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May 29, 2026

Mr. William C. Allen, Project Manager
U.S. Nuclear Regulatory Commission
11545 Rockville Pike
Rockville, MD 20852

SUBJECT: CONFIRMATORY SURVEY OF BUILDING SUBSTRUCTURES AT THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT, CRYSTAL RIVER, FLORIDA; DOCKET NUMBER 50-302; RFTA 26-001; DCN 5390-SR-01-0

Dear Mr. Allen:

The Oak Ridge Institute for Science and Education (ORISE) is pleased to provide the enclosed final report detailing the confirmatory radiological survey activities and results performed at the Crystal River Unit 3 Nuclear Generating Plant located in Crystal River, Florida.

Please feel free to contact me at Erika.Bailey@orau.org if you have any comments or concerns.

Sincerely,

Erika N. Bailey

Erika N. Bailey
Survey & Technical Projects Manager
ORISE

TJV:enb

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CONFIRMATORY SURVEY OF BUILDING SUBSTRUCTURES AT THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT, CRYSTAL RIVER, FLORIDA

Timothy J. Vitkus

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

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CONFIRMATORY SURVEY OF BUILDING SUBSTRUCTURES AT THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT, CRYSTAL RIVER, FLORIDA

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FINAL REPORT

MAY 2026

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ACRONYMS

AF	area factor
ALARA	as low as reasonably achievable
Am-241	americium-241
BOP	balance of plant
C-14	carbon-14
CFR	Code of Federal Regulations
cm	centimeter(s)
Co-60	cobalt-60
cpm	counts per minute
CR3	Crystal River Unit 3
CREC	Crystal River Energy Complex
CS	confirmatory survey
D&D	decontamination and decommissioning
DCGL	derived concentration guideline level
$DCGL_{EMC}$	derived concentration guideline level - elevated measurement comparison
DQO	data quality objective
EPA	U.S. Environmental Protection Agency
Eu-152	europium-152
Eu-154	europium-154
Fe-55	iron-55
FSS	final status survey
GPS	global positioning system
H-3	tritium
HTD	hard-to-detect
ISFSI	independent spent fuel storage installation
LTP	License Termination Plan
m	meter
m^2	square meter
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
MWt	megawatt-thermal
NaI[Tl]	thallium-doped sodium iodide
Ni-59	nickel-59
Ni-63	nickel-63
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	picocurie per gram
PSP	project-specific plan
PSQ	principal study question
PSR	partial site release
Pu-239	plutonium-239
ROC	radionuclide of concern
SOF	sum-of-fractions

Sr-90	strontium-90
SU	survey unit
Tc-99	technetium-99
TEDE	total effective dose equivalent
TPU	total propagated uncertainty

CONFIRMATORY SURVEY OF THE BUILDING SUBSTRUCTURES AT THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT, CRYSTAL RIVER, FLORIDA

1. INTRODUCTION

The Crystal River 3 Nuclear Generating Plant (CR3) is part of the larger Duke Energy Florida-owned Crystal River Energy Complex (CREC) that is located near Crystal River, Florida. CR3 was a single-unit pressurized light-water reactor with a maximum power level of 2,609 megawatt-thermal (MWt) that went into commercial operation in March 1977. The final reactor shutdown occurred in September 2009, followed by permanently ceasing all operations in 2013, with all fuel having been permanently removed from the reactor vessel in May 2011 (ADP 2025). In January 2019, the site submitted a request for a partial site release (PSR) of non-impacted land areas, in accordance with 10 Code of Federal Regulation (CFR) 50.83(b). The U.S. Nuclear Regulatory Commission (NRC) approved the PSR request in January 2020. Then, in February 2025, the site submitted a second request for a PSR for additional portions of the CREC outside the boundaries of the nuclear operations area, which the NRC approved on December 18, 2025, reducing the CR3 licensed/owner-controlled area to approximately 265 acres.

