$0
	Class B/C Waste	$0
TOTAL		**$36.2 Million**

Columns may not add due to rounding

ZION STATION RESTORATION PROJECT
LICENSE TERMINATION PLAN
CHAPTER 8, REVISION 2
SUPPLEMENT TO THE ENVIRONMENTAL REPORT

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LIST OF ACRONYMS AND ABBREVIATIONS

ACS	American Community Survey
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
AMSL	Above Mean Sea Level
BMP	Best Management Practices
ComEd	Commonwealth Edison
CCDD	Clean Construction Demolition Debris
DOE	Department of Energy
DOT	Department of Transportation
DSAR	Defueled Safety Analysis Report
EPA	Environmental Protection Agency
ES	Environmental Statement
FSAR	Final Safety Analysis Report
GEIS	Generic Environmental Impact Statement
GTCC	Greater-Than- Class- C
HASP	Health and Safety Plan
HSA	Historical Site Assessment
ICMP	Illinois Coastal Management Program
IDNR	Department of Natural Resources
IEPA	Illinois Environmental Protection Agency
IRSF	Interim Radioactive Storage Facility
ISFSI	Independent Spent Fuel Storage Installation
LTP	License Termination Plan
NOI	Notice of Intent
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSSD	North Shore Sanitary District
ODCM	Off-site Dose Calculation Manual
OSHA	Occupational Health and Safety Administration
PCB	Polychlorinated Biphenyls
PDSAR	Post Shutdown Decommissioning Activity Report
PWR	Pressurized Water Reactors
RCRA	Resource Conservation and Recovery Act
REMP	Radiological Environmental Monitoring Program
RGPP	Radiological Groundwater Protection Program
SMC	Storm Water Management Commission

SWPPP	Storm Water Pollution Prevention Plan
TEDE	Total Effective Dose Equivalent
VCC	Vertical Concrete Cask
WCS	Waste Control Specialist
WDO	Watershed Development Ordinance
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

8. SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1. Introduction

In accordance with the requirements of 10 CFR 50.82 (a)(9)(ii)(A) and the guidance of Regulatory Guide 1.179, *"Standard Format and Contents for License Termination Plans for Nuclear Power Reactors"* (Reference 8-1), this chapter provides a supplement to the environmental report describing any new information or significant environmental change associated with the site-specific decommissioning and site closure activities performed at the Zion Nuclear Power Station (ZNPS) site.

8.1.1. Purpose

This chapter supplements the Commonwealth Edison Company, "*Environmental Report - Zion Nuclear Power Station*" as supplemented (Reference 8-2), describing any new information or significant environmental changes associated with the site-specific decommissioning and license termination activities presented in this License Termination Plan (LTP). The supplement includes a detailed description of the remaining decommissioning and site closure activities, the interaction between those activities and the environment, and the likely environmental impact of those activities. The supplement discusses whether the activities and their impacts are bounded by the impacts predicted by the United States Atomic Energy Commission (AEC) "*Final Environmental Statement related to operation of Zion Nuclear Power Station Units 1 and 2*", - December 1972 (AEC Environmental Statement) (Reference 8-3) issued in December 1972; NUREG-0586, Supplement 1, Volume 1 "*Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities*" (Reference 8-4); and the Zion Nuclear Power Station, "*Post Shutdown Decommissioning Activity Report*" (PSDAR) (Reference 8-5). This chapter discusses decommissioning activities, with a focus on those activities to be performed from time of submittal of this LTP until the license transfer back to Exelon.

8.1.2. Background

ZNPS is comprised of two 1,100-Mwe Pressurized Water Reactors (PWR), Units 1 and Unit 2, with supporting facilities, which was owned and operated by Commonwealth Edison Company (now Exelon) from 1973 to 1998.

The station was granted a construction permit by the AEC in December 1968 for both Units. Commercial operation was achieved in 1973 for Unit 1 and 1974 for Unit 2. Due to a variety of factors, including economic analysis associated with proposed steam generator replacements, Commonwealth Edison made the decision to shut down ZNPS. Permanent cessation of operations at ZNPS occurred on February 13, 1998. Certification of Permanent Defueled Status for both Units was achieved in March 09, 1998.

In accordance with the requirements of 10 CFR 50.82, Commonwealth Edison Company (now Exelon) submitted the initial revision of the PSDAR to the Nuclear Regulatory Commission (NRC) on February 14, 2000. The reactors at ZNPS remained in a SAFSTOR condition until September of 2010. At this point, the license for the facility was transferred from Exelon (the licensee at that time) to Zion*Solutions* LLC. This was accomplished to allow Zion*Solutions* to begin the process of the physical decommissioning of the ZNPS. Integral to the license transfer,

the PSDAR was amended on March 18, 2008 to address the acceleration of decommissioning activities, changes to the decommissioning schedule and cost milestones. The amended PSDAR established the DECON method as the current decommissioning approach and described the accelerated decommissioning schedule with a lower revised cost estimate to reflect current knowledge and waste disposal options.

The environmental impacts of decommissioning operations at ZNPS were previously assessed in both revision of the PSDAR. The assessments included the evaluation of impacts against those noted in the AEC Environment Statement and NUREG-0586. The reference facility in NUREG-0586 is a 1,175-MWe PWR owned by Portland General Electric and designed by Westinghouse. As the Zion PWRs are similar in size and also designed by Westinghouse, the two ZNPS units fall within the envelope of the generic environmental assessment.

The amended PSDAR concluded that the decommissioning of the ZNPS would be accomplished with no significant adverse environmental impacts and that the environmental impacts associated with the site-specific decommissioning activities for ZNPS would be bounded by previously issued environmental impact statements.

8.2. Site Location and Description

ZNPS is located in northeastern Illinois on the west shore of Lake Michigan, about 40 miles north of Chicago, Illinois and about 42 miles south of Milwaukee, Wisconsin. The site is located in the eastern portion of the City of Zion in Lake County, Illinois, about 3.2 miles south of the Illinois-Wisconsin State line. See Figure 8-1 for a map showing the site location, including nearby prominent features such as highways, rivers and lakes. The map coordinates for ZNPS are longitude 87 degrees, 48.1 minutes West and latitude 42 degrees, 26.8 minutes North.

The site comprises approximately 331 acres which is owned and controlled by Zion*Solutions*, LLC under a lease agreement with Exelon Generation, Inc. integral to the *"Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement"* (Reference 8-6). The site is accessed by Shiloh Boulevard which enters the site on the north side. The site is bordered on the west by an industrial area located immediately east of the Chicago and Northwestern railway. The site is bordered on the north by the City of Zion Hosah Park, and further north as well as south by the Illinois State Beach Park along the Lake Michigan shoreline. Figure 8-2 is an aerial view of the local area showing the site boundaries. Figure 8-3 presents a topographic map of the site with contour intervals; the site grade is at 591 feet Above Mean Sea Level (AMSL).

8.2.1. Site Description After Unrestricted Release

This section provides a summary of the final condition of the site at the conclusion of decommissioning and site closure activities. The "End State" is defined as the configuration of the remaining below ground buildings, structures, piping and open land areas at the time of license termination.

Section 8.5 of Exhibit C, Lease Agreement, titled "Removal of Improvements; Site Restoration" integral to the Asset Sale Agreement requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least three feet below grade (designated as an elevation of 588 foot AMSL. All systems, components, piping, buildings and structures above 588 foot elevation will be removed during decommissioning and disposed of as waste. The

demolition debris will be segregated for recycling, reuse, or disposal. The decommissioning approach also calls for the beneficial reuse of concrete from building demolition as clean fill. Only concrete that meets the definition of Clean Construction and Demolition Debris (CCDD) and, where radiological surveys demonstrate that the concrete is free of plant derived radionuclides above background will be considered for use as fill.

In both Containment basements (Unit 1 and Unit 2), all concrete will be removed from the inside of the steel liner, leaving only the remaining exposed liner below the 588 foot elevation and the structural concrete outside of the liner. In the Auxiliary and Turbine building basements, all internal walls and floors will be removed, leaving only the reinforced concrete floors and outer walls of the building structures. For the Fuel Handling Building, the only portion of the structure that will remain is the lower 12 feet of the fuel pool below the 588 foot elevation and the concrete structure of the Fuel Transfer Canals after the steel liner has been removed. There are four additional below ground structures that will remain including the lower concrete portions of the Waste Water Treatment Facility (WWTF), Main Steam Tunnels, Circulating Water Inlet Piping and Circulating Water Discharge Tunnels.

