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February 7, 2018

Mr. John Hickman
U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Division of Decommissioning, Uranium Recovery, and Waste Programs
Reactor Decommissioning Branch, Mail Stop: T8F5
11545 Rockville Pike
Rockville, MD 20852

SUBJECT: INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE WASTE WATER TREATMENT FACILITY AND SELECT LAND AREAS AT THE ZION NUCLEAR POWER STATION, ZION, ILLINOIS DOCKET NUMBERS 50-295 AND 50-304; RFTA 18-004 DCN 5271-SR-06-0

Dear Mr. Hickman:

The Oak Ridge Institute for Science and Education (ORISE) is pleased to provide the attached report detailing the independent confirmatory survey activities associated with the Waste Water Treatment Facility and select land areas at the Zion Nuclear Power Station in Zion, Illinois. This report provides the summary and results of ORISE on-site activities performed during the period of September 24–27, 2018.

Please feel free to contact me at 865.574.6273 or Erika Bailey at 865-576-6659 if you have any questions.

Sincerely,

Nick A. Altic, CHP
Health Physicist/Project Manager
ORISE

NAA:lw

Attachment

Electronic distribution:	K. Conway, NRC	B. Lin, NRC	R. Edwards, NRC
	D. Hagemeyer, ORISE	E. Bailey, ORISE	File/5271

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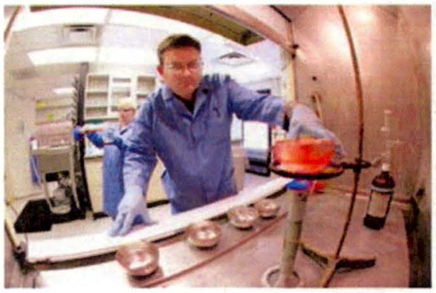

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE WASTE WATER TREATMENT FACILITY AND SELECT LAND AREAS AT THE ZION NUCLEAR POWER STATION, ZION, ILLINOIS

N. A. Altic, CHP
ORISE

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

February 2019

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This document was prepared for the U.S. Nuclear Regulatory Commission (NRC) by the Oak Ridge Institute for Science and Education (ORISE) through an interagency agreement between the NRC and the U.S. Department of Energy (DOE). ORISE is managed by Oak Ridge Associated Universities under DOE contract number DE-SC0014664.

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INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE WASTE WATER TREATMENT FACILITY AND SELECT LAND AREAS AT THE ZION NUCLEAR POWER STATION, ZION, ILLINOIS

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FINAL REPORT

FEBRUARY 2019

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ACRONYMS

AA	alternate action
cm	centimeter
cm^3	cubic centimeters
cpm	counts per minute
DCGL	derived concentration guideline level
$DCGL_{BC}$	Base Case DCGL
$DCGL_{Op}$	Operational DCGL
DER	duplicate error ratio
DOE	U.S. Department of Energy
DQO	data quality objective
DS	decision statement
EPA	U.S. Environmental Protection Agency
Exelon	Exelon Generation Company
FSS	final status survey
HTD	hard-to-detect
LTP	license termination plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
mrem/yr	millirem per year
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
pCi/m^2	picocuries per square meter
PSQ	principal study question
Q-Q plot	quantile-quantile plot
ROC	radionuclide of concern
SA	surface area
SFP	spent fuel pool
SOF	sum-of-fractions
SU	survey unit
TAP	total absorption peak
TEDE	total effective dose equivalent
UCL	upper confidence level
VSP	Visual Sample Plan
WWTF	Waste Water Treatment Facility
ZNPS	Zion Nuclear Power Station
ZS	Zion*Solutions*, LLC

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INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE WASTE WATER TREATMENT FACILITY AND SELECT LAND AREAS AT THE ZION NUCLEAR POWER STATION, ZION, ILLINOIS

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory survey activities of the remaining structures and surfaces of the Waste Water Treatment Facility (WWTF) and select land areas at the Zion Nuclear Power Station. Confirmatory survey activities were conducted during the period of September 24–27, 2018 and consisted of surface scans, *in situ* gamma spectrometry measurements, and volumetric concrete sampling.

All individual confirmatory samples collected from the WWTF were below their respective operational derived concentration guideline level ($DCGL_{Op}$) and, therefore, are also below the base case derived concentration guideline level ($DCGL_{BC}$). The confirmatory concrete mean sample results and associated uncertainty for gamma-emitting radionuclides of concern are low relative to that of the final status survey (FSS), and thus are not in agreement. This difference is attributed to the sensitivity between the two analysis methods (confirmatory data are based on volumetric concrete samples and FSS data are based on *in situ* gamma spectrometry measurements). Additional investigation is unnecessary provided the overall magnitude of the confirmatory survey results (less than 2% of the $DCGL_{BC}$).

The confirmatory soil sample results from both land areas were all below their respective $DCGL_{Op}$ and, therefore, are also below the $DCGL_{BC}$. Initial review of the confirmatory sample results identified a discrepancy between the ORISE and three FSS Co-60 data points. However, additional investigation determined that comparison of results for these samples are inconclusive. Therefore, there is not a discernable bias between the ORISE and FSS measurements for these samples.

Based on the parameters established by project-specific data quality objectives, confirmatory survey data did not identify issues that would preclude the FSS data for demonstrating compliance with the release criteria for these areas.

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE WASTE WATER TREATMENT FACILITY AND SELECT LAND AREAS AT THE ZION NUCLEAR POWER STATION, ZION, ILLINOIS

1. INTRODUCTION

The Zion Nuclear Power Station (ZNPS) consists of two reactors, Units 1 and 2, which operated commercially from 1973 to 1997 and 1974 to 1996, respectively. Cessation of nuclear operations was certified in 1998 after both reactor units were defueled and the fuel assemblies had been placed in the spent fuel pools. Both units were then placed in safe storage pending the commencement of site decommissioning and dismantlement. In 2010, the U.S. Nuclear Regulatory Commission (NRC) operating license was transferred from Exelon Generation Company (Exelon) to Zion*Solutions,* LLC (ZS) to allow the physical decommissioning process that began in 2010 and is expected to be completed within ten years. The end state and primary decommissioning objective at ZNPS is the transfer of all spent nuclear fuel to the independent spent fuel storage installation and to reduce residual radioactivity levels below the criteria specified in 10 CFR 20.1402, permitting release of the site for unrestricted use. Upon successful completion of the decommissioning activities, control and responsibility for the site will be transferred back to Exelon and the independent spent fuel storage installation maintained under Exelon's Part 50 license (EC 2015).

As part of decommissioning, all above-grade structures, with minor exceptions, will be demolished. Structures below the 588-ft elevation (referenced from mean sea level), consisting of primarily exterior sub-grade walls and floors, will remain. These basement structures will be backfilled as part of the final site restoration. In order to demonstrate compliance with the release criteria in 10 CFR 20.1402, ZS will implement final status survey (FSS) activities of remaining basement structures, along with associated embedded piping and penetrations, buried piping, and surface and subsurface soil. FSS methodologies are outlined in Chapter 5 of ZS's license termination plan (LTP) (ZS 2017). The primary FSS method for basement structure survey units (SUs) is *in situ* gamma measurements using a portable, high-resolution gamma spectrometer. FSS methods are based on methods outlined in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NRC 2000).

The NRC requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory survey activities of the remaining structures and surfaces of the Waste Water

Treatment Facility (WWTF) and select land areas at the ZNPS. This report summarizes the confirmatory survey activities and results for these areas.

2. SITE DESCRIPTION

The ZNPS is located in Lake County, Illinois on the easternmost portion of the city of Zion. It is approximately 64 kilometers (40 miles) north of Chicago, Illinois and 68 kilometers south of Milwaukee, Wisconsin. The owner-controlled site is approximately 134 hectares and is situated between the northern and southern parts of Illinois Beach State Park on the western shore of Lake Michigan (EC 2015 and ZS 2017). Figure 2.1 provides an overview of ZNPS. The site and its surrounding environs is relatively flat with the elevation of the developed portion of the site at approximately 591 feet above mean sea level; for reference, the elevation of Lake Michigan—which bounds the site on the east—is approximately 577.4 feet at low water level (ZS 2017).