The CR3 site is continuing decontamination and decommissioning (D&D) and final status surveys (FSS) within licensed/owner-controlled areas in accordance with Revision 4 of the CR3 License Termination Plan (LTP). The LTP, which has been submitted to the NRC for review and approval, details the commitments necessary to ultimately request the release of the site for unrestricted use, excluding the independent spent fuel storage installation (ISFSI) area, in accordance with 10 CFR 20.1402. As part of ongoing D&D activities at the site, certain areas are becoming available for confirmatory survey (CS) activities following the site's FSS activities. The NRC requested that the Oak Ridge Institute for Science and Education (ORISE) perform CS activities within the Turbine, Control Complex, Intermediate, East Auxiliary, West Auxiliary, and Reactor Buildings basement substructures.

A project-specific plan (PSP) was developed that detailed the objectives and procedures for the confirmatory investigations and the methods for assessing the confirmatory data (ORISE 2025). The intent of the confirmatory investigation was to collect independent data for NRC's use in determining whether the CR3 FSS data accurately represented residual radiological conditions and whether the levels satisfy requirements for unrestricted release. ORISE performed the confirmatory survey of the selected survey units during the period January 5 through 8, 2026.

2. SITE DESCRIPTION

The CREC is situated on a 1,917-hectare (4,738-acre) site that is located on the Gulf of America in Citrus County Florida at 15760 West Power Line Street. The site is approximately 12 kilometers (km) (7.5 miles) northwest of the City of Crystal River and 110 km (70 miles) north of Tampa. The non-impacted land area PSR involved 1,559 hectares (3,852 acres), and the second PSR released an additional 250 hectares (618 acres) of the CREC complex. The remaining 107-hectare (approximately 265 acres) owner-controlled area encompassing CR3 is within the yellow boundary polygon presented in Figure 2.1. The CR3 Turbine, Control Complex, Intermediate, Auxiliary, and Reactor Buildings substructures that are to remain in place and subject to CS activities are shown in Figure 2.2.

These building substructures contain Class 1, Class 2, and Class 3 survey units (SUs) as determined by the licensee through process knowledge or characterization activities in accordance with the *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (NRC 2000).