An evaluation was performed regarding the disposition of the Intake/Discharge structures. The alternative to leave in place was determined to be the least disruptive to the environment and was recommended. The impact of leaving the intake and discharge piping in place is discussed in an AMEC, Inc. report titled *"Final Environmental Analysis of Alternatives Regarding Intake/Discharge Structure Disposition at the Former Zion Nuclear Generating Station, Zion, Illinois"* (Reference 8-7).

Remaining below grade structures such as basement foundations will be filled with clean concrete debris, soil, sand or other suitable media. The end-state will also include a range of buried, embedded piping and penetrations. All buried piping that is abandoned in place will be capped and/or filled with grout. The restored areas on the site will be back-filled, graded and returned to natural contours. Several structures will remain at license transfer as requested by Exelon. These structures are as follows;

- North Access Control Security Gate
- Owner Controlled Fence Line
- Commonwealth Edison Electrical Switchyard (note: the Switchyard will remain in active use after decommissioning in support of the existing Commonwealth Edison offsite electrical transmission and distribution system) including the microwave tower
- Sanitary sewage system Lift Station (note: the Lift Station is required to remain to support the Independent Spent Fuel Storage Installation [ISFSI] Monitoring Building).
- Paved roadways and rail lines, including the lines and rail spur constructed in 2011, allowing for rail service at the site via connection to the nearby Union Pacific railway

After all demolition and remediation activities are compete, Zion*Solutions* will use the final survey process described in Chapter 5 of this LTP to demonstrate that the ZNPS and surrounding open land areas, with the exception of the ISFSI facility, comply with radiological criteria for unrestricted use specified in 10 CFR 20.1402. As part of the decommissioning process, all reactor fuel and greater than Class C waste will be loaded into casks and transferred to the ISFSI.

It is expected that the fuel will remain on-site in dry storage within the ISFSI until it is transferred to the Department of Energy (DOE). The ISFSI, which occupies approximately 5 acres, has been constructed in the southwest corner of the ZNPS site, immediately south of the Switchyard.

Following the conclusion of radiological remediation activities and prior to initiating final survey, isolation and control measures will be implemented. The control measures will be implemented to ensure the final radiological condition is not compromised by the potential for re-contamination as result of access by personnel or equipment. Open land areas, access roads and boundaries will be posted with signs restricting access. Isolation and control measures will be implemented through approved plant procedures and will remain in force throughout final survey activities and until there is no risk of recontamination from decommissioning or the survey area has been released from the license.

Several services, such as the City of Zion water and sanitary sewer services and Commonwealth Edison electrical service will remain in operation to support the ISFSI monitoring and security operations. There are no potable wells on site. Water service will be provide through the City of Zion municipal water supply which draws water from Lake Michigan via a water intake about one mile north of the site.

8.3. Remaining Dismantlement and Decommissioning Activities

Key dismantlement and decommissioning activities that have been completed include: activities associated with the removal of system piping and components; the segmentation and packaging of the internals from both Unit 1 and Unit 2 reactors; the on-going transfer of spent nuclear fuel from the Fuel Handling Building to the ISFSI and the demolition and disposal of several ancillary structures, including the Interim Radioactive Storage Facility (IRSF) and the storage tank farms located east of the Turbine Building.

Chapter 3 of this LTP provides details on the dismantlement, demolition and remediation activities currently performed and remaining activities to be executed to achieve the End State condition.

8.4. Impacts to the Post-Shutdown Decommissioning Activities Report (PSDAR)

The PSDAR, amended in March 2008, described the planned decommissioning operations at the site and concluded that the potential environmental impacts associated with decommissioning the site have already been postulated in, and will be bounded by the previously issued environmental impact statements, specifically:

- Final Environmental Statement,
- NUREG-0586,
- NUREG-1496 *"Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities"* (Reference 8-8), dated July 1997.

Chapter 3 of this LTP identified the dismantlement and decontamination activities that are scheduled to be completed prior to unrestricted release of the site (excluding the ISFSI footprint)

and the transfer of the license back to Exelon. These identified activities are not significantly different than what was proposed in the PSDAR. Although additional details regarding major decommissioning activities will be defined during ongoing decommissioning planning efforts, no significant impacts beyond those identified in the PSDAR have been identified. Subsequent sections in this chapter provide additional evaluation and information regarding the environmental effects of decommissioning.

8.5. Zion Station Environment Description

8.5.1. Geography and Demography

8.5.1.1. Site Location and Description

The site location and description were previously discussed in Section 8.2. In addition, the site occupies portions of Sections 22, 23, 26, and 27 in Township 46 North, Range 23 East. The site is located on a narrow strip of lake deposits which borders the Lake Michigan shoreline. A series of low, parallel beach ridges, oriented north-south, separated by marshy depressions, cross the site. The topography at the site ranges from approximately 580 feet to 605 feet AMSL and represents recessional beach lines deposited along the Lake Michigan shoreline subsequent to the most recent period of glaciations. The beach ridges are composed primarily of sand.

8.5.1.2. Population

The U.S. Census Bureau American Community Survey (ACS) latest 5-year report (Reference 8-9) provides the most reliable census data for the City of Zion, nearby cities, and Lake County. The estimated total population in 2012 was: 24,400 for the City of Zion; 6,752 for Winthrop Harbor, located 3 miles to the north of Zion; 88,982 for Waukegan, located 7 miles south of Zion; and 701,282 for Lake County.

8.5.1.3. Site Access, Land, and Water Use

The ZNPS property is located in the extreme eastern portion of the City of Zion in Lake County, Illinois on the west shore of Lake Michigan. Although the site encompasses approximately 331 acres, it is relatively isolated as the property is bordered to the north and south by Illinois Beach State Park, a small industrial area followed by railroad tracks to the west and Lake Michigan to the east. The center of the community of Zion is approximately 1.6 miles from the plant location on the site. There are no schools or hospitals within one mile of the site and there are no residences within 2,000 feet of any ZNPS structures (Zion *"Historical Site Assessment"* (HSA) [Reference 8-10]).

The area of Lake Michigan adjacent to ZNPS is used by recreational boaters. The nearest marina/public boat launch is located approximately 2.5 miles north of the site. There are also several fishing charter services in Winthrop Harbor that are located approximately 3 miles north of the site. Lake Michigan is also used for commercial barge and ship traffic, however this traffic does not ordinarily operate within 5 miles of the site.

8.5.2. Climate

Zion's climate is continental with cold winters, warm summers, and frequent short fluctuations in temperature, humidity, cloudiness, and wind direction. The average temperature in the summer is 72 degrees F and the average temperature in the winter is 24 degrees F. Because the eastern edge of Zion is bounded by Lake Michigan, inland lake breezes can cool the air along the lake shore by 10 to 15 degrees F in the summer and can warm the air by as much as 20 degrees F in the winter (Conestoga-Rovers and Associates, *"Hydrogeologic Investigation Report, Fleetwide Assessment, Zion Station"*, Revision 1 [Reference 8-11]).

The average annual rainfall is 32.0 inches and the average annual snowfall is 41.0 inches (Illinois Department of Commerce and Economic Opportunity, *"Zion Illinois, General Information, Climate"* [Reference 8-12]). Winter storms, with snowfalls exceeding 6 inches, occur about once every 2 years in the northeastern part of Illinois (*"Illinois State Water Survey (1971-2000)"* [Reference 8-13]).

Wind speeds in the northeastern portion of Illinois, encompassing Zion, typically average 8 to 10 mph (*"Illinois Climate Network, 1991-2000 Data Set"* [Reference 8-14]). High winds (on the order of 70 mph) can be expected once in 50 years from storms (Commonwealth Edison Company, *"Zion Station Defueled Safety Analysis Report"* [DSAR] [Reference 8-15]). The Zion area has rarely experienced tornados. On September 28, 1972, a category F4 tornado 5.8 miles away from the Zion city center injured 20 people and caused between $500,000 and $5,000,000 in damages. On April 21, 1967, a category F4 tornado 20.9 miles away from the city center killed one person and injured 100 people and caused between $500,000 and $5,000,000 in damages (*"City of Zion Illinois Data"* [Reference 8-16]).

8.5.3. Geology and Seismology

The near-surface geology of northeastern Illinois is comprised of unconsolidated glacial deposits which range from 90 to 150 feet in thickness. The surface deposits overlay a series of sedimentary rock layers deposited in the Paleozoic Era. The thickness of the Paleozoic sedimentary rocks in northeastern Illinois is approximately 4,000 feet. These sedimentary bedrock layers dip gently toward the east at an incline of approximately 10 feet per mile and overlay on the Precambrian basement rock (Commonwealth Edison Company *"Zion Nuclear Power Station Final Safety Analysis Report"* (FSAR) [Reference 8-17]).

