ZIONSOLUTIONS LLC
An EnergySolutions Company

September 30, 2019

ZS-2019-0090

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: Revised Final Status Survey (FSS) Final Report – Phase 2
References:

1) Gerard van Noordennen, ZionSolutions, Letter to U.S. Nuclear Regulatory Commission, Final Status Survey Final Report- Phase 1," dated November 1, 2018.
2) Gerard van Noordennen, ZionSolutions, Letter to U.S. Nuclear Regulatory Commission, Final Status Survey Final Report- Phase 2, Part 1," dated March 11, 2019.
3) John B. Hickman, U.S. Nuclear Regulatory Commission, Letter to John Sauger, Energy*Solutions,* " Zion Nuclear Power Station Units 1 and 2 – Review of Final Status Survey Report - Phase 1," dated March 22, 2019.
4) Gerard van Noordennen, ZionSolutions, Letter to U.S. Nuclear Regulatory Commission, "Final Status Survey (FSS) Report Phase 1 Request for Additional Information," dated May 14, 2019.
5) Gerard van Noordennen, Zion*Solutions*, Letter to U.S. Nuclear Regulatory Commission, Revised Final Status Survey Report- Phase 1," dated June 21, 2019.

The Zion Station Restoration Project License Termination Plan (LTP) describes the process of releasing land for unrestricted use. LTP Section 5.11 states 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.

During the FSS Phase 1 report review (Reference 1), NRC staff identified several issues prohibiting completion of the review and stated that the Phase 2, Part 1 report (Reference 2) review would be deferred until resolution of issues documented in Reference 3 was completed. A Zion*Solutions* (ZS) response to the NRC concerns identified was provided in Reference 4.

ZS submitted Final Status Survey (FSS) Report, Phase 2, Part 1 for NRC review on March 11, 2019, as documented in Reference 2. Revisions to this report have been made to address the NRC request for ZS to perform an additional review of the Phase 2 submittal.

Attachment 1 contains the revised Zion Station Restoration Project FSS Final Report – Phase 2. This Phase 2 Final Report encompasses the remaining below-grade basement structures including embedded pipe and penetrations. This report contains a compilation of 31 survey units. Table 1-1 of the FSS Final Report – Phase 2provides a listing of all the survey units addressed in this report, along with their classifications and size. Figure 1-1 of the revised FSS Final Report – Phase 2 depicts the locations of the survey units in relation to the ZNPS site as well as survey unit boundaries. Zion*Solutions* anticipates three additional FSS Final Report submittals. With potential changes in the decommissioning schedule, it is possible that interim submittals will be filed with the NRC with the

NMSS01
NMSS

goal of providing Release Records as soon as possible to support review and the potential release of site open lands. The FSS Final Report for buried pipe will be submitted as Part 2 of the Phase 2 scope. The Phase 3 FSS Final Report will include the open land survey units encompassing the south and east portion of the site, and the Phase 4 FSS Final Report will encompass the north and west portion.

Zion*Solutions* hereby requests the NRC review the attached revised FSS Final Report – Phase 2 for acceptance of this portion of the site final radiological survey by February 1, 2020.

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

Respectfully,

Gerard van Noordennen

Gerard van Noordennen
Vice President Regulatory Affairs

Attachments:

Attachment 1: Revised Final Status Survey Report - Phase 2

Attachment 2: Preflight Report

Enclosures

Enclosure 1: CD contains Revised Final Status Survey Final Report - Phase 2 & Supporting Documents

cc: John Hickman, U.S. NRC Senior Project Manager
Regional Administrator, U.S. NRC, Region III
Service List (Cover letter only, no enclosures)

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

cc:

Ken Robuck
President and CEO
Energy*Solutions*
299 South Main Street, Suite 1700
Salt Lake City, UT 84111

John Sauger
President and Chief Nuclear Officer
Reactor D&D
Energy*Solutions*
121 W. Trade Street, Suite 2700
Charlotte, NC 28202

Bruce Hinkley
Senior Vice President
D&D Operations
Energy*Solutions*
121 W. Trade Street, Suite 2700
Charlotte, NC 28202

Gerard van Noordennen
Senior VP Regulatory Affairs
Energy*Solutions*
121 W. Trade Street, Suite 2700
Charlotte, NC 28202

Jerry Houff
General Manager
Zion*Solutions*, LLC
2701 Deborah Avenue
Zion, IL 60099

Russ Workman
General Counsel
Energy*Solutions*
299 South Main Street, Suite 1700
Salt Lake City, UT 84111

Steven A. Reynolds
Manager, Nuclear Facility Inspection
Division of Nuclear Safety
Illinois Emergency Management Agency
245 W. Roosevelt Road, Units 55 & 56
West Chicago, Illinois 60185

Kelly F. Grahn, Senior Health Physicist
Illinois Emergency Management Agency
Division of Nuclear Safety
245 W Roosevelt Road, Building 8, Suite 55
West Chicago, IL 60185

William P. Mazzeno
Emergency Management Coordinator
Lake County Emergency Management
Agency
1303 N. Milwaukee Avenue
Libertyville, IL 60048-1308

John E. Matthews
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Avenue, NW
Washington, DC 20004

Attachment 1

Revised Final Status Survey Report – Phase 2

ZION STATION RESTORATION PROJECT

FINAL STATUS SURVEY
FINAL REPORT – PHASE 2

SEPTEMBER 2019

PREPARED BY / DATE: Robert Massengill 2019-09-01

Robert Massengill, Radiological Engineer

REVIEWED BY / DATE: 2019-09-01

David A. Montt, Radiological Engineer

APPROVED BY / DATE: 2019-09-01

D. Wojtkowiak, C/LT Manager

TABLE OF CONTENTS

1. INTRODUCTION 9

1.1 Executive Summary 9

2. FINAL STATUS SURVEY PROGRAM OVERVIEW 14

2.1 Survey Planning 16

2.1.1 State the Problem 16
2.1.2 Identify the Decision 17
2.1.3 Identify Inputs to the Decision 17
2.1.4 Define the Study Boundaries 17
2.1.5 Develop a Decision Rule 17
2.1.6 Specify Tolerable Limits on Decision Errors 17
2.1.7 Optimize the Design for Obtaining Data 18

3. SITE INFORMATION 30

3.1 Site Description 30

3.2 Survey Unit Description 31

3.2.1 Survey Units 01100 and 01110 (Unit 1 Containment above 565 foot elevation. and Unit 1 Containment Under Vessel Areas) 31
3.2.2 Survey Unit 01111 (Unit 1 Containment Incore Sump Discharge Pipe) 32
3.2.3 Survey Unit 1112 (Unit 1 Containment Penetrations) 33
3.2.4 Survey Units 02100 and 02110 (Unit 2 Containment above 565 foot elevation and Unit 1 Containment Under Vessel Areas) 34
3.2.5 Survey Unit 02112 (Unit 2 Containment Penetrations) 34
3.2.6 Survey Unit 03202 (Spent Fuel Pool/Transfer Canal) 35
3.2.7 Survey Unit 05100 (Auxiliary Building Basement) 36
3.2.8 Survey Unit 05119 (Auxiliary Building 542 ft. Embedded Floor Drains) 38
3.2.9 Survey Unit 05120 (Auxiliary Building Penetrations) 38
3.2.10 Survey Unit 06100 (Turbine Building Basement) 40
3.2.11 Survey Unit 06105B (Turbine Building Embedded Pipe) 41
3.2.12 Survey Unit 09200 and 6205A (Unit 1 and Unit 2 Circulating Water Discharge Tunnels and Piping) 41
3.2.13 Survey Unit 06107 and 06108 (Unit 1 and Unit 2 Buttress Pits) 41
3.2.14 Survey Unit 06201 and 06202 (Unit 1 and Unit 2 Diesel Fuel Oil Storage Tank Rooms) 41
3.2.15 Survey Unit 06209 and 06210 (Unit 1 and Unit 2 Steam Tunnel Embedded Floor Drains) 42

3.2.16 Survey Unit 06211 and 06212 (Unit 1 and Unit 2 Tendon Tunnel Embedded Floor Drains) ... 42
3.2.17 Survey Unit 06213 and 06214 (Unit 1 East and West Main Steam Valve Houses) ... 42
3.2.18 Survey Unit 06215 and 06216 (Unit 2 East and West Main Steam Valve Houses) ... 42
3.2.19 Survey Unit 08100, 08401, 08102A/B (Crib House/Forebay, including the Unit 1 and Unit 2 Circulating Water Intake Pipes) ... 43
3.2.20 Survey Unit 09100 (Waste Water Treatment Facility) ... 43
3.3 Summary of Historical Radiological Data ... 44
3.3.1 Historical Site Assessment and Characterization Surveys ... 44
3.4 Conditions at the Time of Final Status Survey ... 63
3.5 Identification of Potential Contaminants ... 65
3.6 Radionuclides of Concern and Mixture Fractions ... 65
3.7 Radiological Release Criteria ... 66
4. FINAL STATUS SURVEY PROTOCOL ... 66
4.1 Data Quality Objectives ... 66
4.1.1 State the Problem ... 66
4.1.2 Identify the Decision ... 67
4.1.3 Identify Inputs to the Decision ... 67
4.1.4 Define the Boundaries of the Survey ... 70
4.1.5 Develop a Decision Rule ... 70
4.1.6 Specify Tolerable Limits on Decision Errors ... 70
4.1.7 Optimize Design ... 71
4.2 Survey Unit Designation and Classification ... 75
4.3 Background Determination ... 75
4.4 Final Status Survey Sample Plans ... 75
4.5 Survey Design ... 76
4.5.1 Determination of Number of Data Points ... 76
4.5.2 Measurement Locations ... 78
4.6 Instrumentation ... 81
4.6.1 Instrumentation Efficiencies ... 81
4.6.2 Instrumentation Sensitivities ... 82
4.6.3 Instrument Maintenance and Control ... 83
4.6.4 Instrument Calibration ... 83
4.7 Survey Methodology ... 84
4.7.1 ISOCS Surveys ... 84

4.7.2 Embedded Pipe Surveys 84
4.7.3 Penetrations Surveys 86
4.7.4 Quality Control Surveys 86
5. SURVEY FINDINGS 87
5.1 Survey Data Conversion 87
5.2 Survey Data Verification and Validation 92
5.3 Anomalous Data/Elevated Scan Results and Investigation 93
5.3.1 Unit 1 and Unit 2 Containment Under Vessel (Survey Units 01110 and 02110) 93
5.3.2 Unit 1 Containment Penetrations (Survey Unit 01112) 95
5.3.3 Unit 2 Containment Penetrations (Survey Unit 02112) 95
5.3.4 Auxiliary Building Basement (Survey Unit 05100) 96
5.3.5 Spent Fuel Pool/Transfer Canal (Survey Unit 03202) 97
5.3.6 Turbine Building Basement and Circulating Water Discharge Tunnels (Survey Units 06100 and 09200B) 97
5.3.7 Unit 1 and Unit 2 570 foot Diesel Generator Rooms (Survey Units 06201 and 06202) 98
5.3.8 Unit 1 East and West Steam Tunnel Valve Houses (Survey Units 06213 and 06214) . 98
5.4 Evaluation of Number of Sample/Measurement Locations in Survey Units 98
5.5 Comparison of Findings with Derived Concentration Guideline Levels 99
5.5.1 Basement Surface Area Adjustments 99
5.5.2 Compliance Equation 102
5.6 Description of ALARA to Achieve Final Activity Levels 105
5.7 USNRC/Independent Verification Team Findings 106
6. SUMMARY 106
7. REFERENCES 107
8. APPENDICES 108

LIST OF FIGURES

Figure 1-1 Phase 2 Survey Unit Release Record Designation 12
Figure 2-1 Characterization/License Termination Group Organizational Chart 15
Figure 3-1 Auxiliary Building 542’ Elevation 37
Figure 3-2 Auxiliary Building Penetration Map 39
Figure 3-3 Embedded Drain Pipe FS Survey 58
Figure 3-4 Turbine Building Demolition 60
Figure 4-1 ISOCS Systematic Measurements of Auxiliary Building 542 ft. Elevation 72
Figure 4-2 Investigation of ISOCS Systematic Measurement at Location No. 278 on the Auxiliary Building 542 ft. Elevation 73

LIST OF TABLES

Table 1-1 Survey Units Encompassed in Phase 2 Report 10
Table 2-1 Base Case DCGLs ($DCGL_B$) for Basements (pCi/m^2) 19
Table 2-2 Operational DCGLs ($OpDCGL_B$) for Basements (pCi/m^2) 19
Table 2-3 Base Case DCGLs for Embedded Pipe ($DCGL_{EP}$) 20
Table 2-4 Operational DCGLs for Embedded Pipe ($OpDCGL_{EP}$) 20
Table 2-5 Base Case DCGLs for Penetrations ($DCGL_{PN}$) 21
Table 2-6 Operational DCGLs for Penetrations ($OpDCGL_{PN}$) 21
Table 2-7 Recommended Survey Coverage 22
Table 2-8 Dose Significant Radionuclides and Mixture 24
Table 3-1 Initial Suite of Radionuclides 65
Table 4-1 Investigation Levels 68
Table 4-2 Surrogate Ratios 68
Table 4-3 Surrogate Calculation Results 69
Table 4-4 Synopsis of Survey Design 74
Table 4-5 Adjusted Minimum Number of ISOCS Measurements per FSS Unit 77
Table 4-6 Number of Measurements for FSS 80
Table 4-7 Recommended Scan Coverage 84
Table 5-1 Basic Statistical Properties of Phase 2 Survey Unit Non-Parametric Measurements 89
Table 5-2 Surface Survey Units Contributing to Each Basement 100
Table 5-3 Adjusted Basement Surface Areas for Area-Weighted SOF Calculation 101
Table 5-4 Adjusted Basement Surface Areas for Area-Weighted SOF Calculation 104

LIST OF ACRONYMS AND ABBREVIATIONS

ABEDCT	Auxiliary Building Equipment Drain Collection Tank
ALARA	As Low As Reasonably Achievable
AMCG	Average Member of the Critical Group
BcDCGL	Base Case Derived Concentration Guideline Level
BcSOF	Base Case Sum of Fractions
CsI	Cesium Iodide
CoC	Chain of Custody
DQA	Data Quality Assessment
DQO	Data Quality Objective
DCGL	Derived Concentration Guideline Level
EMC	Elevated Measurement Comparison
FSS	Final Status Survey
HSA	Historical Site Assessment
HTD	Hard-to-Detect
HUT	Hold Up Tank
IC	Insignificant Contributor
ID	Internal Diameter
IMB	Inner Missile Barrier
ISOCS	In Situ Object Counting System
LTP	License Termination Plan
LBGR	Lower Bound of the Gray Region
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
MPC	
NaI	Sodium Iodide
NIST	National Institute of Standards and Technology
NPDES	National Pollutant Discharge Elimination System
ODCM	Off Site Dose Calculation Manual

OMB	Outer Missile Barrier
OpDCGL	Operational Derived Concentration Guideline Level
OpSOF	Operational Sum of Fractions
QAPP	Quality Assurance Project Plan
QC	Quality Control
RCA	Radiologically Controlled Area
RE	Radiological Engineer
ROC	Radionuclides of Concern
ROR	Radiological Occurrence Report
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SOF	Sum of Fractions
TEDE	Total Effective Dose Equivalent
UCL	Upper Confidence Level
USNRC	United States Nuclear Regulatory Commission
VCC	Vertical Concrete Cask
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZSRP	Zion Station Restoration Project

1. INTRODUCTION

1.1 Executive Summary

The purpose of this Phase 2 Final Status Survey (FSS) Final Report is to provide a summary of the survey results and overall conclusions which demonstrate that the Zion Nuclear Power Station (ZNPS) facility, or portions of the site, meet the 25 mrem per year release criterion as established in Nuclear Regulatory Commission Regulation 10 CFR 20.1402 *"Radiological Criteria for Unrestricted Use"*.

This report documents that FSS activities were performed consistent with the guidance provided in the "Zion Nuclear Power Station, Units 1 and 2 - Issuance of Amendments 191 and 178 for the Licenses to Approve the License Termination Plan" (LTP) (Reference 1); NUREG-1575, *"Multi-Agency Radiation Survey and Site Investigation Manual"* (MARSSIM) (Reference 2); ZS-LT-01, *"Quality Assurance Project Plan for Characterization and FSS"* (QAPP) (Reference 3); ZS-LT-300-001-001, *"Final Status Survey Package Development"* (Reference 4); ZS-LT-300-001-003, *"Isolation and Control for Final Status Survey"* (Reference 5); ZS-LT-300-001-004, *"Final Status Survey Data Assessment"* (Reference 6); as well as various other station implementing procedures.

Revision 2 of the Zion LTP, along with the accompanying Safety Evaluation Report (SER) was approved on September 28, 2018.

This Phase 2 FSS Final Report encompasses the below grade basement structures for the Unit 1 and Unit 2 Containments, Turbine Building, Auxiliary Building, Spent Fuel Pool (SFP)/Transfer Canal, Forebay, Crib House, and the Waste Water Treatment Facility (WWTF). The FSS results provided herein assess and summarize that any residual radioactivity results in a Total Effective Dose Equivalent (TEDE) to an Average Member of the Critical Group (AMCG) that does not exceed 25 mrem per year, and the residual radioactivity has been reduced to levels that are As Low As Reasonably Achievable (ALARA). The release criterion is translated into site-specific Derived Concentration Guideline Levels (DCGLs) for assessment and summary.

This FSS Final Report has been written consistent with the guidance provided in the LTP; NUREG-1757, Vol. 2, "*Consolidated Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria*" (Reference 7); MARSSIM; and the requirements specified in ZS-LT-300-001-005, "*Final Status Survey Data Reporting*" (Reference 8).

To facilitate the data management process, this FSS Final Report has incorporated multiple Release Records pertaining to basement FSS units. Release Records are complete and unambiguous records of the as-left radiological status of each specific survey unit. Sufficient data and information are provided in each Release Record to enable an independent re-creation and evaluation at some future time of both the survey activities and the derived results.

This report contains a compilation of all thirty-one (31) below grade basement structure survey units that are within the Phase 2 scope. Table 1-1 provides a listing of all the survey units addressed in this report, along with their classifications and size. Figure 1-1 depicts the locations of the survey units in relation to the ZNPS site as well as survey unit boundaries.

For the below grade structures, compliance with the unrestricted release criteria was demonstrated mainly through the use of Canberra In Situ Object Counting System (ISOCS) for direct measurements of building surfaces, hand held instruments for scans/static measurements of penetrations, and pipe survey instruments for embedded pipe.

All FSS activities essential to data quality have been implemented and performed under approved procedures. Trained individuals, using properly calibrated instruments and laboratory equipment (sensitive to the suspected contaminants), performed the FSS of the Phase 2 survey units. The survey data for all Phase 2 survey units demonstrate that the dose (TEDE) from residual radioactivity is less than the maximum annual dose (TEDE) of 25 mrem/year to the member of the public hypothesized. This dose limit corresponds to the release criterion for license termination of facilities to be released for unrestricted use as specified in 10 CFR 20.1402. It also provides the basis and support for the release of these areas from the 10 CFR 50 licenses. Finally, meeting this release criterion satisfies the ALARA requirement of 10 CFR 20.1402.

Table 1-1 – Survey Units Encompassed in Phase 2 Report

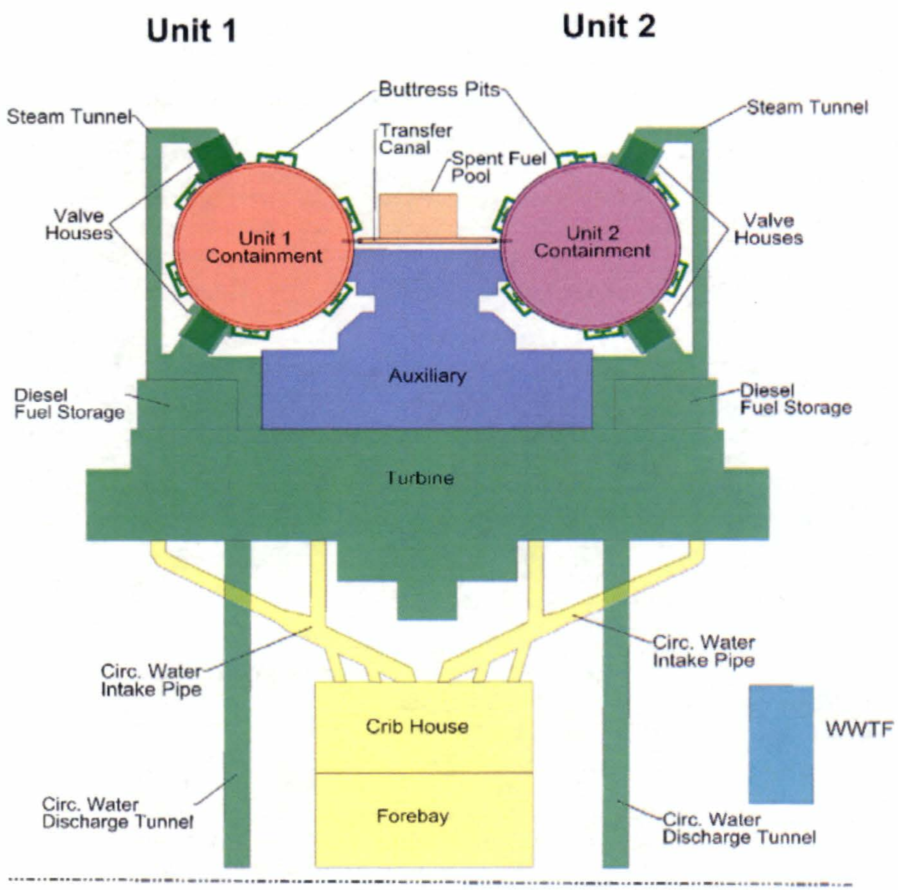
Survey Unit	Name	Class (6)	Size (m^2)
01100(1)	Unit 1 Containment above 565 ft.	1	2,465
01110(1)	Unit 1 Containment Under Vessel Area	1	294
01111	Unit 1 Containment Incore-Sump Drain	1	0.86
01112	Unit 1 Containment Penetrations	1	255
02100(2)	Unit 2 Containment above 565 ft.	1	2,465
02110(2)	Unit 2 Containment Under Vessel Area	1	294
02112	Unit 2 Containment Penetrations	1	253
03202	SFP/Transfer Canal	1	723
05100	Auxiliary Building 542 ft. Floor and Walls	1	7,226
05119	Auxiliary Building Embedded Floor Drains	1	294

Table 1-1 (continued) Survey Units Encompassed in Phase 2 Report

Survey Unit	Name	Class (6)	Size (m^2)
05120	Auxiliary Building Penetrations	1	15.41
06100(3)	Turbine Building Basement and Steam Tunnels	3	27,135
06105A(3)	Circulating Water Discharge Pipe	3	1,075
09200(4)	Unit 1 & Unit 2 Circulating Water Discharge Tunnels	3	4,868
06105B(4)	Turbine Building Embedded Pipe	3	238
06107(4)	Unit 1 Turbine Building Buttress Pit	3	1,596
06108(4)	Unit 2 Turbine Building Buttress Pit	3	1,596
06201(4)	Unit 1 Turbine Building 570' Diesel Fuel Storage	1	813
06202(4)	Unit 2 Turbine Building 570' Diesel Fuel Storage	1	813
06209(4)	Unit 1 Steam Tunnel Floor Drain	3	47
06210(4)	Unit 2 Steam Tunnel Floor Drain	3	46
06211(4)	Unit 1 Tendon Tunnel Floor Drain	3	51
06212(4)	Unit 2 Tendon Tunnel Floor Drain	3	42
06213(4)	Unit 1 Steam Tunnel East Valve House	1	304
06214(4)	Unit 1 Steam Tunnel West Valve House	1	304
06215(4)	Unit 2 Steam Tunnel East Valve House	3	240
06216(4)	Unit 2 Steam Tunnel West Valve House	3	240
08100(5)	Crib House	3	8,435
08401(5)	Forebay	3	5,407
08102A&B(5)	Unit 1 and Unit 2 Circulating Water Intake Pipes	3	4,412
09100	Waste Water Treatment Facility	1	1,124

(1) Both survey units included in Release Record for Unit 1 Containment
(2) Both survey units included in Release Record for Unit 2 Containment
(3) The Release Record for the Turbine Building basement also includes the surface area of the Unit 1 and Unit 2 Steam Tunnels, the unit 1 and Unit 2 Circulating Water Discharge pipe and the Unit 1 and Unit 2 Circulating Water Discharge Tunnels
(4) Included as an "Appendixes" to the Turbine Building basement Release Record
(5) The Release Record for the Crib House also includes the FSS for the Forebay and the Unit 1 and Unit 2 Circulating Water Intake Pipes.
(6) Denote Final Survey Unit Classification

Figure 1-1 – Phase 2 Survey Unit Release Record Designation



Survey Unit #	Survey Unit Name	Release Record Color Designation
01100A	Unit 1 Containment above 565 ft	Red
01110A	Unit 1 Containment Under-Vessel Area	Red
01111A	Unit 1 Containment IC-Sump Drain	Red
01112A	Unit 1 Containment Penetrations	Red
02100A	Unit 2 Containment above 565 ft	Purple
02110A	Unit 2 Containment Under-Vessel Area	Purple
02112A	Unit 2 Containment Penetrations	Purple
03202A	SFP/Transfer Canal	Orange
05100A	Auxiliary Building 542 ft floor and walls	Blue
05119A	Auxiliary Building Embedded Floor Drains	Blue
05120A	Auxiliary Building Penetrations	Blue
06100A	Turbine Building Basement	Green
06100B	Turbine Building U1 & U2 Steam Tunnel	Green
06105A	Circulating Water Discharge Pipe	Green
06105B	Turbine Building Embedded Pipe	Green
06107A	Turbine Building Buttress Pit U1	Green
06108A	Turbine Building Buttress Pit U2	Green
06201A	Turbine Building 570' Diesel Fuel Storage U1	Green
06202A	Turbine Building 570' Diesel Fuel Storage U2	Green
06209A	Unit 1 Steam Tunnel Floor Drain	Green
06210A	Unit 2 Steam Tunnel Floor Drain	Green
06211A	Unit 1 Tendon Tunnel Floor Drain	Green
06212A	Unit 2 Tendon Tunnel Floor Drain	Green
06213A	Unit 1 Steam Tunnel East Valve House	Green
06214A	Unit 1 Steam Tunnel West Valve House	Green
06215A	Unit 2 Steam Tunnel East Valve House	Green
06216A	Unit 2 Steam Tunnel West Valve House	Green
08100A	Crib House	Yellow
08102A	Circulating Water Intake Pipes U1	Yellow
08102B	Circulating Water Intake Pipes U2	Yellow
08401A	Forebay	Yellow
09100A	Waste Water Treatment Facility (WWTF)	Cyan
09200B	Circulating Water Discharge Tunnels U1 & U2	Green

Figure 1-1
Survey Unit Release Record Designation

Date: 02/2019

1.2 Phased Submittal Approach

To minimize the incorporation of redundant historical assessment and other FSS program information, and to facilitate potential phased releases from the current licenses, FSS Final Reports are provided in a phased approach. Zion*Solutions* estimates that a total of five (5) FSS Final Reports will be generated and submitted to the United States Nuclear Regulatory Commission (USNRC) during the decommissioning project.