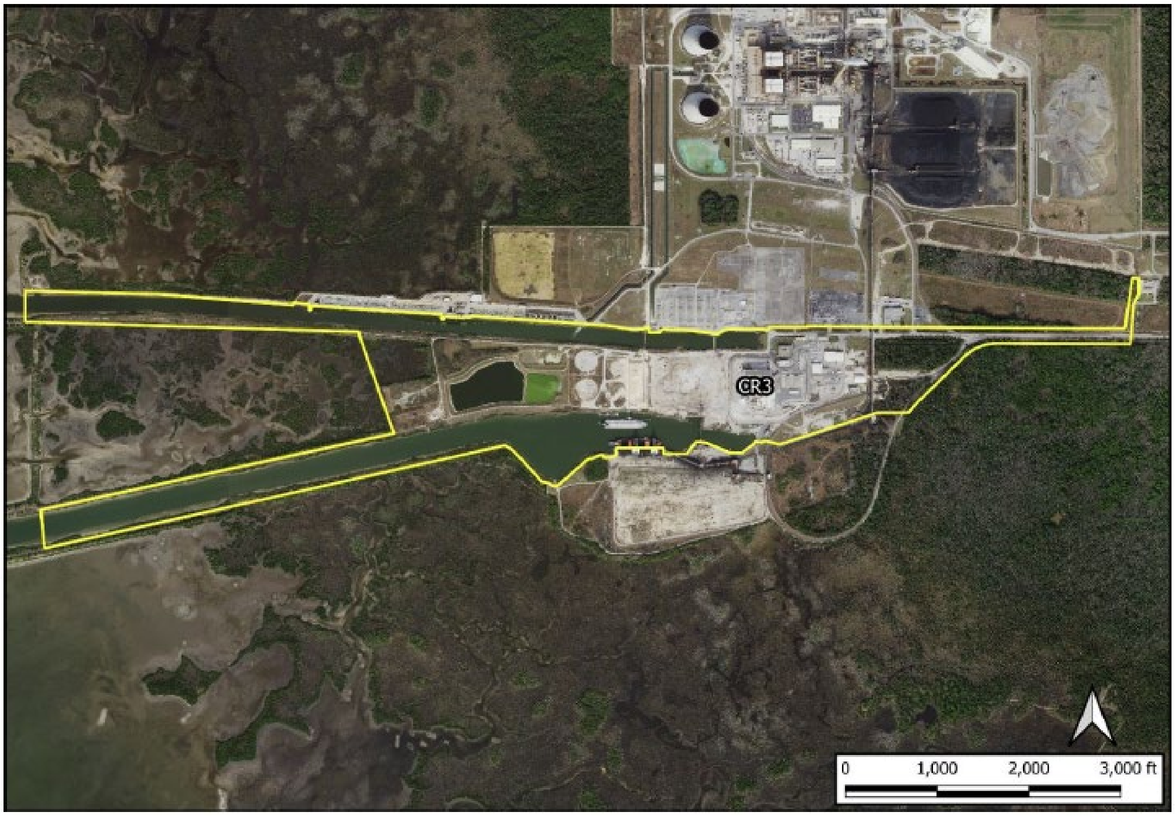

Figure 2.1. Map of Proposed Owner-Controlled Area (ADP 2025)