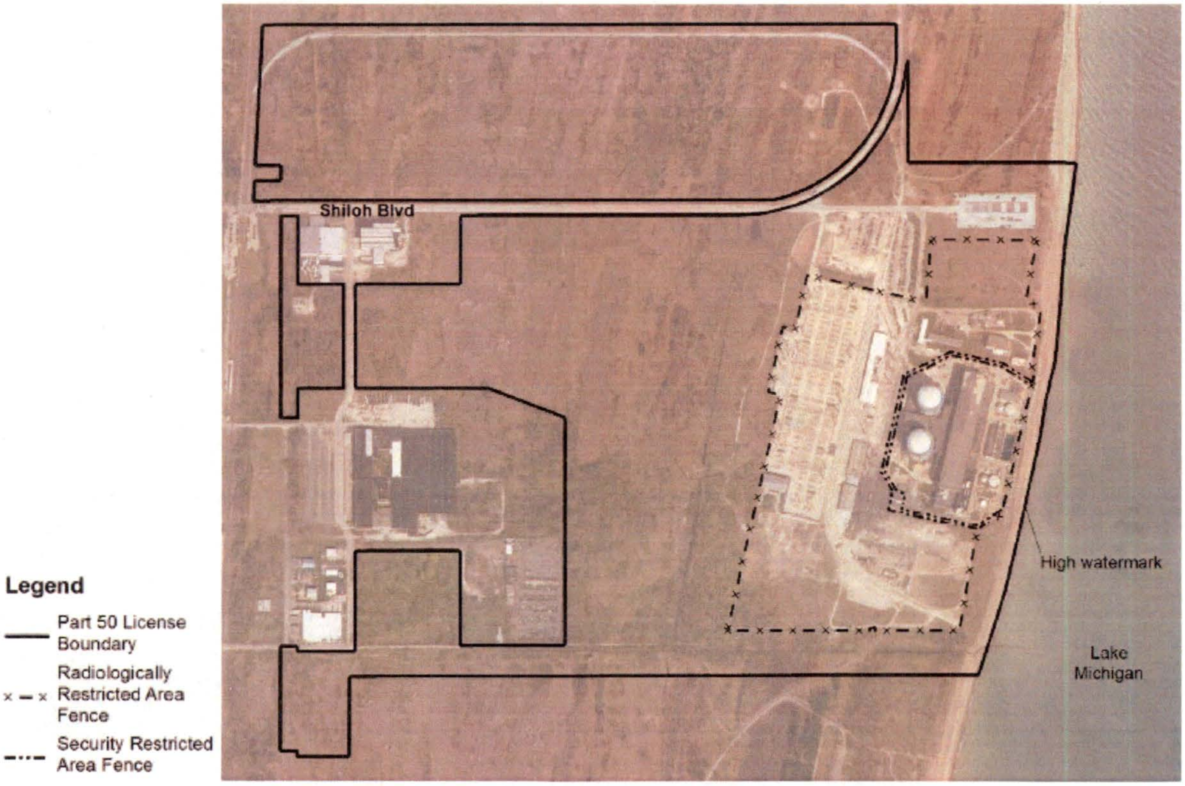

Figure 2.1. ZNPS Overview (adapted from ZS 2017)

The Waste Water Treatment Facility (WWTF) was designed to treat non-radioactive and low level radioactive liquid from ZNPS sources, including building roof run-off and the Turbine Building Fire Sump, which received liquid waste from the Turbine Building Equipment and Floor Drains, and the Fuel Pool Cooling Tower Blowdown. The WWTF was designed to remove suspended solids and oil to ensure compliance with the facility National Pollutant Discharge Elimination System permit. Since the wastewater discharge rates were variable, an equalization tank was installed. The WWTF also includes other equipment, such as mixing tanks, mixers, oil skimmers, flocculators, oil coalescers, clarifiers, sludge drying beds and filters. Discharge from the WWTF was by gravity to the Forebay. During ZNPS operations, liquid waste with detectable low-level radioactive contamination was processed by the WWTF. Figure 2.2 displays the WWTF location in comparison to the remainder of the site.

The land area survey units, selected by the NRC for confirmatory survey, have previously had various buried piping removed leaving open excavations. The land area survey units are located in the northwest quadrant, within the industrial area security fence line, and north of the southern rail spur. Overburden soil removed from the excavation was placed in various soil laydown areas for survey and sampling. The excavations will subsequently be backfilled to grade.

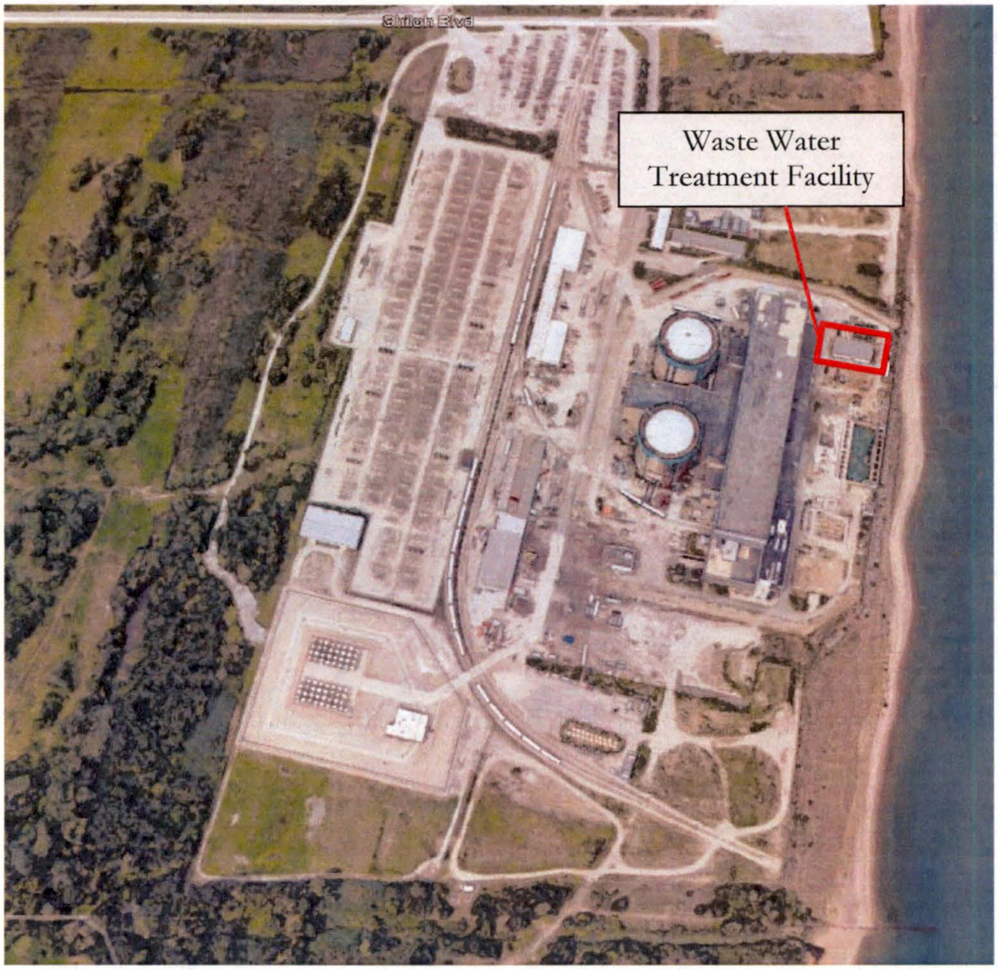

Figure 2.2. ZNPS Waste Water Treatment Facility (Google Earth)

3. DATA QUALITY OBJECTIVES

The data quality objectives (DQOs) described herein are consistent with the *Guidance on Systematic Planning Using the Data Quality Objectives Process* (EPA 2006) and provide a formalized method for planning radiation surveys, improving survey efficiency and effectiveness, and ensuring that the type, quality, and quantity of data collected are adequate for the intended decision applications.

The seven steps of the DQO process are as follows:

1. State the problem
2. Identify the decision/objective
3. Identify inputs to the decision/objective

4. Define the study boundaries
5. Develop a decision rule
6. Specify limits on decision errors
7. Optimize the design for obtaining data

Confirmatory survey DQOs were originally presented in ORISE 2018 and are represented here for completeness.

3.1 State the Problem

The first step in the DQO process defined the problem that necessitated the study. The NRC requested that ORISE perform confirmatory surveys at the ZNPS. The objective of the confirmatory surveys was to provide the NRC with independent confirmatory data for the NRC's consideration in the evaluation of the FSS results. The problem statement was formulated as follows:

> Confirmatory surveys are necessary to generate independent radiological data for the NRC's consideration in the evaluation of the FSS design, implementation, and results for demonstrating compliance with the release criteria.

3.2 Identify the Decision

The second step in the DQO process identified the principal study questions (PSQs) and alternate actions (AAs), developed a decision statement (DS), and organized multiple decisions, as appropriate. This was done by specifying AAs that could result from a "yes" response to the PSQ and combining the PSQ and AAs into a DS. The PSQ, AAs, and combined DSs were organized based on the SU type (i.e., the associated FSS methodology) and are presented in Table 3.1.

Table 3.1. ZNPS Confirmatory Survey Decision Process

Principal Study Question	Alternate Actions
Do confirmatory survey results agree with the final radiological survey data for the WWTF and land areas?	**Yes:** Compile confirmatory data and report results to the NRC for their decision making. Provide independent interpretation that confirmatory field surveys did not identify anomalous areas of residual radioactivity, quantitative field and laboratory data satisfied the NRC-approved decommissioning criteria, and/or that statistical sample population examination/assessment conditions were met. **No:** Compile confirmatory data and report results to the NRC for their decision making. Provide independent interpretation of confirmatory survey results identifying any anomalous field or laboratory data and/or when statistical sample population examination/assessment conditions were not satisfied for the NRC's determination of the adequacy of the FSS data.
Decision Statement	
Confirmatory survey results did/did not identify anomalous results or other conditions that preclude the FSS data from demonstrating compliance with the release criteria.	

3.3 IDENTIFY INPUTS TO THE DECISION

The third step in the DQO process identified both the information needed and the sources of this information, determined the basis for action levels, and identified sampling and analytical methods that would meet data requirements. For this effort, information inputs included the following:

- Concrete characterization data for WWTF
- Remedial action support survey data for land areas
- FSS data for the WWTF and land areas
- Derived concentration guideline levels (DCGLs), discussed in subsection 3.3.1
- ORISE confirmatory survey surface scans
- ORISE volumetric sample analysis results for concrete and soil samples

3.3.1 Radionuclides of Concern and Release Guidelines

The primary radionuclides identified for the ZNPS are beta-gamma emitters—fission and activation products—resulting from reactor operations. At ZNPS, there are four distinct source terms:

basement structures, soils, buried piping, and groundwater. Furthermore, basement structures are composed of four structural source terms: surfaces, embedded piping, penetrations, and fill. ZS has developed site-specific DCGLs that correspond to a residual radioactive contamination level, above background, which could result in a total effective dose equivalent (TEDE) of 25 millirem per year (mrem/yr) to an average member of the critical group. These DCGLs—defined in ZS's LTP as Base Case DCGLs ($DCGL_{BC}$s)—are radionuclide-specific and independently correspond to a TEDE of 25 mrem/yr for each source term. In order to ensure that total dose from all source terms is less than the NRC's release criteria, the $DCGL_{BC}$s are further reduced to Operational DCGLs ($DCGL_{Op}$s). The $DCGL_{Op}$s are scaled to an expected dose from prior investigations and are used for remediation and FSS design purposes. The initial suite of radionuclides present at ZNPS has been reduced based on an insignificant dose contribution from a number of radionuclides. The $DCGL_{BC}$s and $DCGL_{Op}$s, accounting for insignificant dose contributors, for the basement structure source terms—excluding fill material—and subsurface soil are presented in Table 3.2, and Table 3.3, respectively.