The Phase 1 FSS Final Report, which was submitted to the USNRC in October of 2018, encompassed the release of eight (8) Class 3 open land survey units.

The Phase 2A FSS Final Report will address buried pipe.

The Phase 3 FSS Final Report will include the open land survey units encompassing the southern portion of the site, and the Phase 4 FSS Final Report will include the open land areas encompassing the northern portion of the site.

1.3 Phase 2 Report

This Phase 2 FSS Final Report addresses the remaining basement structures. Specifically, this report includes the FSS results for the following:

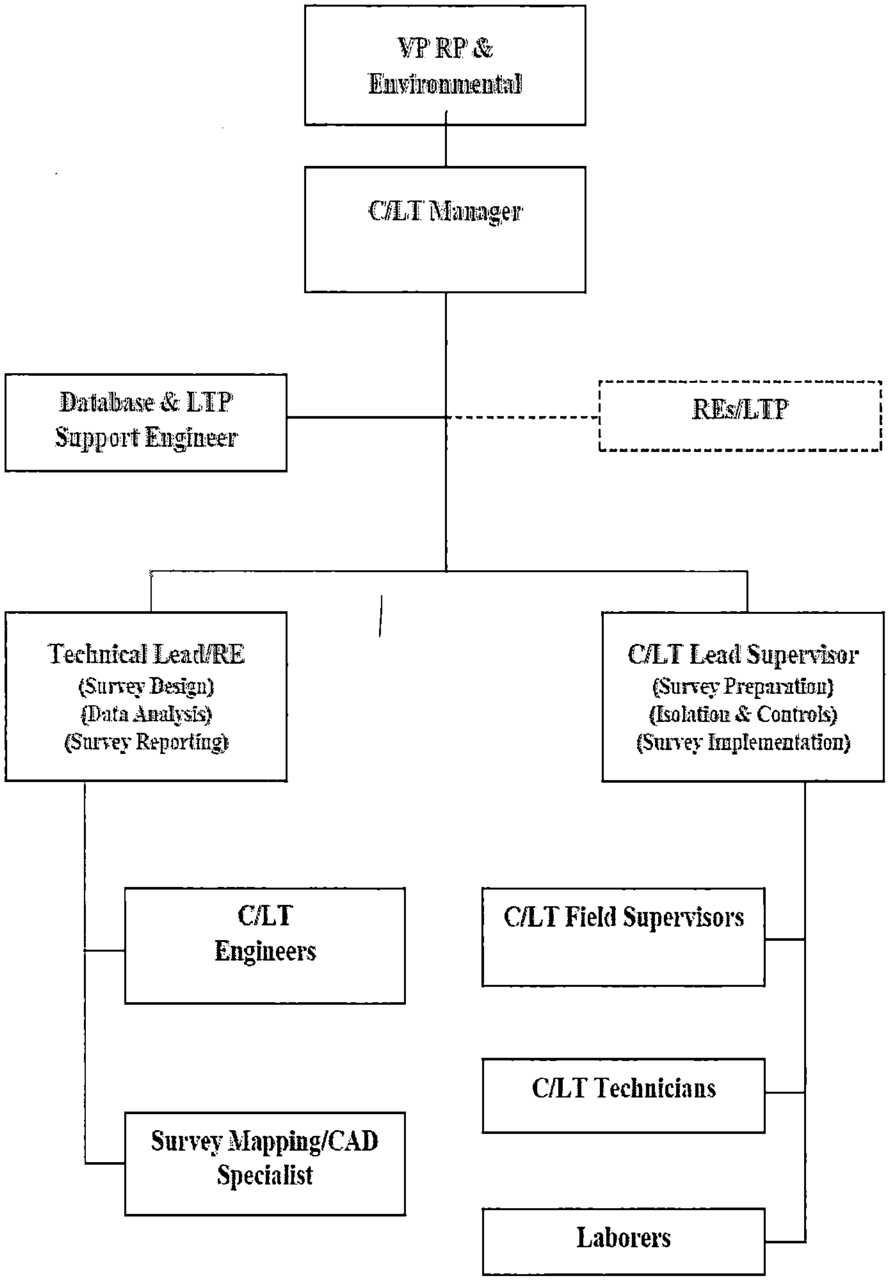
- Unit 1 Containment (including above 565 foot elevation, Under Vessel, Incore Sump drain, and penetrations),
- Unit 2 Containment (including above 565 foot elevation, Under Vessel, and penetrations),
- Spent Fuel Pool/Transfer Canal,
- Auxiliary Building (including 542 foot elevation, embedded floor drains, and penetrations),
- Turbine Building Basement (the main report includes the Turbine Building Basement Structure, which includes the area of the Steam Tunnels, Diesel Generator Rooms, Tendon Tunnels and Valve Houses, as well as the Circulating Water Discharge Pipe and Discharge Tunnels) with addendums addressing additional FSS performed in the Unit 1 and Unit 2 570 foot elevation Diesel Generator Cubicles Basement, Turbine Building Embedded Pipe, Unit 1 and Unit 2 Buttress Pits, Unit 1 and Unit 2 Steam Tunnel Floor Drains, Unit 1 and Unit 2 Circulating Water Intake Pipes, the Unit 1 and Unit 2 Tendon Tunnel Floor Drains, and the Unit 1 and Unit 2 Main Steam Valve Houses,
- Crib House (including the Forebay), and the
- Waste Water Treatment Facility.

2. FINAL STATUS SURVEY PROGRAM OVERVIEW

The FSS Program consists of the methods used in planning, designing, conducting, and evaluating FSS at the ZNPS site to demonstrate that the premises are suitable for release in accordance with the criteria for decommissioning in Title 10 CFR 20, Subpart E. Final Status Surveys (FSS) serve as key elements to demonstrate that the TEDE to an AMCG from residual radioactivity does not exceed 25 mrem per year, and that all residual radioactivity at the site is reduced to levels that are ALARA.

To implement the FSS Program, Zion*Solutions* established the C/LT Group, within the Radiation Protection division, with sufficient management and technical resources to fulfill project objectives. The C/LT Group is responsible for the safe completion of all surveys related to characterization and final site closure. Approved site procedures and detailed Technical Support Documents (TSD) direct the FSS process to ensure consistent implementation and adherence to the LTP and all applicable requirements. Figure 2-1 provides an organizational chart of the C/LT Group.

Figure 2-1 – Characterization/License Termination Group Organizational Chart



2.1 Survey Planning

Following the cessation of commercial operation, the development and planning phase for the decommissioning was initiated in 1999 by the "ComEd *Zion Station Historical Site Assessment*" (HSA) (Reference 9) and the initiation of the characterization process. The characterization process is iterative and will continue until, in some cases, up to the time of completing FSS. The HSA consisted of a review of site historical records regarding plant incidents, radiological survey documents, and routine and special reports submitted by Exelon to various regulatory agencies. Along with these assessments, interviews with current and past site personnel, reviews of historical site photos, and extensive area inspections were performed to meet the following objectives:

- Develop the information necessary to support FSS design, including the development of Data Quality Objectives (DQO) and survey instrument performance standards.
- Develop the initial radiological information to support decommissioning planning, including building decontamination, demolition, and waste disposal.
- Identify any unique radiological or health and safety issues associated with decommissioning.
- Identify the potential and known sources of radioactive contamination in systems, surface or subsurface soils, groundwater, and on structures.
- Divide the ZNPS site into manageable areas or units for survey and classification purposes.
- Determine the initial classification of each survey area or unit as non-impacted or impacted. Impacted survey areas or units are Class 1, 2, or 3, as defined in MARSSIM.

Data Quality Objectives are qualitative and quantitative statements derived from the DQO process that clarify technical and quality objectives, define the appropriate type of data, and specify the tolerable levels or potential decision errors used as the basis for establishing the quality and quantity of data required to support inference and decisions. This process, described in MARSSIM and procedure ZS-LT-300-001-001, "*Final Status Survey Package Development*," is a series of graded planning steps found to be effective in establishing criteria for data quality and guiding the development of FSS Sample Plans. Data Quality Objectives developed and implemented during the initial phase of planning directed all data collection efforts.

The DQO approach consists of the following seven steps:

2.1.1 State 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 required to perform the survey. The problem associated with FSS is to determine whether a given survey unit meets the radiological release criterion of 10 CFR 20.1402.

2.1.2 Identify the Decision

This step 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 principle study question. Alternative actions identify the 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 present in the survey unit exceed the established DCGL values? The alternative actions may include no action, investigation, resurvey, remediation, and reclassification.

2.1.3 Identify Inputs to the Decision

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

2.1.4 Define the Study Boundaries

The step includes identification of the target population of interest, the spatial and temporal features of that population, the 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. The medium of interest 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. Temporal boundaries include activities impacted by time-related events including weather conditions, season, and operation of equipment under different environmental conditions, resource loading, and work schedule.

2.1.5 Develop a Decision Rule

The step 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.

2.1.6 Specify Tolerable Limits on Decision Errors

This step incorporates hypothesis testing and probabilistic sampling distributions to control the decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition (the null hypothesis) to an alternative condition (the alternative hypothesis). Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided to reject it.

2.1.7 Optimize the Design for Obtaining Data

The final step in the DQO process leads to the development of an adequate survey design. By using an on-site analytical laboratory, sampling and analysis 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.

As stated, the primary objective of the DQO process was to demonstrate that the level of residual radioactivity found in the soils in the land area survey units, including any areas of elevated activity, was equal to or below the site-specific DCGLs that correspond to the 25 mrem/yr release criterion.

Each radionuclide-specific Base Case DCGL (BcDCGL) is equivalent to the level of residual radioactivity 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/year or less after all FSS is completed, the BcDCGLs are reduced based on an expected, or *a priori*, fraction of the 25 mrem/year dose limit from each source term. These reduced values are designated as Operational DCGLs (OpDCGL) (LTP Chapter 5, section 5.2.4) and these OpDCGLs are then used as the DCGL for the FSS design of the survey unit (calculation of surrogate DCGLs, investigations levels, etc.). Details of the OpDCGLs derived for each dose component and the basis for the applied *a priori* dose fractions are provided in Zion*Solutions* TSD 17-004, *"Operational Derived Concentration Guideline Levels for Final Status Survey"* (Reference 10).

Table 2-1 (reproduced from LTP Table 5-3) provides a listing for the BcDCGLs for the basement FSS units contained in this Phase 2 report.

Table 2-1 – Base Case DCGLs ($BcDCGL_B$) for Basements (pCi/m^2)

ROC	Auxiliary Building	CTMT	SFP/ Transfer Canal (1)	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

(1) The BcDCGL for the SFP/Transfer Canal set equal to the lower of either the Auxiliary Building or Containment BcDCGL. The Containment BcDCGLs were lower for all ROC, therefore the SFP/Transfer Canal BcDCGLs were set equal to Containment BcDCGLs.

Table 2-2 (reproduced from LTP Table 5-4) provides a listing for the OpDCGLs for the basement FSS units contained in this Phase 2 report.

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

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

Table 2-3 (reproduced from LTP Table 5-11) provides a listing for the BcDCGLs for Embedded Piping FSS units contained in this Phase 2 report.

Table 2-3 – Base Case DCGLs for Embedded Pipe ($BcDCGL_{EP}$)

ROC	Auxiliary Bldg. Basement Embedded Floor Drains (pCi/m²)	Turbine Bldg. Basement Embedded Floor Drains (pCi/m²)	Unit 1 & Unit 2 Containment Incore Sump Embedded Drain Pipe (pCi/m²)	Unit 1 & Unit 2 Steam Tunnel Embedded Floor Drains (pCi/m²)	Unit 1 & Unit 2 Tendon Tunnel Embedded Floor Drains (pCi/m²)
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

Table 2-4 (reproduced from LTP Table 5-12) provides a listing for the OpDCGLs for Embedded Piping FSS units contained in this Phase 2 report.

Table 2-4 – Operational DCGLs for Embedded Pipe ($OpDCGL_{EP}$)

ROC	Auxiliary Bldg. Basement Embedded Floor Drains (pCi/m²)	Turbine Bldg. Basement Embedded Floor Drains (pCi/m²)	Unit 1 & Unit 2 Containment Incore Sump Embedded Drain Pipe (pCi/m²)	Unit 1 & Unit 2 Steam Tunnel Embedded Floor Drains (pCi/m²)	Unit 1 & Unit 2 Tendon Tunnel Embedded Floor Drains (pCi/m²)
H-3	N/A	N/A	6.62E+08	N/A	3.22E+08
Co-60	7.33E+09	2.52E+08	4.38E+08	1.63E+09	2.12E+08
Ni-63	2.78E+11	7.84E+09	1.12E+10	5.04E+10	5.44E+09
Sr-90	2.41E+08	2.78E+06	3.98E+06	1.79E+07	1.94E+06
Cs-134	5.10E+09	5.72E+07	8.40E+07	3.69E+08	4.08E+07
Cs-137	2.68E+09	7.56E+07	1.10E+08	4.88E+08	5.34E+07
Eu-152	N/A	N/A	1.02E+09	N/A	4.96E+08
Eu-154	N/A	N/A	8.88E+08	N/A	4.32E+08

Table 2-5 (reproduced from LTP Table 5-13) provides a listing for the BcDCGLs for Penetration FSS units contained in this Phase 2 report.

Table 2-5 – 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 $BcDCGL_{PN}$ 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.

Table 2-6 (reproduced from LTP Table 5-14) provides a listing for the OpDCGLs for Penetration FSS units contained in this Phase 2 report.

Table 2-6 – 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

The development of information to support decommissioning planning and execution was accomplished through a review of all known site radiological and environmental records. Much of this information was consolidated in the HSA, Zion*Solutions* TSD 14-028, "*Radiological Characterization Report*" (Reference 11), and in files containing copies of records maintained pursuant to Title 10 CFR 50.75(g) (1). These documents are discussed further in applicable sections of this report.

An initial objective of site characterization and assessment was to correlate the impact of a radiological event to physical locations on ZNPS site and to provide a means to correlate subsequent survey data. To satisfy these objectives, the entire 331 acre site was divided into survey areas. Survey area size determination was 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 contamination were classified as non-impacted. These areas had no radiological impact from site operations and are identified in the HSA. Survey areas with reasonable potential for contamination were classified as impacted.

Classification, as described in MARSSIM, is the process by which an area or survey unit is described according to its radiological characteristics and potential for residual radioactivity. Residual radioactivity could be evenly distributed over a large area, appear as small areas of elevated activity, or a combination of both. In some cases, there may be no residual radioactivity in an area or survey unit. Therefore, the adequacy and effectiveness of the FSS process depends upon properly classified survey units to ensure that areas with the highest potential for contamination receive a higher degree of survey effort.

The impacted survey areas established by the HSA were further divided into survey units.

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 2-7, which is reproduced from Table 5.9 from MARSSIM.

Table 2-7 – 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 (typically <10%)	Number of sample/measurement locations for statistical test

Prior to FSS, each survey unit's classification was reviewed and verified in accordance with the LTP and its implementing procedures. A classification change to increase the class may be implemented without notification to regulatory authorities. A classification change to decrease the class may be implemented only after accurate assessment and notification to regulatory authorities as detailed in the LTP and its implementing procedures. Final classification was performed in conjunction with the preparation of the FSS Sample Plan. The Sample Plan reconciles all outstanding characterization data into the final characterization.

2.2 Survey Design

Final Status Surveys for the ZNPS site are designed following Zion*Solutions* procedures, the LTP, and MARSSIM guidance. FSS design utilizes the combination of traditional scanning surveys, systematic sampling protocols and investigative/judgmental methodologies to evaluate survey units relative to the applicable release criteria within each survey plan.

To aid in the development of an initial suite of potential radionuclides of concern (ROC) for the decommissioning of ZNPS, the analytical results of representative characterization samples collected at the site were reviewed. In general, the samples associated with these results were collected from within various waste/process streams and sent off site to meet the analysis criteria of 10 CFR 61, Subparts C and D. This initial suite of potential radionuclides was further refined by the Containment and Auxiliary Building concrete core data analysis. This analysis determined that Co-60, Cs-134, Cs-137, Ni-63, and Sr-90 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. Since activated concrete will be removed and disposed of as waste, the final suite of ROC for all areas outside of the Containments does not include H-3, Eu-152, and Eu-154.

The final suite of potential radionuclides and the mixture is provided in Table 2-8 (reproduced from LTP Table 5-2).

Table 2-8 – 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).

Characterization results determined that Co-60 and/or Cs-137 would be the primary ROC for the majority of survey design. Cs-137 characterization data for the survey units discussed in this report were used to determine the expected variability, number of samples required, and investigation levels for FSS design.

The dose contribution from each ROC was accounted for using the Sum of Fractions (SOF) to ensure that the total dose from all ROC did not exceed the dose criterion. The SOF or unity rule was applied to the data used for the survey planning, and data evaluation and statistical tests for soil sample analyses since multiple radionuclide-specific measurements were performed or the concentrations inferred based on known relationships. The application of the unity rule served 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 $OpDCGL_W$ (used for the nonparametric statistical test) becomes one (1). The use and application of the unity rule was performed in accordance with section 4.3.3 of MARSSIM.

Survey design objectives included a verification of the survey instrument's ability to detect the radiation(s) of interest relative to the DCGL. As standard practice to ensure that this objective was consistently met, radiation detection instruments used in FSS were calibrated on a yearly frequency with a National Institute of Standards and Technology (NIST) traceable source in accordance with Zion*Solutions* procedures. Instruments were response checked before and after use. Minimum Detectable Count Rates (MDCR) were established and verified prior to FSS. Control and accountability of survey instruments were maintained and documented to assure quality and prevent the loss of data.

The level of effort associated with planning a survey is based on the complexity of the survey, structural interferences/limitations, and the nature of the hazards. Guidance for preparing FSS plans was provided in procedure ZS-LT-300-001-001 "*Final Status Survey Package Development*".

The FSS of basement structures was primarily performed using the ISOCS. Basement structures are defined as basement surfaces (concrete and steel liners). As described in the LTP section 5.4.5, remaining floor and wall concrete surfaces were remediated to levels below the $OpDCGL_B$ as measured by ISOCS. After remediation, FSS was conducted to demonstrate that the residual radioactivity in building basements corresponded to a dose below the 25 mrem/year criteria. 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 up to 52 m^2 (e.g. see Release Records for Unit 1 and Unit 2 Turbine Building 570 foot Diesel Fuel Storage presented in Appendix 10) which essentially eliminates the need for scan surveys except in the case of penetrations and embedded piping.
- 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 provide results that were 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.

ISOCS geometries are provided in Zion*Solutions* TSD 14-022, "*Use of In-Situ Gamma Spectroscopy for Source Term Survey of End State Structures*" (Reference 12). Continuing characterization concrete core data was used to validate that the proper geometries were applied to ISOCS measurements.

Based on the contamination potential of each FSS unit, along with the corresponding areal coverage, the number of ISOCS measurements required in each FSS unit was calculated as the quotient of the ISOCS Field Of View (FOV) divided into the surface area required for areal coverage. Table 5-19 of the LTP presents the FSS units, the classification based on contamination potential, the surface area to be surveyed and the minimum number of ISOCS measurements that were 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 LTP section 5.6.4.1. This calculation was applied to the 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 was performed, the number of measurements met or exceeded that required by the statistical test.

For embedded pipe and penetration surveys, the level of effort associated with planning a survey was based on the complexity of the survey and nature of the hazards. Guidance for preparing FSS plans was provided in procedure ZS-LT-300-001-001 "*Final Status Survey Package Development.*" The FSS plans for the survey of pipes and penetrations employed sample designs that combined hand-held scanning with static measurements and pipe detector survey methodologies.

The survey method for large diameter pipes and penetrations (>12") differed from smaller penetrations due to measurement sensitivity (i.e. Minimum Detectable Concentration) differences in the two size regimes. The larger diameter penetrations were surveyed using a similar approach as for traditional building surface surveys whereas the smaller diameter pipes and penetrations were surveyed with a single detector advanced through the length of the pipe interior in nominal 1-foot increments.

For pipe surveys, the detector efficiencies were determined for each instrument using a wide range of pipe interior diameters and geometries with NIST traceable planar sources. These pipe detectors and instruments were utilized predominantly on pipes and penetrations with diameters less than or equal to 12 inches. They were also used for larger diameter penetrations whose length was significantly greater than the typical depth of wall and floor penetrations (e.g., greater than 10 feet in length). For penetrations greater than 12 inches in diameter, hand held scanning instruments (proportional beta and beta scintillator detectors) were used to scan and perform static counts and the efficiencies for these utilized either conservative efficiencies for these instruments, or the actual efficiency for specific instrument and detector combination calibration records.

Designated samples were sent to an off-site laboratory for Hard-to-Detect (HTD) radionuclide specific analysis. Laboratory DQO and analysis results are summarized in Release Records and reported as actual calculated results. Sample report summaries within the Release Records includes unique sample identification, analytical method, radioisotope, result, uncertainty of two standard deviations, laboratory data qualifiers, units, and required Minimum Detectable Concentration (MDC).

Another consideration of survey design was the use of surrogates. In lieu of analyzing every sample for HTD radionuclides, the development and application of Surrogate Ratio DCGLs as described in MARSSIM, section 4.3.2 was applied to estimate HTD radionuclides. Surrogate ratios allow for expedient decision making in characterization, remediation planning, or FSS design.

A surrogate is a mathematical ratio where an Easy-to-Detect (ETD-gamma emitter) radionuclide (i.e., Cs-137) concentration is related to a HTD radionuclide (i.e., Sr-90) concentration. From the analytical data, a ratio is developed and applied in the survey scheme for samples taken in the area. Details and applications of this method are provided in section 5.2.11 of the LTP.

Due to the lack of significant activity revealed during background studies, assessments and characterization, it was determined that background subtraction would not be applied during FSS.

2.3 Survey Implementation

Final Status Survey implementation of the Turbine Building Phase 2 survey units commenced in March of 2016. FSS implementation for the remaining Phase 2 survey units commenced in December of 2017. Implementation was the physical process of the FSS Sample Plan execution for a given survey unit. Each Sample Plan was assigned to an Radiological Engineer (RE) for implementation and completion in accordance with the LTP, Zion*Solutions* procedures and the QAPP for Characterization and FSS. A walk-down and turnover survey was performed for each FSS survey unit in accordance with the Isolation and Control requirements of procedure ZS-LT-300-001-003. A turnover survey was performed within each FSS survey unit and consisted of surveys for loose surface contamination as well as the acquisition of several ISOCS measurements.

The tasks included in the implementation were:

- Verification and validation of personnel training as required by Training Department and Radiation Protection procedures.
- Monitoring instrument calibration and routine performance checks, as detailed in ZS-RP-108-000-000, "*Radiological Instrumentation Program*" (Reference 13) and ZS-RP-108-004-012, "*Calibration and Initial Set Up of the 2350-1*" (Reference 14).
- Implementation of applicable operating and health and safety procedures.
- Implementation of isolation of control of the survey unit in accordance with ZS-LT-300-001-003, "*Isolation and Control for Final Status Survey.*"

- Determination of the amount of surveys and sampling required to meet DQOs as described in ZS-LT-300-001-001, "*Final Status Survey Package Development.*"
- Determination that the ISOCS geometries used were in accordance with Zion*Solutions* TSD 14-022 Revision 2, Addendum 1, "*Use of In-Situ Gamma Spectroscopy for Source Term Survey of End State Structures*".
- Validation proper operation of the ISOCS in accordance with ZionSolutions TSD 17-003, *"Evaluation of Efficiency Calibration Geometries for In-Situ Gamma Spectrometry During Final Status Surveys"* (Reference 15)
- Determination of ISOCS measurement locations, core sample locations and creation of survey unit maps displaying the locations in accordance with ZS-LT-300-001-001.
- Proper techniques for collecting and handling FSS samples in accordance with Job Aid LT-JA-004, "*FSS Sample Collection*" (Reference 16).
- Maintaining Quality Assurance/Quality Control requirements (i.e., replicate measurements or samples) in accordance with the QAPP for Characterization and FSS.
- Sample Chain of Custody (CoC) maintained in accordance with ZS-LT-100-001-004, "*Sample Media Preparation for Site Characterization*" (Reference 17).
- Sample submission to approved laboratories in accordance with ZS-WM-131, "*Chain of Custody Protocol*" (Reference 18).
- Application of the DCGLs to sample results in accordance with the Data Quality Assessment (DQA) process as detailed in ZS-LT-300-001-004, "*Final Status Survey Data Assessment.*"
- Determination of investigation methodology and corrective actions, if applicable.

The FSS implementation and completion process resulted in the generation of field data and analysis data consisting of measurements taken with handheld radiation detecting equipment, observations noted in field logs, and radionuclide specific analysis. Data were stored electronically on the Zion*Solutions* common network.

2.4 Survey Data Assessment

Prior to proceeding with data evaluation and assessment, the assigned RE ensured consistency between the data quality and the data collection process and the applicable requirements.