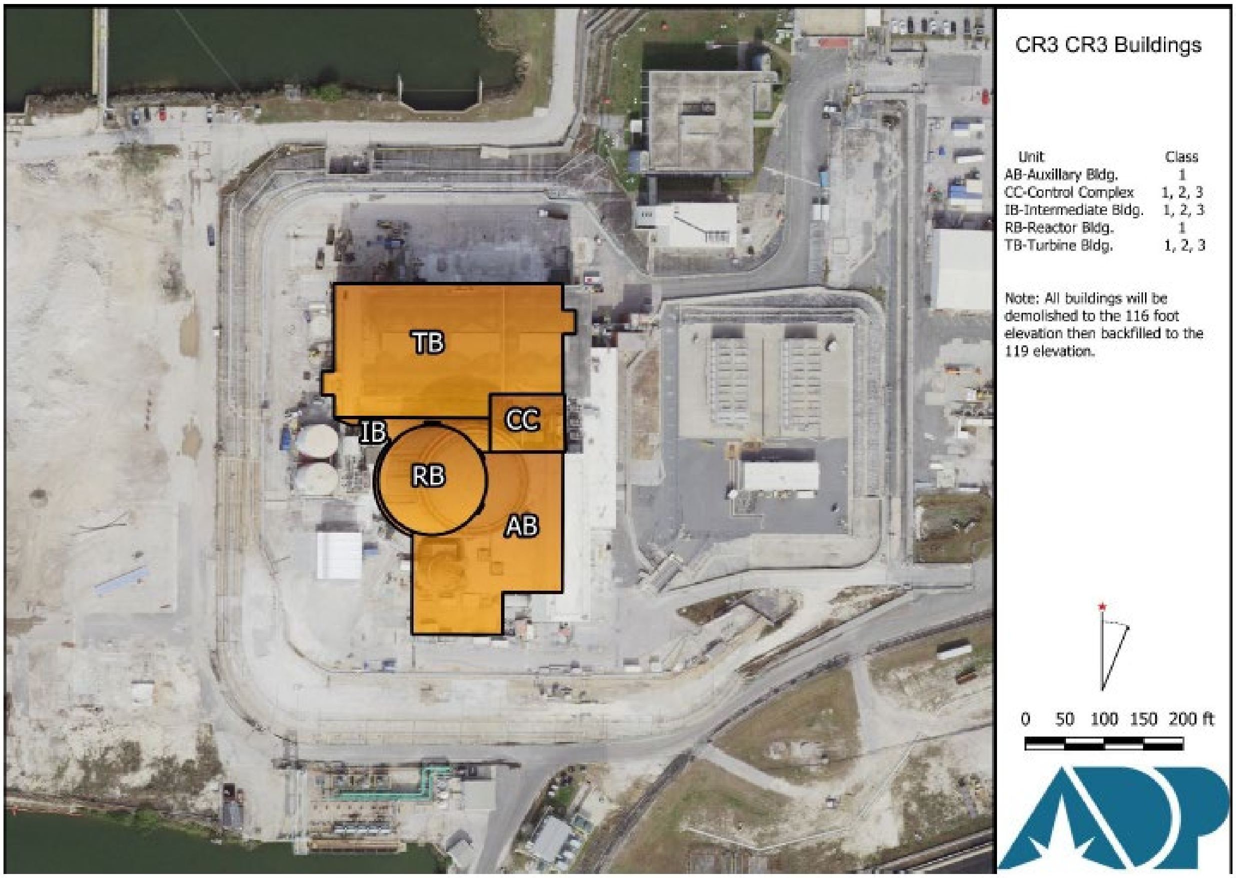

Figure 2.2. CR3 Building Substructures to Remain in Place (ADP 2025)

3. DATA QUALITY OBJECTIVES

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

CS DQOs were originally presented in ORISE 2025 and are represented here for completeness.

3.1 STATE THE PROBLEM

The first step in the DQO process defines the problem that necessitated the study, identifies the planning team, and examines the project budget and schedule. The planning team, project budget, and schedule were presented in ORISE 2025 and are not repeated here. The CR3 FSS activities were conducted in accordance with MARSSIM (NRC 2000) to demonstrate that any residual radiological contamination on or within substructures remaining on site satisfied the approved radiological-derived concentration guideline levels (DCGLs) necessary for release without radiological restrictions. The FSS activities included surface scans, surface activity measurements, and volumetric sampling/removable activity smear sampling.

The CS objectives are to provide NRC with independent radiological data to assist the NRC in evaluating the FSS results. Therefore, the problem statement is as follows:

> Confirmatory surveys activities are necessary to generate independent radiological data to assist the NRC with their assessment of the CR3 FSS design, implementation, and results,

and that the results adequately account for surficial and volumetric residual activity to ensure compliance with the release criteria.

3.2 IDENTIFY THE DECISION/OBJECTIVE

The second step in the DQO process identifies the principal study questions (PSQs) and alternative actions, develops decision statements, and organizes multiple decisions as appropriate. This second step specifies alternative actions that could result from a "Yes" response to the PSQs and combining the PSQs and alternative actions into decision statements. Given that the problem statement introduced in Section 3.1 is fairly broad, multiple PSQs arose. PSQs, alternative actions, and combined decision statements are presented in Table 3.1.

Table 3.1. Confirmatory Survey Decision Process for CR3 Substructures

Principal Study Questions	Alternative Actions
PSQ1: Are confirmatory survey results below applicable unrestricted release limits?	**Yes:** Compile confirmatory data and report results to NRC for their decision-making. Provide interpretation of confirmatory field surveys and verify that: 1) surveys did not identify anomalous areas of residual radioactivity, 2) quantitative field and analytical data were less than the applicable unrestricted release limits. **No:** Compile confirmatory data and report results to the NRC for their decision-making. Provide interpretation of confirmatory field surveys and verify that: 1) confirmatory surveys identified anomalous areas of residual radioactivity, 2) quantitative field and analytical data exceeded the unrestricted release criteria.
PSQ2: Has the licensee accurately or otherwise conservatively represented the radionuclides of concern (ROCs) and the fractional abundances used to calculate total activity results via either gross activity or surrogate measurement methods?	**Yes:** Compile radionuclide-specific analytical results for each sample, indicate detectable and non-detectable results, calculate fractional estimates, compare with licensee's LTP assumptions, and provide to the NRC for their evaluation and determination that the licensee adequately represented the radionuclide mixture and relationship used in calculating FSS data. **No:** Compile radionuclide-specific analytical results in comparison to the licensee's assumptions and indicate parameters that are not in agreement between confirmatory results and licensee, provide to the NRC for their determination of any non-conservative impact to reported FSS results and guideline compliance.