<table>
<tr><th colspan="10">Table 3.2. ZNPS Basement Surfaces DCGLs[a]</th></tr>
<tr><th rowspan="2">ROC</th><th rowspan="2">Auxiliary Building</th><th colspan="2">Containment</th><th rowspan="2">SFP/Transfer Canal</th><th colspan="2">Turbine Building</th><th rowspan="2">Crib House/ /Forebay</th><th rowspan="2">WWTF</th></tr>
<tr><th>Above 565 ft</th><th>Under-vessel</th><th>Floors & Walls[b]</th><th>Circ Water Discharge Tunnel</th></tr>
<tr><td colspan="9">Base Case DCGLs (pCi/m²)</td></tr>
<tr><td>H-3</td><td>5.30E+08</td><td colspan="2">2.38E+08</td><td>2.38E+08</td><td colspan="2">1.29E+08</td><td>1.93E+08</td><td>1.71E+07</td></tr>
<tr><td>Co-60</td><td>3.04E+08</td><td colspan="2">1.57E+08</td><td>1.57E+08</td><td colspan="2">7.03E+07</td><td>5.52E+07</td><td>2.83E+07</td></tr>
<tr><td>Ni-63</td><td>1.15E+10</td><td colspan="2">4.02E+09</td><td>4.02E+09</td><td colspan="2">2.18E+09</td><td>3.25E+09</td><td>2.89E+08</td></tr>
<tr><td>Cs-134</td><td>2.11E+08</td><td colspan="2">3.01E+07</td><td>3.01E+07</td><td colspan="2">1.59E+07</td><td>2.13E+07</td><td>2.31E+06</td></tr>
<tr><td>Cs-137</td><td>1.11E+08</td><td colspan="2">3.94E+07</td><td>3.94E+07</td><td colspan="2">2.11E+07</td><td>2.96E+07</td><td>2.93E+06</td></tr>
<tr><td>Sr-90</td><td>9.98E+06</td><td colspan="2">1.43E+06</td><td>1.43E+06</td><td colspan="2">7.74E+05</td><td>1.16E+06</td><td>1.03E+05</td></tr>
<tr><td>Eu-152</td><td>6.47E+08</td><td colspan="2">3.66E+08</td><td>3.66E+08</td><td colspan="2">1.62E+08</td><td>1.23E+08</td><td>7.55E+07</td></tr>
<tr><td>Eu-154</td><td>5.83E+08</td><td colspan="2">3.19E+08</td><td>3.19E+08</td><td colspan="2">1.43E+08</td><td>1.12E+08</td><td>5.74E+07</td></tr>
<tr><td colspan="9">Operational DCGLs (pCi/m²)</td></tr>
<tr><td>H-3</td><td>1.71E+08</td><td>3.25E+07</td><td>2.37E+08</td><td>4.98E+07</td><td>1.10E+07</td><td>5.39E+07</td><td>7.43E+07</td><td>3.28E+06</td></tr>
<tr><td>Co-60</td><td>9.81E+07</td><td>2.15E+07</td><td>1.56E+08</td><td>3.28E+07</td><td>5.98E+06</td><td>2.94E+07</td><td>2.13E+07</td><td>5.43E+06</td></tr>
<tr><td>Ni-63</td><td>3.71E+09</td><td>5.50E+08</td><td>4.00E+09</td><td>8.41E+08</td><td>1.85E+08</td><td>9.11E+08</td><td>1.25E+09</td><td>5.55E+07</td></tr>
<tr><td>Cs-134</td><td>6.81E+07</td><td>4.12E+06</td><td>2.99E+07</td><td>6.30E+06</td><td>1.35E+06</td><td>6.65E+06</td><td>8.20E+06</td><td>4.44E+05</td></tr>
<tr><td>Cs-137</td><td>3.58E+07</td><td>5.39E+06</td><td>3.92E+07</td><td>8.24E+06</td><td>1.79E+06</td><td>8.82E+06</td><td>1.14E+07</td><td>5.63E+05</td></tr>
<tr><td>Sr-90</td><td>3.22E+06</td><td>1.96E+05</td><td>1.42E+06</td><td>2.99E+05</td><td>6.58E+04</td><td>3.24E+05</td><td>4.47E+05</td><td>1.98E+04</td></tr>
<tr><td>Eu-152</td><td>2.09E+08</td><td>5.00E+07</td><td>3.64E+08</td><td>7.66E+07</td><td>1.38E+07</td><td>6.77E+07</td><td>4.74E+07</td><td>1.45E+07</td></tr>
<tr><td>Eu-154</td><td>1.88E+08</td><td>4.36E+07</td><td>3.17E+08</td><td>6.67E+07</td><td>1.22E+07</td><td>5.98E+07</td><td>4.31E+07</td><td>1.10E+07</td></tr>
</table>

[a]Recreated from ZS 2017
[b]The Operational DCGLs for floors and walls will be applied to surfaces in the Circulating Water Intake Pipe and Circulating Water Discharge Pipe
SFP = spent fuel pool
WWTF = Waste Water Treatment Facility

Table 3.3. ZNPS Sub-surface Soil DCGLs[a]

ROC	DCGL
Base DCGLs (pCi/g)	
Co-60	3.44
Cs-134	4.44
Cs-137	7.75
Ni-63	763.02
Sr-90	1.66
Operational DCGLs (pCi/g)	
Co-60	0.881
Cs-134	1.137
Cs-137	1.984
Ni-63	195.333
Sr-90	0.425

[a]Recreated from ZS 2017
pCi/g = picocuries per gram ROC = radionuclide of concern

Because each of the individual $DCGL_{BC}$s represent a separate radiological dose, the sum-of-fractions (SOF) approach must be used to evaluate the total dose from the SU and demonstrate compliance with the dose limit. SOF calculations are performed as follows:

$$SOF = \sum_{j=1}^{n} \frac{C_{mean,j}}{DCGL_{BC,j}} + \frac{(C_{Elv,j} - C_{mean,j})}{\left(DCGL_{BC,j} \times \left(\frac{SA_{SU}}{SA_{Elv.}}\right)\right)} \qquad \text{(Equation 3-1)}$$

Where $C_{mean,j}$ is the mean concentration of radionuclide of concern (ROC) "j"; $C_{Elv,j}$ is an elevated area of ROC "j"; $DCGL_{BC,j}$ is the $DCGL_{BC}$ for ROC "j"; SA_{SU} is the adjusted surface area (SA) of the FSS unit; and $SA_{Elv.}$ is the surface area of the elevated measurement. Per Section 5.5.4 of the LTP, areas of elevated activity—for building surfaces—are defined as any area identified by measurement/sample (systematic or judgmental) that exceeds the $DCGL_{Op}$ but is less than the $DCGL_{BC}$. Any area that exceeds the $DCGL_{BC}$ will be remediated. Note that gross concentrations are considered here for conservatism.

3.4 DEFINE THE STUDY BOUNDARIES

The fourth step in the DQO process defined target populations and spatial boundaries, determined the timeframe for collecting data and making decisions, addressed practical constraints, and determined the smallest subpopulations, area, volume, and time for which separate decisions must be made. Confirmatory surveys were performed in the WWTF and two land area survey units. ZS assigned SU ID number 09100A for the WWTF. The first land area investigated contains survey units, including 10209E, and is also known as Land Area 7. The second land area investigated transects multiple survey units, including 10204B & C, 10206D & E, and 10207D & E, and is also known as Land Area 6. All survey units were considered a Class 1, and thus have a potential for contamination above the $DCGL_{BC}$. ORISE staff was not able to perform direct radiation surveys of the excavation surfaces as personnel access was prohibited due to safety considerations. On-site confirmatory activities were conducted during the period of September 24–27, 2018.

3.5 DEVELOP A DECISION RULE

The fifth step in the DQO process specified appropriate population parameters (e.g., mean, median), confirmed action levels were above detection limits, and developed an if…then… decision rule statement. Decision rules for this survey are based on independent scan surveys and volumetric

concrete and soil sample results to assess whether there is a statistical bias relative to the FSS data. Typically, decision rules are based on a statistical comparison of the ORISE survey data and the FSS data using an appropriate statistical test. However, the difference in sample sizes was significant. For example, 70 FSS *in situ* gamma spectroscopy measurements were collected in the WWTF SU, whereas ORISE collected 16 samples. The approximately 4:1 sample size ratio significantly reduces the statistical power to detect a difference between the two sample groups. Therefore, alternative assessment methods were employed.