The DQA process is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement with the FSS Sample Plan objectives. A key step in the data assessment process converts all of the survey results to DCGL units, if necessary. The individual measurements and sample concentrations are compared to the DCGL for evidence of small areas of elevated activity or results that are statistical outliers. When practical, graphical analyses of survey data that depicts the spatial correlation of the measurements was used.

The DQO process was employed to determine the ROC for each FSS unit in this report. During FSS, concentrations for HTD ROC H-3 (for Containments), Ni-63 and Sr-90 were inferred using a surrogate approach. Cs-137 is the principle surrogate radionuclide for both H-3 and Sr-90. Co-60 is the principle surrogate radionuclide for Ni-63. The mean, maximum and 95% Upper Confidence Level (UCL) were calculated in TSD 14-019, *"Radionuclides of Concern for Soil and Basement Fill Model Source Terms"* (Reference 19) and are presented in LTP Table 5-15. The maximum ratios were used to infer HTD concentrations during FSS unless area specific ratios were determined. In these cases, the ratios used and their basis are described in the individual Release Record.

In accordance with LTP Chapter 5, section 5.6.4.1.1, the Type I decision error 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 $OpDCGL_B$. The Lower Bound of the Gray Region (LBGR) was set at the expected fraction of the $OpDCGL_B$. The expected fraction of the $OpDCGL_B$ in the Class 1 and Class 2 FSS units was set at 50% and the expected fraction of the $OpDCGL_B$ in the Class 3 FSS units was set at 1%. LTP, Table 5-19 presents the basement surface FSS units and the adjusted number of ISOCS measurements that will be taken in each for FSS.

2.5 Quality Assurance and Quality Control Measures

Quality assurance and control measures were employed throughout the FSS process to ensure that all decisions were based on data of acceptable quality. Quality assurance and control measures were applied to ensure:

- The plan was correctly implemented.
- The DQA process was used to assess results.
- DQOs were properly defined and derived.
- All data and samples were collected by individuals with the proper training and in adherence to approved procedures and sample plans.
- All instruments were properly calibrated and routinely performance checked.
- All collected data was validated, recorded, and stored in accordance with approved procedures.
- All required documents were properly maintained.
- Corrective actions were prescribed, implemented and tracked, as necessary.

Independent laboratories used for analysis of the samples collected during FSS maintain Quality Assurance Plans designed for their facility. Zion*Solutions* reviewed those plans, as required by ZS-QA-10, "*Quality Assurance Project Plan*" (Reference 20) and the QAPP for Characterization and FSS, prior to selection. In addition, regular vendor performance reviews, audits and/or surveillances of these laboratories were performed to ensure an adequate level of quality.

The Zion*Solutions* Quality Assurance (QA) department provided oversight of the C/LT Group on a consistent basis throughout the project at the Zion Station Restoration Project (ZSRP). QA surveillances have scrutinized the LTP, C/LT procedures, Sample Plans, and C/LT records. The responses to the QA surveillances are captured in the Corrective Action Program (CAP).

3. SITE INFORMATION

3.1 Site Description

Zion Nuclear Power Station, owned by Exelon Nuclear Generation, LLC (Exelon), is located in Zion, 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 owner-controlled site consists of approximately 331 acres, and within the owner-controlled area is an approximate 87-acre, fence-enclosed nuclear facility. 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 no residences are within 2,000 feet of any ZNPS structures.

Westinghouse Electric Corporation, Sargent and Lundy Engineers, and the Commonwealth Edison Company (ComEd) jointly participated in the design and construction of ZNPS. The plant was comprised of two pressurized water reactors with supporting facilities. The primary coolant system for each unit employed a four-loop pressurized water reactor nuclear steam supply system housed in a steel-lined, reinforced concrete containment structure. Each unit employed a pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation, designed for a power output of 3,250 MWt. The equivalent warranted gross and approximate net electrical outputs of the plant were 1085 MWe and 1050 MWe, for Unit 1 and Unit 2 respectively.

ZNPS was previously operated by Commonwealth Edison until it was permanently shut down on February 13, 1998. On March 9, 1998, ComEd certified to the USNRC that all fuel assemblies had been permanently removed from both reactors and placed in the Spent Fuel Pool. The USNRC acknowledged the certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessels in a letter dated May 4, 1998. In 2000, the licenses were transferred from ComEd to Exelon. In 2008, the licenses were transferred to Zion*Solutions* to coordinate and execute the decommissioning of the site. The Post Shutdown Decommissioning Activities Report (PSDAR) (Reference 21) was submitted, in accordance with 10 CFR 50.82(a), in February 2000 and accepted by the USNRC. An amended PSDAR was submitted in March 2008 to accommodate the transfer of the 10 CFR 50 licenses to Zion*Solutions* and to revise cost estimates and the decommissioning schedule. The Defueled Safety Analysis Report (DSAR) (Reference 22) was updated in October 2016. An evaluation of the systems, structures, and components (SSCs) was performed to determine the function these systems would perform in a defueled condition. With the relocation of the spent fuel to the Independent Spent Fuel Storage Installation (ISFSI), the license basis for the majority of the SSCs was changed and only minimal SSCs were needed to support the ongoing active decommissioning. The remaining SSCs needed to support active decommissioning had controls established in the DSAR and the Offsite Dose Calculation Manual (ODCM) (Reference 23).

On November 2, 2011, site characterization commenced. At the time these surveys were performed, the site-specific Zion*Solutions* characterization plans and procedures were still under development. Consequently, due to schedule restraints, Zion*Solutions* contracted the Energy*Solutions* Commercial Services Group (ESCSG) to perform characterization of the ISFSI location, the Vertical Concrete Cask (VCC) Construction Area, and the pathway for the new rail track. The results of these surveys were validated and integrated into the subsequent site-specific characterization program, which was approved in February 2012. Initial scheduled site characterization efforts concluded on November 11, 2013. The results of site characterization are presented in LTP Chapter 2 as well as TSD 14-028.

3.2 Survey Unit Description

The following information is a description of each survey unit at the time of FSS from April of 2016 (for the Turbine Building) through August of 2018 (for the WWTF). During this period, thirty-one (31) FSS survey units were completed and are presented in this Phase 2 Final Report.

3.2.1 Survey Units 01100 and 01110 (Unit 1 Containment above 565 foot elevation. and Unit 1 Containment Under Vessel Areas)

The Unit 1 Containment basement survey units (survey unit 01100 and survey unit 01110) are impacted Class 1 basement FSS units. The Containment basement structure is located within Class 1 open land survey units 12107, 12108 and 12109.

Final Status Survey unit 01100 encompasses the Unit 1 Containment above the 565 foot elevation. The Unit 1 Containment structure housed the Unit 1 Reactor Vessel, Steam

Generators and Pressurizer. The HSA noted several occasions of radioactive liquid spill events during plant operation.

Final Status Survey unit 01110 housed the Unit 1 Incore flux monitoring tubes and associated supports. This survey unit is the concrete structure around and beneath the reactor void space (565 foot elevation and below) to remain at license termination. It provided personnel access to the area under the reactor vessel and housed the Incore sump for collection and recovery of liquids released into the area.

The Incore area extends below the containment slab and consists of a cylindrical area directly under the reactor vessel biological shield and a sloped tunnel. The Incore area walls are 1 foot 11.5 inches thick (23.5 inches) with a 2 foot 6 inches under vessel area floor thickness. There is also an access tunnel with 15 inch thick walls, floor and roof.

In accordance with the planned end state configuration, the concrete floor of the 568 foot elevation was removed to the ½-inch steel liner. In this end state configuration, survey unit 01100 consisted of the interior side of the steel liner walls below the 588 foot elevation and the 565 foot elevation liner floor. The survey unit also contains the Cavity Flood sump and Recirculation sump.

Based upon completion of the Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct classification of survey units 01100 and 01110 was determined to be Class 1.

3.2.2 Survey Unit 01111 (Unit 1 Containment Incore Sump Discharge Pipe)

The Unit 1 Containment Incore Sump Discharge Pipe is 1.61 inch Internal Diameter (ID) embedded pipe located in the concrete of the Incore Access Tunnel in Unit-1. The Incore area extends below the containment slab and consists of a cylindrical area directly under the reactor vessel biological shield and a sloped tunnel. The sump is 2' x 2' x 2' approximately 1 foot from the bend line. The pipe enters the wall at the floor above the sump. The pipe has an estimated length of 26.74' (8.15 meters) and a total surface area of 1.05 m^2.

Survey unit 01111 was classified in accordance with Zion*Solutions* procedure ZS-LT-300-001-002, "*Survey Unit Classification*" (Reference 24).

Based on information from the HSA, the Incore Sump Discharge Pipe is located in a Class 1 area. The Under Vessel Incore area was subjected to operational conditions as well as the exercising of the Incore detectors. The Unit 1 Incore Sump Discharge Pipe contained radioactive material and was classified as a Class 1 system.

3.2.3 Survey Unit 1112 (Unit 1 Containment Penetrations)

The Unit 1 Containment Building contained, as documented in Zion*Solutions* TSD 14-016, *"Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Zion End State"* (Reference 25), sixty-one (61) penetrations identified as being present within the survey unit.

The End State condition depicted in TSD 14-016 was altered due to D&D activities and observations made during survey design and walk-down. Eight (8) penetrations listed for Unit 1 Containment in TSD 14-016 were above the basement End State 588 foot elevation and were removed prior to FSS. One of the penetrations identified in TSD 14-016, was the Unit 1 Containment Incore Sump Drain (P-125, with an ID of 1.6 inches, addressed in the Release Record for survey unit 01111 [See Appendix 2]). Lastly, the Spent Fuel Transfer Tube, P-049, was also removed, leaving an 8 foot square opening, to permit ISOCS and personnel access and egress from the Unit 1 Containment. Therefore, the total number of penetrations surveyed as part of this survey unit, was reduced to sixty-one (61).

The penetrations ranged in size from six (6) inches to fifty (50) inches in diameter. A summary of the original end state lengths and surface areas for the Unit 1 Containment Building Penetrations are depicted in TSD 14-016.

Penetrations and embedded pipe are defined in LTP Chapter 5, section 5.5.5 which states, "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 Unit 1 Containment Building Penetrations were initially classified as Class 1, Class 2, or Class 3, based on historical assessment, exposure to radioactive materials and system use. Based upon completion of the Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct final classification of penetrations within Unit 1 were validated. As a conservative measure, the classifications of all Unit 1 Containment Building Penetrations were changed to Class 1. Consequently, sufficient measurements were taken in all Unit 1 Containment Building Penetrations to ensure 100% areal coverage of all accessible internal surfaces within the penetrations.

3.2.4 Survey Units 02100 and 02110 (Unit 2 Containment above 565 foot elevation and Unit 1 Containment Under Vessel Areas)

Survey units 02100 and 02110 are located in the Unit 2 Containment basement and are impacted Class 1 basement FSS units. The Unit 2 Containment basement structure is located within Class 1 open land FSS unit's 12201, 12104 and 12105.

Final Status Survey unit 02110 housed the Unit 2 Incore flux monitoring tubes and associated supports. This survey unit is the concrete structure around and beneath the reactor void space (565 foot elevation and below) to remain at license termination. It provided personnel access to the area under the reactor vessel and housed the Incore sump for collection and recovery of liquids released into the area.

In accordance with the planned end state configuration, the concrete floor of the 568 foot elevation has been removed to expose the ½-inch steel liner. In this end state configuration, FSS unit 02100A consisted of the interior side of the steel liner walls below the 588 foot elevation and the 565 foot elevation liner floor. The survey unit also contains Cavity Flood Sump and the Recirculation Sump. The bottoms of both sumps are located at the 559 foot elevation.

Prior to remediation, the configuration of FSS unit 02110 included the concrete and embedded steel support rings interior to the steel liner below the 565 foot elevation. Prior to remediation, the circular concrete walls directly under the Reactor Vessel were 23.5 inches thick and the concrete floor was 30 inches thick. The access tunnel had concrete walls, floor and roof that were 15 inches thick.

The ZSRP performed extensive remediation of the concrete located in the Under Vessel area below the 565 foot elevation in Unit 2 Containment. Scabbling and hammering demolition techniques were used to remove at least six inches of concrete from the floor and walls located directly under the reactor vessel and at least six inches of concrete from the walls and slanted floor of the access tunnel. In some places, sufficient concrete was removed to expose the steel liner. Parts of the 0.5″ steel support rings were also removed. Also, during remediation, Pipe P325, the Unit 2 Containment Incore Sump Drain header, which was embedded in the concrete of the tunnel walls, was completely removed and disposed of as radioactive waste.

Based upon completion of the Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct classification of survey units 02100 and 02110 was determined to be Class 1.

3.2.5 Survey Unit 02112 (Unit 2 Containment Penetrations)

The Unit 2 Containment Building contained, as documented in Zion*Solutions* TSD 14-016, sixty-two (62) penetrations within the survey unit.

The End State condition depicted in TSD 14-016 was altered due to D&D activities and observations made during survey design and walk-down. Eight (8) penetrations listed for Unit 2 Containment in TSD 14-016 were above the basement end state 588 ft. elevation and were removed prior to FSS. Lastly, the Spent Fuel Transfer Tube, P-249, was also removed, leaving an 8 foot square opening, to permit ISOCS and personnel access and egress from the Unit 2 Containment. Therefore, the total number of penetrations surveyed as part of this survey unit was sixty-one (61).

The penetrations ranged in size from six (6) inches to fifty (50) inches in diameter. A summary of the original End State lengths and surface areas for the Unit 2 Containment Building Penetrations as depicted in TSD 14-016.

The Unit 2 Containment Building Penetrations were initially classified as Class 1, Class 2, or Class 3, based on historical assessment, exposure to radioactive materials and system use. Based upon completion of the Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct final classification of penetrations within Unit 2 were validated. As a conservative measure, the classifications of all Unit 2 Containment Building Penetrations were changed to Class 1. Consequently, sufficient measurements were taken in all Unit 2 Containment Building Penetrations to ensure 100% areal coverage of all accessible internal surfaces within the penetrations.

3.2.6 Survey Unit 03202 (Spent Fuel Pool/Transfer Canal)

The Fuel Handling Building was located between the Unit 1 and Unit 2 Containments and adjacent to the Auxiliary Building. The structure was designed for the storage of new and spent fuel. Major support systems that were located in the Fuel Handling Building included the SFP Heat Exchangers and SFP Skimmer Pumps. The SFP was a 63 ft. long by 33 ft. wide by 40 ft. deep pool located in the east half of the building. The pool was filled with borated water and contained 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 were located in the western portion of the building. A cask decontamination pit was 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 were radiologically contaminated internally. The SFP, the decontamination pit, and the equipment cubicles were all posted as "Contaminated Areas."

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 were removed and disposed of as radioactive or non-radioactive waste, as appropriate.

The Fuel Handling Building structure located above the 588 foot elevation was completely demolished. The remaining structure following demolition consisted of the lower portion of the SFP and the Fuel Transfer Canal foundation floors and walls. The west wall of the SFP was reduced to ~ 6 feet in height to allow heavy equipment to enter the SFP floor and remove the steel liner. The east wall was also completely removed to provide access to the Transfer Canal liner.

The area of this structural survey unit is approximately 7,783 ft^2 or 723 m^2.

3.2.7 Survey Unit 05100 (Auxiliary Building Basement)

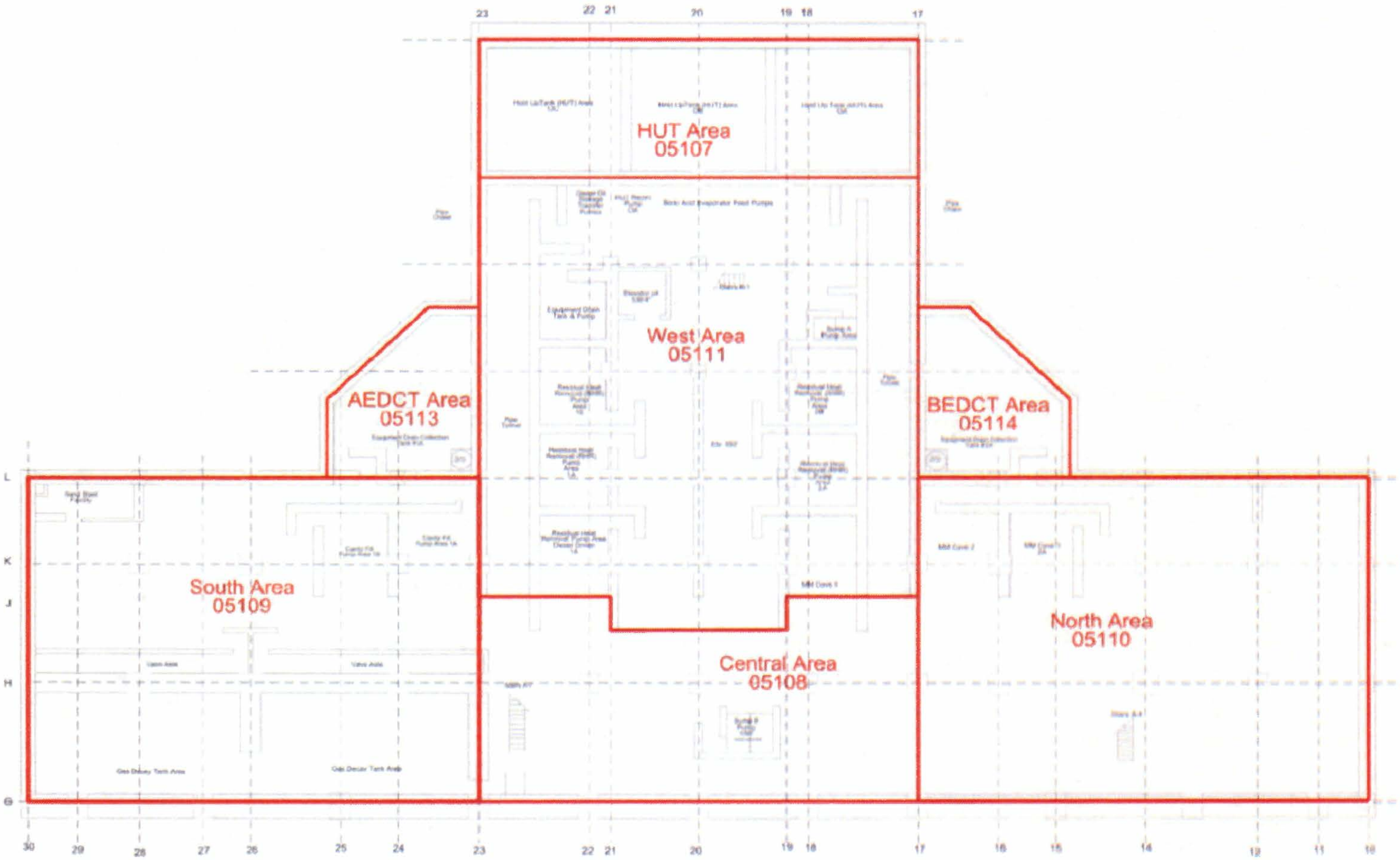
The Auxiliary Building footprint contained numerous systems and components including the following: Hold Up Tank (HUT) system components, Boric Acid Evaporator feed pumps, Drain collection tanks and piping, Safety Injection system, Residual Heat Removal system components, Containment spray, Chemical drain, Blowdown Heat Exchanger, Waste gas, Charging pumps, Refuel water storage tanks, Letdown heat exchanger, etc.

The Auxiliary Building basement survey unit is a Class 1 basement FSS unit. The Auxiliary Building basement survey unit is comprised of the combined exterior wall and floor surfaces of each remaining building basement from the 542 foot, 560 foot and 579 foot elevations, following demolition. This survey unit consists of the Auxiliary Building basement floor at the 542 foot elevation, the two horizontal surfaces beneath the Unit 1 and Unit 2 primary piping penetrations at the 560 foot elevation, the two horizontal surfaces above the Auxiliary Building Equipment Drain Collection Tank (ABEDCT) areas at the 579 foot elevation, and all the associated walls below the 588 foot elevation.

The Auxiliary Building housed numerous systems containing radioactively contaminated support systems. System leakage and maintenance activities over the operating life of the reactor resulted in the radiological contamination of most of the interior surfaces of the structures. Based on the building design basis and the operating history, all internal survey units in Auxiliary Building were assigned an initial classification of Class 1 in accordance with the HSA.

A map of the 542 foot elevation of the Auxiliary Building is provided in Figure 3-1.

Figure 3-1 – Auxiliary Building 542' Elevation



Auxiliary Building 542'
Prior to Demolition

3.2.8 Survey Unit 05119 (Auxiliary Building 542 ft. Embedded Floor Drains)

The Auxiliary Building 542 ft. elevation embedded equipment and floor drain survey unit consists of 28 different pipes ranging from 4-inch to 6-inch in diameter. The floor drain system consists of approximately 2,721 linear feet of floor drain pipe, embedded 4 feet deep in the concrete floor in 28 pipe headers that are accessed by 125 drain openings and terminate in one of the two Auxiliary Building sumps. Sump A serviced 10 pipe headers in the west portion of the basement and Sump B serviced 18 pipe headers in the east portion of the basement.

3.2.9 Survey Unit 05120 (Auxiliary Building Penetrations)

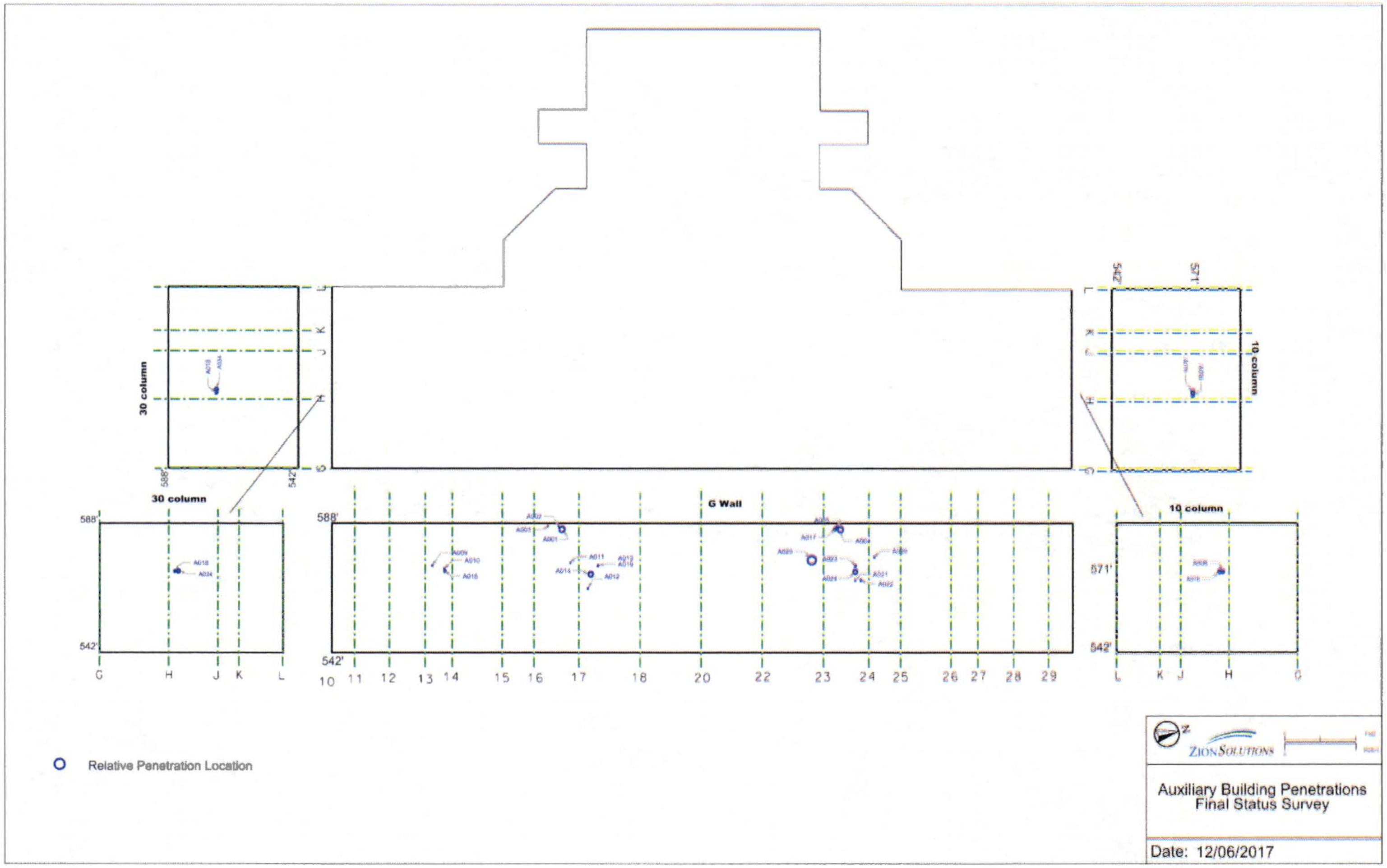
The Auxiliary Building penetrations survey unit consists of one hundred and five (105) penetrations that accessed the Auxiliary Building between the 542 foot, 560 foot and the 579 foot elevation. However, seventy-nine (79) of the 105 penetrations were identified as being both an Auxiliary Building and Containment Building penetration. Since the Containment DCGLs are more limiting than the Auxiliary Building DCGLs, the penetrations identified as being both Auxiliary and Containment were addressed in the Release Records for the Containment penetrations.

The remaining twenty-six (26) penetrations that accessed the Auxiliary Building between the 542, 560 and the 579 foot elevations, but did not access the Containments, were A001-A025, and A034.

The Auxiliary Building penetrations housed numerous primary contaminated systems. The location of the penetrations, their function, and the operational history of the Auxiliary Building to support the initial classifications are described in TSD 14-016.

See Figure 3-2 for a map of the relative locations of the Auxiliary Building penetrations.

Figure 3-2 – Auxiliary Building Penetration Map



3.2.10 Survey Unit 06100 (Turbine Building Basement)

The Turbine Building footprint is located within Class 1 open land survey units 12205A, 12205B, 12205C, 12205D and 12205E.