At ZNPS, in the vicinity of the major buildings, the surface deposits are comprised of three layers, or units, of irregular thickness. In descending order, the following overburden stratigraphic units have been identified and characterized during the various site investigations, Zion*Solutions* TSD 14-003, Conestoga Rovers & Associates (CRA) Report, *"Zion Hydrogeologic Investigation Report"* (Reference 8-18):

- Upper sand unit (also known as the Shallow Aquifer): 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). This unit includes both native and fill sand. Depth ranges from the ground surface to an elevation of approximately 555 feet AMSL.

- Upper silty clay unit: Hard silt, silty clay, clayey silt, and sandy silt which contain some sand and gravel and occasional cobbles and boulders. Depth ranges from approximately 525 feet to 555 feet AMSL.
- Lower sand unit: 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. Depth ranges from approximately 480 feet to 525 feet AMSL. This unit is discontinuous.

The lower unconsolidated sand unit layer overlies an upper bedrock layer. This upper bedrock layer is the Niagara Dolomite, a consolidated layer of carbonaceous marine sediments laid down in the Silurian Period. It is about 200 feet thick in the vicinity of ZNPS.

There is no indication of faulting beneath the site. The area within 100 miles of the site is considered to be one of minor seismic activity. Few events of moderate significance have occurred in the region in the last 150 to 200 years.

Information on recent earthquakes near Lake County was obtained from the Illinois State Geological Survey (Reference 8-19). This review indicated a small 2.4-magnitude earthquake on January 30, 2012 at an epicenter of 42.340 latitude and -88.243 longitude, 2 miles east of McHenry and approximately 30 miles west of ZNPS. A previous earthquake of 3.8 magnitude occurred on February 10, 2010; this seismic event was located about 2 miles northwest of Lily Lake in Kane County, southwest of Zion, approximately 70 miles from ZNPS.

8.5.4. Hydrology and Hydrogeology

Hydrology and hydrogeology information was primarily obtained from two Conestoga Rovers & Associates reports (References 8-11 and 8-18). Groundwater is encountered at a depth less than 20 feet 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 a significant thickness (~30 to 50 feet) of glacial silts and clays that act as an aquitard.

Lake Michigan it the major regional discharge zone for groundwater. The groundwater flow in the region is generally towards the lake. Based on borehole observations and the hydrogeological setting, groundwater flow at ZNPS proceeds predominantly easterly to southeasterly toward Lake Michigan, with a more complex localized flow around deep foundations, utilities and the retaining wall that was installed during construction.

The Upper Sand Unit is a high permeability unit that is in hydraulic communication with Lake Michigan, the regional discharge feature, and which generally allows unrestricted lateral groundwater flow, with the exception of the areas around plant structures and the cutoff wall: these deep structures local alters the local flow patterns, however ultimate discharge of groundwater is to the Lake. Vertical groundwater flow is limited by the underlying Silt-Clay Unit which has a low permeability and is approximately 30 feet thick (Dames and Moore *"Foundation Investigation, Proposed Nuclear Power Plant, Zion, Illinois"* [Reference 8-20]).

8.6. Environmental Effects of Decommissioning

8.6.1. Summary

The evaluation of the environmental effects (or impacts) of the decommissioning of ZNPS follows the approach outlined in NUREG-0586. The methodology is described in NUREG-0586, Supplement 1. This approach includes identification of environmental issues as either generic or site-specific. If the issue is considered to be generic, it is assigned a significance level of either "Small", "Moderate", or "Large." If identified as generic, the environmental impact is considered to be bounded by the evaluation in the GEIS which concludes that the impact significance is "Small." In this event, site specific evaluation by licensees is generally not required.

For those environmental issues or decommissioning activities that require site-specific evaluation, a standard approach is followed. It is summarized as follows:

1) The issue or activity is summarized including a summary of the impacts as reported in the original Environmental Statement (ES) and PSDAR. Note that many decommissioning activities are not identified in these documents.
2) Applicable regulations, permits, limits or other regulatory requirements are identified.
3) Potential impacts from decommissioning activities relating to the environmental issue are described.
4) An evaluation is performed. This includes analysis and professional judgment to estimate or determine whether the activity is likely to make a noticeable impact on the environment considering the available information. If an impact is likely, existing and additional mitigation measures that can be taken are evaluated. If an impact cannot be avoided, a determination is made as to whether the impact is likely to seriously damage the resource or attribute.
5) A conclusion is reached.

A conclusion is derived from the evaluation steps summarized above. The conclusion identifies the level of significance of the impacts. Site-specific issues are not bounded by the GEIS evaluation.

Table 8-1 was used as the basis for the site specific environmental impact assessment for ZNPS. It is excerpted from Table 6.1 of NUREG-0586, Supplement 1. The first step in this process is to screen the issues to identify site-specific issues. Decommissioning activities specific to ZNPS are then reviewed and the activities that may require site-specific evaluation are identified. The screening identified the following;

- Offsite land use activities: changes in demographics and zoning that have occurred in the past 40 years.
- Aquatic ecology affected by activities beyond the operational area; changes in designation of sensitive areas (local wetlands and expansion of Illinois State Beach).
- Terrestrial Ecology affected by activities beyond the operational area: changes in designation of sensitive areas (local wetlands and expansion of Illinois State Beach).

- Threatened and endangered species: changes in local flora and fauna and designation of threatened and endangered species that have occurred in the past 40 years.
- Environmental Justice: changes in demographics and socioeconomic status in the past 40 years.
- Cultural and Historic Resource impacts beyond the operational areas: changes in local historic landmark designations and other cultural resources.

The following decommissioning activities were identified which required evaluation of impacts across several environmental attributes or issues.

- ISFSI construction: land use impacts (onsite).
- Vertical Concrete Cask (VCC) construction for the ISFSI: land use impacts (onsite).
- Rail line upgrade and extension (onsite and offsite).
- Circulating Water inlet and outlet piping disposition: aquatic ecology (within and beyond the operational area).
- Placement of clean construction demolition debris (CCDD) and sand mix in major building basements: terrestrial ecology and transportation.

8.6.2. Radiological Effects of Decommissioning

8.6.2.1. Occupational Radiation Exposure

During decommissioning, Zion*Solutions* has and will continue to implement a Radiation Protection Program in accordance with the license specifications and the requirements of 10 CFR Part 20. The objectives of the Radiation Protection Program are to control radiation hazards, avoid accidental radiation exposures, maintain occupational worker exposures to less than the administrative limit of less than 2,000 mrem/yr Total Effective Dose Equivalent (TEDE) and, to maintain doses to workers and the public As Low As Reasonably Achievable (ALARA).

On March 9, 1998, Commonwealth Edison (ComEd), the licensee at the time, placed both units at ZNPS in a SAFSTOR condition (a period of safe storage of the stabilized and defueled facility). The reactors at Zion remained in a SAFSTOR condition until September of 2010, when active decommissioning activities commenced. This period of time allowed for the decay of most short-lived radionuclides which subsequently, reduced radiation levels at the facility. This fact, combined with the effective implementation of the Radiation Protection Program and ALARA measures minimizes the projected and actual occupational radiation dose exposure during the decommissioning of ZNPS. It is anticipated that the most significant contributors to occupational dose from remaining dismantlement activities is the segmenting, packaging and shipping of the reactor vessel internals and the reactor vessel.

The GEIS estimates that 1,115 Rem will be needed to decommission a PWR similar in size to the Zion units. Current occupational dose expended and dose expected to complete decommissioning for both units is less than 1,000 Rem. This is well below the GEIS estimate of 2,230 Rem for two units.

As the occupational dose for the decommissioning will meet the regulatory standards of 10 CFR 20, it is therefore bounded by the criteria in the GEIS and the impact is considered as "Small".

8.6.2.2. Offsite Radiation Exposure and Monitoring

Zion*Solutions* implements a regulatory compliant Radiological Environmental Monitoring Program (REMP) at ZNPS, which provides annual reports with an accurate assessment of the radiological environment in and around the environs of the site. The REMP program provides assurance that the radioactive gaseous and liquid effluent releases during plant operations do not exceed the concentration limits of 10 CFR 20, the dose limits of 10 CFR 50, Appendix I, or the fuel cycle dose limits of 40 CFR 190. Zion*Solutions* will continue to adhere to these limits throughout the course of the decommissioning. Consequently, the public dose from decommissioning is bounded by the criteria in the GEIS and the impact is considered as "Small".

At ZNPS, the Circulating Water Discharge Tunnels are the main authorized effluent release pathway for the discharge of treated and filtered radioactive liquid waste to Lake Michigan. Liquid effluents are monitored and sampled prior to release from onsite storage tanks.