The mean and associated uncertainty for the confirmatory measurements were compared directly to that of the FSS data. Agreement between the two sample sets was evaluated by the overlapping of confidence intervals. The parameter of interest for each survey area was the mean ROC concentration and the associated 95% confidence interval of the mean. The SOF of the confirmatory survey data will be compared directly to the $DCGL_{BC}$. The aforementioned information is combined to formulate the decision rule for the WWTF, which is stated as follows:

> *If the mean ROC concentrations of the confirmatory and FSS sample populations overlap at the 95% confidence level and results are below the applicable limit, conclude that confirmatory survey data agrees with the FSS data—otherwise, perform further evaluation(s) and provide technical comments to the NRC.*

Based on review of the previous Radiological Assessment soil sampling data for SU 10209E, the ROC concentrations were expected to be at or near the analytical minimum detectable concentration (MDC). Therefore, a formal statistical comparison was unnecessary. Rather, the confirmatory survey focused on identifying locations that exceed the applicable soil DCGLs. Additionally, FSS samples were selected for analysis by the ORISE laboratory to evaluate potential bias between measurement systems. These confirmatory duplicate samples were evaluated using the duplicate error ratio (DER) (DOE 2013). The DER is calculated by:

$$DER = \frac{|O-F|}{\sqrt{U_O{}^2+U_F{}^2}} \qquad \text{(Equation 3-2)}$$

Where:

O = ORISE sample result
F = FSS sample result
U_O = ORISE sample one sigma uncertainty
U_F = FSS sample one sigma uncertainty

The decision rule for the land area survey units is stated as follows:

> *If ROC concentrations in confirmatory samples are less than the applicable DCGL and confirmatory duplicate sample analytical results are in agreement with the ORISE results, then recommend acceptance otherwise, perform further evaluation(s) and provide technical comments to the NRC.*

3.6 SPECIFY LIMITS ON DECISION ERRORS

The sixth step in the DQO process examined the consequences of making an incorrect decision and established bounds of decision errors. Decision errors were controlled both during the in-process investigations and during data quality assessment and were based on three orders of control.

The first order of control was related to sample size, which impacts the degree to which the estimated sample mean is bound. Visual Sample Plan (VSP), version 7.9, was used to determine the confirmatory survey sample size using the FSS/characterization data as planning inputs. The constraint on the estimated mean was not larger than the difference between the DCGL and the reported FSS data mean. Numerically, the constraint on the estimated mean was represented by (half-width of the estimated mean ≤ 1.00 – FSS Mean SOF). The confirmatory survey mean was estimated at the 95% confidence level. (See section 4.3 of this report for additional details.)

The volumetric concrete samples collected by ORISE do not represent 100% areal coverage of the WWTF. Thus, the presence or absence of all of the potential locations not represented by the arithmetic mean of the confirmatory data set could not be identified and accounted for using Equation 3-1. Therefore, the 95-percent upper confidence level (UCL95) of the mean was used for the SOF calculations. The UCL accounts for the uncertainty in the estimate of the mean due to sampling error, and the probability of a Type I decision error is minimized (i.e., incorrectly concluding that the ROC concentration is less than $DCGL_{BC}$).

Agreement between sample results constituted the second order of control. A DER ≤3 indicates that, at a 99% confidence interval, the sample results do not differ significantly when compared to their respective one standard deviation (sigma) uncertainty (DOE 2013).

The third order of control was to optimize the analytical minimum detectable concentrations (MDCs) with respect to ORISE sample count times for laboratory measurements. Measurement

MDCs for volumetric sample analyses were 10% or less of their respective $DCGL_{BC}$ presented in Section 3.3.1, with the exception of Sr-90, which averaged less than 30% $DCGL_{BC}$.

3.7 OPTIMIZE THE DESIGN FOR OBTAINING DATA

The seventh step in the DQO process was used to review DQO outputs, develop data collection design alternatives, formulate mathematical expressions for each design, select the sample size to satisfy DQOs, decide on the most resource-effective design of agreed alternatives, and document requisite details. ORISE originally planned to perform in situ gamma spectrometry measurements in the WWTF. However, equipment malfunction while on site prevented the collection of in situ measurements. Specifically, the detector was operating outside of established quality control parameters. After receiving approval from the NRC, ORISE modified the measurement approach to collecting volumetric concrete samples in lieu of performing in situ measurements. Specific survey procedures are presented in Section 4.

4. PROCEDURES

The ORISE survey team conducted independent confirmatory survey activities, including surface scans, and volumetric sampling activities, within the accessible survey area specifically requested by the NRC. Survey activities were conducted in accordance with the *ORAU Radiological and Environmental Survey Procedures Manual* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2016a and ORAU 2018).

4.1 REFERENCE SYSTEM

ORISE used specific X-Y coordinates from the southwest corner of the respective SU floors and lower left corner of walls to reference measurement and sampling locations that were documented on detailed survey maps.

4.2 SURFACE SCANS

Thallium doped sodium iodide (NaI[Tl]) detectors were used to evaluate direct gamma radiation levels on basement and land area surfaces. Accessible WWTF surfaces received high-density scan coverage. High-density surface scans of the excavated overburden material in Land Area 7 were performed.

All detectors were coupled to Ludlum Model 2221 ratemeter-scalers with audible indicators and were also coupled to data-loggers to electronically record all scanning data in the WWTF. During the surface scans of Land Area 7, the ratemeter-scaler was coupled to a GPS receiver allowing for real-time count rate and spatial data capture. NaI detectors used for scans of Land Area 7 were collimated with lead to reduce the influence from gamma sources from nearby locations. Locations of elevated response that are audibly distinguishable from localized background levels, suggesting the presence of residual contamination, were marked for further investigation with volumetric sampling. Due to temporal boundaries and at the direction of the NRC inspector, the overburden soil in Land Area 6 was not scanned, but rather evaluated with soil sampling.

4.3 SAMPLE NUMBER DETERMINATION

Measurement locations were determined both randomly and judgmentally, as necessary. VSP was used to assess the sample size required for decision making and to randomly place locations throughout the survey area. The NRC has provided the preliminary FSS data for the WWTF. The resulting SOF is 0.01, based on the UCL of the FSS data and the $DCGL_{BC}$. The associated variability is 0.01. As Figure 4.1 indicates, seven measurements are required to estimate the mean within ±0.01 of the true mean using simple random sampling (as opposed to 6 required for ranked set sampling). To improve confidence, 16 random concrete samples were collected from the WWTF.

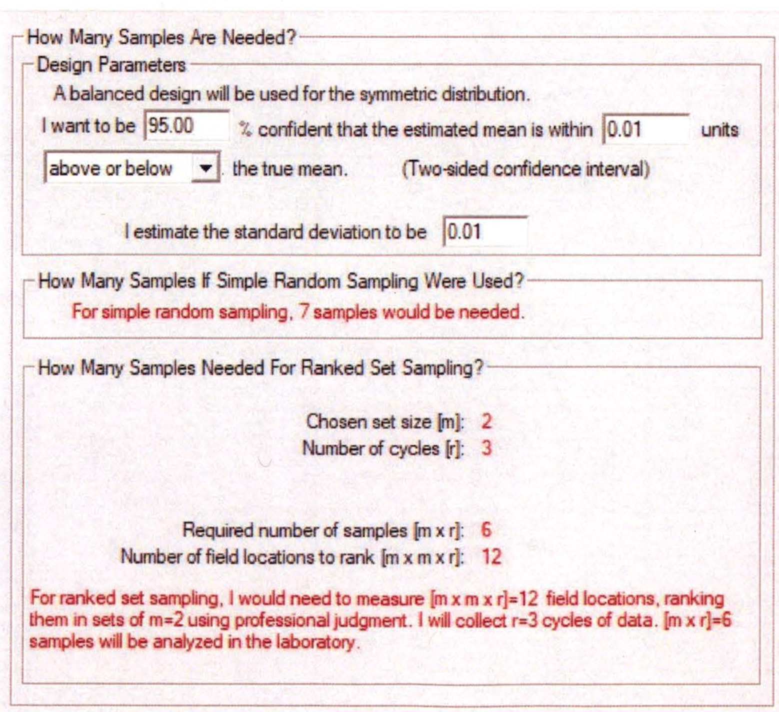

Figure 4.1. Sample Size Determination Using VSP

Eight soil samples were collected from the bottom surface of the trench in each land area. The number of samples was selected such that the estimated mean SOF would fall within 0.25 units of the true mean.

4.4 VOLUMETRIC SAMPLING

Concrete samples were collected from randomly and judgmentally selected locations using an electric concrete coring machine with a 15.24 centimeter (cm) diameter coring bit. Samples were collected to a depth of up to 15 cm (6 inches), or until refusal, and the sample depth was noted at each location. Concrete cores were sectioned into 5-cm (2-inch) increments, for a maximum of three increments. The increments were analyzed individually starting from the top portion (i.e., the 0 to 5-cm portion). The coring bit was decontaminated using water and a laboratory grade detergent to prevent cross-contamination. Samples were containerized in zip-top plastic bags and marked with a unique identifier.

Surface (0 to 15-cm) soil samples were collected using hand-held trowels/shovels from systematic locations from the bottom of the Land Area 6 and 7 trench excavations and judgmentally selected locations from the overburden material in Land Area 7. The eight soil samples from each land area were equally spaced along the accessible bottom surface of the trench in each land area. Sampling activities in Land Area 6 were limited to the Eastern most portion of the trench, parallel to the train tracks. Sampling equipment was decontaminated between sampling locations to prevent cross-contamination. Samples were containerized using plastic bags and marked with a unique identifier.

5. SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data collected on site were transferred to the ORISE facility for analysis and interpretation. Sample custody was transferred to the Radiological and Environmental Analytical Laboratory in Oak Ridge, Tennessee. Sample analyses were performed in accordance with the *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2017). The concrete samples were analyzed by gamma spectrometry for gamma-emitting fission and activation products. Additionally, the concrete samples were processed via wet chemistry and then analyzed for Sr-90 and Ni-63 by low-background proportional counting and liquid scintillation respectively, after

separation. Volumetric sample results in units of picocuries per gram (pCi/g) were converted to units of picocuries per meter squared (pCi/m^2) based on the concrete sample depth. ProUCL, Version 5.1, was used to calculate the UCL95 for both the confirmatory and FSS data set. The mean ROC concentration and associated 95% confidence level were plotted for direct comparison.