Table 3.1. Confirmatory Survey Decision Process for CR3 Substructures	
Principal Study Questions	**Alternative Actions**
PSQ3: Do the confirmatory results support the MARSSIM classification of the FSS SUs?	**Yes:** Confirmatory results support the classification of the FSS SU(s). Compile confirmatory survey data and present results to the NRC for their decision-making. **No:** Confirmatory results do not support the classification of the FSS SU(s). Summarize the discrepancies and provide technical comments to the NRC for their decision-making.
Decision Statements	
• The confirmatory survey data agree/do not agree with the licensee's assumed inventory of ROCs and/or that fractional amounts used in developing surrogate ratios or gross activity do/do not accurately represent residual activity, and that the FSS data are/are not adequate for demonstrating that the residual surface activity levels are below the unrestricted release limits. • Confirmatory survey results do/do not support the site's MARSSIM classification of the FSS SUs.	

3.3 IDENTIFY INPUTS TO THE DECISION/OBJECTIVE

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

- FSS survey packages and final data;
- DCGLs, surface activity limits, and removable contamination limits, further discussed in subsection 3.3.1;
- ORISE confirmatory results for surface scans and surface activity levels;
- ORISE volumetric sample and removable activity results.

3.3.1 Radionuclides of Concern and Release Guidelines

CR3 structures and systems are potentially impacted as a result of reactor operations. The site developed a list of primary radionuclides of concern (ROCs) through characterization activities. Site-specific soil DCGLs were developed that correspond to a residual radioactive contamination level, above background, that ensures the total effective dose equivalent (TEDE) to an average member of the critical group will not exceed 25 millirems per year (mrem/yr). These DCGLs are radionuclide-specific and independently correspond to a TEDE of 25 mrem/yr for each source term (see Table 3.2). In addition to the site-specific DCGLs, Table 3.2 also includes ALARA (as low as reasonably achievable) DCGLs that were adopted from the NRC/EPA Memorandum of Understanding Industrial Soil Consultation Soil Trigger Levels. These ALARA DCGLs are equivalent to the site-specific DCGLs for all ROCs except H-3 and Eu-152, in which case the lower of the DCGLs is applied. For H-3 and Eu-152, the ALARA DCGL is lower. These soil concentration DCGLs are applicable to other media, specifically asphalt and concrete, with consideration of differences in material density.

Table 3.2. Resident Farmer Scenario-Based Site-Specific DCGLs[a]

ROC	Subsurface Soil DCGLv 0 to 1 meter depth[b] (pCi/g)	ALARA DCGLv 0 to 1 meter depth[b] (pCi/g)
H-3	1,480	423
C-14	16.4	16.4
Fe-55	33,600	33,600
Co-60	3.4	3.4
Ni-59	2,685	2,685
Ni-63	980	980
Sr-90	3.6	3.6
Tc-99	61.8	61.8
Cs-137	9.6	9.6
Eu-152	7.8	7
Eu-154	7.2	7.2
Pu-239	59	59
Am-241	87.4	87.4

[a]Found in Table 6-2 in the site's LTP.
[b]Also applicable to asphalt and concrete with consideration of differences in material density.
DCGLv = derived concentration guideline level for volumetric concentrations
pCi/g = picocuries per gram

Because each individual DCGL in Table 3.2 corresponds to the TEDE criterion, the sum-of-fractions (SOF) approach must be used to evaluate the total dose from each SU and demonstrate compliance with the dose limit. The SOF calculation is performed as follows:

$$SOF = \sum_{j=1}^{n} \frac{C_j}{DCGL_j} \qquad \text{Eq. (3-1)}$$

Where:

C_j is the concentration of ROC "j"

$DCGL_j$ is the DCGL for ROC "j"