The gaseous pathway analysis is subject to the meteorological conditions during the time of the release. Due to plant shutdown and cessation of noble gas and other radionuclide generation, gaseous effluents do not present a significant release or exposure pathway. Routine grab air sampling is performed to determine the dose due to radioactive gaseous releases.

The direct radiation exposure is measured continuously with the use of passive monitoring devices. The dose is integrated over three months to accumulate a statistically significant exposure.

The design basis for the ISFSI precludes airborne radioactive releases during spent fuel storage and provides adequate shielding to minimize exposure. Radiation monitoring for the ISFSI is performed in accordance with the Radiation Protection Program implemented at ZNPS. In accordance with the worst case scenario in the design basis, the projected doses at the site boundary are substantially below the limits established in 10 CFR 72.106(b) where there is total loss of the confinement barrier. Exposure from the ISFSI to the nearest permanent resident will not exceed 25 mrem/year as specified in 10 CFR 72.104 and 40 CFR Part 190.

8.6.2.3. Environmental Effects of Accidents and Decommissioning Events

Decommissioning accident analysis is integral to the licensing design basis for ZNPS. While decommissioning radioactively contaminated structures, systems and components at ZNPS, it is necessary to assure the safety of the public in the surrounding area and workers. Worker safety is addressed in the Radiation Protection and Safety programs for the Zion Station Restoration Project (ZSRP) which rely on ALARA principles and the Zion*Solutions* ZS-SA-01, "*Zion Restoration Project Health and Safety Plan*" (HASP) (Reference 8-21). The safety of the public is principally related to potential hazards associated with an airborne release of radioactive materials during decommissioning operations.

During decommissioning, ZSRP will perform decontamination and dismantlement of structures, systems, and components in addition to maintenance, waste management, and surveillance. The

accidents discussed in NUREG-0586, Supplement 1 associated with immediate dismantling would also be applicable during the decommissioning of ZNPS. However, the potential consequences associated with those accidents would be less at ZNPS due to the reduction of the total radionuclide inventory at ZNPS due to:

- Decontamination efforts made before decommissioning,
- Prior radioactive waste shipments, and
- Radioactive decay.

Consequently, the potential decommissioning accidents at ZNPS are bounded by the accident evaluation presented in NUREG-0586, Supplement 1.

Operational accidents during decommissioning could result from equipment failure, human error, and service conditions. With the spent nuclear fuel removed from the reactors, operational accidents during decommissioning can be categorized as follows:

- Radioactive waste transportation accidents,
- Explosions and/or fires associated with explosive and/or combustible materials,
- Loss of contamination control,
- Natural phenomena, and
- Human caused events external to ZNPS.

These potential operational accidents during decommissioning are addressed in NUREG-0586, Supplement 1 for immediate dismantlement and consequently, are bounding for the decommissioning of ZNPS.

8.6.2.4. Storage and Disposal of Low-level Radioactive Waste

The decommissioning of ZNPS has, and will continue to require the disposal of large volumes of low level radioactive waste, including contaminated equipment, tools, clothing and bulk debris materials such as concrete, metal, and asphalt. Materials that cannot be free released are, and will continue to be dispositioned as low-level radioactive waste. Through the proper implementation of the Waste Management Program, Process Control Program and associated procedures, Zion*Solutions* ensures the appropriate segregation, classification, processing, packaging, shipment and control of solid, liquid and gaseous radioactive wastes.

The majority of the Class A low-level radioactive waste from ZNPS will be shipped to the Energy*Solutions* disposal site in Clive, Utah. The radioactive materials are typically packaged in SuperSacs and then placed into Energy*Solutions* owned 100 ton, high-capacity SuperGondola railcars for transport on Union Pacific rail lines to the disposal site. Oversized or overweight components, such as the Reactor Vessel Head, are shipped using multiple axle tractor/trailer rigs or special rail cars. Rail and truck shipments are made in accordance with Department of Transportation (DOT) regulations. Class B and C low-level radioactive waste from ZNPS will be shipped to the Waste Control Specialists disposal site in Andrews, Texas.

Zion*Solutions* completed the construction of the ISFSI in August 2013 and started spent nuclear fuel cask loading in December 2013. Zion*Solutions* anticipates completing the transfer of all its

spent nuclear fuel, in sixty-one (61) VCC to the ISFSI by early 2015. The multi-purpose fuel canisters within the casks are seal-welded and leak tight; therefore no leakage is expected during normal operation, off-normal conditions, or design basis accidents. The storage of the fuel at the ISFSI does not generate any gaseous, liquid, or solid radioactive waste. The spent nuclear fuel will remain in storage at the ISFSI under the Part 50 license until the fuel is transferred to a permanent repository. Greater-Than-Class C low-level radioactive waste will be stored in four seal-welded leak tight canisters within storage casks co-located at the ISFSI with the spent fuel.

8.6.2.5. Radiological Criteria for License Termination

Following the completion of decontamination, dismantlement and remediation activities, radiological surveys will be performed to demonstrate that the dose from any residual radioactivity remaining in as-left structure basements and soils at ZNPS (excluding the ISFSI) to the unrestricted release criteria as specified in 10 CFR 20.1402. Once the balance of the site is remediated and the as-left radiological conditions are demonstrated to be below the unrestricted release criteria, the 10 CFR Part 50 license will be reduced to the area around the ISFSI and the site will be transferred back to Exelon under the 10 CFR Part 50 license. LTP Chapter 5 and Chapter 6 provide the methodology for demonstrating compliance with the unrestricted release criteria.

8.6.3. Non-radiological Effects of Decommissioning

8.6.3.1. Onsite Land Use

The environmental impact associated with onsite land uses have been determined by the NRC, within section 4.3.1 of NUREG-0586, Supplement 1 to be generically considered as a "Small" impact.

The decommissioning project is located and executed within the boundary of the existing ZNPS property previously used for power generation; all work will be conducted in previously developed footprint. Some onsite roads have been refurbished and a reinforced heavy haul path was constructed to support the transfer of VCCs to the ISFSI. No barge slips are being constructed. The rail was originally installed during the construction of the station and was part of the operation of the facility. The onsite rail line was modified and refurbished to support decommissioning activities. Containers will be unloaded and loaded onsite. Onsite land activities such as vehicle parking and equipment/container laydown, storage, staging and waste loading are and continue to occur in a manner similar to when the facility was operational. Several structures such as the Switchyard, the ISFSI, the ISFSI warehouse, the microwave tower, and the Sewage Lift Station, as well as all roadways and rail lines, will remain at license termination as requested by Exelon.

Section 8.5 of Exhibit C, Lease Agreement, titled "Removal of Improvements; Site Restoration" integral to the Asset Sale Agreement requires the demolition and removal of all on-site buildings, structures, and components to a depth of at least three feet below grade. The major structures that will remain at license termination are the basements of the Unit 1 Containment Building, Unit 2 Containment Building, Auxiliary Building, Turbine Building, WWTF, the lower portion of the Spent Fuel Pool, Crib House and Forebay, Unit 1 and Unit 2 Steam Tunnels and the Circulating Water Intake and Discharge Tunnels below the 588 foot elevation. All systems,

components as well as all structures above the 588 foot elevation (with the exception of the structures previously noted) will be removed during the decommissioning process and disposed of as a waste stream. In both Containment basements, all concrete will be removed from the interior side of the steel liner, leaving only the remaining exposed liner below the 588 foot elevation and the structural concrete outside of the liner. In the Auxiliary Building, all interior walls and floors will be removed, leaving only the exterior walls and basement floor. In the Turbine Building basement, the remaining structures will consist of reinforced concrete floors and exterior foundation walls and the sub-grade portions of the pedestals below the 588 foot elevation. For the Fuel Handling Building, the only portion of the structure that will remain is the lower 12 feet of the Spent Fuel Pool below 588 foot elevation and the concrete structure of the Fuel Transfer Canals once the steel liner has been removed. Other below ground structures that will remain are the lower concrete portions of the WWTF, Main Steam Tunnels, and Circulating Water Inlet Piping and Discharge Tunnels.

The decommissioning approach for ZSRP also calls for the beneficial reuse of concrete from building demolition as clean fill. Uncontaminated concrete that meets the definition of CCDD and where radiological surveys demonstrate that the concrete is free of plant derived radionuclides above background will be used. Demolition debris found to be contaminated or potentially contaminated based on process knowledge will be disposed of as low-level radioactive waste. Consequently, the burial of demolition debris contaminated with residual radioactivity will not have the potential to affect land use and ground or surface water quality. Similarly, painted concrete will only be used if the chemical analysis demonstrates that the chemical constituents are below USEPA and IEPA regulatory criteria.