6. FINDINGS AND RESULTS

The results of the confirmatory survey activities are discussed in the subsections below.

6.1 WASTE WATER TREATMENT FACILITY

6.1.1 Surface Scans

Overall, NaI detector scan responses ranged from approximately 2,900 to 8,900 counts per minute (cpm) for the floor of the WWTF and 3,200 to 7,200 cpm for the lower wall. Figure 6.1 presents the quantile-quantile plot (Q-Q plot) for these scan data. Scan data are approximately normal, as indicated by the fairly flat lines as depicted in Figure 6.1, indicating a background distribution. Nine data points for the floor scan appear to be from a separate population distribution; however, the surveyor did not identify these points as contamination.

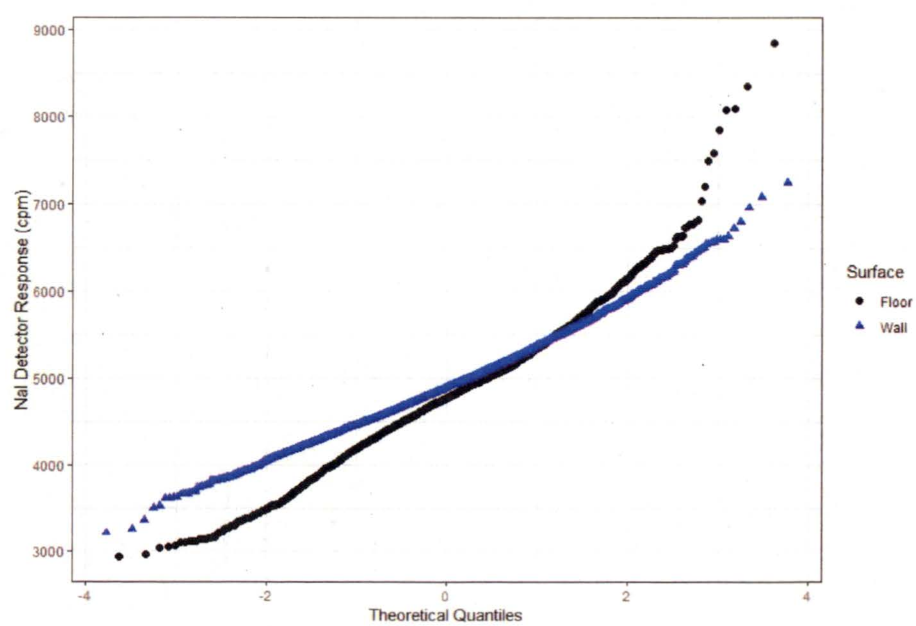

Figure 6.1. Q-Q Plot for the WWTF Wall and Floor

6.1.2 ROC Concentrations in Concrete Samples

Individual concrete sample results for gamma emitting radionuclides of concern are provided in Appendix B, Table B.1. Eight of the 16 concrete samples were randomly selected and analyzed for hard-to-detect (HTD) radionuclides (Sr-90 and Ni-63); individual results for these samples are also presented Table B.1. Concrete sample analytical results were converted to units of pCi/m^2 based on a sample depth of 5.08 cm and a concrete density of 2.35 grams per cubic centimeter (g/cm^3). Figure A.1 provides a depiction of the concrete sampling locations. Table 6.1 provides a summary of the confirmatory concrete samples collected from the WWTF. Only one sample result identified a gamma-emitting ROC above the analytical MDC—sample 5271M0056. Analysis of the subsequent increment (5.08 to 10.16 cm) at this location yielded a result below the MDC. None of the HTD radionuclides were detected above their respective MDC.

None of the individual ROC concentrations are above their respective $DCGL_{Op}$ and, therefore, less than the respective $DCGL_{BC}$. The average SOF in the WWTF SU for gamma-emitting radionuclides is less than 0.01 based on the $DCGL_{BC}$ (or 0.02 when based on the $DCGL_{Op}$), which is less than the SOF—based on the $DCGL_{BC}$—for gamma-emitting radionuclides presented in the FSS data (0.01).The mean HTD radionuclide (consisting of Ni-63 and Sr-90) SOF, when compared to respective $DCGL_{BC}$ values, was determined to be 0.02 compared to 0.00 (i.e., less than 0.01) reported in the FSS data. The primary difference between the HTD SOF values is due to the difference in the respective analysis methods (volumetric sampling for the ORISE confirmatory data versus the surrogate approach for the FSS measurements). MDCs for Sr-90 analyses are above the $DCGL_{Op}$, however are still sufficient for confirmatory survey decision-making.

Table 6.1. Summary of WWTF Random Volumetric Concrete Samples								
ROC	Statistic (pCi/m²)						Fraction[b]	
	Mean	Median	SD	Min	Max	UCL95[a]	Op.	BC
Co-60	2.91E+02	1.79E+02	2.38E+03	-3.34E+03	4.30E+03	2.89E+03	<0.01	<0.01
Cs-134	1.56E+03	2.45E+03	2.81E+03	-3.34E+03	6.57E+03	4.62E+03	0.01	<0.01
Cs-137	2.95E+03	2.03E+03	4.49E+03	-5.01E+03	1.55E+04	7.83E+03	0.01	<0.01
Sr-90	-8.21E+03	-5.97E+03	6.47E+03	-1.67E+04	0.00E+00	2.17E+03	0.11	0.02
Ni-63	2.16E+04	1.97E+04	4.70E+04	-3.58E+04	1.15E+05	9.43E+04	<0.01	<0.01
						SOF[c]	0.14	0.03

[a]UCL is based on the Chebyshev Inequality
[b]Op. represents the UCL95 divided by the Operational DCGL; BC represents the UCL95 divided by the Base Case DCGL
[c]Discrepancy in summation due to rounding
SD = standard deviation
SOF = Sum of Fractions

Gamma-emitting ROC mean concentrations and their associated uncertainties for both ORISE and the FSS data are plotted in Figure 6.2. The error bars in Figure 6.2 represent the uncertainty in the mean concentration, where the upper end is simply the UCL95. As indicated in Figure 6.2, all ROC confidence intervals for the ORISE data are biased low relative to the FSS data. The primary difference between the confidence intervals is due to the difference in MDCs in the respective analysis methods (volumetric sampling for the ORISE confirmatory data versus *in situ* gamma spectrometry for the FSS measurements).

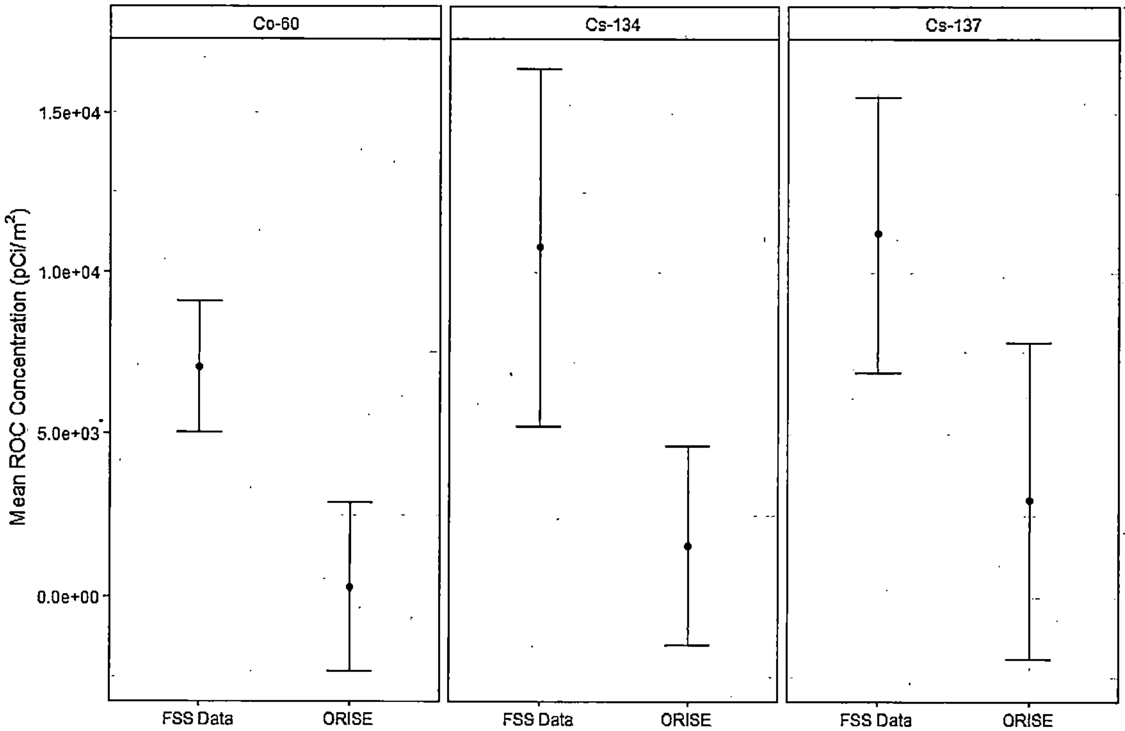

Figure 6.2. Comparison of FSS Data and ORISE Confirmatory Mean Concentrations and Uncertainties for Gamma-emitting Radionuclides in the WWTF

6.2 LAND AREA SURVEYS

6.2.1 Surface Scans

Overall, NaI detector scan responses ranged from approximately 1,000 to 4,000 cpm for the overburden soil pile in Land Area 7. Figure 6.3 presents the Q-plot for these scan data and the gamma walk-over map for this area is presented in Figure A.2. Scan data for this area are approximately normal, as indicated by the fairly flat line depicted in Figure 6.3, indicating a background distribution. Two locations, representing areas at the upper end of the background, were selected for judgmental soil sampling.