The Turbine Building housed 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 internal structures that supported the Condensers, Turbine and Generators are solid concrete below the 588 foot elevation. The Circulating Water Intake and Discharge pipes are embedded in concrete above the 560 foot elevation. The floors of the Unit 1 and Unit 2 Steam Tunnels are at the 570 foot elevation and the floors of the Unit 1 and Unit 2 Diesel Generator Oil Storage rooms are at the 567 foot elevation. The Turbine Building sits on top of the Circulating Water Discharge Tunnels. The floor of the Unit 1 and Unit 2 Turbine Building basement is at the 560 foot elevation and has a Common Area between them. The Unit 1 and 2 areas are mirror images of each other.

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, including the turbines, generator, moisture separator re-heaters, feed water heaters and coolers. In parallel with this effort, the surveys were performed for the unconditional release of materials, equipment and structural surfaces throughout the building.

The Unit 1 and Unit 2 Circulating Water Discharge Pipe and Discharge Tunnels provided for the discharge of cooling water, primarily from the Main Condensers but also from ancillary system cooling systems to Lake Michigan. The Circulating Water Tunnels were also the main authorized effluent release path to Lake Michigan for the release of treated and filtered radioactive liquid effluent. The tunnels run under the Turbine Building where two 12 foot diameter Circulating Water Discharge pipes opens into the tunnels from above. The tunnels dip down under the Circulating Water Intake Pipes and then up again to the Valve House where it connects to the 14 foot diameter tunnels to Lake Michigan.

The Turbine Building structure was demolished to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. The Circulating Water Intake Piping, Circulating Water Discharge Pipe and Discharge Tunnels were abandoned in place. Following the performance of FSS (as detailed in this Release Record) and a confirmatory survey by Oak Ridge Institute for Science and Education (ORISE), the Turbine Building void was backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the 588 foot elevation.

3.2.11 Survey Unit 06105B (Turbine Building Embedded Pipe)

The embedded drain piping in the Turbine Building consisted of 4-inch, 6-inch, 8-inch and 10-inch diameter pipe that was approximately 1,250 linear feet in length. The floor of the Turbine Building was at 560 foot elevation, with the drain piping embedded in the concrete approximately 2 feet deep.

3.2.12 Survey Unit 09200 and 6205A (Unit 1 and Unit 2 Circulating Water Discharge Tunnels and Piping)

The Unit 1 and Unit 2 Circulating Water Discharge Pipe and Discharge Tunnels are part of the Turbine Building survey unit. The Circulating Water Discharge system discharged cooling water, primarily from the Main Condensers but also from ancillary system cooling systems to Lake Michigan. The Circulating Water Tunnels were also the main authorized effluent release path to Lake Michigan for the release of treated and filtered radioactive liquid effluent. The tunnels run under the Turbine Building where two 12 foot diameter Circulating Water Discharge pipes opens into the tunnels from above. The tunnels dip down under the Circulating Water Intake Pipes and then up again to the Valve House where it connects to the 14 foot diameter tunnels to Lake Michigan.

3.2.13 Survey Unit 06107 and 06108 (Unit 1 and Unit 2 Buttress Pits)

The Unit 1 and Unit 2 Tendon Buttress Pits are mirror images of each other. Both Tendon Buttress Pits are bound by the interior surface of the structure from 591 foot elevation down to the 565 foot elevation. The survey unit consisted of the Containment concrete exterior and buttress, tendon steel end caps protruding from the buttress faces, perimeter walls and the wall dividing the Tendon Buttress Pits into chambers. The Tendon Buttress Pits were 26 feet deep and 4 feet wide with angled side walls that limited access.

There were six entrances to each buttress pit. In the end-state condition, the portions of the structures above the 588 foot elevation was removed and disposed of as waste. The remaining void below the 588 foot elevation was backfilled with a combination of clean demolition debris and clean fill.

3.2.14 Survey Unit 06201 and 06202 (Unit 1 and Unit 2 Diesel Fuel Oil Storage Tank Rooms)

Unit 1 and Unit 2 Diesel Fuel Oil Storage Tank Rooms were located on the 570 foot elevation and housed the oil for the Unit 1 and Unit 2 Diesel Generators. Both Unit 1 and Unit 2 Diesel Fuel Oil Storage Tank Rooms are mirror images of each other. The rooms were adjacent to the Turbine Building on the West side of the "G" (Column) Wall which divided the Turbine Building and Auxiliary Building. The entrance into both Diesel Fuel Oil rooms was from the Unit 1 and Unit 2 Steam Tunnel stair wells from the 560 foot elevation of the Turbine Building. The area of each Diesel Fuel Oil Storage Tank Room was approximately 813 m^2.

3.2.15 Survey Unit 06209 and 06210 (Unit 1 and Unit 2 Steam Tunnel Embedded Floor Drains)

The Unit 1 and Unit 2 Steam Tunnel embedded floor drain piping consisted of 479 linear feet of 4-inch ID pipe embedded in 2-feet of concrete in the floor of each Steam Tunnel.

3.2.16 Survey Unit 06211 and 06212 (Unit 1 and Unit 2 Tendon Tunnel Embedded Floor Drains)

The Unit 1 and Unit 2 Tendon Tunnel embedded floor drain piping consisted of 524 linear feet of 4-inch ID pipe embedded in 7-inches of concrete in the floor of each Tendon Tunnel.

3.2.17 Survey Unit 06213 and 06214 (Unit 1 East and West Main Steam Valve Houses)

Both the Unit 1 East and West Main Steam Valve Houses were located on the 570 foot elevation and housed the Main Steam Isolation valves for the Unit 1 reactor. Both Valve Houses were located adjacent to the Unit 1 Containment Building.

The Unit 1 East and West Main Steam Valve Houses were classified, as the Turbine Building, in accordance with ZSRP LTP Chapter 5, section 5.5.2.1.2 as Class 3 survey units. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS. Upon completion of Survey Unit Classification basis for final classification, which included a review of the Zion Station Historical Site Assessment (HSA), the classification for survey design remained as Class 3. During the performance of FSS, it was observed that many survey measurements exceed 50% of the OpDCGLs and several measurements exceeded a SOF (OpSOF) of one when compared against the OpDCGLs. Consequently, the Unit 1 East and West Main Steam Valve Houses, survey units 06213 and 06214 were reclassified from a Class 3 to a Class 1 survey unit and the survey was redesigned accordingly.

3.2.18 Survey Unit 06215 and 06216 (Unit 2 East and West Main Steam Valve Houses)

Both the Unit 2 East and West Main Steam Valve Houses were located on the 570 foot elevation and housed the Main Steam Isolation valves for the Unit 2 reactor. Both Valve Houses were located adjacent to the Unit 2 Containment Building.

The Unit 2 East and West Main Steam Valve Houses were classified, as the Turbine Building, in accordance with ZSRP LTP Chapter 5, section 5.5.2.1.2 as a Class 3 survey unit. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS. Upon completion of Survey Unit

Classification basis for final classification, which included a review of the HSA, the classification for survey design remained as Class 3.

3.2.19 Survey Unit 08100, 08401, 08102A/B (Crib House/Forebay, including the Unit 1 and Unit 2 Circulating Water Intake Pipes)

The Crib House/Forebay basement survey unit was an impacted Class 3 basement FSS unit. The Crib House/Forebay basement structure is located within Class 1 open land Survey Unit's 12204A, 12204B and 12204C.

The 552 foot and 559 foot elevation of the Crib House contained the upper pump housings of the six circulatory pumps, three for Unit 1 and three for Unit 2 that provide cooling water from Lake Michigan to various heat exchangers and condensers in the Turbine Building and Auxiliary Building. In addition, the intake/outtake plenum under the Crib House contained the cooling water outlet from these systems back into the lake. The 552 foot elevation also contained the Crib House sumps and sump system components which served as the collection point for the Crib House drain piping.

The Forebay structure was built to house and protect offshore water intakes providing cooling water from Lake Michigan to the Circulating Water Pumps, which in turn supplied various heat exchangers and condensers in the Turbine Building. It consisted of poured concrete walls, plate steel reinforcements and steel flow restriction gates along with associated conduits, piping and mechanical actuators. These walls and components began at approximately 596 foot elevation and extended to approximately 537 foot elevation with a mean Lake Michigan level of 577 foot elevation.

The Circulating Water Pumps took suction on the Forebay and pumped cooling water into the Circulating Water Intake Pipes. The Circulating Water Intake Pipes entered the east side of the Turbine Building, beneath the Condenser Water Boxes. The interior surface area of the Circulating Water Intake piping was 4,412 m^2, but the only portions that were accessible were the two vertical lengths of 9 foot diameter piping (in each unit) from the 588 foot elevation to the 558 foot elevation (approximate surface area of 158 m^2).

3.2.20 Survey Unit 09100 (Waste Water Treatment Facility)

The 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. The WWTF was designed to remove radioactive material, suspended solids and oil to ensure compliance with the station's ODCM permitted release criteria, and the station's 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 cation/anion resin beds, charcoal beds, mixing tanks, mixers, oil skimmers, monitoring and auto-isolation equipment, flocculate's, 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 were considered to be potentially contaminated.

All systems, component and materials associated with the WWTF 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 basic decommissioning end-state for the WWTF was the walls, floor, and sumps/pits below the 588 foot elevation.

3.3 Summary of Historical Radiological Data

The site historical radiological data for this Phase 2 FSS Final Report incorporates the results of the HSA issued in 1999 and supplemented in 2006, and includes the initial characterization surveys completed in 2013.

3.3.1 Historical Site Assessment and Characterization Surveys

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 DCGLs, 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 included:

- The identification of potential, likely, or known sources of radioactive and chemical contaminants based on existing or derived information.
- Distinguishing portions of the site that may need further action from those that pose little or no threat to human health.
- Providing an assessment of the likelihood of contaminant migration.
- Providing information useful to subsequent continuing characterization surveys.
- Providing an initial classification of areas and structures as non-impacted or impacted.

- Providing a graded initial classification for impacted soils and structures in accordance with MARSSIM guidance.
- Delineating initial survey unit boundaries and areas based upon the initial classification.

The survey units established by the HSA were used as initial survey units for characterization. Survey unit sizes were adjusted in accordance with the guidance provided in MARSSIM section 4.6 for the suggested physical area sizes for survey units for FSS.

Site characterization of the ZNPS was performed in accordance with ZS-LT-02, "*Characterization Survey Plan*" (Reference 26), which provided 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 Characterization Survey Plan was used to meet three primary objectives:

- Provide radiological inputs necessary for the design of FSS.
- Develop the required inputs for the LTP.
- Support the evaluation of remediation alternatives and technologies and estimate waste volumes.

For the survey units of interest in this report; the HSA and site continuing characterization activities were the basis for the information provided below.

3.3.1.1 Survey Units 01100 and 01110 (Unit 1 Containment above 565 foot and Unit 1 Containment Under Vessel Areas)

The following is a summary of processes and incidents pertaining to the Unit 1 Containment that were obtained from the HSA:

- 07/24/1973: Had a Reactor Coolant System spill from the pressurizer sprays (ROR – No number).
- 09/12/1975: An estimated 1000-2000 gallons of Radioactive Water Storage Tank water sprayed through U1 Containment (USNRC IR 75-13/75-12).
- 10/07/1976: Noble gas levels up to 100 Maximum Permissable Concentration (ROR 76-055).
- October 1977: Containment liner coatings and concrete paint noted to be degrading (USNRC IR 77-23/-).

- 11/02/1983: Note of high noble gas activity resulting in contamination of ~65 persons (USNRC IR 83-21/83-22 and 83-27/83-28 and ROR 83-97).
- January 1985 to March 1985: Component cooling leak (on 1CC-9428) in first quarter 1985 which led to a spill of ~10,000 gallons of CC water to the containment floor (USNRC IR 85-12/85-13).
- 10/01/1989: Identified flooding of U1 Containment though 4 open S/Gs (Zion RP/Decon Log).
- 02/19/1997: It was identified that U1 Containment coatings (Outer Missile Barrier) contained an alkyd primer covered by a carboline 305 product (PIF 97-0909).
- August 1998: General exposure rates from 10-150 mR/hr and contamination up to 50,000 dpm/100cm^2.
- 03/18/1976: Legal overexposure (8.05 rem) occurred in this area (USNRC IR 76-12).
- 03/25/1982: Legal overexposure (3.880 rem) occurred in this area (USNRC IR 82-09).

During the time that initial characterization was performed, all radioactive systems and components were still located inside Containment. Consequently, ambient radiation dose rates inside the Containment prohibited the direct assessment of concrete and steel structural surfaces below the 588 foot elevation by scanning or direct measurement.

On March 12, 2012, a characterization survey of the Incore surfaces was conducted (Survey 2012-0810). All smears (10) collected in the area were greater than 1,000 dpm/100 cm^2; the highest loose surface contamination indication was 80,000 dpm/100 cm^2. The maximum dose rate recorded in the area was 25 mR/hr.

From June of 2012 through January of 2013, site characterization was performed in Unit 1 Containment. The characterization survey consisted of a series of concrete core samples taken in the 568 foot concrete floor, the 541 foot Incore tunnel floor and Incore tunnel walls. 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. 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 the Unit 1 Containment, eight inside the missile shield and eight outside of the missile shield. Three (3) concrete core samples were obtained from each of the Incore tunnel Under

Vessel areas. 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.

The results of the site characterization surveys performed from 2010 to 2013 are documented in TSD 14-028 and in Chapter 2 of the LTP.

For the Unit 1 568 foot elevation, the sample analysis indicated that the majority of the radionuclide source inventory resided within the first ½-inch of concrete and that Cs-137 was the dominant radionuclide. For the Unit 1 Under Vessel area, the maximum dose rate recorded was 26 mR/hr and the maximum loose surface contamination smear indicated 80,000 dpm/100cm^2, which was taken at the Incore Access Tunnel plate that supports the Incore tubes (Survey 2012-0810). Sample B1-01110-CJF-CCV-001 showed the majority of activity above MDC for Co-60, Cs-137, and Eu-152 to a depth of 15.5 inches (entire core). The majority of Eu-154 source term was in the first 10 inches. Sample B1-01110-CJF-CCV-002 showed the majority of activity above MDC for Co-60, Eu-152, and Eu-154 was to a depth of 4 inches. The majority of Cs-137 source term was in the first 1/2 inch. Sample B1-01110-CJW-CCV-003 showed the majority of activity above MDC for Co-60, Cs-137, Eu-152, and Eu-154 was to a depth of 3.5 inches (entire core).

The top ½-inch puck from ten (10) of the nineteen (19) cores from Unit 1 were sent to Eberline Laboratory for gamma spectroscopy and HTD analyses for radionuclides such as H-3, C-14, Ni-63, Sr-90, and alpha emitters. Significant HTD radionuclides identified by the analysis of the concrete core samples included Ni-63, H-3 and Sr-90. The other radionuclides were less than their respective MDCs.

On November 11, 2016, the last routine surveys were conducted in Unit 1 Containment, before heavy demolition started and access was no longer possible. Survey 2016-2053, conducted in the Outer Missile Barrier (OMB) area of Unit 1, showed that the maximum dose rate was 1.0 mR/hr, and in survey 2016-1988 all smears were less than 20,000 dpm/100cm^2 with a maximum dose rate of 5 mR/hr. Survey 2016-3470, conducted in the Inner Missile Barrier (IMB) area of Unit 1, showed a maximum loose contamination level of 8,000 dpm/100cm^2 and a maximum dose rate of 1.7 mR/hr.

Following demolition and prior to attempting FSS, LTP section 5.3.4.4 required that continuing characterization be performed of the concrete walls and floor of the Under Vessel area in Unit 1 Containment and to assess the radiological condition of the exposed steel liner above the 565 foot elevation after the contaminated concrete has been removed. The continuing characterization of the steel liner above the 565 foot elevation consisted of sufficient smear samples and beta scans of accessible surfaces to ensure that the liner was adequately decontaminated prior to FSS. The results of the continuing characterization are addressed in detail in the Release Record for the Unit 1 Containment.

Continuing characterization was performed in the Under Vessel between November 11, 2017 and December, 2017. The survey consisted of scanning the exposed concrete surfaces and the acquisition of sixteen (16) concrete core samples with three (3) of those samples taken on the upper wall, five collected from the mid-wall, four (4) samples taken on the lower wall and three (4) samples taken on the floor. In addition, three (3) samples were taken from the metal in the embedded steel support ring. The results of the continuing characterization are addressed in detail in the Release Record for the Unit 1 Containment.

During the removal of the concrete from the floor above the 565 foot elevation, several instances occurred where the steel liner was punctured by the ram-hoe used to break apart the concrete. During these occurrences, work was stopped and the area was surveyed for loose surface contamination. In all cases, swipe samples taken of the puncture locations showed loose surface contamination of less than 1,000 dpm/100 cm^2. No conditions were encountered that indicated any potential cross-contamination of media outside of the liner. A patch was welded over each puncture location to prevent any future potential cross-contamination.

A RE performed a visual inspection and walk-down of the survey units on February 27, 2018 prior to performing FSS. The purpose of the walk-down was to assess the physical condition of the survey units, evaluate access points and travel paths and identify potentially hazardous conditions. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002, "*Survey Unit Classification*" as part of the survey design for FSS.

Based upon completion of the Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct classification of survey units 01100 and 01110 was determined to be Class 1.

3.3.1.2 Survey Unit 01111 (Unit 1 Containment Incore Sump Discharge Pipe)

No additional characterization data was available for the Unit 1 Containment Incore Sump Discharge Pipe (No. P125). The Incore sump discharge piping originates in a Class 1 area. The Under Vessel Incore area was subjected to operational conditions as well as the exercising of the Incore detectors. The region was subject to the same conditions as the remainder of the Containment interior. Consequently, the embedded pipe was assigned a classification of Class 1.

3.3.1.3 Survey Unit 01112 (Unit 1 Containment Penetrations)

Survey unit 01112 was classified in accordance with Zion*Solutions* procedure ZS-LT-300-001-002. The Unit 1 Containment Building Penetrations were initially classified as Class 1, Class 2, or Class 3, based on historical assessment, exposure to radioactive materials and system use. As a conservative measure, the classifications of all Unit 1 Containment Building Penetrations were changed to Class 1.

The Unit 1 Containment Penetrations were building to building pipe pathways for various primary and secondary systems. The location of the penetrations, their function, and the operational history of the Unit 1 Containment Building support the initial classifications. Those shared with the Auxiliary Building are addressed as Containment penetrations as the results are more conservative using Containment Penetration OpDCGLs.

From June of 2012 through January of 2013, site characterization was performed in Unit 1 Containment. The results of the site characterization surveys performed from 2010 to 2013 are documented in TSD 14-028, *"Radiological Characterization Report"* and in Chapter 2 of the LTP. During the time that initial characterization was performed, all radioactive systems and components were still located inside Containment. Consequently, ambient radiation dose rates inside the Containment prohibited the direct assessment of penetrations or system interior surfaces by scanning or direct measurement.

On November 11, 2016, the last routine surveys were conducted in Unit 1 Containment, before heavy demolition started and access was no longer possible. Surveys 2016-2053 and 2016-1988, conducted in the Outer Missile Barrier (OMB) area of Unit 1, showed a maximum contamination level of 20,000 dpm/100cm^2 and that the maximum dose rate was 5.0 mR/hr. Survey 2016-3470, conducted in the IMB area of Unit 1, showed a maximum loose contamination level of 8,000 dpm/100 cm^2 and a maximum dose rate of 7.0 mR/hr.

A RE performed the visual inspection and walk-down of the survey unit on February 27, 2018 and all Unit 1 Containment survey units were accepted for turnover by C/LT. The purpose of the walk-down was to assess the physical condition of the survey unit, evaluate access points and travel paths and identify potentially hazardous conditions.

3.3.1.4 Survey Units 02100 and 02110 (Unit 2 Containment above 565 foot and Unit 1 Containment Under Vessel Areas)

The following is a summary of processes and incidents pertaining to the Unit 2 Containment that were obtained from the HSA:

- May 1980: Due to a valve connection error, a freeze seal was applied to a line connected to the Unit 2 refueling cavity. The freeze seal blew out, causing the spillage of ~2000 gallons of refueling cavity water into the lower portion of Unit 2 Containment (USNRC IR 80-12/80-12).
- 08/24/1996 to 10/04/1996 Inspection: Discussion of unplanned spraying of ~3000 gallons of demineralized water into Unit 2 568 ft. IMB (USNRC IR 96-14/96-14).
- January 1997: 25-100 mR/hr General Area, 80,000 dpm/100cm^2.

- 12/28/1976 to 03/08/1977: Inspection noted flaking paint in Unit 2 Containment (USNRC IR -/77-11).
- 08/14/1984: The Containment exceeded the tech spec limit of 120 degrees F. Actual level reached 120.48 degrees F (LER 2-84-020).[1] – Note: The 568 ft. level is the pre-remediation floor elevation; the 565 ft. level is the new floor level following the removal of 3 ft. of concrete.
- 06/09/1986: Flooding was noted in the Unit 2 Tendon Tunnels from a possible water main problem. On 06/13/1986, Tendon Tunnel drains backed up, supposedly from high lake water level (Zion RP/Decon Logs).
- 06/27/1986: Unit 2 tripped due to lightning strike on one or more of the Containment lightning rods. The surge followed a path from Containment liner to ground via the electrical penetrations (LER 2-86-016 and USNRC 86-13/86-12).
- 05/07/1990: During refueling, a piece of grid strap was observed falling from the assembly. All cladding appeared to remain intact (LER 2-90-006).
- 05/13/1992: Approximately 4200 gallons of Reactor Coolant System water was inadvertently sprayed into Containment though the 2A Charging System header – A General Site Emergency was declared (USNRC IR 92-10/92-10 and Zion RP/Decon Log).
- November 1996: 40-50% of concrete floor coatings in Unit 2 Containment showed extensive failure. Unqualified coatings (~1200 ft^2) were observed on various components including instrument racks, struts, filter housings, valve bodies, and piping (USNRC IEN 97-24).

During the time that initial characterization was performed, all radioactive systems and components were still located inside Containment. Consequently, ambient radiation dose rates inside the Containment prohibited the direct assessment of concrete and steel structural surfaces below the 588 foot elevation by scanning or direct measurement.

On November 9, 2010, an initial characterization survey of the Incore surfaces was conducted (Survey 2980). Fourteen (14) out of twenty (20) smears collected in the area were greater than 1,000 dpm/100 cm^2; the highest loose surface contamination indication was 127,000 dpm/100 cm^2. The maximum dose rate recorded in the area was 35 mR/hr.

From June of 2012 through January of 2013, site characterization was performed in Unit 2 Containment. The characterization survey consisted of a series of concrete core samples taken in the 568 foot concrete floor, the 541 foot. Incore tunnel floor and Incore tunnel walls. 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. 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 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 Incore tunnel Under Vessel areas. 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.

The results of the site characterization surveys performed from 2010 to 2013 are documented in TSD 14-028 and in Chapter 2 of the LTP.

For the Unit 2 568 foot elevation, the sample analysis indicated that the majority of the radionuclide source inventory resided within the first ½-inch of concrete and that Cs-137 was the dominant radionuclide. For the Unit 2 Under Vessel area, the maximum dose rate recorded was 15 mR/hr and the maximum loose surface contamination smear indicated 90,000 dpm/100cm^2, which was taken at the plate that supports the Incore tubes (Survey 2013-0046). Sample B1-02110-CJF-CCV-001 showed the majority of activity above MDC for Co-60, Cs-137, and Eu-152 was to a depth of 14 inches (entire core). The majority of Eu-154 source term was in the first 10 inches. Sample B1-02110-CJF-CCV-002 showed the majority of activity above MDC for Co-60, Eu-152, and Eu-154 was to a depth of 4.5 inches (entire core). The majority of Cs-137 source term was in the first 0.5 inches. Sample B1-02110-CJW-CCV-003 showed the majority of activity above MDC for Co-60, Cs-137, Eu-152, and Eu-154 was to a depth of 5.5 inches (entire core).

The top ½-inch puck from eight (8) of the nineteen (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. Significant HTD radionuclides identified by the analysis of the concrete core samples included Ni-63, H-3 and Sr-90. The other radionuclides positively detected at concentrations greater than their respective MDC included; C-14, Tc-99, Pu-238, Pu-239/240, Am-241, Am-243 and Cm-243/244.

Following demolition and prior to attempting FSS, LTP section 5.3.4.4 required that continuing characterization be performed of the concrete walls and floor of the Under Vessel area in Unit 2 Containment and to assess the radiological condition of the exposed steel liner above the 565 foot elevation after the contaminated concrete has been removed. The continuing characterization of the steel liner above the 565 foot elevation consisted of sufficient smear samples and beta scans of accessible surfaces to ensure that the liner was adequately decontaminated prior to FSS. The results of the continuing characterization are addressed in detail in the Release Record for the Unit 2 Containment.

Continuing characterization was performed in the Under Vessel are between November 11, 2017 and December, 2017. The survey consisted of scanning the exposed concrete surfaces and the acquisition of sixteen (16) concrete core samples with three (3) of those samples taken on the upper wall, five collected from the mid-wall, four (4) samples taken on the lower wall and three (4) samples taken on the floor. In addition, three (3) samples were taken from the metal in the embedded steel support ring. The results of the continuing characterization are addressed in detail in the Release Record for the Unit 2 Containment.

During the removal of the concrete from the floor above the 565 foot elevation, several instances occurred where the steel liner was "punctured" by the ram-hoe used to break apart the concrete. During these occurrences, work was stopped and the area was surveyed for loose surface contamination. In all cases, swipe samples taken of the puncture locations showed loose surface contamination of less than 1,000 dpm/100 cm^2. No conditions were encountered that indicated any potential cross-contamination of media outside of the liner. A patch was welded over each puncture location to prevent any future potential cross-contamination.

A RE performed a visual inspection and walk-down of the Unit 2 Containment basement survey units 02100 and 02110 on January 9, 2018 as part of the initial turnover for performing FSS. The purpose of the walk-down was to assess the physical condition of the survey units, evaluate access points and travel paths and identify potentially hazardous conditions. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS.

Based upon completion of the Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct final classification of survey units 02100 and 02110 was determined to be Class 1.