As during the operation of the facility, decommissioning activities have not been conducted in wetlands. The wetlands around the plant have been protected in accordance with environmental regulations and permits.

There is no information pertaining to any significant environmental changes associated with the site-specific decommissioning activities. Site closure will comply with applicable USEPA and IEPA regulatory requirements.

In accordance with the guidance presented in the GEIS, the potential impacts to land use onsite are considered as "Small".

8.6.3.2. Offsite Land Use (in the Vicinity)

Only areas within the existing site boundary will be used to support decommissioning and license termination activities (such as temporary storage and staging areas). Appropriate isolation and control measures will be instituted to prevent the spread of contamination. These measures will also be monitored to ensure their effectiveness. Thus, no environmental impacts associated with the use of offsite lands are anticipated from the decommissioning activities at ZNPS.

Of the 331 acre site, about 87 acres are located within the fence-enclosed "Radiologically-Restricted Area". The remainder, which lies mostly to the west of the station switchyard, which belongs to ComEd, is an open marshy area. This area is undeveloped except for overhead transmission lines and corridors maintained by ComEd.

The land area immediately west of the site, located between the site and the railway is zoned light industrial by the City of Zion (The City of Zion, Illinois, *"Comprehensive Plan 2010"* [Reference 8-22]). This area is about five blocks long, extending from 29th Street on the south to Shiloh Boulevard on the north and is about four blocks wide in the east-west direction centered on Deborah Avenue. It is currently occupied by several warehouses and associated truck shipping operations, an industrial cleaning-service company, several auto service garages, a salvage yard, a former manufacturing facility and a number of vacant lots.

A significant factor which affects land use in the near vicinity of ZNPS is the Illinois Beach State Park. The Illinois Beach State Park has been expanded since the construction of ZNPS, at which time it only comprised an area located south of the site along the Lake Michigan shoreline. The present day Illinois Beach State Park is comprised of a section north and a section south of ZNPS. The Park is part of a state-owned coastal management area, which extends from Winthrop Harbor (about three miles north of the ZNPS site) to about three miles south of the ZNPS site. The area from Winthrop Harbor Marina on the north to the southern end of the Illinois Beach State Park has been incorporated into the Illinois Coastal Management Program (ICMP). The ICMP has identified this area as a unique public resource requiring special attention for preservation, protection and restoration of areas impacted by shoreline erosion, invasive species and damage caused by previous industrial activities (Illinois Coastal Management Program, *"Illinois Beach State Park and North Point Marina, Including the Dead River and Kellogg Creek Watersheds, 2011"* [Reference 8-23]).

The ZNPS site and the surrounding land on all sides are identified as "environmentally sensitive" in the Zion City planning document. The Lake County Regional Framework Plan identifies the western portion of the ZNPS site and adjacent land, as wetlands and areas with "environmental limitations". In the Lake County Plan, this area is also assigned a "high priority for open space". This includes the strip of land between the Chicago and Northwestern Railway on the west and the Illinois Beach State Park on the east, bisected by the Deborah Avenue light industrial area and the ZNPS site.

Decommissioning activities are not being performed in areas defined as "environmentally sensitive" within the site boundary, nor in land which adjoins similar offsite land areas, City of Zion industries, or the State beach park areas. Consequently, the offsite land areas are not affected by the decommissioning activities and the potential impacts to land use offsite are considered "Small".

8.6.3.3. Water Use

In accordance with section 4.3.2 of NUREG-0586, Supplement 1, the environmental impact associated with water use has been determined to be generally applicable with a "Small" impact.

ZNPS is located on the shores of Lake Michigan. The lake is 307 miles long from north to south and has an average width of 70 miles. The predominant water usage during the operation of ZNPS was the use of water from Lake Michigan as secondary cooling water for the reactor systems. With the plant shutdown and fuel removed from the reactor, the cooling water system is currently used for the cooling of the Spent Fuel Pool, building environmental systems such as air conditioning and heating, and fire suppression. The use of water from Lake Michigan during decommissioning activities is significantly less than the usage during operations.

Water from Lake Michigan is also extensively used for municipal and domestic water supplies. There is multiple potable water intakes located in Lake Michigan in the vicinity of ZNPS. The nearest intake is located about 1 mile north of the ZNPS site and approximately 3,000 feet out into the lake. The City of Zion provides potable water services to support ZNPS. The sewage system is connected to the North Shore Sanitary District (NSSD). Potable water use during decommissioning operations is not expected to be greater than the potable water use experience during operations. Water will continue to be processed in accordance with the site National Pollutant Discharge Elimination System permit (NPDES). Consequently, in accordance with the GEIS, the potential impacts to water use are considered "Small".

8.6.3.4. Water Quality

This section evaluates potential project effects on those portions of the natural environment related to surface water and groundwater. Surface water generally refers to streams, rivers, ponds, reservoirs and lakes. At ZNPS, the nearby bodies of water are Lake Michigan and surface streams near the site, including Kellogg Creek (1.25 miles north), Dead River (3 miles south), and Bull Creek (0.2 mile south) and surrounding wetlands.

The environmental impact evaluation associated with surface and groundwater quality in section 4.3.3 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

At ZNPS, all non-radiological water discharges to Lake Michigan are controlled under an NPDES permit which is issued by the Illinois Environmental Protection Agency (IEPA). Zion*Solutions* has filed a Notice of Intent (NOI) with the IEPA, implemented a Storm Water Pollution Prevention Plan (SWPPP) and obtained a Watershed Development Ordinance (WDO) permit from the Lake County Stormwater Management Commission (SMC) for the demolition of the site structures. In addition, impacts to the lake and nearby creeks will be greatly reduced through implementation of appropriate Best Management Practices (BMP) for soil erosion and sedimentation control.

Radiological impacts are minimized through adherence to Off-site Dose Calculation Manual (ODCM) limits and assessed through the Radiological Environmental Monitoring Program (REMP) and the Radiological Groundwater Protection Program (RGPP). Potential groundwater impacts are monitored by the routine sampling of eleven (11) permanent onsite RGPP wells at ZNPS.

As the water from Lake Michigan is no longer used to cool operating reactor systems at ZNPS, the thermal impact to Lake Michigan has been reduced.

No adverse impacts on surface water and groundwater are expected from the implementation of decommissioning activities. Consequently, the potential impacts to surface and groundwater quality are bounded by the GEIS and considered "Small".

8.6.3.5. Air Quality

The environmental impact evaluation associated with air quality in section 4.3.4 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

ZNPS complies with all applicable Federal and State air quality regulations, including the requirements of the IEPA, Bureau of Air, and will implement BMP to minimize fugitive dust during demolition and decommissioning activities. Air emission sources such as the diesel generators are no longer in service and the auxiliary boiler has been removed. A minor emission source for the above ground storage gasoline tank (1000 gallons) is permitted under the IEPA Registration of Smaller Sources (ROSS) program. This tank will be removed when the decommissioning is complete.

Fugitive dust will be generated from various decommissioning activities, including the demolition of concrete building structures and the excavation of soil. Careful planning and controlled demolition and dismantlement techniques, with appropriate assessments by Zion*Solutions* Radiation Protection, Environment, and Health and Safety staff, will be conducted to ensure excessive and harmful dust emissions are not generated. As necessary, measures such as dust suppression by misting water will be used to mitigate dust emissions.

Demolition equipment will be operated and maintained in accordance with manufacturer's specifications which will prevent increased exhaust emissions. Appropriate Health and Safety assessments and controls will also be established during expected extended periods of operation to ensure that personnel and the environment are not adversely impacted by excessive exhaust emissions.

No adverse impacts on air quality are expected from the implementation of decommissioning activities. Consequently, the potential impacts to air quality are bounded by the GEIS and considered "Small".

8.6.3.6. Aquatic Ecology

The environmental impact evaluation associated with aquatic ecology in section 4.3.5 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

The aquatic habitat at ZNPS includes the area from the intake structure integral to the Crib House at the shoreline to the diffuser structure that extends out into Lake Michigan approximately 870 feet from the lakeshore. Habitats associated with this area were previously disturbed during the initial construction of the facility. However, the implementation of decommissioning activities is not expected to disturb existing aquatic habitats, their flora and fauna in the lake and also nearby streams and wetlands.