As explained in Chapter 6 of the LTP, the significant-dose contributing ROCs in the reactor building itself were carbon-14 (C-14), cobalt-60 (Co-60), and cesium-137 (Cs-137), with the remaining ROCs being deselected as insignificant based on their contributing less than 10% to the TEDE. The site estimated the combined dose total from the deselected ROCs to be 0.20 mrem. The remaining building substructures have been termed balance of plant (BOP), with the only dose-significant ROC being Cs-137; with the remaining ROCs deselected based on having an insignificant combined dose total of 0.42 mrem. Volumetric ALARA DCGLs for the significant ROCs were converted to surface activity units as described in Section 6.3.5.4 of the LTP. These surface activity DCGLs for the dose significant radionuclides are presented in Table 3.3. The quantity DCGL × AF is referred to as the elevated measurement comparison, denoted $DCGL_{EMC}$. Discussion of EMCs and the relevant area factors (AFs) is presented in Section 6.4.6. of the LTP (ADP 2025) and are not reproduced in this document.

Table 3.3. ALARA Surface Activity DCGLs for Dose Significant ROCs[a]

ROC	DCGL (dpm/100 cm²)
C-14	5.46E+05
Co-60	1.13E+05
Cs-137	3.20E+05

[a]Found in Table 6-14 in the site's LTP.
dpm = disintegrations per minute

As only BOP substructures were investigated during this CS, the only applicable DCGLs for data comparison from Table 3.3 were the surface activity and volumetric limits for Cs-137 of 3.20+E05 and 9.6 pCi/g, respectively.

3.4 DEFINE THE STUDY BOUNDARIES

The fourth step in the DQO process defines target populations and spatial boundaries, determines the timeframe for collecting data and making decisions, addresses practical constraints, and determines the smallest subpopulations, area, volume, and time for which separate decisions were to be made.

NRC identified the Turbine, Control Complex, Intermediate, Auxiliary, and Reactor Building substructures for CS activities, with the additional boundary condition that the site must have completed the FSS before an SU was selected for CS. Temporal and/or access restriction boundaries were also factors as to which SUs or portions of SUs were to have CS. The temporal boundary was the scheduled period budgeted for the on-site visits, while having safe or physical access to targeted confirmatory areas was an access-restriction boundary, e.g., confined space status, erected scaffolding, or the availability of aerial lifts.

A temporal boundary of three days on-site to complete the confirmatory field activities was established by Request for Technical Assistance 26-001. On-site CS activities were performed from January 5–8, 2026.

3.5 DEVELOP A DECISION RULE

The fifth step in the DQO process specifies appropriate population parameters (e.g., mean, median), develops action levels, confirms detection limits are less than action levels, and develops "if…then…" decision rule statements. Multiple PSQs were introduced in Table 3.1, resulting in multiple decision rules. Additional discussions on this matter are provided in Section 3.5.3 and Section 6. Decision rules for each PSQ are discussed below.

CS results are not intended to demonstrate compliance with the release criterion directly, but rather demonstrate that the FSS results are appropriate for the intended use. The CS PSQs were concerned with identifying residual radioactivity above the DCGL, identifying anomalous ROC mixtures relative to those presented in the LTP, and whether SUs were correctly classified.

3.5.1 PSQ1: Residual Contamination Levels

The decision for PSQ1 requires comparison of the CS results with the Table 3.2 and Table 3.3 volumetric and surface activity DCGLs to select the appropriate alternative action. Decision rules that arise from the CS PSQ1s are:

> If CS measurement results are less than the ROC-specific gross activity DCGLs, then provide results to the NRC indicating that CS results were less than the 25 mrem/yr free release criteria. If confirmatory surface activity levels exceed the DCGLs, results are provided to NRC for their assessment and decision-making.