3.3.1.5 Survey Unit 02112 (Unit 2 Containment Penetrations)

Survey unit 02112 was classified in accordance with procedure ZS-LT-300-001-002.

The Unit 2 Containment Building Penetrations were initially classified as Class 1, Class 2, or Class 3, based on historical assessment, exposure to radioactive materials and system use. As a conservative measure, the classifications of all Unit 2 Containment Building Penetrations were changed to Class 1. Consequently, sufficient measurements were taken in all Unit 2 Containment Building Penetrations to ensure 100% areal coverage of all accessible internal surfaces within the penetrations.

The Unit 2 Containment Penetrations were building to building pipe pathways for various primary and secondary systems. The location of the penetrations, their function, and the operational history of the Unit 2 Containment Building support the initial classifications.

Those shared with the Auxiliary Building are addressed as Containment penetrations as the results are more conservative using Containment Penetration OpDCGLs.

From June of 2012 through January of 2013, site characterization was performed in Unit 2 Containment. The results of the site characterization surveys performed from 2010 to 2013 are documented in TSD 14-028 and in Chapter 2 of the LTP. During the time that initial characterization was performed, all radioactive systems and components were still located inside Containment. Consequently, ambient radiation dose rates inside the Containment prohibited the direct assessment of penetrations or system interior surfaces by scanning or direct measurement.

On July 20, 2016, the last routine surveys were conducted in Unit 2 Containment, before heavy demolition started and access was no longer possible. Survey 2016-2226, conducted in the OMB area of Unit 2, showed that all smears were less than 1,000 dpm/100cm^2 and that the maximum dose rate was 1.5 mR/hr. Survey 2016-2224, conducted in the IMB area of Unit 2, showed a maximum loose contamination level of 6,400 dpm/100cm^2 and a maximum dose rate of 5.1 mR/hr.

A RE performed the visual inspection and walk-down of the survey unit on December 12, 2017. The purpose of the walk-down was to assess the physical condition of the survey unit, evaluate access points and travel paths and identify potentially hazardous conditions. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS.

3.3.1.6 Survey Unit 03202 (Spent Fuel Pool/Transfer Canal)

The SFP/Transfer Canal survey unit 03202 is an impacted Class 1 basement survey unit. The potential for the presence of residual contamination at concentrations in excess of the release criteria existed throughout the Fuel Handling Building. Based on the building design basis, the operating history, as well as the areas within the building that were controlled as contaminated, all internal survey units within the Fuel Handling Building were considered to be potentially contaminated. The HSA states that there were two documented overflows of the SFP. The first occurred in April of 1991 and the second occurred in August of 1994. In addition, the HSA also notes that there was a fire in the Transfer Canal area in 1971 (pre-operational) and a potential leakage path from the pool through the "tell tales" drains.

Initial site characterization was performed at the Zion site in 2013. At that time, the survey of many inaccessible or not readily accessible building surfaces was deferred due to physical obstructions and/or the presence of prohibitive background from commodities. The end state structure for the Fuel Handling Building was the underlying concrete of the SFP/Transfer Canal after the steel liner had been removed. Characterization was deferred until decommissioning had progressed to the point when the surface of interest was exposed.

On June 6, 2016, access was granted into the Fuel Handling Building to acquire characterization data prior to demolition. Demolition of the SFP had progressed to the point where the steel liner was removed exposing the underlying concrete on the bottom of the pool. The FSS staff attempted to scan the bottom of the pool, however the remaining source term in the Transfer Canal created radiation levels that were too high to support scan surveys. Four (4) concrete core samples were acquired in the Spent Fuel Pool on the 576 foot elevation, three on the wall and one on the floor. The analysis of these samples indicated a maximum Cs-137 concentration of 10.50 pCi/g and a maximum Co-60 concentration of 6.12 pCi/g.

An additional continuing characterization surveys were performed on April 2, 2018. The objective of the continuing characterization survey was to assess the depth of any activation in the concrete and, to ensure the correct geometry was used for the ISOCS measurements. The survey consisted of a series of scans of the exposed concrete surfaces and the acquisition of eight concrete core samples. During the beta scan surveys, several elevated areas were identified. The results indicted radiation levels between 1.0 - 1.5 mR/hr on nearby ledges from the adjacent Auxiliary Building structure. These radiation levels were identified as causing elevated survey results and multiple scan alarms along the east end of the SFP basement footprint and shielding were placed on the ledge to lower the ambient background.

Following the acquisition of continuing characterization samples; the onsite contractor continued with the remediation of the exposed concrete of the SFP/Transfer Canal by scabbling with heavy machinery. After the completion of remediation, 19 judgmental ISOCS measurements were taken of the exposed concrete in an effort to determine if remediation was sufficient. The results verified that the gamma shine coming from the elevated ledge areas would not impact the successful implementation of FSS of the SFP/Transfer Canal basement survey unit as long as the shielding remained in place.

The RE performed a visual inspection and walk-down of the survey unit on April 2, 2018 prior to performing FSS. The purpose of the walk-down was to assess the physical condition of the survey unit, evaluate access points and travel paths and identify potentially hazardous conditions. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS.

Based upon completion of Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of final Survey Unit Classification Worksheet, it was concluded that the correct classification of survey unit 03202 was Class 1.

3.3.1.7 Survey Unit 05100 (Auxiliary Building Basement)

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 time that initial characterization was performed, all radioactive systems and components were still located inside the Auxiliary Building. Consequently, ambient radiation dose rates inside some of the cubicles on the 542 foot elevation prohibited the direct assessment of concrete surfaces by scanning or direct measurement.

During the initial 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 twenty (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. 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 concrete pucks were analyzed on the on-site gamma spectroscopy system. The on-site gamma spectroscopy results of the concrete cores taken from the 542 foot elevation of the Auxiliary Building indicated that Co-60, Cs-134 and Cs-137 were the only plant-derived gamma emitting radionuclides identified. 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 was 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 twenty (20) cores from the Auxiliary Building were sent to Eberline Laboratory for gamma spectroscopy and HTD analyses. The mixture percentages for the initial suite of radionuclides for the Auxiliary Basement concrete were developed in TSD 14-019 using the results of all core sample analyses, including the cores sent to Eberline. 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 included; C-14, Tc-99, Sr-90, Ag-108m, Pu-238, Pu-239/240, Am-241 and Am-243.

In December of 2017, as part of continuing characterization activities, a total of thirty-two (32) additional concrete cores were taken and analyzed throughout the Auxiliary Building basement 542 foot elevation.

Cores were collected from the floor and lower walls to a depth of 6 inches, or refusal. The cores were cut into ½ inch thick pucks and onsite gamma spectroscopic analysis was performed on both sides of each puck throughout the length of the core. Additionally, ½ inch pucks from eight (8) of the sample locations, that exhibited the highest gamma activity, were sent to an off-site laboratory for HTD radionuclide analyses.

Analysis of the data indicated that the results of the gamma spectroscopic analysis of the ½ inch pucks indicated that the activity concentrations corresponded to an OpSOF of less than 0.1 SOF for twenty-nine (29) of the thirty-two (32) sample locations below a depth of 2 inches, illustrating that the majority of the source term was surficial and not at depth.

A RE performed a visual inspection and walk-down of the survey unit on December 4, 2017, prior to performing FSS. The purpose of the walk-down was to assess the physical condition of the survey unit, evaluate access points and travel paths and identify potentially hazardous conditions. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS.

Based upon completion of Survey Unit Classification Basis for final classification, which included a review of the historical information, the results of the Characterization Survey data and, completion of a final Survey Unit Classification Worksheet, the correct final classification of survey units 05100 was Class 1.

3.3.1.8 Survey Unit 05119 (Auxiliary Building 542 foot Embedded Floor Drains)

Throughout the operation of ZNPS, and during the period of SAFSTOR, the Auxiliary Building 542 foot elevation floor drain system received contaminated liquids from equipment operation, spills and flooding. During the operation of the facility, storage tank overflow into the Auxiliary Building basement resulted in water flooding up to 2 feet deep. The HSA documents five (5) occurrences between 1990 and 1996 where the Lake Discharge Tanks overflowed to the Auxiliary Building 542 foot elevation floor.

Operational surveys showed significant dose rates at the drain scuppers. In addition, from May 20, 2016 to June 27, 2016, a characterization survey was performed on 2,539 feet of pipe that was accessible. The results of the characterization survey are documented in Zion*Solutions* TSD 16-008, *"Radiological Characterization Report for Auxiliary Building 542 foot Embedded Floor Drain Pipe"* (Reference 27). The results of the characterization surveys indicated that the Auxiliary Building 542 foot embedded floor drains were radiologically contaminated with gamma measurements up to 2.61E+09 pCi/m^2. The results of the characterization survey, combined with the known introduction of contaminated liquids into the pipe, the analysis of contaminated liquids in the sumps and collection tanks prior to processing and the documented spills and flooding in the Auxiliary Building basement would indicate the presence of a significant radioactive source inventory in the pipe. Consequently, all embedded floor pipe in the Auxiliary Building 542 foot elevation floor were classified as MARSSIM Class 1.

Figure 3-3 depicts a FSS embedded piping survey in the Auxiliary Building.

Figure 3-3 – Embedded Drain Pipe FSS Survey

3.3.1.9 Survey Unit 05120 (Auxiliary Building Penetrations)

The Auxiliary Building housed numerous systems containing radioactively contaminated support systems. System leakage and maintenance activities over the operating life of the reactor resulted in the radiological contamination of most of the interior surfaces of the structures. Based on the building design basis and the operating history, all internal survey units in Auxiliary Building were assigned an initial classification of Class 1 in accordance with the HSA.

The location of the penetrations, their function, and the operational history of the Auxiliary Building to support the initial classifications are described in TSD 14-016.

As part of the survey unit turnover process, a RE performed the visual inspection and walk-down of the survey unit on March 27, 2018. The purpose of the walk-down was to assess the physical condition of the survey unit, evaluate access points and travel paths and identify potentially hazardous conditions and determine if the survey unit was acceptable for performing Final Status Surveys. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS.

3.3.1.10 Survey Unit 06100 (Turbine Building Basement)

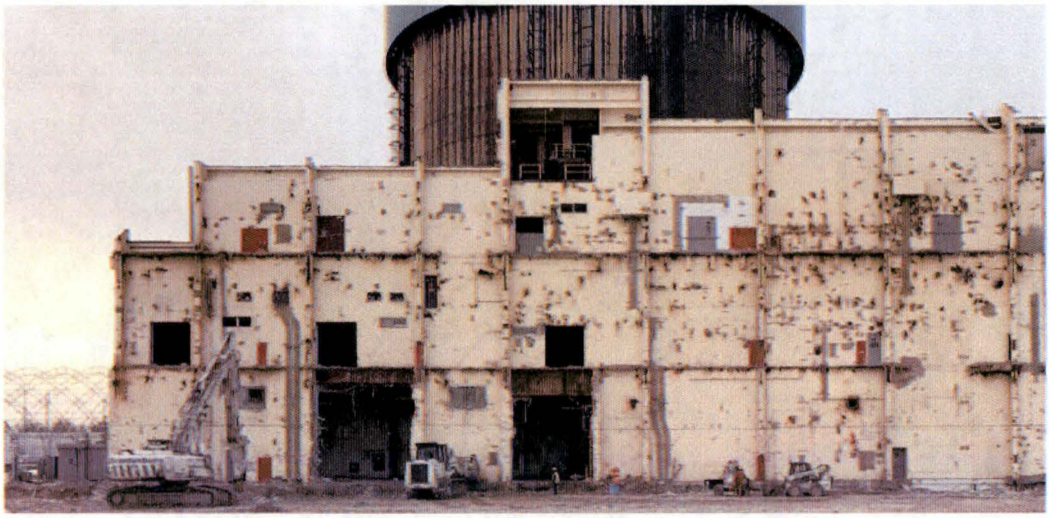
The Turbine Building was initially classified as a Class 2 structure by the HSA. LTP Section 5.5.2.1.2 changed the classification of the Turbine Building basement from Class 2 to Class 3. The LTP states "The FSS units for the basements of the Turbine Building, the Crib House/Forebay, WWTF and the Circulating Water Discharge Tunnels are designated as Class 3 as defined in MARSSIM, section 2.2 in that the FSS units are expected to contain levels of residual activity at a small fraction of the DCGLs, based on site operating history and previous radiation surveys."

In November of 2012, site characterization of the Turbine Building commenced with the acquisition of a series of concrete core samples that were taken in the 560 foot elevation Turbine Building concrete floor as well as the 570 foot elevation Steam Tunnel concrete floors.

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. 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. 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.

FSS surveys of the Turbine Building occurred in March of 2016. Figure 3-4 is a photograph of the Turbine Building prior to demolition.

Figure 3-4 – Turbine Building Demolition



The Turbine Building structure was demolished to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. The Circulating Water Intake Piping, Circulating Water Discharge Pipe were filled with grout and the Intake/Discharge pipe and Discharge Tunnels were abandoned in place. Following the performance of FSS (as detailed in the Release Record for the Turbine Building) and a confirmatory survey by ORISE, the Turbine Building void was backfilled using clean concrete debris and clean fill from off-site to the 588 foot elevation.

3.3.1.11 Survey Unit 06105B (Turbine Building Embedded Pipe)

See section 3.3.1.10 above.

3.3.1.12 Survey Unit 06107 and 06108 (Unit 1 and Unit 2 Buttress Pits)

Survey units 06107 and 06108 were classified in accordance with procedure ZS-LT-300-001-002. The Tendon Buttress Pits are part of the Turbine Building survey unit.

Based on information from the HSA, the initial classification for Turbine Building basement survey unit was Class 3. Although the Tendon Buttress Pits were inaccessible during site characterization in 2013, the results of environmental monitoring of radiological effluents indicate that the residual radioactivity in the Tendon Buttress Pits was minimal, supporting a Class 3 classification.

3.3.1.13 Survey Unit 06201 and 06202 (Unit 1 and Unit 2 Diesel Fuel Oil Storage Tank Rooms)

The Turbine Building basement was classified in accordance with Zion*Solutions* procedure ZS-LT-300-001-002. The Unit 1 and Unit 2 Diesel Fuel Oil Storage Tank Rooms are part of the Turbine Building basement survey unit. The Turbine Building was classified as a mixture of Class 2 and Class 3 structural survey units in accordance with the Zion Station HSA. LTP Chapter 5, section 5.5.2.1.1 changed the classification of the Turbine Building from Class 2 to Class 3.

During decommissioning, the primary pathway into and out of the basement of the Auxiliary Building became the Unit 1 and Unit 2 570 foot elevation Diesel Fuel Oil floors. Ramps were constructed through each into the 542 foot elevation to allow for the transit of heavy equipment and removal of radioactive commodities. Due to the introduction of radioactive material into both of these areas, they were both reclassified as Class 1 survey units in accordance with ZS-LT-300-001-002.

3.3.1.14 Survey Unit 06209 and 06210 (Unit 1 and Unit 2 Steam Tunnel Embedded Floor Drains)

The Unit 1 and Unit 2 Steam Tunnel Embedded Floor Drains were classified in accordance with procedure ZS-LT-300-001-002. The Turbine Building was initially classified as a Class 2 structure by the HSA. LTP Section 5.5.2.1.2 changed the classification of the Turbine Building basement from Class 2 to Class 3.

3.3.1.15 Survey Unit 06211 and 06212 (Unit 1 and Unit 2 Tendon Tunnel Embedded Floor Drains)

The Unit 1 and Unit 2 Tendon Tunnel Embedded Floor Drains were classified in accordance with procedure ZS-LT-300-001-002.

Based on information from the HSA, the initial classification for this Survey Unit was Class 3. Although the Tendon Tunnels were inaccessible during site characterization in 2013, the results of environmental monitoring of radiological effluents indicate that the residual radioactivity in this FSS unit was minimal, supporting the initial classification.

3.3.1.16 Survey Unit 06213 and 06214 (Unit 1 East and West Main Steam Valve Houses)

See section 3.3.1.10 above.

3.3.1.17 Survey Unit 06215 and 06216 (Unit 2 East and West Main Steam Valve Houses)

See section 3.3.1.10 above.

3.3.1.18 Survey Unit 08100, 08401, 08102A/B (Crib House/Forebay, including the Unit 1 and Unit 2 Circulating Water Intake Pipes)

The Crib House/Forebay survey unit 08100 was initially classified in the HSA. The interior 552 foot and 559 foot elevation concrete surfaces in the basement structure survey unit were originally designated as "non-impacted". The Crib House/Forebay survey unit was not located in a radiologically controlled area. In addition, no radiological postings or labeled radioactive material were identified in or around the Crib House/Forebay structure and an RWP was not required for entry. The HSA classified the exterior of the building as well as the grounds surrounding the building as MARSSIM Class 2. The HSA classified the Circulating Water Intake Pipe interior surfaces as "Class 3".

Fixed residual radioactive material was discovered on the 594 foot elevation of the Crib House in 1985. In addition, the Circulatory Water system was the normal effluent release pathway for the facility during operation. Based upon use, location and previous findings, the MARSSIM classification for the interior of the Crib House was changed from its original classification of "non-impacted" to a MARSSIM impacted Class 2 classification in accordance with procedure ZS-LT-300-001-002.

As part of site characterization, Zion*Solutions* acquired and analyzed twenty (20) concrete core samples taken from the 559 foot elevation of the Crib House in March and April of 2012. Sample locations were selected at random. Prior to acquiring the core samples, the area was scanned to ensure the absence of surface radioactive contamination at each sample location. Scans were performed with a Ludlum 43-93 100 cm^2 alpha-beta scintillation detector. Gross beta background ranged from 150 cpm to 300 cpm. No activity greater than background was observed by scan. 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.

On November 17 and 18, 2014, six (6) samples were taken of sediment from the Forebay and Crib House basement while divers were used to install cofferdams and plugs. Analysis of the samples indicated the presence of Cs-137 at concentrations ranging from less than MDC to 1.09E-01 pCi/g. No other plant-derived radionuclides were detected.

The Crib House above the 588 foot elevation was surveyed for unconditional release in January and February of 2015. Once it was demonstrated that the Crib House internal surfaces were suitable for unconditional release, a demolition contractor salvaged clean equipment out of the Crib House. Unrestricted release surveys were performed in March of 2015 on equipment removed prior to and during the demolition of the upper levels of the Crib House. All smear and direct readings were less than MDC. The Crib House structural concrete and cinder block above the 588 ft. elevation was also surveyed to demonstrate that the material was free of plant-derived radionuclides at concentrations greater than background. The concrete and cinderblock Crib House structure above the 588 foot elevation was demolished and stockpiled to be used as clean hard fill.

In accordance with Section 5.5.2.1 of the LTP Chapter 5, the classification of the Crib House/Forebay was changed from Class 2 to Class 3. At the time of LTP submittal, the Forebay and the Circulating Water Intake Piping were completely underwater and not accessible. Process knowledge and the results of environmental monitoring of radiological conditions at effluent outfalls in the past indicate that the probability of residual radioactivity in these FSS units exceeding 50% of the $OpDCGL_B$ for the Crib House/Forebay was very low.

A RE performed a visual inspection and walk-down of the survey unit on June 25, 2016 prior to performing FSS. The purpose of the walk-down was to assess the physical condition of the survey unit, evaluate access points and travel paths and identify potentially hazardous conditions. A final classification assessment was performed in accordance with procedure ZS-LT-300-001-002 as part of the survey design for FSS.

3.3.1.19 Survey Unit 09200 and 6105A (Unit1 and Unit 2 Circulating Water Discharge Tunnels and Piping)

During plant operations, the Circulating Water Discharge Tunnels and Piping were the main authorized effluent release pathway for the discharge of treated and filtered radioactive liquid effluent to Lake Michigan. The liquid effluent release pathway was monitored and the results presented in the annual Radiological Environmental Monitoring Program (REMP) report in accordance with the ODCM.

All commodities were removed from the Turbine Building basement with the exception of the underground Circulating Water pipe, Circulating Water Discharge Tunnels and buried Service Water pipe running between the Crib House location and the Auxiliary Building beneath the Turbine Building concrete floor.

3.3.1.20 Survey Unit 09100 (Waste Water Treatment Facility)

The design purpose of the WWTF was to receive the discharges from the Fire Sump and the heater bay roof drains. Due to contamination reaching the Fire Sump, many portions of the WWTF contained trace levels of contaminants.

Based on the building design basis and the operating history, the WWTF was given an initial classification of Class 3 in accordance with the HSA. However, during decommissioning activities, the facility was used as a radioactive material storage area to keep the materials out of the inclement weather that is common to the area. Therefore, LTP Chapter 5, section 5.5.2.1 and Table 5-18 identifies the WWTF as a Class 1 area requiring 100% areal coverage.

3.4 Conditions at the Time of Final Status Survey

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

- **Containments –** Both Unit 1 and Unit 2 Containment Buildings were comprised of concrete walls and floors with all interior surfaces of the containment "shell" covered by a 0.25 inch steel liner. The 30 inch thick layer of concrete covering the liner on the 565 foot elevation floor was removed. The floor of the Under Vessel area is located at the 541 foot elevation. A 30 inch layer of concrete was present above the liner in the Under Vessel area and a 15 inch layer of concrete was on the walls in the Under Vessel area.

- **Auxiliary Building** – The Auxiliary Building has no steel liner. The majority of the remaining End State inventory in the Auxiliary Building Basement was surface and volumetric contamination in the concrete floor and lower walls of the 542 foot elevation. The majority of the remaining End State inventory in the Auxiliary Building basement was surface and volumetric contamination in the concrete floor and lower walls of the 542 foot elevation.

  The upper walls above 545 foot elevation were also contaminated but at significantly lower concentrations than the floors. Upper wall contamination was primarily in the vicinity of floors that had been removed during demolition. Loose surface contamination was also present on remaining concrete surfaces due to the deposition of airborne radioactivity generated during operations, commodity removal and the demolition of interior concrete structures.

- **SFP/Fuel Handling Building and Fuel Transfer Canals –** The only portion of the Fuel Handling Building Basement that remained 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 was removed from both the SFP and the Transfer Canals.

- **Turbine Building –** The Turbine Building structure was demolished to a depth of 3 feet below grade in accordance with the requirements of the Asset Sale Agreement. The Circulating Water Intake Piping, Circulating Water Discharge Pipe and Discharge Tunnels were abandoned in place. Following the performance of FSS (as detailed in the Release Record contained in Appendix 10) and a confirmatory survey by ORISE, the Turbine Building void was backfilled using concrete debris suitable for reuse as clean hard fill and/or clean fill to the 588 foot elevation.

- **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 was expected in these basements.

3.5 Identification of Potential Contaminants

Zion*Solutions* TSD 11-001, *"Potential Radionuclides of Concern During the Decommissioning of the Zion Station"* (Reference 28) was prepared and approved in November 2011. The purpose of this document was to establish the basis for an initial suite of potential 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 LTP Table 5-1 and reproduced in Table 3-1 below.

Table 3-1 – Initial Suite of Radionuclides

Radionuclide	Half Life (years)	Radionuclide	Half Life (years)	Radionuclide	Half Life (years)
H-3	1.24E+01	Tc-99	2.13E+05	Np-237	2.14E+06
C-14	5.73E+03	Ag-108m	1.27E+02	Pu-238	8.77E+01
Fe-55	2.70E+00	Sb-125	2.77E+00	Pu-239/240	2.41E+04
Ni-59	7.50E+04	Cs-134	2.06E+00	Pu-241	1.44E+01
Co-60	5.27E+00	Cs-137	3.00E+01	Am-241	4.32E+02
Ni-63	9.60E+01	Eu-152	1.33E+01	Am-243	7.38E+03
Sr-90	2.91E+01	Eu-154	8.80E+00	Cm-243/244	1.81E+01
Nb-94	2.03E+04	Eu-155	4.96E+00		

3.6 Radionuclides of Concern and Mixture Fractions

LTP Chapter 2 provides detailed characterization data that describes current contamination levels in the basements. 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. TSD 14-019 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 contributors (IC) 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 2-8 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.

3.7 Radiological Release Criteria

Prior to FSS process proceeding, the BcDCGLs were established to demonstrate compliance with the 25 mrem/year unrestricted release criterion. The BcDCGLs were 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 the LTP describes in detail the approach, modeling parameters and assumptions used to develop the BcDCGLs.

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).

4. FINAL STATUS SURVEY PROTOCOL

4.1 Data Quality Objectives

The DQO process as outlined in Section 2 of this report was applied for each FSS Sample Plan and contains basic elements common to all FSS Sample Plans at ZSRP. An outline of those elements presented in the ZSRP FSS Sample Plans are as follows:

4.1.1 State the Problem

The problem: To demonstrate that the level of residual radioactivity in a survey unit does not exceed the release criteria of 25 mR/year TEDE and that the potential dose from residual radioactivity is ALARA.

Stakeholders: The primary stakeholders interested in the answer to the problem were Zion*Solutions* LLC, Exelon Nuclear Generation LLC (Exelon), the Illinois Environmental Management Agency (IEMA) and the United States Nuclear Regulatory Commission (USNRC).

The Planning Team: The planning team consisted of the assigned RE with input from other C/LT personnel as well as the Safety Department. The primary decision maker was the RE with input from the C/LT Manager.

Schedule: The approximate time projected to mobilize, implement, and access an FSS unit.

Resources: The following resources were necessary to implement an FSS Sample Plan:

- RE to prepare the plan and evaluate data.
- C/LT Field Supervisor to monitor and coordinate field activities.
- Survey Mapping/CAD Specialist to prepare survey maps, layout diagrams, composite view drawings, and other graphics as necessary to support design and reporting.
- C/LT Technicians to perform survey activities, collect survey measurement data, and collect media samples.
- Chemistry/Analysis laboratory Staff to analyze samples as necessary.

4.1.2 Identify the Decision

Principal Study Question: Are the residual radionuclide concentrations found in the building surfaces equal to or below the applicable site-specific OpDCGLs?

Alternate Actions: Alternative actions include failure of the survey unit, remediation, reclassification, and resurvey.

The Decision: If the survey unit failed to demonstrate compliance with the release criteria, then the survey unit was not suitable for unrestricted release. The DQA process was reviewed to identify the appropriate additional action or combination of actions.