Various fresh water fish species, macro-invertebrate populations, and vegetation exist within these aquatic environments and were identified during a study contracted by Zion*Solutions*. Zion*Solutions* contracted an independent environmental analysis to assist with the decision for removing or leaving the Forebay and Circulating Water Discharge Tunnels at ZNPS. This analysis was documented in the previously cited report by AMEC, Inc. pertaining to the Discharge Piping. The report concludes that no action should be taken for the removal of the Forebay and Circulating Water Discharge Tunnels as no action decision resulted in the least impact to the environment, including aquatic ecology considerations.

Plans for the demolition of structures at ZNPS do not include the removal of waste or equipment by barge. Consequently, there is no impact to the beach or shoreline from this type of activity.

Zion*Solutions* will continue to maintain its NPDES permit and decommissioning operations will be performed within applicable NPDES limits. Furthermore, protection of the onsite and adjacent wetlands is, and will continue to be a priority when planning any onsite dismantlement or waste management operation. In addition, the SWPPP is implemented with BMPs to prevent impacts to the aquatic systems.

Exotic species can threaten native species and ecosystems due to aggressive growth, reproduction or survival rate, and diseases or parasites they may transmit to native species. The decommissioning of ZNPS will not introduce any exotic plants or animals into the environment.

The potential impacts to the aquatic ecology within the site boundary are bounded by the GEIS and considered to be "Small". The potential impacts to the aquatic ecology beyond the site boundary have also been evaluated and considered to be "Small".

8.6.3.7. Terrestrial Ecology

The environmental impact evaluation associated with terrestrial ecology in section 4.3.6 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

Exotic/invasive species are known to occur in only few locations near ZNPS. These exotic/invasive species include common reed *(Phragmites australis)* and purple loosestrife *Lythrum salicaria)*, both of which are found in the swale and wetland habitats that are located behind the sand dunes along Lake Michigan. Zion*Solutions* continues to be a partner and support the work of Illinois Department of Natural Resources (IDNR) to remove invasive species (Lime Grass) along the beach. No known exotic or invasive species occur within the decommissioning project area. To minimize the introduction of exotic or invasive species, appropriate BMPs are, and will continue to be followed.

The land around ZNPS was initially disturbed by the construction of the facility and no longer resembles the dune formations prevalent in surrounding areas. This alteration results in a less desirable habitat for many species that rely on dune formations for habitat. Given the short-term nature of the work associated with decommissioning, and the fact that the project area is separated from the Illinois Beach State Park, no direct impacts to sensitive species are anticipated. Additionally, upon completion of construction activities, the land is going to be brought to existing grade and stabilized with guidance on native vegetation, therefore minimizing any long-term impacts to sensitive species. In addition, the planned demolition activities do not include the removal of waste or equipment by barge and consequently, there is no anticipated impact to the beach or shoreline.

Floodplain management requires that long-term and short-term adverse impacts associated with modification of floodplains be avoided to the extent possible. Diverse wetland habitats, including marsh, fen, panne, sedge meadow, and ponds occur within the ZNPS property. Wetlands have been delineated and permits have been obtained for work activities that are in the vicinity of wetlands and wetland buffer zones. Compliance with these permits and the implementation of BMPs mitigates the potential impact on wetlands from decommissioning activities.

The potential impacts to terrestrial ecology are bounded by the GEIS and considered "Small".

8.6.3.8. Threatened or Endangered Species

The only "Threatened" or "Endangered" species that has been observed at ZNPS is the Blanding's turtle. The Blanding's Turtle (Emydoidea, blandingii) is listed as a "Threatened" species in the state of Illinois. During the decommissioning process, Blanding's turtles have been observed, rescued, and protected. Blanding's turtle awareness signs have been posted and inspections are performed to ensure that the Blanding's turtles are protected in accordance with the IDNR recommendations and the IEPA, NPDES, and SWPPP. During the refurbishment of a rail crossing located north of ZNPS, Zion*Solutions* worked with local stakeholders and the IDNR to rescue a den of snakes which included several Western Fox snakes. A hibernaculum was established nearby for the relocation of the rescued snakes. The Western Fox snake is not listed as "Threatened" or "Endangered" but they are considered an important part of the ecosystem.

Other listed species such as the Piping Plover birds and the Massasauga rattlesnake have not been observed on site. No adverse impact to any listed species is anticipated since they are not present in locations expected to be impacted by decommissioning activities. Monitoring and awareness programs have been put in place to for the Blanding's turtles and other protected species that may be identified during the decommissioning activities.

The potential impacts to "Threatened" or "Endangered" species are bounded by the GEIS and considered "Small".

8.6.3.9. Occupational Issues/Safety

The environmental impact evaluation associated with occupational issues in section 4.3.10 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact. While decommissioning involves increased industrial activities and safety focus, similar programs addressing worker safety were implemented during the operation of the facility and also during repair and refueling outages. The occupational issues and safety impacts assessed are those related to human health and safety, including impacts from physical, chemical, ergonomic, and biological hazards. Radiological impacts, were previously discussed in section 8.6.2.1.

Zion*Solutions* is committed to decommissioning ZNPS safely and has established a HASP to implement a program to effectively control hazards in the work environment and prevent occupational injuries and illnesses. The HASP and Zion*Solutions* Health and Safety Program complies with federal and state regulations including Illinois Department of Labor and the U.S. Occupational Health and Safety Administration (OSHA) requirements. The HASP applies to all Zion*Solutions* employees as well as visitors and contract personnel working under direct Zion*Solutions* supervision.

Numerous safety practices and communications are conducted at the site and include, but are not limited to:

- Safety is emphasized as the first topic of discussion at meetings.
- All workers are provided a Health and Safety booklet.
- Worker training and required certifications are reviewed prior to assignment to tasks requiring specific worker qualifications. Certain specialty subcontractors are mobilized, as

necessary, such as the asbestos abatement firm contracted for the removal of all asbestos from the Turbine Building.

- Safety Data Sheets are obtained and reviewed for chemicals bought onsite.
- Health and Safety staff are involved in reviewing and approving decommissioning work packages and participating in pre-job walkdowns, work condition assessments and reviews.
- Daily and weekly safety messages are issued as well as Safety Bulletins to communicate awareness of significant safety issues and lessons learned.
- Safety stand-downs are held whenever serious safety events occur to communicate and reinforce safety events and lessons learned site-wide.

Therefore, occupational issues/safety is evaluated to be bounded by the GEIS and the impact is considered "Small".

8.6.3.10. Cost

A detailed discussion of the site decommissioning project costs is presented in Chapter 7 of this LTP.

8.6.3.11. Socioeconomic Impacts

ComEd's original decision to permanently cease plant operations was not subject to NRC review or approval. On January 14, 1998, the Unicom Corporation and ComEd Boards of Directors authorized the permanent cessation of operations at ZNPS for economic reasons. The economic growth and job opportunities in the Chicago Metropolitan area and the nuclear industry at the time of shutdown in 1998, minimized the effects of unemployment that resulted from the plant shutdown.

In September of 2010, decommissioning activities began and a demolition permit was obtained from the City of Zion. Some of the labor requires specialized skills and equipment from out-of-state. Overall, the decommissioning activities have a short-term positive economic impact on the local community due to the permit fee, and the impact on the local workforce and opportunities for local business.

Therefore, socioeconomic impacts are evaluated to be bounded by the GEIS and the impact is considered "Small".

8.6.3.12. Environmental Justice

While low-income and minority populations are present in the vicinity of the former ZNPS, the percentages of low-income and minorities within the ZNPS census tract are lower than those in other City of Zion census tracts. No impact to the greater population, including special groups, is expected.

An existing rail spur was refurbished to transport large components from ZNPS. The refurbished rail spur will be used to transport waste over an existing route. Decommissioning activities will cause increases in truck traffic to and from ZNPS to transport equipment and debris. The truck traffic will use existing main street routes. Since approximately 90% of the waste will be

removed by rail, the increase in truck traffic will be temporary. There will be no environmental justice impact relative to rail and truck transportation as a result of decommissioning.

There is no reason to believe that low-income and minority populations will be adversely impacted by the decommissioning project. Per the GEIS and this evaluation, the potential site specific impact is considered "Small".

8.6.3.13. Cultural, Historic, and Archaeological Resources

The AEC Environmental Statement included documentation from the State of Illinois, Department of Conservation which stated the following:

"This letter will certify that the Illinois Department of Conservation has reviewed the land sections to be affected by the Zion Nuclear Power Station and has determined that no archaeological, architectural, or historical resources are evident within the same."

ZNPS had an existing rail spur that was refurbished to ship waste and large components off-site. Land disturbance for the removal of large components is minimized since removal is primarily conducted via site rail system.