3.5.2 PSQ2: ROC Mixture and Fractional Abundances

PSQ2 was developed to assess the licensee's determination of the significant ROCs assumed present in residual contamination and the fractional abundances used to calculate total activity results via either gross activity or surrogate methods were appropriate. The CS PSQ2 decision rules are:

> If the on-site investigations of SUs selected for CS do not identify any radiological anomalies and/or if radionuclide specific laboratory analyses of samples collected agree with the licensee's-determined ROC mixture and relative abundances, then provide confirmatory results to NRC to support decision making. If analytical results indicate non-conservative ROC abundances, such as deselected ROCs present at concentrations that could contribute greater than 10% of the TEDE or ROC abundances that indicate the gross activity DCGLs are not conservative, provide results with technical assessments to NRC for decision making.

3.5.3 PSQ3: MARSSIM Classification

PSQ3 was developed to assess whether the licensee had properly classified the SU and performed the FSS in accordance with MARSSIM guidance. The CS PSQ3 decision rule is: Do the confirmatory results support the MARSSIM classification of the FSS SUs?

> The CS results are compared with the DCGLs, residual contamination levels in Class 2 SUs must be less than the DCGLs and within Class 3 SUs should be no more than a fraction of the DCGL, generally 10% or less and CS results satisfying these conditions indicate the SU was properly classified. If sample concentrations or surface activity levels/removable levels

indicate that a Class 2 or Class 3 should be reclassified to a higher classification, then summarize confirmatory data for NRC's evaluation.

3.6 SPECIFY LIMITS ON DECISION ERRORS

The sixth step in the DQO process examines the consequences of making an incorrect decision and establishes bounds on decision errors. Decision errors were controlled during the survey design, on-site field investigations, and during the data assessment. There were two orders of control, each discussed in the subsections below.

3.6.1 Scan Surveys

The CS was planned to rely on judgmental surface scanning to identify candidate locations for selecting both confirmatory judgmental surface activity measurement and volumetric sample locations. The confirmation of ROCs present and the relative ratios between hard-to-detect (HTD) and detectable ROCs relied on identifying sample locations that contained residual contamination where activity levels were high enough to readily quantify the activity and minimize the analytical uncertainty. The preliminary step in sample location selection was the scanning surveys that were performed.

There are relevant decision errors associated with radiation scanning. NUREG-1507 (NRC 2020) describes scanning decision errors used in the calculation of the scan minimum detectable concentration (MDC_{SCAN}). These errors are defined here as the true positive proportion (correctly concluding that contamination is present) and false positive proportion (incorrectly concluding that contamination is present). The false positive proportion is often harder to assign, though the assigned statistics have a considerable impact on the survey design. If contamination levels above DCGLs are difficult to detect, greater diligence is necessary to identify the contamination. Surveyors are instructed to investigate small anomalous responses, and then a relatively large false positive proportion is assigned. Alternatively, if the contamination at or above the DCGL is easily detected and it is not important for surveyors to pause often, then a relaxed false positive proportion, i.e., fewer pauses, can be tolerated.

For this CS, it was assumed that the NRC desired high confidence that scanning locations identified for investigation would both represent the contamination profile(s) and bias the CS such that the highest residual contamination was identified for comparison with the DCGLs. Therefore, the confirmation gamma-scanning assessments were based on a high true positive proportion of 0.95, or 5% possibility of overlooking a small area of elevated activity that may provide such data. The design was also based on a false positive proportion of 0.50, or a 50% possibility that the surveyor will investigate a potential anomaly when none is present.

3.6.2 Field and Analytical MDCs

The second order of control was to optimize the confirmatory field measurement and laboratory analytical MDCs. The analytical MDCs request was for preferred MDCs of less than 10% of each DCGL, but if not reasonably achievable, then no greater than 50% of each DCGL to ensure the analytical results were sufficient for decision-making. Detector type, count times, and other related factors were selected that ensured surface activity measurement MDCs were less than the DCGLs.