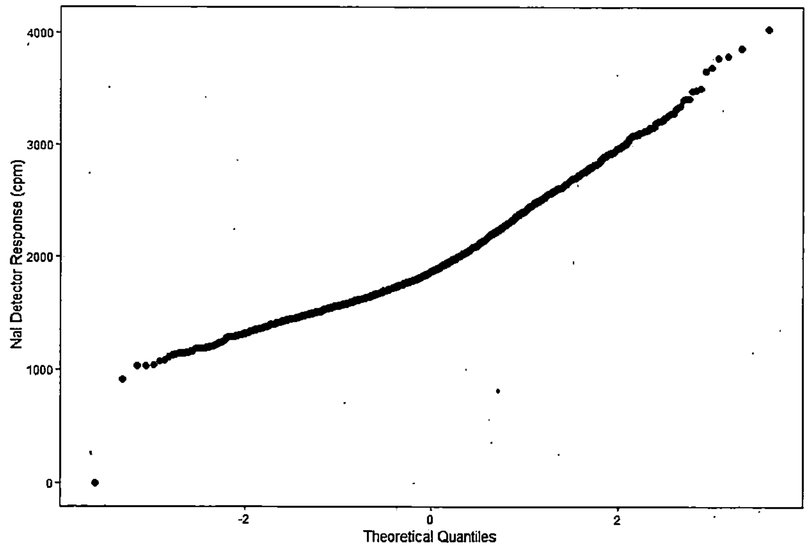

Figure 6.3. Q-Plot for Land Area 7

6.2.2 ROC Concentrations in Soil

Individual soil sample analytical results for Land Area 6 and 7 are presented in Appendix B, Table B.2 and B.3, respectively. A summary of the soil sample results is provided in Table 6.2. None of the individual results were above their respective $DCGL_{Op}$ for subsurface soil.

Table 6.2. Summary of ROC Concentrations in Land Area Soil Samples

ROC	Statistic (pCi/g)					Fraction[a]	
	Mean	Median	SD	Min	Max	Op.	BC
Land Area 6							
Co-60	0.001	-0.002	0.006	-0.012	0.011	0.01	0.00
Cs-134	0.002	0.002	0.013	-0.015	0.017	0.01	0.00
Cs-137	0.020	0.016	0.014	0.000	0.041	0.02	0.00
Land Area 7							
Co-60	0.009	0.009	0.012	-0.005	0.026	0.03	0.01
Cs-134	0.008	0.016	0.017	-0.018	0.024	0.02	0.00
Cs-137	0.04	0.04	0.03	0.003	0.097	0.05	0.01

[a]Op. represents the max result divided by the Operational DCGL; BC represents the max result divided by the Base Case DCGL

Figure 6.4 presents a comparison plot of the confirmatory duplicate sample results. In Figure 6.4 ORISE results are represented on ordinate and the FSS results are represented on the abscissa. The solid line represents a perfect agreement between the ORISE result and FSS data. Points below the line indicate that the FSS data are biased high and points above the line represent the opposite. As

indicated by Figure 6.4, the Cs-137 results are in good agreement based on the linear trend. Three data points for Co-60 have a significant departure from the solid line. Further evaluation of these three points is discussed in detail below.

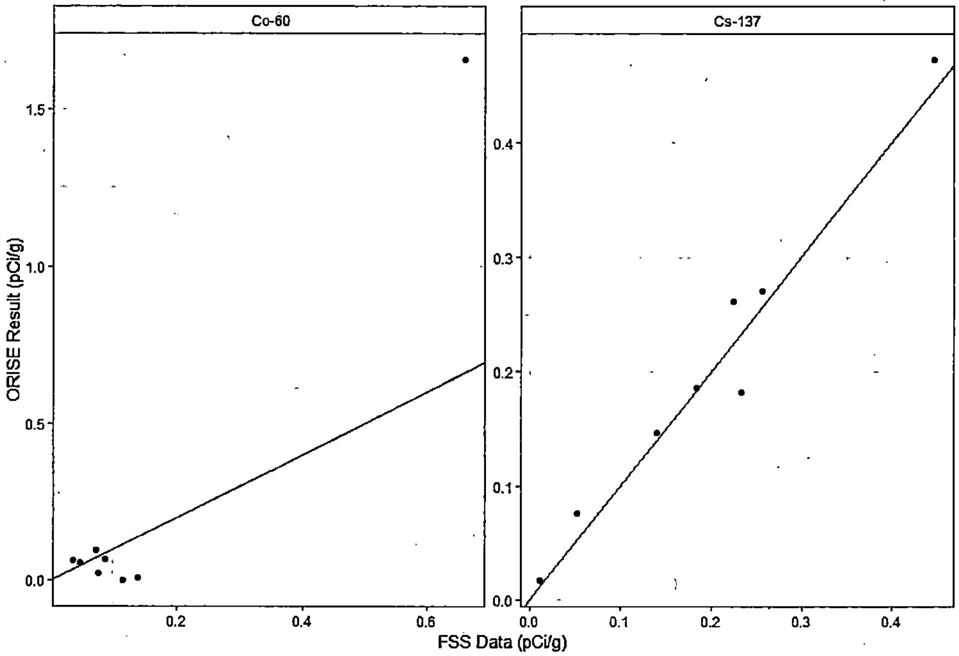

Figure 6.4. Plot of Confirmatory Duplicate Sample Results

Individual confirmatory duplicate results are provided in Appendix B, Table B.4. A summary of the duplicate error ratios is provided in Table 6.3 below. As indicated by the DER, all confirmatory duplicate sample results for Cs-137 and the majority for Co-60 are in agreement at the 99% confidence level. The three sample results that had a DER greater than 3.0 were the three Co-60 results discussed at the end of the previous paragraph. Two of the three samples did not contain a Co-60 concentration above the ORISE analytical MDC. The DER is inconclusive when one or both of the results are less than three to five times the MDC (DOE 2013). The ORISE result for the third sample with a DER above 3.0 had a Co-60 concentration approximately 2.5 times the FSS data. ORISE laboratory staff identified a discrete particle of Co-60 in the ORISE sample, which had a smaller mass than the FSS sample (as part of the original analysis, ORISE lab staff had to re-package the sample into a calibrated geometry). The remaining material from the repackaging was analyzed, and Co-60 was not identified in the sample. As a result of the discrete particle present in the re-packaged sample with a smaller mass, the resulting Co-60 concentration was biased high relative to that of the FSS result.

Table 6.3. DER Summary for Confirmatory Duplicate Samples			
ROC	Min	Max	(n > 3.0)[a]
Co-60	0.9	15.2	3
Cs-137	0.1	2.1	0

[a]Number of confirmatory duplicate samples above a DER of 3.0

7. SUMMARY AND CONCLUSIONS

At the NRC's request, ORISE conducted confirmatory survey activities of the WWTF and Land Areas 6 and 7 at ZNPS during the period of September 24–27, 2018. The survey activities included gamma surface scans, volumetric concrete sampling at 16 random/systematic locations in the WWTF, and volumetric soil sampling in the land areas.

All individual confirmatory samples collected from the WWTF were below their respective $DCGL_{Op}$ and, therefore, are also below the $DCGL_{BC}$. Only one concrete sample had Cs-137 concentration above the analytical MDC. All other radionuclides of concern were below their respective MDCs. The confirmatory concrete mean sample results and associated uncertainty for gamma-emitting radionuclides of concern are low relative to that of the FSS, and thus are not in agreement. This difference was attributed to the sensitivity between the two analysis methods. The confirmatory data are based on volumetric concrete samples and FSS data are based on *in situ* gamma spectrometry measurement, which has a higher detection limit than laboratory analysis of the concrete samples. Additional investigation is unnecessary provided the overall magnitude of the confirmatory survey results.

The confirmatory soil sample results from both land areas were all below their respective $DCGL_{Op}$ and, therefore, are also below the $DCGL_{BC}$. Review of the confirmatory duplicate sample results identified a discrepancy between the ORISE and the FSS Co-60 data, based on exceeding a DER of 3.0. Additional investigation of these duplicate samples determined that the DER calculation was inconclusive due to either sample homogeneity or one result less than MDC. Therefore, there is not a discernable bias between the ORISE and FSS laboratory measurement system for these samples.

Based on the parameters established by project-specific DQOs, confirmatory survey data did not identify issues that would preclude the FSS data for demonstrating compliance with the release criteria.