4.1.3 Identify Inputs to the Decision

Information Needed: The survey unit requiring evaluation of residual activity and its surface area. The characterization surveys and HSA were preliminary sources of information for FSS. New measurements of sample media were needed to determine the concentration and variability for those radionuclides potentially present at the site at the time of FSS.

Historical Information: The classification as originally identified in the HSA and the verification of that classification during characterization. The information included a summary of site processes or incidents that occurred in the survey unit.

Radiological Survey Data: The current radiological survey data from characterization, Remedial Action Support Surveys (RASS), Radiological Assessments (RAs), or turnover surveys. This information was used to develop a sample size for FSS.

Radionuclides of Concern: The ROC for the FSS of Containments and all remaining survey units (identified as Auxiliary Building) are presented in Section 2.2, Table 2-8, of this report.

Basis for the Action Level: The action levels for the survey units discussed in this Phase 2 report were provided in Table 5-25 of the LTP and reproduced in Table 4-1.

Table 4-1 – Investigation Levels

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

During FSS, concentrations for HTD ROC Ni-63 and Sr-90 (as well as H-3 for both Containments) were inferred using a surrogate approach. As presented in the LTP Chapter 5, section 5.2.11, Cs-137 is the principle surrogate radionuclide for both Sr-90 and H-3 and Co-60 is the principle surrogate radionuclide for Ni-63. The mean, maximum and 95% UCL of the surrogate ratios for concrete core samples taken in the Auxiliary Building basements and Containments were calculated in TSD 14-019 and Table 5-15 of the LTP and are reproduced in Table 4-2. The maximum ratios were used in the surrogate calculations during FSS unless specific ratios were determined for a survey unit based on sample analysis.

Table 4-2 – 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

For the FSS of the relevant survey units in this report, the surrogate OpDCGLs for Co-60 and Cs-137 were computed based on the maximum ratios from Table 4-2. The equation for calculating a surrogate DCGL is as follows:

Equation 1

$$Surrogate_{DCGL} = \frac{1}{\left[\left(\frac{1}{DCGL_{Sur}}\right) + \left(\frac{R_2}{DCGL_2}\right) + \left(\frac{R_3}{DCGL_3}\right) + \cdots \left(\frac{R_n}{DCGL_n}\right)\right]}$$

Where:

$DCGL_{Sur}$ = Surrogate radionuclide DCGL

$DCGL_{2,3\ldots n}$ = DCGL for radionuclides to be represented by the surrogate

R_n = Ratio of concentration (or nuclide mixture fraction) of radionuclide "n" to surrogate radionuclide

Using the OpDCGLs presented in Table 2-2 for basements, Table 2-4 for piping and Table 2-6 for penetrations, and using the maximum ratios from Table 4-2, the following table presents the results of surrogate calculations performed for each survey unit addressed in this report.

Table 4-3 – Surrogate Calculation Results

Survey Unit Number	Survey Unit Name	Cs-137	Co-60	Gross Gamma(1)
			(pCi/m^2)	
01100	Unit 1 Containment above 565 ft.	2.88E+06(2)	1.18E+06	N/A
01110	Unit 1 Under Vessel	2.10E+07(2)	8.55E+06	N/A
01111	Unit 1 Containment Incore Sump Drain	5.87E+07(2)	2.40E+07	5.40E+07
01112	Unit 1 Containment Penetrations	2.06E+07(2)	8.41E+06	N/A
02100	Unit 2 Containment above 565 ft.	2.88E+06(2)	1.18E+06	N/A
02110	Unit 2 Under Vessel	2.10E+07(2)	8.55E+06	N/A
02112	Unit 2 Containment Penetrations	2.06E+07(2)	8.41E+06	N/A
03202	SFP/Transfer Canal	7.81E+06	4.08E+06	N/A
05100	Auxiliary Building 542 ft. floor and walls	3.50E+07	1.70E+07	N/A
05119	Auxiliary Building Embedded Floor Drains	N/A	N/A	N/A
05120	Auxiliary Building Penetrations	4.62E+07	5.63E+06	N/A
06100	Turbine Building Basement and Steam Tunnels	1.70E+06	8.75E+05	N/A
06105A	Circulating Water Discharge Pipe	1.70E+06	8.75E+05	N/A
06105B	Turbine Building Embedded Pipe	7.17E+07	3.71E+07	N/A
09200	Circulating Water Discharge Tunnels	8.36E+06	4.31E+06	N/A
06107	Unit 1 Turbine Building Buttress Pit	1.70E+06	8.75E+05	N/A
06108	Unit 2 Turbine Building Buttress Pit	1.70E+06	8.75E+05	N/A
06201	Unit 1 Turbine Building 570 ft. Diesel Fuel Storage	1.70E+06	8.75E+05	N/A
06202	Unit 2 Turbine Building 570 ft. Diesel Fuel Storage	1.70E+06	8.75E+05	N/A
06209	Unit 1 Steam Tunnel Floor Drain	4.63E+08	2.38E+08	1.75E+08
06210	Unit 2 Steam Tunnel Floor Drain	4.63E+08	2.38E+08	1.75E+08
06211	Unit 1 Tendon Tunnel Floor Drain	5.06E+07	2.64E+07	8.68E+06
06212	Unit 2 Tendon Tunnel Floor Drain	5.06E+07	2.64E+07	8.68E+06
06213	Unit 1 Steam Tunnel East Valve House	1.70E+06	8.75E+05	N/A
06214	Unit 1 Steam Tunnel West Valve House	1.70E+06	8.75E+05	N/A
06215	Unit 2 Steam Tunnel East Valve House	1.70E+06	8.75E+05	N/A
06216	Unit 2 Steam Tunnel West Valve House	1.70E+06	8.75E+05	N/A
08100	Crib House/Forebay	1.08E+07	5.23E+06	N/A
08401	Forebay	1.08E+07	5.23E+06	N/A
08102	Circulating Water Intake Pipes	1.08E+07	5.23E+06	N/A
09100	Waste Water Treatment Facility (WWTF)	5.33E+05	2.91E+05	N/A

(1)-Indicates Gross Gamma surrogate value derived for piping surveys.
(2)-For the Unit 1 & Unit 2 Containments, Cs-137 was the principle surrogate radionuclide for both H-3 and Sr-90.

Investigation Levels: The investigation levels were based the survey unit classification and the Table 4-1 values and are provided in the individual release records.

Sampling and Analysis Methods to Meet the Data Requirements: Final Status Survey planning and design hinges on coherence with the DQO process to ensure, through compliance with explicitly defined inputs and boundaries, that the primary objective of the survey is satisfied. The DQO process is described in the ZSRP LTP as outlined in Appendix D of MARSSIM.

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. In designing the survey plan, the underlying assumption, or null hypothesis was that residual activity in the survey unit exceeded the release criteria. Rejection of the null hypothesis would indicate that residual activity within the survey unit did not exceed the release criteria. Therefore, the survey unit would satisfy the primary objective of the FSS sample plan.

The primary objective of the FSS sample plan is to demonstrate that the level of residual radioactivity in a survey unit did not exceed the release criteria specified in the LTP and that the potential dose from residual radioactivity is As Low As Reasonably Achievable (ALARA).

4.1.4 Define the Boundaries of the Survey

Boundaries of the Survey: The actual physical boundaries as stated for each survey unit.

Temporal Boundaries: Estimated times and dates for the survey. Scanning and sampling in a survey unit was normally performed only during daylight and dry weather.

Constraints: The most common constraints were the weather, standing water and/or ice in a survey unit.

4.1.5 Develop a Decision Rule

Decision Rule: If any measurement data result exceeded the release criteria, the DQA process would then be used to evaluate alternative actions.

4.1.6 Specify Tolerable Limits on Decision Errors

The Null Hypothesis: Residual radioactivity in the survey unit exceeds the release criteria.

Type I Error: This is also known as the "α" error. This is the error associated with incorrectly concluding the null hypothesis has been rejected. In accordance with LTP section 5.6.4.1.1, the Type I error was set at 0.05 (5%).

Type II Error: This is also known as the "β" error. This is the error associated with incorrectly concluding the null hypothesis has been accepted. In accordance with LTP section 5.6.4.1.1, the Type II error was set at 0.05 (5%).

The Lower Bound of the Gray Region: The LBGR was set at 50% of the OpDCGL. In using the unity rule, the OpDCGL becomes one (1) and the LBGR is set as 0.5.

4.1.7 Optimize Design

Type of Statistical Test: The Sign Test was selected as the non-parametric statistical test for FSS. The Sign Test is conservative as it increases the probability of incorrectly accepting the null hypothesis (i.e., the conclusion will be that the survey unit does not meet the release criteria) and does not require the selection or use of a background reference area.

Number of Systematic Measurements: The number of systematic measurements was determined by the survey unit classification. The required areal coverage for a Class 1 basement survey unit was 100%. The LTP required that sufficient measurements be taken in a Class 1 FSS unit to ensure that 100% of the surface area was surveyed (ISOCS FOV overlapped to ensure that there were no un-surveyed corners and gaps). In cases where the physical configuration or measurement geometry made the acquisition of a 28 m^2 FOV difficult or prohibitive, then the FOV for the ISOCS measurement was reduced provided that the adjusted number of samples remained constant and the minimum areal coverage represented by the FSS unit classification was achieved. To ensure that were no un-surveyed corners and gaps, the number of measurements that were taken in the basement FSS units was adjusted by overlaying the center-point of the ISOCS measurement on a 4 m x 4 m (16 m^2) grid system.

In embedded piping and in long penetrations, measurements were typically acquired with a sodium iodide (NaI) detector that was transported into the pipe/penetration using a push-pull locomotion. The FOV for each measurement was conservatively assumed as 1-foot. Consequently, to achieve 100% areal coverage during survey, a measurement was acquired at 1-foot intervals.

ZSRP LTP Chapter 5, section 5.1 states that concrete core samples would be collected during FSS at 10% of the locations where an ISOCS measurement was collected with the locations selected at random to confirm the HTD to surrogate radionuclide ratios. The concrete core locations were 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.

The coordinates for all of the ISOCS measurement locations were conspicuously marked to designate where to position the survey rig to the center-point of the instrument FOV. The ISOCS detector was then positioned either vertically or horizontally and adjacent to the surface at the center-point of each designated floor or wall measurement location. Each survey measurement location would then be reproducible utilizing permanent markings on the survey unit floor and walls and annotated within the survey comments.

Figure 4-1 depicts an example of the ISOCS measurement location marking system that was employed.

Figure 4-1 – ISOCS Systematic Measurements of Auxiliary Building 542 foot Elevation

Number of Judgmental/Investigational Measurements and Locations: The selection of judgmental samples was at the discretion of the RE. The judgmental measurement locations were typically chosen to measure an area of interest. Areas of interest for judgmental measurements included cracks and crevasses in the surface in question, drains and low points, areas of discoloration, etc. The individual release records identifies when judgmental samples were utilized.

If during the course of performing a FSS, measurement results were encountered that were not as expected for the surface undergoing survey, then an investigation was performed to determine the cause of the discrepancy. Investigational measurements were acquired as part of a documented investigation within the individual survey unit. Investigational measurements were collected within a survey unit to bound areas of elevated activity or to verify that conditions had not changed within a FSS survey unit as a result of adjacent remediation activities.

An example of a location where an investigational measurement was taken is provided in Figure 4-2. ISOCS measurement location No. 278 was identified as a location for an investigation ISOCS measurement due to the uneven surface encountered due to grout settling in a sump drain location. The original ISOCS measurement was taken with the standard geometry identified in the survey plan. An investigational ISOCS measurement was collected at the same location as the original but with a reduced standoff distance to account for the uneven physical geometry.

Figure 4-2 – Investigation of ISOCS Systematic Measurement at Location No. 278 on the Auxiliary Building 542 foot Elevation

Number of Scan Areas and Locations: The frequency of scanning and the specific locations are provided in the release record that is specific for that survey unit.

Number of Samples for Quality Control: The number of quality control samples was 5% percent of sample set. The locations for duplicate samples and replicate scan areas were selected randomly using a random number generator.

Power Curve: The Prospective Power Curve, developed using characterization data and MARSSIM 2000 software, showed adequate power for the survey design in each of the survey units.

A synopsis of the survey designs are provided in Table 4-4.

Table 4-4 – Synopsis of Survey Design

FEATURE	DESIGN BASIS													
Survey Unit #	01100	01110	01111	01112	02100	02110	02112	03202	05100	05119	05120	06100	06105B	06107
Description	U1 CTMT 565 ft	U1 CTMT Under Vessel	U1 CTMT Incore Sump Drain	U1 CTMT PN	U2 CTMT 565 ft	U2 CTMT Under Vessel	U2 CTMT PN	SFP/ Transfer Canal	Aux Bldg.	Aux Bldg. 542 ft EP	Aux Bldg. PN	Turb. Bldg. (1)	Turb. Bldg. 560 ft. EP	U1 Buttress Pit
Area (m^2)	294	2,465	0.86	255	294	2,465	253	723	7,226	294	15	27,135	238	1,596
Number of Measurements	19	155	22	369	19	155	369	76	453	2636	66	28	133	7 judgmental
Spacing	28 m^2 FOV	28 m^2 FOV	1-ft. Interval	1-ft. Interval	28 m^2 FOV	28 m^2 FOV	1-ft. Interval	28 m^2 FOV	28 m^2 FOV	1-ft. Interval	1-ft. Interval	Random	Random	Biased
DCGLs	Operational DCGLs Presented in Tables 2-2 (structures), Table 2-4 (embedded pipe) and Table 2-6 (penetrations)													
Classification (Initial/Final)	1/1	1/1	1/1	1,2 & 3/1	1/1	1/1	1,2 & 3/1	1/1	1/1	1/1	1,2 & 3/1	2/3	2/3	2/3
Investigation Level	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL
Scan Area Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	3% Coverage	3% Coverage	8% Coverage
QC	5% Duplicate Measurements													

FEATURE	DESIGN BASIS												
Survey Unit #	06108	06201	06202	06209	06210	06211	06212	06213	06214	06215	06216	08100	09100
Description	U2 Buttress Pit	U1 570 ft. DG Rooms	U2 570 ft. DG Rooms	U1 Steam Tunnel EP	U2 Steam Tunnel EP	U1 Tendon Tunnel EP	U2 Tendon Tunnel EP	U1 East Valve House	U1 West Valve House	U2 East Valve House	U2 West Valve House	Crib House/ Forebay (2)	WWTF
Area (m^2)	1,596	813	813	47	46	51	42	304	304	240	240	18,254	1,124
Number of Measurements	6 judgmental	51 judgmental	51 judgmental	48	48	52	46	26	26	20 judgmental	20 judgmental	14	71
Spacing	Biased	28 m^2 FOV	28 m^2 FOV	2-ft. Interval	2-ft. Interval	10-ft. Inter	10-ft. Inter	28 m^2 FOV	28 m^2 FOV	28 m^2 FOV	28 m^2 FOV	Random	28 m^2 FOV
DCGLs	Operational DCGLs Presented in Tables 2-2 (structures), Table 2-4 (embedded pipe) and Table 2-6 (penetrations)												
Classification (Initial/Final)	2/3	2/1	2/1	2/3	2/3	2/3	2/3	2/1	2/1	2/3	2/3	2/3	3/1
Investigation Level	>0.5 OpDCGL	>OpDCGL	>OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL	>OpDCGL	>OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL	>0.5 OpDCGL	>OpDCGL
Scan Area Coverage	7% Coverage	100% Coverage	100% Coverage	10% Coverage	10% Coverage	10% Coverage	10% Coverage	100% Coverage	100% Coverage	100% Coverage	100% Coverage	4% Coverage	100% Coverage
QC	5% Duplicate Measurements												

(1) The survey design for survey unit 06100 (Turbine Building) also includes the survey design for the Unit 1 and Unit 2 Circulating Water Discharge Pipe (06105A) and the Unit 1 and Unit 2 Circulating Discharge Tunnels (09200).

(2) The survey design for survey unit 08100 (Crib House) also includes the survey design for the Forebay (08401) and the Unit 1 and Unit 2 Circulating Water Intake Pipes (08102A/B).

4.2 Survey Unit Designation and Classification

Procedure ZS-LT-300-001-002 defines the decision process for classifying an area in accordance with the LTP and MARSSIM. Survey unit classifications are provided in Table 4-5. The justification for each survey unit classification is delineated in the individual release records for each survey unit contained in Appendices 1-12 of this report.

During the FSS the following areas were reclassified:

- Survey units 06213 and 06214 (Unit 1 East and West Valve Houses) were re-classified from Class 3 to Class 1 based on FSS findings.
- Several penetrations in both Unit 1 and Unit 2 Containment were originally given Class 2 and Class 3 classifications in TSD 14-016. All penetrations were given a final classification of Class 1.
- The Turbine Building was originally classified as Class 2 in accordance with the HSA. The classification was changed in accordance with LTP Chapter 5, section 5.5.2.1 to Class 3.
- Due to the fact that radioactive material was transported out of the Auxiliary Building through the Unit 1 and Unit 2 570 foot Diesel Generator Rooms, both were reclassified from Class 3 to Class 1.
- The WWTF was originally classified as Class 3 in accordance with the HSA. The classification was changed in to Class 1.

4.3 Background Determination

During FSS area scanning of embedded pipe and penetrations, ambient backgrounds were determined and the technician established the Alarm Set Point (ASP) based on the background for that scan area. Each applicable survey unit Release Record discusses scan area readings and presents the results of the scan.

4.4 Final Status Survey Sample Plans

The level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. Guidance for preparing FSS plans is provided in procedure ZS-LT-300-001-001. The FSS plan uses an integrated sample design that combines scanning surveys and sampling.

4.5 Survey Design

4.5.1 Determination of Number of Data Points

The FSS of basement surfaces at ZSRP was planned, designed, implemented and assessed as specified in MARSSIM and LTP section 5.6. A survey package was generated for each FSS unit. The same area preparation, area turnover and control measures specified in LTP section 5.6.3 also applied to basement FSS units. The QA requirements specified in LTP section 5.9 also applied 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 was oriented perpendicular to the surface of interest. In most cases, the exposed face of the detector was positioned at a distance of 3 meters above the surface. A plumb or stand-off guide attached to the detector was 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 was reduced, thereby reducing the FOV, which increased the number of measurements to ensure that the required FSS areal coverage was achieved. In most cases, the measurements were acquired using the ISOCS with a geometry that evaluated residual activity at depth.

Table 4-5, which is reproduced from LTP Chapter 5, Table 5-19, presents the adjusted minimum number of ISOCS measurements per FSS Survey Unit.

Table 4-5 – 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).

For embedded pipe and penetration surveys, the required areal coverage of the embedded pipe or penetration is provided in Chapter 5, section 5.5.5of the LTP.

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 OpDCGLs as listed in LTP Chapter 5, Tables 5-12 and 5-14 (reproduced as Tables 2-4 and 2-6 in this report). However, the survey methods are the same for both.

Shallow penetrations or short lengths of embedded pipe that were directly accessible were surveyed using hand-held portable detectors, such as a gas-flow proportional or scintillation detector.

Lengths of embedded pipe or penetrations that could not be directly accessed by hand-held portable detectors were surveyed using applicable sized NaI or Cesium Iodide (CsI) detectors that was inserted and transported through the pipe using flexible fiber-composite rods or attached to a flexible video camera/fiber-optic cable.

The pipe detectors were inserted into each pipe/penetration using a simple "push-pull" methodology, whereby the position of the detector inside the pipe was easily determined in a reproducible manner through the use of tape measures and/or distance encoders. This ensured that a timed one-minute measurement was acquired for every foot increment of pipe travelled.

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 conservatively provided 100% areal coverage of all accessible pipe/sleeve interior surfaces.

The detector output represented the gamma activity in gross cpm. This gamma measurement value in cpm was then converted to dpm using an efficiency factor based on the calibration source. The total activity in dpm was 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 represented a commensurate and conservative gamma surface activity.

The gamma surface activity for each FSS measurement was 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 were inferred to the applicable gamma radionuclide concentration to derive a concentration for each ROC for each measurement taken. The measurement concentration for each ROC was then divided by the applicable OpDCGL to produce a dose fraction for each ROC. The individual ROC dose fractions were then summed to produce an OpSOF for the measurement. There was no Elevated Measurement Comparison (EMC) applicable to embedded pipe or penetrations. Consequently, a measurement OpSOF that exceeded one would have required investigation.

4.5.2 Measurement Locations

The LTP required that sufficient measurements be taken in a Class 1 FSS unit to ensure that 100% of the surface area was surveyed (ISOCS FOV overlapped to ensure that there were no un-surveyed corners and gaps). In cases where physical configuration or measurement geometry made the acquisition of a 28 m^2 FOV difficult or prohibitive, then the FOV for the ISOCS measurement was reduced provided that the adjusted number of samples remained constant and the minimum areal coverage represented by the FSS unit classification was achieved. To ensure that there were no un-surveyed corners and gaps, the number of measurements that were taken in Class 1 FSS basement survey units was adjusted by overlaying the center-point of the 28 m^2 FOV for the ISOCS measurement on a 4 m x 4 m (16 m^2) grid system. During the establishment of the grid system within a survey unit, some measurement locations were not feasible. When a measurement could not be performed, then the RE selected a suitable replacement location to ensure 100% areal coverage was achieved. The FSS Plan was revised for that survey unit to reflect the changes to the number and/or locations of FSS measurements.

For embedded piping and penetrations, each piping system or penetration was identified by plant drawings. TSD 14-016 was used to obtain a description and classification for each piping/penetration system. Specific information with regard to embedded piping and penetration surveys are provided in the release record for that survey unit. Generally, one-minute timed static measurements were taken throughout the accessible portion of that pipe or penetration. The frequency of the measurements was also provided in the survey unit release record. For example, if the embedded piping/penetration system was identified as a Class 1 system, then one measurement would be taken for every foot of pipe interior surface to provide 100% areal coverage for that survey unit.

The total number of measurements actually acquired for each FSS survey unit is provided in Table 4-6.

Table 4-6 – Number of Measurements for FSS

Survey Unit	Non-Parametric Measurements	Quality Control Measurements	Judgmental/ Investigation Measurements	Total Measurements
01100	164	9	0	173
01110	60	3	0	63
01111	22	3	0	25
01112	369	19	0	388
02100	164	9	0	173
02110	54	3	0	57
02112	369	20	0	389
03202	76	4	0	80
05100	425	23	5	453
05119	2,636	180	0	2,816
05120	66	5	0	71
06100	28	3	24	55
06105A	0	0	4	4
9200	0	1	14	15
06105B	134	14	0	148
06107	7	1	0	8
06108	6	1	0	7
06201	51	3	0	54
06202	51	3	0	54
06209	68	4	0	72
06210	60	3	0	63
06211	58	3	0	61
06212	44	3	0	47
06213	0	2	26	28
06214	0	2	26	28
06215	0	1	20	21
06216	0	1	20	21
08100	14	3	17	34
8102 A&B	0	1	5	6
8401	0	1	14	15
09100	70	4	0	74

4.6 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 OpDCGL. 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 measurements and sample analysis) 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.

Specific implementing procedures 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 and executed in the FSS sample plan.

4.6.1 Instrumentation Efficiencies

The source term geometry for ISOCS efficiency calibration, (i.e., concentration depth profile and areal distribution of the residual radioactivity in structures), is required to generate efficiency curves (i.e., efficiency as a function of energy) for the ISOCS gamma spectroscopy measurements. The basis for the majority of the ISOCS efficiency calibrations are documented in Zion*Solutions* TSD 14-022. The typical ISOCS geometry utilized a 28 m^2 FOV; however other geometries used had FOVs that ranged from 10 to 52 m^2, depending on physical access restraints.

Although hand held instruments were used to scan building surfaces in rare instances, these were used principally to survey penetrations and embedded pipe. For embedded pipe, the detectors used included NaI and CsI scintillation detectors. These detectors were housed in protective housings for embedded pipe and paired with instruments designed specifically to log one minute static counts.

These pipe detectors ranged in physical size from less than one inch in diameter (CsI used on the 1.6 inch ID Unit 1 Containment Incore Sump Drain line, e.g.) to up to 3 by 3 inch NaI detectors in special ball shaped housings, paired with scalers/data loggers to cover a wide range of pipe diameters. Each detector was calibrated for specific pipe diameter ranges, with the lowest efficiency in each range used to survey all pipe diameters in that range resulting in accurate to slightly conservative results. These detectors were principally used to detect gamma-emitters and were calibrated using Cs-137 NIST traceable sources. Efficiencies were used to quantify the gamma emitters, and those results were used to infer concentrations of HTD ROC using the maximum ratios from LTP Chapter 5, Table 5-12.

The FSS of penetrations used alpha/beta plastic scintillator detectors and proportional detectors, and these were calibrated to alpha and beta emitters using NIST traceable Th-230 and Tc-99 sources. Although both Cs-137 and Co-60 emit betas, the Co-60 beta is nearly identical in energy and abundance to Tc-99. Efficiencies used were determined from calibration records for each instrument. Proportional and NaI/CsI pipe detector efficiency values were relatively consistent, and as a result, a standard efficiency was used. For the plastic alpha/beta plastic scintillation detectors, efficiencies were wider ranging and for these, the actual calibration efficiencies for each specific detector/scaler data logger pair were used.

4.6.2 Instrumentation Sensitivities

Zion*Solutions* TSD 14-022 provides the initial justification for the selection of reasonably conservative geometries and efficiency calibrations for the ISOCS based on the physical conditions of the remediated surface and the anticipated depth and distribution of activity. All ISOCS measurements were acquired using an approved geometry. One source to detector distance was utilized to ensure the required areal coverage.

Detector response or Quality Control (QC) checks were performed:

1.) Daily or in conjunction with field use.

2.) After replacement of spectroscopy signal chain components (detector preamplifier, Inspector, or Inspector power component).

3.) Loss of power to the ISOCS system, as this may have caused calibration values to not be saved

Background checks were primarily intended to determine if the detector and or associated housing was contaminated. Background checks are a boundary test of the low energy end (12.5-250 keV) and high energy region (251-2048 keV) region of the spectrum. The "Be" and "Ab" flags denote if the check is above or below the low (roughly 470-630 cpm) or high (roughly 460 to 700 cpm) energy region of the spectrum.