Land that was disturbed for projects beyond the operational areas (within the owner controlled site) was performed in accordance with the IEPA, NPDES permits and the Lake County SMC, Watershed Development Ordinance which included soil erosion controls and stormwater pollution prevention plans. Additional IEPA and SMC permits were obtained during 2014 for the demolition phase that will take place within the "Radiologically-Restricted Area". These permits included a review by the Illinois Historic Preservation Agency which identified no historic, architectural or archaeological sites exist on the ZNPS site.

Based on the historical information in the AEC Environmental Statement, the results of the reviews of historic, cultural and archaeological resources performed in 2013 and 2014, current transportation methods for large components, and soil erosion control work practices, the decommissioning will have no significant impact on cultural and historic resources. Consequently, as bounded by the GEIS and based upon this evaluation, the potential impacts to Cultural, Historic, and Archaeological Resources are considered "Small".

8.6.3.14. Aesthetics

The environmental impact evaluation associated with aesthetics in section 4.3.15 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

The impact of decommissioning on site aesthetics (e.g. visual skyline) is limited in terms of land disturbance and duration. These impacts are temporary and will cease when decommissioning is completed.

The location of the ISFSI is set back several hundred yards from the lake frontage and located adjacent to the existing switchyard. Once all of the major plant structures and buildings on the lake front are removed, aesthetics will improve by providing a more open view of Lake Michigan. Due to the proximity of the Illinois State Beach Park on the north and south of ZNPS,

restoration of the site to a natural grade will result in a contiguous open view of the Lake Michigan shoreline.

Aesthetics will improve once the site is returned to open space. The final determination on usage for the lake front property will be made by Exelon upon completion of the decommissioning and transfer of the license back to Exelon.

Therefore, the environmental impact associated with aesthetics is evaluated to be bounded by the GEIS and the impact is considered "Small".

8.6.3.15. Noise

The environmental impact evaluation associated with noise in section 4.3.16 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

ZNPS is located on the shore of Lake Michigan with the Illinois State Beach Park on the north and south perimeters of the Owner-Controlled Property. There are no residences within 2,000 feet of the station structures and no schools or hospitals within one mile. The center of the nearest community, Zion, Illinois is located approximately 1.6 miles to the west of the plant.

Noise generation will primarily result from demolition activities involving heavy construction equipment. The noise from the shipment of waste will be minimal since the primary transportation method for shipment of low level radioactive waste will be by rail. Noise associated with decommissioning and shipment of waste is intermittent and temporary and will occur primarily during daylight hours. The ISFSI construction was completed in 2013. The ISFSI is a passive facility and there will be minimal noise generated from its operation. Once the decommissioning is complete, noise levels in the vicinity of the ZNPS site will be reduced to levels below those experienced during the operation of the facility.

Due to the distance of the station from sensitive receptors, there will be limited temporary impacts on noise levels during decommissioning and demolition activities. During the decommissioning, Zion*Solutions* agrees to comply with any noise limitations imposed by the City of Zion.

Therefore, the environmental impact associated with noise is evaluated to be bounded by the GEIS and the impact is considered "Small".

8.6.3.16. Irretrievable Resources

The environmental impact evaluation associated with irretrievable resources in section 4.3.18 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

During the demolition and structural dismantlement of the station, recycling and asset recovery efforts will be made. Some metals (e.g. from turbine, transformer components, etc.) have been released as clean scrap. Low level radioactive waste has been and will be continue to be shipped to the Energy*Solutions* disposal site in Clive, Utah. This facility has sufficient space for the disposal of this waste. In addition, any Class B/C waste that is generated will be shipped to the Waste Control Specialist (WCS) facility in Andrews, Texas.

As stated in the GEIS, irretrievable resources that would occur during the decommissioning process are the materials used to decontaminate the facility (e.g., rags, solvents, gases, and tools), and fuel used for construction machinery and for transportation of materials to and from the site. These resource commitments are considered to be minor and are neither detectable nor destabilizing.

Therefore, the environmental impact associated with irretrievable resources is evaluated to be bounded by the GEIS and the impact is considered "Small".

8.6.3.17. Traffic and Transportation

The environmental impact evaluation associated with transportation issues in section 4.3.17 of NUREG-0586, Supplement 1 has been determined to be generally applicable to ZNPS with a "Small" impact.

The number of shipments and the volume of waste shipped are greater during decommissioning than during the operation of the facility. Non-radiological impacts of transportation include increased traffic and wear and tear on roadways. Because the majority of the waste will be transported by rail, the average number of daily shipments from the site will be relatively small. Consequently, it is anticipated that there will be no significant effect on traffic flow or road wear. The impacts of a transportation accident would be neither detectable nor destabilizing.

Therefore, the environmental impact associated with traffic and transportation is evaluated to be bounded by the GEIS and the impact is considered as "Small".

8.6.3.18. Placement of Clean Construction Demolition Debris (CCDD) and Sand Mix in Major Building Basements: Terrestrial Ecology and Transportation

Zion*Solutions* evaluated the use of CCDD for basement fill end-state and submitted a Request for Concurrence for Basement Fill End-State (ZS-2014-0272) to the Illinois EPA in August of 2014. This request included the results of a sampling plan for concrete candidate fill material to be used for the basement fill end-state. On October 3, 2014 Zion*Solutions* received a Letter of Concurrence from the Illinois EPA (ZS-2014-0349) for the use of CCDD for the basement fill end-state.

8.7. Overview of Regulatory Governing Decommissioning Activities and Site Release

8.7.1. Federal Requirements

Decommissioning activities that are subject to federal regulations include:

- Spent fuel storage at the ISFSI.
- Handling, packaging, and shipment of radioactive waste.
- Worker radiation protection.
- License termination and final site release.
- Worker health and safety.
- Liquid effluent releases.

- Hazardous waste generation/disposition.
- Handling and removal of asbestos.
- Characterization and removal of polychlorinated biphenyls (PCBs).
- Handling and removal of lead paint.

8.7.1.1. Nuclear Regulatory Commission

The majority of radiological activities falls under Title 10 of the Code of Federal Regulation and are administered by the NRC. Applicable Title 10 regulations include:

- Part 20 – Radiation protection.
- Part 50 – Decommissioning activities.
- Part 51 – Environmental protection.
- Part 61 – Disposal of radioactive waste.
- Part 71 – Packaging and transportation of radioactive waste (regulations in 49 CFR Parts 171 through 174 also apply).
- Part 72 – Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related Greater-Than- Class- C (GTCC) waste.
- Part 73 – Physical Protection of Plants and Materials.

8.7.1.2. U.S. Environmental Protection Agency

The Environmental Protection Agency (EPA) regulations outlined in Title 40 of the Code of Federal Regulations apply as follows:

- Part 61 – Asbestos Handling and Removal
- Parts 122 to 125 –NPDES
- Part 141 – Safe Drinking Water Standards
- Part 190 – Radiation Protection Standards for Nuclear Power Operations
- Parts 260 to 272 – Resource Conservation and Recovery Act (RCRA)
- Part 280 – Underground Storage Tanks
- Part 761 – Toxic Substance Control Act (TSCA) for Polychlorinated Biphenyls (PCBs)
- Part 129-132 – Clean Water Act

8.7.2. State and Local Requirements

Permits and approvals from or notifications to state and local agencies are required for safety and environmental protection purposes. Decommissioning activities and related site operations that fall under State and local jurisdiction include but are not limited to the following:

- Lake County Stormwater Management Commission, Watershed Development Ordinance

- Illinois Environmental Protection Agency
- Illinois Historic Preservation Agency
- Clean Construction or Demolition Debris, Illinois Environmental Protection Act, Section 3.160(b)
- City of Zion Demolition Permit

This information provided above is a general overview of the applicable regulations and not intended to be all-inclusive.

8.8. Conclusion

As previously evaluated in the Zion PSDAR, the non-radiological environmental impacts from decommissioning ZNPS are temporary and not significant. The potential issues identified as "site-specific" in NUREG-0586, Supplement 1 (such as "Threatened" and "Endangered" species and environmental justice) have been evaluated and there is no significant impact. The potential environmental impacts associated with decommissioning ZNPS have already been predicted in and will be bounded by the previously issued environmental impacts statements (PSDAR, NUREG-0586, and Zion Environmental Statement). Therefore, there are no new or significant environmental change associated with decommissioning.