8. REFERENCES

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APPENDIX A
FIGURES

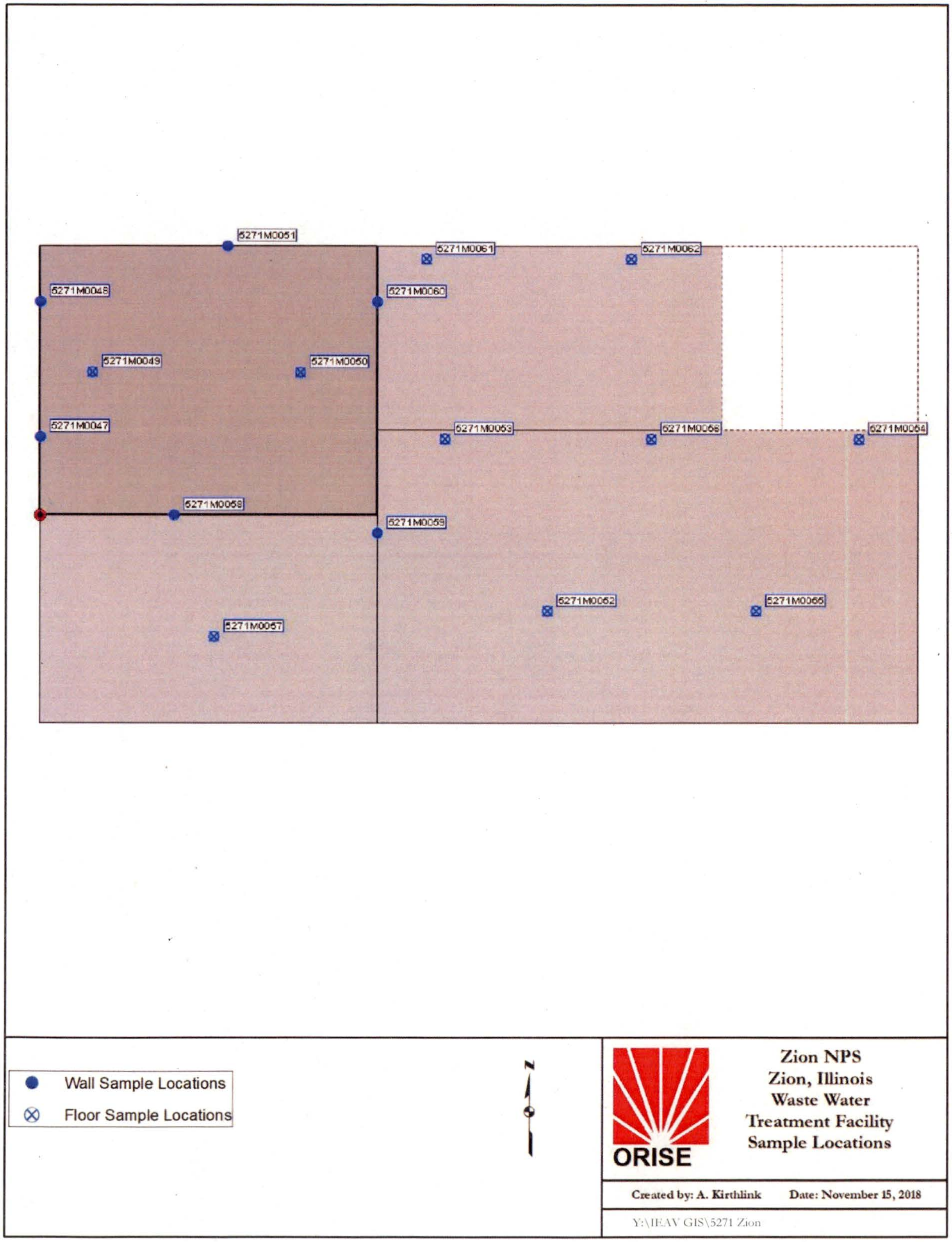
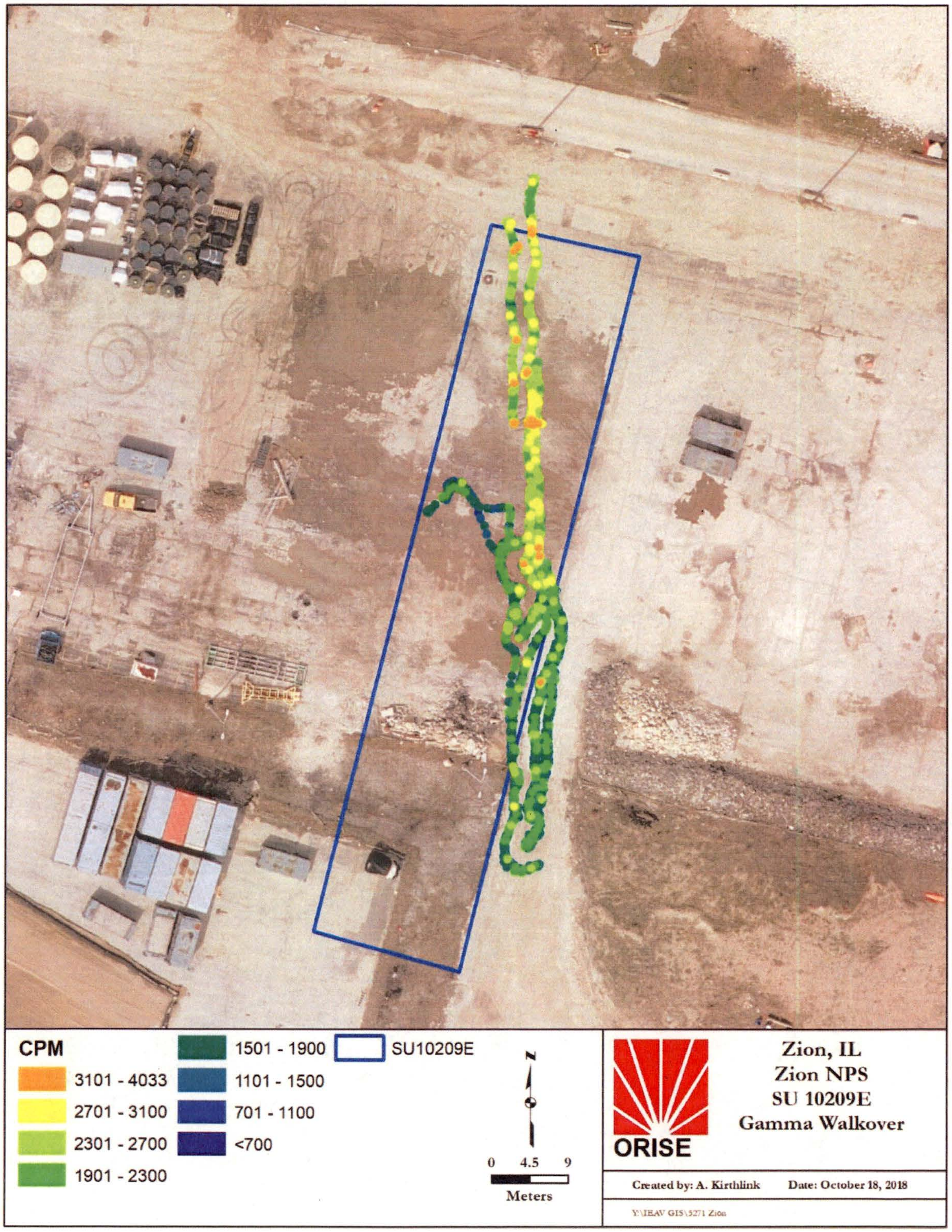


Figure A.1. WWTF Concrete Sampling Locations Figure



A.2. Gamma Walkover Survey of Land Area 7 Overburden

APPENDIX B
DATA TABLES

Table B.1. ROC Concentrations in WWTF Concrete (pCi/m²)[a]										
Sample ID[b]	Co-60		Cs-134		Cs-137		Sr-90		Ni-63	
	Result	MDC	Result	MDC	Result	MDC	Result	MDC	Result	MDC
5271M0047A	1.31E+03	1.01E+04	2.87E+03	1.31E+04	3.82E+03	7.28E+03	--	--	--	--
5271M0048A	2.15E+03	1.31E+04	-1.91E+03	1.36E+04	2.03E+03	6.21E+03	-2.39E+03	2.63E+04	1.15E+05	1.95E+05
5271M0049A	-8.36E+02	1.21E+04	6.57E+03	1.50E+04	-1.43E+03	1.22E+04	-7.16E+03	2.98E+04	8.36E+03	1.87E+05
5271M0050A	-4.78E+02	7.04E+03	-2.03E+03	1.48E+04	3.10E+03	1.41E+04	-1.67E+04	2.63E+04	3.34E+04	1.95E+05
5271M0051A	-2.63E+03	1.16E+04	2.39E+03	1.53E+04	5.85E+03	1.11E+04	--	--	--	--
5271M0052A	-3.34E+03	1.22E+04	-2.51E+03	1.25E+04	-5.01E+03	1.23E+04	-4.78E+03	2.98E+04	-9.55E+03	1.97E+05
5271M0053A	2.87E+03	1.24E+04	4.78E+02	1.31E+04	1.07E+03	1.31E+04	--	--	--	--
5271M0054A	8.36E+02	1.12E+04	-3.34E+03	1.34E+04	2.03E+03	9.31E+03	--	--	--	--
5271M0055A	-2.03E+03	1.17E+04	3.46E+03	1.40E+04	4.89E+03	9.43E+03	--	--	--	--
5271M0056A	7.16E+02	1.44E+04	9.55E+02	1.41E+04	1.55E+04	1.00E+04	--	--	--	--
5271M0057A	4.18E+03	1.13E+04	2.51E+03	1.11E+04	1.19E+02	1.04E+04	--	--	--	--
5271M0058A	-3.58E+02	1.21E+04	1.07E+03	1.33E+04	2.39E+03	1.17E+04	0.00E+00	2.75E+04	-3.58E+04	1.90E+05
5271M0059A	4.30E+03	1.35E+04	4.89E+03	1.50E+04	1.67E+03	1.40E+04	-1.43E+04	2.75E+04	3.10E+04	1.87E+05
5271M0060A	-2.27E+03	9.43E+03	3.10E+03	1.13E+04	1.91E+03	9.43E+03	--	--	--	--
5271M0061A	-1.67E+03	1.12E+04	3.34E+03	8.95E+03	9.55E+02	1.13E+04	-3.58E+03	2.98E+04	-1.67E+04	2.03E+05
5271M0062A	1.91E+03	1.40E+04	3.10E+03	1.46E+04	8.24E+03	1.27E+04	-1.67E+04	2.87E+04	4.78E+04	2.03E+05

[a]Values converted to units of pCi/m² based on a concrete density of 2.35 g/cm³ and sample depth of 5.08 cm.
[b]"A" represents 0 to 5.08 cm increment
[c]Indicates analysis was not performed