The QC check sources or instrument response sources had an energy range that spanned approximately 50% of the operational energy spectrum. FSS surveys conducted using the ISOCS required a valid geometry file to be associated with the analysis performed and analysis sequence. As appropriate, a valid and approved Geometry Composer File was included with the ISOCS File Analysis Structure prior to beginning any field surveys.

The detectors used for the penetrations and embedded pipe surveys were calibrated to capture the readily detectable principal radionuclides of interest The CsI detectors were ideal for small diameter pipes as they have a higher efficiency than the NaI detectors and are very efficient for the energies of interest in small sizes. They are typically manufactured in sizes that range from ½ by ½ inch (diameter and depth) to 1 inch diameter by 1.5 inches in depth. NaI was ideal for the large size pipes in detector sizes ranging from one by one inch to three by three inches. Both types are sensitive to the gamma energies emitted by Co-60 and Cs-137 and are calibrated using Cs-137 to which these detectors are slightly less sensitive to, (and slightly less efficient) than for the Co-60 gamma. So when calibrated to Cs-137, the same detector is approximately 15 to 20% more sensitive to Co-60 than Cs-137 which adds a level of conservatism between the assumed and actual efficiencies.

Proportional counters and alpha/beta scintillators were used to scan the inner surfaces of wall and floor penetrations. Proportional detectors utilize P-10 gas (90% argon, 10% methane) fed thru flexible tubes to the detector housing at a constant flow, ideal for optimizing the quantification of ionizations in the detector body created by beta and, to a lesser extent, gamma emissions and interactions in the detector volume. However, the need for flexible tubing connected to a compressed gas cylinder makes there use challenging. Alpha/beta plastic scintillators require no gas hook up and are very portable and were typically used when surveying penetrations at elevation.

4.6.3 Instrument Maintenance and Control

Control and accountability of survey instruments were maintained to assure the quality and prevent the loss of data. All personnel operating radiological instruments, analysis equipment, measurement location equipment etc., were qualified to operate any assigned equipment and recognize off normal results and indications.

4.6.4 Instrument Calibration

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

Instrumentation used for FSS was calibrated and maintained in accordance with approved Zion*Solutions* site calibration procedures. Radioactive sources used for calibration were traceable to the NIST and were obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium detector was used, suitable NIST-traceable sources were used for calibration, and the software set up appropriately for the desired geometry. If vendor services were used, these were obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality. Source checks were performed prior to and after each survey shift to ensure satisfactory instrument performance in accordance with the DQO's.

4.7 Survey Methodology

The LTP specifies the minimum amount of scanning required for each class as summarized in Table 4-7. The total fraction of scanning coverage is determined during the DQO process with the amount, and location(s) based on the likelihood of finding elevated activity during FSS.

Table 4-7 – Recommended Scan Coverage

Area Classification	Surface Scans
Class 1	100%
Class 2	10% to 100%, Systematic and Judgmental
Class 3	Judgmental

4.7.1 ISOCS Surveys

For basement structures, compliance with the unrestricted release criteria was demonstrated through direct measurements using the ISOCS. Zion*Solutions* TSD 14-022 provided the initial justification for the selection of reasonably conservative geometries for efficiency calibrations for the ISOCS based on the physical conditions of the remediated surface and the anticipated depth and distribution of activity. Prior to implementing each sample plan for FSS, the physical condition of the surfaces to be surveyed was assessed to ensure that the geometry was not significantly changed from that assumed in TSD 14-022. ISOCS measurements were acquired using the geometries identified in the individual FSS sample plans. The details pertaining to specific ISOCS geometries are provided in the applicable Release Record for the survey unit.

Table 4-4 provides a summary of the area scanned in each survey unit during FSS.

4.7.2 Embedded Pipe Surveys

Once remediation was completed in a section of pipe to the extent practicable, the residual radioactivity remaining in each accessible section of embedded piping was assessed and quantified by direct survey. The approach used for the radiological survey of the interior surfaces of embedded piping involved the insertion of a detector that was attached to the See Snake camera system and transported through the pipe to the maximum deployment length, or to a location of drain drop. A simple "push-pull" methodology was used, whereby the position of the detector in the piping system could be easily determined in a reproducible manner. Footage was tabulated on the See Snake, then measurements were obtained at each one-foot location while backing out of the pipe section.

The piping detectors were configured in a fixed geometry relative to the surveyed surface, thus creating a situation where a defensible efficiency could be calculated. The detectors were then deployed into the actual pipe and timed measurements were acquired at an interval of one measurement for every foot of pipe. A conservative "area of detection" of one-foot was assumed. It was also conservatively assumed that any activity inside of the pipe was uniformly distributed in the area of detection.

For each detector in each diameter of pipe, an instrument efficiency factor was derived by placing a flexible Cs-137 radiological plane source into a pipe jig, depending on the diameter of the pipe to be surveyed. This created a geometry similar to what would be encountered in the actual pipe. Using the known source activity, an efficiency factor was then derived for the detector in that geometry.

A background value was also determined for the detector/instrument combination to be used prior to deployment. The background value was obtained at the location where the pre-use response check of the instrument was performed. The background value was primarily used to ensure that the detector had not become cross-contaminated by any previous use. Background was not subtracted from any measurement.

Daily prior to use and daily following use, each detector was subjected to an Operational Response Check in accordance with procedure ZS-LT-300-001-006, *"Radiation Surveys of Pipe Interiors Using Sodium/Cesium Iodide Detectors"* (Reference 29). The Daily Operational Response Check compared the background response and the response to check sources ranges established for normal background and detector source response to ensure that the detector was working properly.

Once the detector was determined to be fully functional, it was then deployed to the field for insertion into the targeted piping. A one-minute static measurement was acquired at each foot traversed into the pipe. The detector output represented the gamma activity for each one-minute timed measurement in units of gross cpm. The gamma measurement value in units of cpm was then converted to units of dpm using the efficiency factor for the detector applicable to the diameter of the pipe surveyed.

Each measurement assumed a conservative "area of detection" for the detector of one foot. This assumption is conservative because there is additional instrument response from contamination located in the pipe at distances outside of the "area of detection". Consequently, the total activity from the measurement, in units of dpm is adjusted for the total effective surface area commensurate with the pipe diameter and the assumed "area of detection", resulting in measurement results in units of dpm/100 cm^2. Using the appropriate conversion factors, the result is then converted to units of pCi/m^2. This measurement result represents a commensurate and conservative gamma surface activity for the one-foot of pipe surface where the measurement was taken.

After completion of the FSS measurements in the pipe, the sample plan was reviewed to confirm the completeness of the survey and the survey data was validated in accordance with procedure ZS-LT-300-001-004. Data processing included converting measurement data into reporting units, validating instrument applicability and sensitivity, calculating relevant statistical quantities, and verification that all DQO had been met. In accordance with the procedure, a preliminary Data Assessment was prepared for each section of pipe surveyed.

4.7.3 Penetrations Surveys

The level of effort associated with planning a penetration survey is based on the complexity of the survey and nature of the hazards. Guidance for preparing FSS plans is provided in procedure ZS-LT-300-001-001. The FSS plans for the survey of penetrations employed sample designs that combined hand-held scanning with static measurements and pipe detector survey methodologies.

The survey method for large diameter penetrations (>12") differs from smaller penetrations due to measurement sensitivity (i.e. MDC's) differences in the two size regimes. The larger penetrations were surveyed using a similar approach as for traditional building surface surveys whereas the smaller penetrations were surveyed with a single detector advanced through the penetration. Measurements were conducted at one-foot intervals throughout the length of the penetration using either using a NaI detector or a hand-held detector to ensure 100% areal coverage of the pipe interior surface.

4.7.4 Quality Control Surveys

The method used for evaluating QC replicate samples collected in support of the FSS program is specified in the QAPP for Characterization and FSS. QC replicate data was assessed using criteria taken from the USNRC Inspection Manual, Inspection Procedure 84750, "*Radioactive Waste Treatment and Effluent and Environmental Monitoring*" (Reference 30).

A minimum of 5% of the sample locations used in the FSS design were selected randomly using the Microsoft® Excel "RANDBETWEEN" function and submitted as "replicates." Most replicates taken for FSS were field replicates, that is, samples obtained from one location, homogenized, divided into separate containers, and treated as separate samples. These samples were then used to assess errors associated with sample heterogeneity, sample methodology, and analytical procedures. It is desirable that when analyzed, there is agreement between the replicates resulting in data acceptance. If there was no agreement between the samples, the RE evaluated the magnitude and impact on survey design, the implementation and evaluation of results as well as the need to perform confirmatory sampling. If the RE had determined that the discrepancy affected quality or was detrimental to the implementation of FSS, then a Condition Report would have been issued.

For scan surveys, replicate measurements were taken at a frequency of 5% of the scan or 1-minute count locations randomly. For ISOCS measurements, replicate measurements were taken using the same random location selection process.

To maintain the quality of the FSS, isolation and control measures were implemented throughout FSS activities until there was no risk of recontamination from decommissioning or when the survey area will be released from the licenses. Following FSS, and until the area is released, a semi-annual surveillance will be performed on FSS completed survey units. This includes an inspection of area postings, inspection of the area for signs of dumping or disturbance and some sampling from selected locations. In the event that isolation and control measures were compromised, a follow-up survey may be performed after evaluation.

5. SURVEY FINDINGS

Procedure ZS-LT-300-001-004 provides guidance to C/LT personnel to interpret survey results using the DQA process during the assessment phase of FSS activities.

The DQA process is the primary evaluation tool to determine that data is of the right type, quality and quantity to support the objectives of the FSS sample plan. The five steps of the DQA process are:

- Review the sample plan DQOs and the survey design.
- Conduct a preliminary data assessment.
- Select the statistical test.
- Verify the assumptions of the statistical test.
- Draw conclusions from the data.

Data validation descriptors described in MARSSIM Table 9.3 were used during the DQA process to verify and validate collected data as required by the QAPP for Characterization and FSS.

Hand held instruments utilized for surveying embedded pipe and wall and floor penetrations were calibrated with NIST traceable sources, and the efficiencies used to quantify results taken from those calibrations. Prior to, and following each use, each hand held instrument was operationally verified using check sources to verify the instruments were operating within pre-determined acceptable ranges.

5.1 Survey Data Conversion

During the data conversion, the RE evaluated raw data for problems or anomalies encountered during sample plan activities (sample collection and analysis, handling and control, etc.) including the following:

- Recorded data,
- Missing values,
- Deviation from established procedure, and
- Analysis flags.

Once resolved, initial data conversion, which is part of preliminary data assessment was performed and consisted of converting the data into units relative to the release criteria (e.g., pCi/g) and calculating basic statistical quantities (e.g., mean, median, standard deviation). Table 5-1 provides a summary of the basic statistical properties for Phase 2 systematic sample populations.

Table 5-1 – Basic Statistical Properties of Phase 2 Survey Unit Non-Parametric Measurements

Survey Unit	Description	Class	# of Measurements	Mean OpSOF	Max OpSOF	# OpSOF > 1	Mean BcSOF	Dose to Survey Unit	Radionuclide Statistical Summary					
									Co-60			Cs-137		
									Max	Mean	Std. Dev	Max	Mean	Std. Dev
								(mrem/yr)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)	(pCi/m^2)
01100	Unit 1 CTMT above 565 foot	1	173	0.124	1.156	1	0.019	0.463	9.01E+05	7.40E+04	9.91E+04	9.07E+05	1.59E+05	1.41E+05
01110	Unit 1 CTMT Under Vessel	1	63	0.532	0.738	0	0.196	4.888	3.90E+06	7.74E+05	1.13E+06	1.32E+07	4.37E+06	3.35E+06
01111	Unit 1 CTMT Incore Sump Drain	1	25	0.363	5.793	1	0.049	1.221	2.01E+07	1.26E+06	4.26E+06	2.91E+08	1.82E+07	6.15E+07
01112	Unit 1 CTMT Penetrations	1	388	0.564	8.600	43	0.059	1.468	1.05E+07	6.87E+05	1.70E+06	1.51E+08	9.92E+06	2.46E+07
02100	Unit 2 CTMT above 565 foot	1	173	0.063	0.985	0	0.009	0.219	8.23E+05	4.49E+04	7.74E+04	5.30E+05	6.08E+04	6.71E+04
02110	Unit 2 CTMT Under Vessel Area	1	57	0.147	0.457	0	0.106	2.650	4.60E+06	8.54E+05	1.33E+06	1.11E+07	1.71E+06	2.23E+06
02112	Unit 2 CTMT Penetrations	1	369	0.121	0.685	0	0.008	0.206	8.35E+05	1.48E+05	1.10E+05	1.21E+07	2.13E+06	1.59E+06
03202	SFP/Transfer Canal	1	80	0.139	1.843	2	0.033	0.829	3.15E+05	6.39E+04	5.68E+04	1.40E+07	9.48E+05	2.24E+06
05100	Auxiliary Building Basement	1	453	0.143	2.189	16	0.075	1.868	2.46E+07	8.88E+05	3.17E+06	7.46E+07	3.15E+06	6.86E+06
05119	Auxiliary Building Embedded Floor Drains	1	2816	0.170	0.839	0	0.007	0.170	N/A	N/A	N/A	N/A	N/A	N/A
05120	Auxiliary Building Penetrations	1	71	0.027	0.279	0	0.002	0.053	1.43E+05	1.37E+04	2.93E+04	1.17E+07	1.12E+06	2.40E+06
06100	Turbine Building Basement	3	55	0.246	1.346	1	0.021	0.523	1.73E+05	7.45E+04	6.89E+04	1.69E+06	1.56E+05	3.15E+05
06105A	Circulating Water Discharge Pipe	3	4	0.146	0.417	0	0.012	0.310	2.64E+05	7.89E+04	1.24E+05	1.26E+05	6.41E+04	6.00E+04

Table 5-1 (continued) Basic Statistical Properties of Phase 2 Survey Unit Non-Parametric Measurements

Survey Unit	Description	Class	# of Measurements	Mean OpSOF	Max OpSOF	# OpSOF > 1	Mean BcSOF	Dose to Survey Unit	Radionuclide Statistical Summary					
									Co-60			Cs-137		
									Max	Mean	Std. Dev	Max	Mean	Std. Dev
								(mrem/yr)	(pCi/m²)	(pCi/m²)	(pCi/m²)	(pCi/m²)	(pCi/m²)	(pCi/m²)
09200	Unit 1 & 2 Circulating Water Discharge Tunnels	3	17	0.285	2.252	2	0.127	3.180	9.51E+06	1.18E+06	3.02E+06	2.69E+05	5.40E+04	9.05E+04
06105B	Turbine Building Embedded Pipe	3	148	0.011	0.028	0	0.001	0.011	2.37E+04	9.73E+03	4.10E+03	1.95E+06	8.00E+05	3.37E+05
06107	Unit 1 Turbine Building Buttress Pit	2	8	0.011	0.034	0	0.001	0.023	1.09E+04	4.95E+03	3.54E+03	2.92E+04	6.95E+03	1.01E+04
06108	Unit 2 Turbine Building Buttress Pit	2	7	0.010	0.022	0	0.001	0.021	5.66E+03	2.47E+03	2.09E+03	6.90E+03	5.32E+03	1.74E+03
06201	Unit 1 Turbine Building 570' Diesel Fuel Storage	1	54	0.054	0.177	0	0.004	0.102	3.20E+04	1.17E+04	1.10E+04	3.72E+04	1.25E+04	1.37E+04
06202	Unit 2 Turbine Building 570' Diesel Fuel Storage	1	54	0.043	0.228	0	0.004	0.091	5.51E+04	1.51E+04	1.24E+04	2.51E+05	2.89E+04	4.34E+04
06209	Unit 1 Steam Tunnel Floor Drain	3	72	0.007	0.018	0	0.001	0.020	9.83E+04	3.80E+04	1.95E+04	8.05E+06	3.11E+06	1.60E+06
06210	Unit 2 Steam Tunnel Floor Drain	3	63	0.002	0.003	0	0.000	0.006	1.56E+04	1.19E+04	1.33E+03	1.28E+06	9.73E+05	1.09E+05
06211	Unit 1 Tendon Tunnel Floor Drain	3	61	0.018	0.074	0	0.000	0.009	4.46E+04	1.11E+04	4.77E+03	3.66E+06	9.08E+05	3.91E+05
06212	Unit 2 Tendon Tunnel Floor Drain	3	47	0.014	0.016	0	0.000	0.007	9.69E+03	8.23E+03	5.97E+02	7.94E+05	6.74E+05	4.88E+04
06213	Unit 1 Steam Tunnel East Valve House	1	28	0.448	4.213	2	0.127	3.186	4.26E+04	1.50E+04	1.30E+04	7.06E+06	6.86E+05	1.43E+06
06214	Unit 1 Steam Tunnel West Valve House	1	28	0.239	1.817	1	0.053	1.324	3.41E+04	1.61E+04	1.12E+04	3.03E+06	3.20E+05	6.31E+05
06215	Unit 2 Steam Tunnel East Valve House	3	21	0.096	0.327	0	0.008	0.205	6.14E+04	1.91E+04	1.68E+04	5.52E+05	6.91E+04	1.25E+05

Table 5-1 (continued) Basic Statistical Properties of Phase 2 Survey Unit Non-Parametric Measurements

Survey Unit	Description	Class	# of Measurements	Mean OpSOF	Max OpSOF	# OpSOF > 1	Mean BcSOF	Dose to Survey Unit (mrem/yr)	Radionuclide Statistical Summary: Co-60 Max (pCi/m²)	Co-60 Mean (pCi/m²)	Co-60 Std. Dev (pCi/m²)	Cs-137 Max (pCi/m²)	Cs-137 Mean (pCi/m²)	Cs-137 Std. Dev (pCi/m²)
06216	Unit 2 Steam Tunnel West Valve House	3	21	0.109	0.304	0	0.009	0.231	4.14E+04	1.77E+04	1.37E+04	4.03E+05	9.95E+04	1.30E+05
08100	Crib House	3	15	0.000	0.000	0	0.000	0.000	1.84E+02	6.12E+01	6.38E+01	2.78E+02	5.82E+01	8.20E+01
8401	Forebay	3	15	0.053	0.064	0	0.020	0.503	8.22E+03	6.71E+03	6.04E+02	6.77E+05	5.52E+05	4.98E+04
08102	Unit 1 & 2 Circulating Water Intake Pipes	3	5	0.002	0.006	0	0.001	0.018	3.16E+04	8.99E+03	1.52E+04	2.08E+03	7.86E+02	8.79E+02
09100	Waste Water Treatment Facility (WWTF)	1	74	0.013	0.236	0	0.013	0.335	4.99E+04	7.06E+03	8.38E+03	3.43E+04	1.12E+04	8.27E+03

5.2 Survey Data Verification and Validation

Items supporting DQO sample design and data were reviewed for completeness and consistency. This included:

- Classification history and related documents,
- Site description,
- Survey design and measurement locations,
- Analytic method and detection limits and validation that the required analytical method(s) were adequate for the ROC,
- Sampling variability provided for the radionuclides of interest,
- QC measurements have been specified,
- Survey and sampling result accuracy have been specified,
- MDC limits,
- Field conditions for media and environment, and
- Field records.

Documentation, as listed, was reviewed to verify completeness and that it was legible:

- Field and analytical results,
- CoC,
- Field Logs,
- Instrument issue, return and source check records,
- Instrument downloads, and
- Measurement results relative to measurement location.

After completion of these previously mentioned tasks, a Preliminary Data Assessment record was initiated. This record served to verify that all data were in standard units in relation to the DCGLs and performed the calculation of the statistical parameters needed to complete data evaluation which at a minimum, included the following:

- The number of observations (i.e., samples or measurements),
- The range of observations (i.e., minimum and maximum values),
- Mean,
- Median, and
- Standard deviation.

In order to adequately evaluate the data set, consideration as additional options included the coefficient of variation, measurements of relative standing (such as percentile), and other statistical applications as necessary (frequency distribution, histograms, skew, etc.). Finalization of the data review consisted of graphically displaying the data in distributions and percentiles plots.

5.3 Anomalous Data/Elevated Scan Results and Investigation

FSS survey data was assessed to determine if the data set in question met the DQO process. This process was documented in accordance with ZS-LT-300-001-004, "*Final Status Survey Data Assessment*."

If during the assessment, it was determined that the data did not meet the DQO's identified in the survey package for that area, then an investigation would have been initiated

The DQO process was used to evaluate the remediation, reclassification and/or resurvey actions to be taken if an investigation level was exceeded. Based upon the failure of the statistical test or the results of an investigation, LTP Chapter 5, Table 5-26 presents the actions that would be required.

5.3.1 Unit 1 and Unit 2 Containment Under Vessel (Survey Units 01110 and 02110)

In accordance with LTP Chapter 5, section 5.1, a concrete core was required to be taken at 10% of the locations selected for an FSS ISOCS measurement. For the FSS of the Under Vessel areas in both Containments, the survey design required the acquisition of a minimum of six FSS confirmatory concrete cores. The LTP also assumed that HTD concentrations would be inferred. Section 5.2.11 of the LTP states, "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 maximum ratios used to infer HTD concentrations during compliance are presented in Table 5-15 from LTP Chapter 5, section 5.2.11.

As previously stated, a concrete core was required to be taken at 10% of the locations selected for an FSS ISOCS measurement in the Unit 1 and Unit 2 Containment Under Vessel areas. The purpose of the core samples was to ensure that the ratios used to infer the HTD concentrations remained valid. During the remediation process, it was acknowledged that the concrete surfaces that were represented by the continuing characterization concrete samples were remediated twice and the actual concrete that was sampled (original concrete surface to a depth of ½ inch) had been removed and disposed of as radioactive waste. At least a foot of concrete was removed from both Under Vessel floors and up to 6 inches of concrete removed from the walls.

Due to the amount of remediation that occurred on the Unit 1 and Unit 2 Under Vessel concrete, ZSRP took an additional 19 concrete cores in each that represented the as-left condition of the Under Vessel concrete following concrete remediation. A review of the analysis of the post remediation concrete core data indicated that almost all of the ratios to Cs-137 for H-3 and Sr-90 exceeded the maximum ratios from LTP section 5.2.11, Table 5-15. A review of the results clearly show that the cause can be attributed to the fact that the majority of the less soluble source term activity for Cs-137 was contained with the near surface concrete that was remediated and removed (within a minimum of 6 inches) while the more soluble ROC (H-3 and Sr-90), while present in lesser concentrations than present in the pre-remediated concrete, became the dominant radionuclide in the relationship with Cs-137. Due to the significant reduction in the concentrations of the gamma-emitting ROC (many at MDC), the H-3 and Sr-90 concentrations were not well correlated with Cs-137 and the use of a ratio with Cs-137 to infer a concentration for the HTD ROC was no longer defensible.

On April 4, 2018, ZSRP submitted a proposal to the USNRC for an alternate approach to use the actual HTD concentrations from the 19 end-state cores to demonstrate compliance as opposed to surrogate ratios. ZSRP proposed to use measured concentrations of each HTD ROC in units of pCi/g for each of the nineteen (19) locations and, assuming a depth of ½ inch (1.27 cm) and a concrete density of 2.35 g/cm^3, converting the concentrations to units of pCi/m^2. While it was acknowledged that the depth of contamination for the HTD ROC was greater than ½ inch, it was also proposed to use the maximum concentration to conservatively compensate for the additional source term at depth. The concentration would then be divided by its respective OpDCGL to derive an OpSOF.

For the FSS of both the Unit 1 and Unit 2 Containment Under Vessel concrete, the maximum measured concentrations of H-3, Ni-63 and Sr-90 in the 19 concrete core samples were used to extrapolate a "worst-case" dose consequence from the presence of HTD ROC. An OpSOF was calculated for each of the ISOCS measurements however, only the gamma results were included. Instead of inferring concentrations for the HTD OpSOF using a surrogate, the maximum measured OpSOF from the HTD ROC was added to the OpSOF for the gamma results. Using this approach, no measurement exceeded an OpSOF of one. The OpSOF (including the addition of the maximum SOF from HTD) for each measurement was used as the sum value for the Sign Test. Passing the Sign Test demonstrates that the mean activity for each ROC is less than the $OpDCGL_B$ at a Type 1 decision error of 0.05. The sample data passed the Sign Test. The null hypothesis was rejected. Compliance with the dose-based unrestricted release criteria was again demonstrated in accordance with the process presented in the LTP as well as the proposed approach for accounting for the presence of HTD ROC.

Demonstrating compliance based on dose consequence from the actual measured concentrations for HTD ROC was a reasonable approach as the spatial distribution of the concrete cores is representative and, due to the extensive remediation and removal of source term, particularly Cs-137, the ratios used to infer H-3 and Sr-90 using the Cs-137 as a surrogate were no longer consistent or reasonably correlated.

In April 2018, ORISE performed confirmatory surveys of the Unit 1 and Unit 2 Containment basements. The confirmatory surveys consisted of several ISOCS measurements and additional concrete core samples. Upon review of the results, the USNRC had questions pertaining to measured activity for H-3 in certain concrete core samples extrapolated over a 6-inch depth verses a ½-inch depth and the potential to exceed the BcDCGL in that scenario. To address the USNRCs concerns, ZSRP agreed to remove a minimum of 1 to 4 inches of additional concrete from around the location of these cores.

ZSRP commenced removal of the additional concrete commencing in June of 2018. Due to the location of the concrete designated for removal, it was more effective to remove all of the concrete, exposing the steel liner. During the execution of this evolution, ZSRP controlled the spread of concrete dust, wiped down adjacent areas after concrete removal and performed an extensive post-work contamination survey. All survey results indicted no detectable loose surface contamination.

Following the completion of concrete removal, it was agreed to acquire and analyze two concrete core samples from each of the zones that were remediated. As no concrete remained in several locations, no concrete core sample was acquired. In other locations where concrete still remained, two concrete core samples were acquired under USNRC observation. While H-3 was still detectable in these samples, the concentrations were significantly less than concentrations that would be of concern. Upon completion of the sampling, the USNRC provided concurrence to backfill both Containment basements.