8.9. References

8-1 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" – January 1999

8-2 Commonwealth Edison Company, "Environmental Report - Zion Nuclear Power Station" – May 1971, Supplement I – November 1971, Supplement II – December 1971, Supplement III – February 1972, Supplement IV – April 1972, Supplement V – May 1972

8-3 United States Atomic Energy Commission, Directorate of Licensing, "Final Environmental Statement related to the Operation of Zion Nuclear Power Station Units 1 and 2", Docket Nos. 50-295 and 50-304 – December 1972

8-4 U.S. Nuclear Regulatory Commission NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities", Supplement 1, Volume 1" – November 2002

8-5 Zion Nuclear Power Station, "Post Shutdown Decommissioning Activity Report" (PSDAR), – March 2008

8-6 "Zion Nuclear Power Station, Units 1 and 2 Asset Sale Agreement" – December 2007

8-7 AMEC, Inc., "Final Environmental Analysis of Alternatives Regarding Intake/Discharge Structure Disposition at the Former Zion Nuclear Generating Station, Zion, Illinois" – October 2013

8-8 U.S. Nuclear Regulatory Commission, NUREG-1496, Volume 2, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities" – July 1997

8-9 US Census Bureau, "2008-2012 American Community Survey" – 2012

8-10 "Zion Station Historical Site Assessment" (HSA) – September 2006

8-11 Conestoga-Rovers and Associates, "Hydrogeologic Investigation Report, Fleetwide Assessment, Zion Station, Zion Illinois", Revision 1 – September 2006.

8-12 Illinois Department of Commerce and Economic Opportunity, "Zion Illinois, General Information, Climate" – September 2012

8-13 "Illinois State Water Survey 1971 – 2000"

8-14 "Illinois Climate Network, 1991 – 2000 Data Set"

8-15 Zion Station, "Defueled Safety Analysis Report" (DSAR) – September 2014

8-16 "City of Zion" www.city-data.com/city/Zion-Illinois

8-17 Commonwealth Edison Company, "Zion Nuclear Power Station - Final Safety Analysis Report" (FSAR) – November 1970

8-18 Zion*Solutions* Technical Support Document 14-003, Revision 3, Conestoga Rovers & Associates (CRA) Report, "Zion Hydrogeologic Investigation Report"

8-19 "Illinois State Geological Survey" www.isgs.illnois.edu

8-20 Dames and Moore, "Foundation Investigation, Proposed Nuclear Power Plant, Zion, Illinois" – October 1967

8-21 Zion*Solutions* ZS-SA-01, Revision 8, "Zion Restoration Project Health and Safety Plan" (HASP)

8-22 The City of Zion, Illinois, "Comprehensive Plan 2010" – January 1992

8-23 Illinois Coastal Management Program, "Illinois Beach State Park and North Point Marina Including the Dead River and Kellogg Creek Watersheds" – 2011

Table 8-1 Summary of the Environmental Impacts from Decommissioning Nuclear Power Facilities

Section	Environmental Issue	GEIS	Impact Significance
8.6.2	Radiological		
	Activities resulting in occupational dose to workers	Yes	Small
	Activities resulting in dose to the public	Yes	Small
	Radiological Accidents	Yes	Small
8.6.3.1	Onsite land use activities	Yes	Small
8.6.3.2	Offsite land use activities	No	Site-specific
8.6.3.3	Water Use	Yes	Small
8.6.3.4	Water Quality		
	Surface water	Yes	Small
	Ground water	Yes	Small
8.6.3.5	Air Quality	Yes	Small
8.6.3.6	Aquatic Ecology		
	Activities within the operational area	Yes	Small
	Activities beyond the operational area	No	Site-specific
8.6.3.7	Terrestrial Ecology		
	Activities within the operational area	Yes	Small
	Activities beyond the operational area	No	Site-specific
8.6.3.8	Threatened and Endangered Species	No	Site-specific
8.6.3.9	Occupational Issues	Yes	Small
8.6.3.11	Socioeconomic	Yes	Small

Table 8-1 Summary of the Environmental Impacts from Decommissioning Nuclear Power Facilities (continued)

Section	Environmental Issue	GEIS	Impact Significance
8.6.3.12	Environmental Justice	No	Site-specific
8.6.3.13	Cultural and Historic Resource Impacts		
	Activities within the operational area	Yes	Small
	Activities beyond the operational area	No	Site-specific
8.6.3.14	Aesthetics	Yes	Small
8.6.3.15	Noise	Yes	Small
8.6.3.16	Irretrievable Resources	Yes	Small
8.6.3.17	Traffic and Transportation	Yes	Small
8.6.3.18	Placement of clean construction demolition debris (CCDD) and sand mix in major building basements: terrestrial ecology and transportation.	No	Site-specific

Note: Cost, section 4.3.11 in GEIS Supplement 1, is not evaluated using environmental significance levels and is not identified as a generic or site-specific issue.

Figure 8-1 Zion Nuclear Power Station Geographical Location

Figure 8-2 Zion Nuclear Power Station Owner Controlled Area

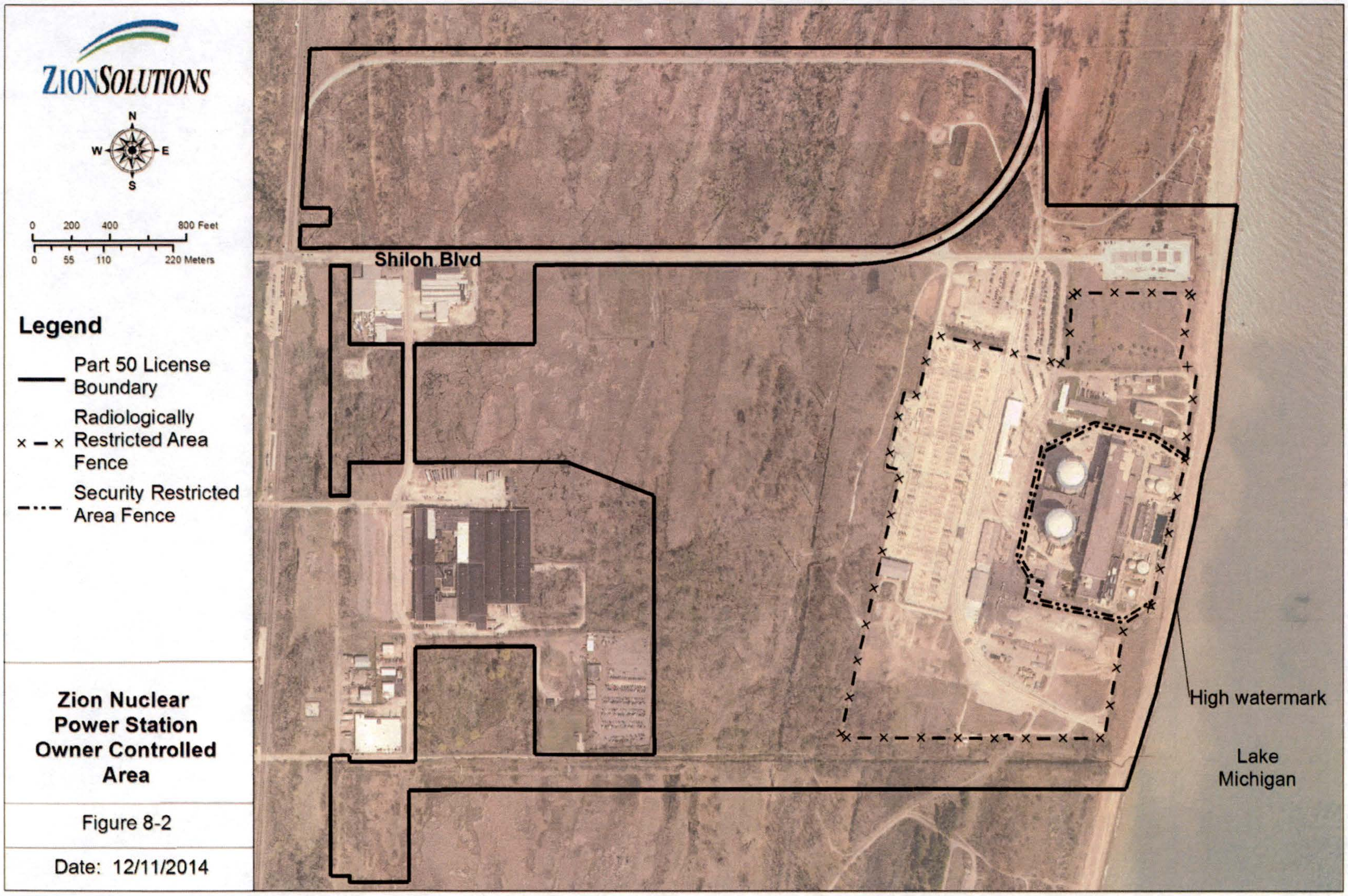

Figure 8-3 Zion Nuclear Power Station Topographical Map