Table B.2. ROC Concentration in Land Area 6 Soil Samples (pCi/g)[a]									
Sample ID	Co-60			Cs-134			Cs-137		
5271S0031	-0.0032	±	0.0096	-0.018	±	0.012	0.054	±	0.013
5271S0032	-0.005	±	0.014	0.019	±	0.013	0.018	±	0.0063
5271S0033	0.006	±	0.013	0.024	±	0.014	0.061	±	0.013
5271S0034	-0.002	±	0.018	-0.003	±	0.012	0.0234	±	0.0098
5271S0035	0.026	±	0.016	-0.014	±	0.013	0.097	±	0.017
5271S0036	0.014	±	0.017	0.018	±	0.019	0.044	±	0.013
5271S0037	0.011	±	0.015	0.022	±	0.022	0.003	±	0.017
5271S0038	0.0214	±	0.0098	0.014	±	0.013	0.0393	±	0.0025

[a]Uncertainties represent the total propagated uncertainty reported at the 95% confidence level

Table B.3. ROC Concentration in Land Area 7 Soil Samples (pCi/g)[a]									
Sample ID	Co-60			Cs-134			Cs-137		
5271S0021	-0.0022	±	0.0053	0.0098	±	0.0073	0.0003	±	0.0003
5271S0022	0.009	±	0.018	-0.009	±	0.012	0.012	±	0.014
5271S0023	-0.004	±	0.012	-0.015	±	0.012	0.011	±	0.013
5271S0024	-0.003	±	0.028	0.015	±	0.028	0.035	±	0.014
5271S0025	-0.002	±	0.013	-0.0053	±	0.0079	0.0164	±	0.0062
5271S0026	0.00	±	0.012	-0.0078	±	0.0083	0.0311	±	0.0086
5271S0027	0.011	±	0.016	0.015	±	0.017	0.041	±	0.012
5271S0028	-0.002	±	0.015	0.017	±	0.014	0.0151	±	0.0088
5271S0029b	-0.012	±	0.019	-0.011	±	0.011	0.0118	±	0.0079
5271S0030b	0.011	±	0.019	-0.0025	±	0.0071	0.0113	±	0.0089

[a]Uncertainties represent the total propagated uncertainty reported at the 95% confidence level
[b]Judgmental sample

Table B.4. Confirmatory Duplicate Sample Results									
Sample ID	FSS Data				ORISE Data				DER
	Result (pCi/g)	Uncertainty (pCi/g)[a]	MDC (pCi/g)	Result/ MDC	Result (pCi/g)	Uncertainty (pCi/g)[b]	MDC (pCi/g)	Result/ MDC	
Co-60									
L1-10207D-EJGS-991SB	0.1160	0.0129	0.0304	3.8	0.000	0.019	0.014	0.0	7.2
L1-10207D-EJGS-993SB	0.6610	0.0343	0.0563	11.7	1.66	0.11	0.04	41.5	15.2
L1-10206D-EJGS-975SB	0.0477	0.0064	0.0168	2.8	0.058	0.019	0.028	2.1	0.9
L1-10207D-EJGS-990SB	0.0371	0.0076	0.0280	1.3	0.065	0.011	0.022	3.0	2.9
L1-10209D-EJGS-327SB	0.0773	--[c]	0.0845	0.9	0.025	0.014	0.021	1.2	--
L1-10209D-EJGS-230SB	0.1390	0.0146	0.0328	4.2	0.009	0.017	0.038	0.2	7.7
L1-10209D-EJGS-322SB	0.0739	0.0114	0.0332	2.2	0.098	0.017	0.022	4.5	1.7
L1-10209D-EJGS-325SB	0.0892	0.0112	0.0318	2.8	0.070	0.016	0.025	2.8	1.4
Cs-137									
L1-10207D-EJGS-991SB	0.0523	0.0118	0.0279	1.9	0.076	0.016	0.023	3.3	1.7
L1-10207D-EJGS-993SB	0.1400	0.0238	0.0602	2.3	0.146	0.021	0.029	5.0	0.2
L1-10206D-EJGS-975SB	0.4460	0.0341	0.0337	13.2	0.473	0.047	0.028	16.9	0.6
L1-10207D-EJGS-990SB	0.2320	0.0219	0.0325	7.1	0.182	0.020	0.019	9.6	2.1
L1-10209D-EJGS-327SB	0.2240	0.0260	0.0393	5.7	0.261	0.024	0.024	10.9	1.3
L1-10209D-EJGS-230SB	0.0118	--	0.0542	0.2	0.017	0.01	0.021	0.8	--
L1-10209D-EJGS-322SB	0.1830	0.0220	0.0366	5.0	0.186	0.019	0.018	10.3	0.1
L1-10209D-EJGS-325SB	0.2550	0.0283	0.0500	5.1	0.270	0.026	0.020	13.5	0.5

[a]Uncertainties represent the total propagated uncertainty reported at the 68% confidence level (1-sigma uncertainty)
[b]Uncertainties represent the total propagated uncertainty reported at the 95% confidence level (2-sigma uncertainty)
[c]Indicates value not reported

APPENDIX C
SURVEY AND ANALYTICAL PROCEDURES

C.1. PROJECT HEALTH AND SAFETY

ORISE performed all survey activities in accordance with the *ORAU Radiation Protection Manual*, the *ORAU Health and Safety Manual*, and the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2014, ORAU 2016b, and ORAU 2016a). Prior to on-site activities, a work-specific hazard checklist was completed for the project and discussed with field personnel. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. Additionally, prior to performing work, a pre-job briefing and walkdown of the survey areas were completed with field personnel to identify hazards present and discuss safety concerns. Should ORISE have identified a hazard not covered in the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2016a) or the project's work-specific hazard checklist for the planned survey and sampling procedures, work would not have been initiated or continued until the hazard was addressed by an appropriate job hazard analysis and hazard controls.

C.2. CALIBRATION AND QUALITY ASSURANCE

Calibration of all field instrumentation was based on standards/sources, traceable to National Institute of Standards and Technology (NIST).

Field survey activities were conducted in accordance with procedures from the following documents:

- *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2016a)
- *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2017)
- *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2018)

The procedures contained in these manuals were developed to meet the requirements of U.S. Department of Energy (DOE) Order 414.1D and the NRC *Quality Assurance Manual for the Office of Nuclear Material Safety and Safeguards* and contain measures to assess processes during their performance.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations.

- Participation in Mixed-Analyte Performance Evaluation Program and Intercomparison Testing Program laboratory quality assurance programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

C.3 RADIOLOGICAL SAMPLE ANALYSIS

C.3.1 Gamma Spectroscopy

Samples were analyzed as received, mixed, crushed, and/or homogenized as necessary, and a portion sealed in a 0.5-liter Marinelli beaker. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry. Net material weights were determined and the samples counted using intrinsic, high purity, germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. All total absorption peaks (TAPs) associated with the radionuclides of concern were reviewed for consistency of activity. Spectra were also reviewed for other identifiable TAPs. TAPs used for determining the activities of radionuclides and the typical associated MDCs for a one-hour count time are presented in Table C.1.

Table C.1. Typical MDCs Total Absorption Peak

Radionuclide	TAP (MeV)[a]	MDC (pCi/g)
Co-60	1.332	0.06
Cs-134	0.795	0.05
Cs-137	0.662	0.05

[a]MeV = mega electron volt

C.3.2 Ni-63 Analysis

Soil samples were spiked with a nickel and cobalt carrier and digested with a mixture of nitric and hydrochloric acids. Unwanted elements, such as iron and cobalt, are then removed by running the slurry via anion exchange chromatography. Nickel is then separated from the slurry using a nickel selective resin cartridge. The purified nickel is then eluted off of the column with a dilute nitric acid solution. Ni-63 activity is then determined via liquid scintillation counting. The typical MDC for a 1 gram sample and 60-minute count time using this procedure is 1.6 pCi/g.

C.3.3 Radioactive Strontium Analysis

Sr-90 concentrations were quantified by total sample dissolution followed by radiochemical separation and counted on a low background proportional counter. Samples were homogenized and dissolved by a combination of potassium hydrogen fluoride and pyrosulfate fusions. The fusion cakes were dissolved, and strontium is coprecipitated on lead sulfate. The strontium was separated from residual calcium and lead by reprecipitating strontium sulfate from EDTA at a pH of 4.0. Strontium was separated from barium by complexing the strontium in DTPA while precipitating barium as barium chromate. The strontium was ultimately converted to strontium carbonate and counted on a low-background gas proportional counter. The typical MDC for a 1 gram sample and 120-minute count time using this procedure is 0.24 pCi/g.

APPENDIX D
MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the author or his employer.

D.1 SCANNING AND MEASUREMENT INSTRUMENT/DETECTOR COMBINATIONS

D.1.1 Gamma

Ludlum NaI Scintillation Detector Model 44-10, Crystal: 5.1 cm × 5.1 cm
coupled to: Ludlum Ratemeter-scaler Model 2221
coupled to: Trimble Geo 7X

D.2 LABORATORY ANALYTICAL INSTRUMENTATION

High-Purity, Extended Range Intrinsic Detector
Canberra/Tennelec Model No: ERVDS30-25195
Canberra Lynx® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)

High-Purity, Intrinsic Detector
EG&G ORTEC Model No. GMX-45200-5
Canberra Lynx® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)

High-Purity, Intrinsic Detector
EG&G ORTEC Model No. GMX-30P4
Canberra Lynx® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)

High-Purity, Intrinsic Detector

EG&G ORTEC Model No. CDG-SV-76/GEM-MX5970-S
Canberra Lynx® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)

Low-Background Gas Proportional Counter
Series 5 XLB
(Canberra, Meriden, CT)
Used in conjunction with:
Eclipse Software
Dell Workstation
(Canberra, Meriden, CT)