5.3.2 Unit 1 Containment Penetrations (Survey Unit 01112)

Based on historical assessment, exposure to radioactive materials and system use, the Unit 1 Containment Building penetrations were initially classified as Class 1, Class 2, or Class 3. As a conservative measure, the classifications of all Unit 1 Containment Building penetrations were changed to Class 1. Consequently, sufficient measurements were taken in all Unit 1 Containment Building penetrations to ensure 100% areal coverage of all accessible internal surfaces within the penetrations.

5.3.3 Unit 2 Containment Penetrations (Survey Unit 02112)

Based on historical assessment, exposure to radioactive materials and system use, the Unit 2 Containment Building penetrations were initially classified as Class 1, Class 2, or Class 3. As a conservative measure, the classifications of all Unit 2 Containment Building penetrations were changed to Class 1. Consequently, sufficient measurements were taken in all Unit 2 Containment Building penetrations to ensure 100% areal coverage of all accessible internal surfaces within the penetrations.

5.3.4 Auxiliary Building Basement (Survey Unit 05100)

Chapter 4 of the ZSRP LTP states that remediation beyond that required to meet the release criteria is unnecessary and that the remaining residual radioactivity in structures was ALARA.

Once the basement was turned over for FSS, no additional remedial activities were performed within the Auxiliary Building. However, during the FSS survey of the Unit 1 Containment penetrations in April of 2018, debris fell out of penetration numbers T123 & T124 and onto the Auxiliary Building basement floor. The floor and adjacent wall were surveyed and indicated 15,000 dpm/100 cm^2 direct frisk and <1,000 dpm/100 cm^2 removable. A direct reading on the adjacent wall indicated 1.4 mR/hr (fixed). The debris was removed and follow-up surveys indicated <1,000 dpm/100 cm^2 (direct frisk) and <1,000 dpm/100 cm^2 (removable). This incident was captured and documented on April 9, 2018 through the generation of Condition Report (CR) ES-ZION-CR-2018-0510.

In May of 2018, remediation work was performed on the SFP pad, which was located adjacent to the west boundary of the Auxiliary Building. The potential for the remedial activities to radiologically impact the Auxiliary Building basement floor and walls existed. Therefore, Remedial Action Support Survey (RASS) B1-05100ZF was developed to verify that the events described in the CR and the SFP remediation did not change the radionuclide inventory identified in the Auxiliary Building basement FSS survey.

The results of the RASS indicated that neither the CR event nor the SFP remedial activities adversely affected the Auxiliary Building basement FSS.

Twenty-five (25) ISOCS measurement locations on the floor and walls were re-assessed using ISOCS direct measurements at the same locations that were analyzed previously. The re-survey results were compared to the original (FSS) measurements. The comparison indicated the following:

- Five (5) locations exhibited Cs-134 levels >20% more than the original levels identified during FSS. However, all five (5) measurements were less than the OpSOF and therefore of minimal dose significance.
- Eleven (11) locations exhibited Cs-137 levels >20% more than the original levels identified during FSS.

The average difference between the twenty-five (25) original ISOCS measurements and the corresponding RASS measurements for Co-60 was +12.3%. The average difference between the twenty-five (25) original ISOCS measurements and the corresponding RASS measurements for Cs-137 was -14.3%. Since Cs-137 has a much greater abundance in the isotopic mix identified in Table 4, the lower readings for Cs-137 had a much greater effect on the OpSOF than Co-60. The mean OpSOF for the 25 sample results for the RASS was 0.13, as compared to the OpSOF for the original FSS results which was 0.17.

Therefore, neither the event described in CR ES-ZION-ACT-2018-0510 nor the SFP remediation had any measurable effect on the FSS data for survey unit 05100.

5.3.5 Spent Fuel Pool/Transfer Canal (Survey Unit 03202)

Following the completion of structural remediation, 19 judgmental ISOCS measurements were taken of the exposed concrete in an effort to determine if remediation was sufficient. Hand scanning still indicated elevated measurements along the east edge of the concrete pad adjacent to the Auxiliary Building. It was speculated that the elevated scan measurements were due to "shine" from the ledges that were previously identified and not from insufficient remediation. Consequently, lead blankets were placed on the ledges (outside of the SFP/Transfer Canal survey unit) prior to taking additional ISOCS measurements.

An additional 19 judgmental ISOCS measurements were collected. These measurements were taken as part of a characterization effort and were not designed to demonstrate compliance. The results verified that the gamma shine coming from the elevated ledge areas would not impact the successful implementation of FSS of the SFP/Transfer Canal basement survey unit as long as the shielding remained in place.

5.3.6 Turbine Building Basement and Circulating Water Discharge Tunnels (Survey Units 06100 and 09200B)

In accordance with ZSRP LTP Chapter 5, section 5.5.2.1 and Table 5-19, the Turbine Building basement survey unit, which includes the surface area of the Unit 1 and Unit 2 Steam Tunnels, the Unit 1 and Unit 2 Main Steam Valve Houses, the Unit 1 and Unit 2 570 ft. Diesel Generator Rooms, The Unit 1 and Unit 2 Tendon Tunnels, the Circulating Water Discharge Tunnels, the Circulating Water Discharge Pipe and the Circulating Water Intake Pipes are classified as MARSSIM Class 3. When the FSS of the Turbine Building occurred in March of 2016, it was performed at risk in accordance with Revision 0 version of the LTP, which was not approved. The survey design for all applicable survey units utilized Basement Inventory Levels (BIL) as the OpDCGLs had not yet been developed. The initial analysis of the FSS data was directly compared against the BILs to determine the SOF of individual measurements and to derive the values used for the Sign Test. In addition, other commitments from Revision 2 of the LTP, such as the requirement to acquire concrete core samples for HTD ROC analysis were not required at the time the surveys were performed. When compared against the BILs, all measurements taken for the FSS of these survey units were less than a SOF of 0.5 and decommissioning decisions were made based upon those results. However, for this Release Record, the measurement results taken in 2016 were compared against the

OpDCGLs from the approved Revision 2 of the LTP. As the OpDCGLs are significantly more conservative than the BILs, 5 measurements taken during the FSS of the Turbine Building in 2016 exceeded 50% of the OpDCGL. No measurements exceeded the BcDCGLs. When the survey was performed in 2016, no investigations were performed as required by LTP Chapter 5, section 5.6.4.6 and, no assessment was made to determine if reclassification was appropriate in accordance with LTP Chapter 5, section 5.6.4.6.1. By the time this discrepancy was identified, the Turbine Building basement void had been completely backfilled and additional investigations were not possible. In addition, it should also be noted that with the exception of the two measurements taken in the Unit 2 Discharge Tunnel, all measurements were less than a Base Case Sum of Fractions (BcSOF) of one when compared against the BcDCGLs. (BcSOF of 0.9411 and 0.6860). Despite these differences in LTP Rev 0 vs Rev 2 compliance, sufficient measurements were acquired to adequately quantify the radiological source term that remains in the Turbine Building footprint and that the dose assigned is representative and conservative. Upon discovery of these differences during preparation for this submittal, a Condition Report (ES-ZION-CR-2019-0020) was initiated to document the issue and to specify follow-up corrective actions.

Prior to backfill, a confirmatory survey of the Turbine Building basement was performed by ORISE with no findings.

5.3.7 Unit 1 and Unit 2 570 foot Diesel Generator Rooms (Survey Units 06201 and 06202)

The Unit 1 and Unit 2 570 foot Diesel Generator rooms were also initially classified as MARSSIM Class 3. However, during the course of decommissioning, unpackaged radioactive material was transported through these areas from the Auxiliary Building. Consequently, the Unit 1 and Unit 2 570 foot Diesel Generator rooms were reclassified during decommissioning to MARSSIM Class 1.

5.3.8 Unit 1 East and West Steam Tunnel Valve Houses (Survey Units 06213 and 06214)

The Unit 1 East and West Steam Tunnel Valve Houses were initially classified as Class 3 survey units. However, when the initial FSS was performed, there were several locations identified by ISOCS as having levels > 50% of the OpDCGLs. The areas were investigated, re-classified and FSS was performed as Class 1 survey units.

5.4 Evaluation of Number of Sample/Measurement Locations in Survey Units

An effective tool utilized to evaluate the number of samples collected in the sampling scheme is the Retrospective Power Curve. The Retrospective Power Curve shows how well the survey design achieved the DQOs. For reporting purposes, all Release Records include a Retrospective Power Curve analysis indicating that the sample design had adequate power to pass the FSS release criteria (i.e. adequate number of samples was collected).

The Sign Test was selected as the statistical test for all Release Records submitted in this report. This test, performed in accordance with ZS-LT-300-001-004, along with the Retrospective Power Curve demonstrates survey design adequacy. If the data passed the Sign Test and Retrospective Power Curve, the null hypothesis is rejected and the survey unit can be released with no further actions required. For reporting purposes, all survey unit Release Records passed the Sign Test, indicating that the survey design was adequate (i.e. adequate number of samples was collected).

5.5 Comparison of Findings with Derived Concentration Guideline Levels

The SOF or "unity rule" was applied to FSS data in accordance with the guidance provided in Section 2.7 of NUREG-1757, Vol. 2, and the LTP. This was accomplished by calculating a fraction of the OpDCGL for each sample or measurement by dividing the reported concentration by the OpDCGL. If a sample had multiple ROC, then the fraction of the OpDCGL for each ROC was summed to provide an OpSOF for the sample.

If a surrogate concentration was inferred as part of the survey design for the FSS, then the inferred HTD ROC concentration using the maximum ratios from LTP Chapter 5, Table 5-15 was used to derive the OpSOF.

A BcSOF was calculated for each ROC by dividing the reported mean concentration by the BcDCGL. A BcSOF of 1 is equivalent to the decision rule, meaning any measurement with a BcSOF of 1 or greater, would not meet the 25 mR/yr release criteria. The mean BcSOF was multiplied by 25 to establish the dose attributed to the survey unit. The mean BcSOF and equivalent dose contribution for each Phase 2 survey unit is provided in Table 5-1.

5.5.1 Basement Surface Area Adjustments

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. Table 5-2 lists the surface survey units that contribute to each basement. This table is reproduced from LTP Chapter 5, Table 5-22.

Table 5-2 – 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, Circulating Water 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 LTP Chapter 6, 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 5-3, which is reproduced from LTP Chapter 6, section 6.6.8.1, Table 6-23.

Table 5-3 – 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 BcSOF for basements that have dose contributions from multiple surface survey units is calculated in accordance with Equation 2 below. For the areas specified in Footnote 1 of Table 5-2, the $SOF_{Bi,B}$ to be used in Equation 2 is based on the mean of the judgmental samples/measurements.

Equation 2

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

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 5-3) for basement (B)
$SOF_{Bi,B}$	=	SOF_B for survey unit (i) in basement (B)

5.5.2 Compliance Equation

There are four distinct source terms for the end-state at Zion: backfilled basements, soil, buried piping and groundwater. Demonstrating compliance with the dose criterion requires the summation of dose from the four source terms. The final compliance dose will be calculated using LTP Chapter 6, Equation 6-11, reproduced below as Equation 3, after FSS has been completed 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 mean BcSOF of FSS results plus the dose from any identified elevated areas. The compliance dose must be less than 25 mrem/yr. The dose contribution from each ROC is accounted for using the BcSOF to ensure that the total dose from all ROC does not exceed the dose criterion.

Equation 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 term for each basement includes the dose contributions from wall and floor surfaces within the basement, the dose contribution from embedded pipe within the basement, the dose contribution from penetrations within the basement and the dose contribution from concrete fill in the basement when clean concrete debris was used as fill. Each (structural surfaces, embedded pipe and penetrations) are surveyed separately during FSS. The dose from clean concrete fill is predetermined in accordance with LTP Chapter 5, Table 5-16, which is conservatively based on a maximum allowable MDC of 5,000 dpm/100cm^2.

Basement surface area adjustments (i.e. increases) as described in the previous section of this report were applied to the structure surface DCGL calculation for certain basements to ensure that the DCGLs accounted for the contribution of residual radioactivity from basements/structures that cannot, on their own, support a water supply well but were hydraulically connected to a basement that could support a well.

Once the surface area adjustments are complete, the result becomes the mean of FSS non-parametric results (plus the dose from any identified elevated areas) for backfilled basements or the variable $BcSOF_B$ in Equation 4 below

Equation 4

$$BcSOF_{BASEMENT} = BcSOF_B + BcSOF_{EP} + BcSOF_{PN} + BcSOF_{CF}$$

where:

$BcSOF_{BASEMENT}$	=	BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for backfilled basements
$BcSOF_B$	=	BcSOF for structural survey unit(s) within the basement (mean of FSS systematic results plus the dose from any identified elevated areas)
$BcSOF_{EP}$	=	BcSOF for embedded pipe survey unit(s) within the basement (mean of FSS systematic results plus the dose from any identified elevated areas)
$BcSOF_{PN}$	=	BcSOF for penetration survey unit(s) within the basement (mean of FSS systematic results plus the dose from any identified elevated areas)
$BcSOF_{CF}$	=	BcSOF for clean concrete fill (if applicable) based on maximum MDC during Unrestricted Release Survey (URS)

The variable $BcSOF_B$ was calculated for the basement survey units specified in Table 5-3, which included Unit 1 and Unit 2 Containments, the Auxiliary Building, the SFP/Transfer Canal, the Turbine Building, the Crib House/Forebay and the WWTF. Table 5-4 presents the values for dose for surface, penetrations, embedded pipe and clean fill and the derived value for $BcSOF_{BASEMENT}$ for each. The maximum $BcSOF_{BASEMENT}$ was for the Unit 1 Containment at 0.402 (which equates to a dose of 10.062 mrem/yr. This value will be used for the variable "Max $SOF_{BASEMENT}$" in the compliance equation (Equation 3).

Table 5-4 – Adjusted Basement Surface Areas for Area-Weighted SOF Calculation

Basement	$BcSOF_B$	$BcSOF_{EP}$	$BcSOF_{PN}$	$BcSOF_{CF}$	$BcSOF_{BASEMENT}$	**Dose (mrem/yr)**
Unit 1 Containment	0.223	0.049	0.059	0.071	0.402	10.062
Unit 2 Containment	0.123	0.000	0.008	0.071	0.203	5.064
Auxiliary Building	0.079	0.007	0.069	0.040	0.195	4.865
SFP/Transfer Canal	0.033	0.000	0.000	0.006	0.039	0.979
Turbine Building	0.037	0.003	0.069	0.063	0.173	4.317
Crib House/Forebay	0.006	0.000	0.000	0.063	0.069	1.723
WWTF	0.013	0.000	0.000	0.256	0.269	6.725

5.6 Description of ALARA to Achieve Final Activity Levels

With the exception of some penetrations, embedded and buried piping, all contaminated and non-contaminated systems were disassembled, removed, packaged and shipped off-site as a waste stream commodity. Once commodity removal was complete, structural surfaces were remediated as necessary to meet the open-air demolition criteria. These criteria provided the removable contamination levels and contact exposure rates that allowed structures to be safely demolished without containment.

Prior to demolition, a contamination verification survey (CVS) was performed to identify areas requiring remediation to meet the open-air demolition limits. Identified areas were remediated to provide high confidence that no FSS ISOCS measurement would exceed the $OpDCGL_B$. Once remediation was complete, structural surfaces located above the 588 foot elevation and non-load-bearing interior concrete walls below the 588 foot elevation were demolished, reduced in size, packaged and shipped off-site to a licensed disposal facility.

Concrete inside the liner above the 565 foot elevation was removed from the interiors of both Containment Buildings prior to demolition. This includes all activated and contaminated concrete. The source term in the Containment Basements remaining after demolition consisted of the remaining concrete in the Under Vessel area(s) and low levels of surface contamination on the exposed liner surfaces. There was minimal contamination in the Turbine Building, Crib House/Forebay, and Circulating Water Piping at levels that were well below the open air demolition criteria. The only portion of the Fuel Handling Building basement that remained following building demolition is the lower 13 foot (~4 m) concrete bottom of the SFP and the Transfer Canal, which is located at the 575 foot elevation. The steel liner was removed from both the SFP and the Transfer Canal.

In summary, the vast majority of residual radioactivity remaining in the structures after concrete removal from the Containment basements and open air demolition was located in the 542 foot elevation floor of the Auxiliary Building. Therefore, the ALARA assessment for the remediation of basement structures focused 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 bounds ALARA assessments for the other buildings which would use the same methods (and cost estimate) but remove less contamination. The full analysis is presented in LTP Chapter 4, section 4.4.2. 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.

5.7 USNRC/Independent Verification Team Findings

The Oak Ridge Institute for Science and Education (ORISE) performed confirmatory survey activities in the Turbine Building in August of 2015. A report, "*Final Report – Independent Confirmatory Survey Summary and Results for the Turbine Building Basement and Open Land Areas at the Zion Nuclear Power Station*" (Reference 31) was issued. The report concluded that the radiological conditions of the Turbine Building met the criteria for unrestricted release and that the survey unit was properly classified.

In 2018, ORISE performed confirmatory surveys of the Unit 1 and Unit 2 Containment basements, the Auxiliary Building basement, the SFP and the WWTF. A report, "*Independent Confirmatory Survey Summary and Results for the Containment and Auxiliary Building at the Zion Nuclear Power Station*" (Reference 32) was issued. The confirmatory surveys concluded that the radiological conditions of the basement survey units met the criteria for unrestricted release and that the survey units were properly classified.

6. SUMMARY

Final Status Survey (FSS) is the process used to demonstrate that the ZNPS structures and soils comply with the radiological criteria for unrestricted use specified in 10 CFR 20.1402. The purpose of FSS Sample Plan is to describe the methods to be used in planning, designing, conducting, and evaluating the FSS.

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 TEDE to an AMCG that does not exceed 25 mrem/year, including that from groundwater sources of drinking water, and 2) the residual radioactivity has been reduced to levels that are ALARA.

All survey units addressed in this Final Report have met the DQOs of their respective FSS plans. The ALARA criteria as specified in Chapter 4 of the LTP were achieved. The EMC is not applicable to structural surfaces.

All identified ROC were used for statistical testing to determine the adequacy of each survey unit for FSS. Evaluation of the data shows that none of the mean ROC concentration values exceeded their respective OpDCGL, therefore, in accordance with the LTP Chapter 5, Section 5.10, the survey unit meets the release criterion.

In each survey unit, the sample data passed the Sign Test, the null hypothesis was rejected and the Retrospective Power Curve showed that adequate power was achieved. All survey units were properly classified.

It is the conclusion of this report that all survey units addressed within are acceptable for unrestricted release.

7. REFERENCES

1. "Zion Nuclear Power Station, Units 1 and 2 - Issuance of Amendments 191 and 178 for the Licenses to Approve the License Termination Plan" (LTP)
2. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM)
3. ZS-LT-01, "Quality Assurance Project Plan for Characterization and FSS" (QAPP)
4. Zion*Solutions* Procedure ZS-LT-300-001-001, "Final Status Survey Package Development"
5. Zion*Solutions* Procedure ZS-LT-300-001-003, "Isolation and Control for Final Status Survey"
6. Zion*Solutions* Procedure ZS-LT-300-001-004, "Final Status Survey Data Assessment"
7. NUREG-1757, Vol. 2, "Consolidated Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria"
8. Zion*Solutions* Procedure ZS-LT-300-001-005, "Final Status Survey Data Reporting"
9. "ComEd Zion Station Historical Site Assessment" (HSA)
10. Zion*Solutions* TSD 17-004, "Operational Derived Concentration Guideline Levels for Final Status Survey"
11. Zion*Solutions* TSD 14-028, "Radiological Characterization Report"
12. Zion*Solutions* TSD 14-022, "Use of In-Situ Gamma Spectroscopy for Source Term Survey of End State Structures"
13. Zion*Solutions* Procedure ZS-RP-108-000-000, "Radiological Instrumentation Program"
14. Zion*Solutions* Procedure ZS-RP-108-004-012, "Calibration and Initial Set Up of the 2350-1"
15. Zion*Solutions* TSD 17-003, "Evaluation of Efficiency Calibration Geometries for In-Situ Gamma Spectrometry During Final Status Surveys"
16. Zion*Solutions* Job Aid LT-JA-004, "FSS Sample Collection"
17. Zion*Solutions* Procedure ZS-LT-100-001-004, "Sample Media Preparation for Site Characterization"
18. Zion*Solutions* Procedure ZS-WM-131, "Chain of Custody Protocol"

19. Zion*Solutions* TSD 14-019, "Radionuclides of Concern for Soil and Basement Fill Model Source Terms"

20. ZS-QA-10, "Quality Assurance Project Plan"

21. Post Shutdown Decommissioning Activities Report (PSDAR)

22. Defueled Safety Analysis Report (DSAR)

23. Offsite Dose Calculation Manual (ODCM)

24. Zion*Solutions* Procedure ZS-LT-300-001-002, "Survey Unit Classification"

25. Zion*Solutions* TSD 14-016, "Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Zion End State"

26. ZS-LT-02, "Characterization Survey Plan"

27. Zion*Solutions* TSD 16-008, "Radiological Characterization Report for Auxiliary Building 542 Ft. Embedded Floor Drain Pipe"

28. Zion*Solutions* TSD 11-001, "Potential Radionuclides of Concern During the Decommissioning of the Zion Station"

29. Zion*Solutions* Procedure ZS-LT-300-001-006, "Radiation Surveys of Pipe Interiors Using Sodium/Cesium Iodide Detectors"

30. USNRC Inspection Manual, Inspection Procedure 84750, "Radioactive Waste Treatment and Effluent and Environmental Monitoring"

31. Final Report – Independent Confirmatory Survey Summary and Results for the Turbine Building Basement and Open Land Areas at the Zion Nuclear Power Station"

32. Final Report - Independent Confirmatory Survey Summary and Results for the Containment and Auxiliary Building at the Zion Nuclear Power Station

8. APPENDICES

1. FSS Release Record, Survey Units 01100 and 01110 (Unit 1 Containment above 565 foot and Unit 1 Containment Under Vessel Areas)
2. FSS Release Record, Survey Unit 01111 (Unit 1 Containment Incore Sump Drain)
3. FSS Release Record, Survey Unit 01112 (Unit 1 Containment Penetrations)
4. FSS Release Record, Survey Units 02100 and 02110 (Unit 2 Containment above 565 foot and Unit 2 Containment Under Vessel Areas)
5. FSS Release Record, Survey Unit 02112 (Unit 2 Containment Penetrations)

6. FSS Release Record, Survey Unit 03202 (SFP/Transfer Canal)
7. FSS Release Record, Survey Unit 05100 (Auxiliary Building 542 foot Floor and Walls)
8. FSS Release Record, Survey Unit 05119 (Auxiliary Building Embedded Floor Drains)
9. FSS Release Record, Survey Unit 05120 (Auxiliary Building Penetrations)
10. FSS Release Record, Survey Unit 06100 (Turbine Building including Steam Tunnels, Circulating Water Discharge Pipe [06105A] and Circulating Water Discharge Tunnels [09200])

a.	06105B	Turbine Building Embedded Pipe
b.	06107	Unit 1 Turbine Building Buttress Pit
c.	06108	Unit 2 Turbine Building Buttress Pit
d.	06201	Unit 1 Turbine Building 570 ft. Diesel Fuel Storage
e.	06202	Unit 2 Turbine Building 570 ft. Diesel Fuel Storage
f.	06209	Unit 1 Steam Tunnel Floor Drain
g.	06210	Unit 2 Steam Tunnel Floor Drain
h.	06211	Unit 1 Tendon Tunnel Floor Drain
i.	06212	Unit 2 Tendon Tunnel Floor Drain
j.	06213 & 06214	Unit 1 East and West Main Steam Valve Houses
k.	06215 & 06216	Unit 2 East and West Main Steam Valve Houses

11. FSS Release Record, Survey Units 08100, 08401 and 08102A/B (Crib House/Forebay and Circulating Water Intake Pipes)
12. FSS Release Record, Survey Units 09100 (Waste Water Treatment Facility)

Attachment 2

Preflight Report for Enclosure 1 to

ZS-2019-0090, Revised Final Status Survey Report – Phase 2

This document serves as preflight report for Enclosure 1 to the letter ZS-2019-0090. 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.

FILE NAME	PREFLIGHT STATUS	REASON
FINAL REPORT_Phase 2A_NRC Final_2019-09-30	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Supporting Documents to the Revised Final Status Survey Final Report Phase 2		
Release Record_01111_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_01112_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_02112_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_03202_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_05119_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_05120_NRC Final_2019-09-26	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06105B_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06107_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06108_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06201_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06202_NRC Final_2019-09-29	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible

FILE NAME	PREFLIGHT STATUS	REASON
Release Record_06209_NRC Final_2019-09-29	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06210_NRC Final_2019-09-27	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06211_NRC Final_2019-09-27	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06212_NRC Final_2019-09-26	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06213_06214_NRC Final_2019-09-29	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_06215_06216_NRC Final_2019-09-29	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_08100_NRC Final_2019-09-29	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
Release Record_09100_NRC Final_2019-09-29	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 01100-01110 Folder		
Release Record_01100-01110_NRC Final_2019-09-29 2311	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 01100-01110 - Eberline Analytical Reports	Error / Failed	Document contains scanned pages < 300 ppi, clear and legible
RR 01100-01110 - ISOCS Analytical Reports	Error / Failed	Document contains scanned pages < 300 ppi, clear and legible
RR 02100-02110 Folder		
Release Record_02100-02110_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 02100-02110 - Eberline Analytical Reports	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible

FILE NAME	PREFLIGHT STATUS	REASON
RR 02100-02110 - ISOCS Analytical Reports	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 5100 Folder		
Release Record_05100_NRC Final_2019-09-27	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 05100 Attachment 7 Part 1	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 05100 Attachment 7 Part 2	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 05100 Attachment 7 Part 3	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 05100 Attachment 7 Part 4	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 6100 Folder		
Release Record_06100_NRC Final_2019-09-28	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible
RR 6100 ISOCS Reports	Error / Failed	Document contains logos, color maps, photos, graphics and/or signatures < 300 ppi, clear and legible

ENCLOSURE 1

CD contains

Revised Final Status Survey Final Report – Phase 2

and Supporting Documents