3.7 Optimize the Design for Obtaining Data

The seventh step in the DQO process involves reviewing the DQO outputs, developing data collection design alternatives, formulating mathematical expressions for each design, selecting the sample size to satisfy DQOs, determining the most resource-effective design among agreed-upon alternatives, and documenting the requisite details. Judgmental sample/measurement results were determined to be adequate to collect the data quantity and quality needed to answer the PSQs. Therefore, the parameter(s) of interest are the individual judgmental surface activity measurement and volumetric sample results. The specific survey procedures implemented during the CS are presented in Section 4.

4. PROCEDURES

The ORISE survey team performed visual inspections, measurements, and sampling activities within the accessible portions of the selected SUs in each building substructure, or as specifically requested by NRC. Survey activities were conducted in accordance with the PSP, the *Oak Ridge Associated Universities (ORAU) Radiological and Environmental Survey Procedures Manual,* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORISE 2025, ORAU 2016, ORAU 2024). Appendices C and D provide additional information regarding survey instrumentation and related processes discussed within this section.

4.1 Reference System

ORISE referenced confirmatory measurement/sampling locations to global positioning system (GPS) coordinates mainly for the subsurface floors when the GPS unit was able to receive adequate satellite signals. The coordinate system was based on NAD 1983 Conus CORSS96 State Plane Florida West FIPS 0902 (meters). Prominent site features were used for reference when position data capture could not be achieved. Measurement and sampling locations were documented on detailed survey maps and appropriate field forms.

4.2 Surface Scans

CS gamma radiation surface scans were performed in judgmentally selected SUs. Substructure surfaces scanned included floors, lower and upper walls, sumps, and floor trenches where drains had been removed. CS surface scans first focused on judgmentally selected areas where potential residual contamination could have migrated to and accumulated or missed during remediation, as well as SU boundaries that separated areas with different classifications.

The scan coverage ranged from low to high density (up to 100%) and was commensurate with the site's SU classification, based on accessibility but primarily on contamination potential, with areas with the greatest residual contamination potential receiving the high-density scans. Table 4.1 summarizes the substructure, SU identifier, the SU classification, and the scan coverage for each confirmatory SU.

Table 4.1. Confirmatory Survey Units, Classification, and Gamma Radiation Surface Scan Density				
Building Substructure	**SU ID**	**SU Type**	**Classification**	**Scan Coverage**
Auxiliary Bldg.	CRAB-01	East (Upper[a] and Lower) and South Walls (Lower)	1	Low Density
Auxiliary Bldg.	CRAB-02	East Floors and Trenches	1	High Density
Auxiliary Bldg.	CRAB-03	Aux. Bldg. Sump	1	High Density
Auxiliary Bldg.	CRAB-04	Laundry and Shower Sump	1	High Density
Auxiliary Bldg.	CRAB-05	Elevator Shaft and Sump	1	High Density
Auxiliary Bldg.	CRAB-06	Aux./Reactor Bldg. Wall(Upper and Lower)	1	High Density
Control Complex	CRCC-01	East Wall (Lower)	1	Low Density
Control Complex	CRCC-02	Floors and Trenches	1	High Density
Intermediate Bldg.	CRIB-01	Floors	1	High Density
Intermediate Bldg.	CRIB-02	Lower and Upper Walls include Inter./Reactor Bldg. Wall	1	Low Density
Turbine Bldg.	CRTB-01	Walls	2	Low Density
Turbine Bldg.	CRTB-02	East Floor	2	High Density
Turbine Bldg.	CRTB-06	West Floor and Demin Sump	2	High Density

[a]Only a small section of the upper East wall was scanned with higher density coverage in the vicinity of what the site called "the window"; a location where significant remediation had occurred.

The surface scans used Ludlum model 44-10, 5.1-centimeter by 5.1-centimeter (2-inch by 2-inch) thallium-doped sodium iodide (NaI[Tl]) scintillation detectors coupled to Ludlum model 2221 ratemeter-scalers with audible indicators, and were coupled to GPS systems that enabled real-time gamma count rate and geo-referenced data capture when a GPS signal was available in the basement substructures. Locations of elevated response that were audibly distinguishable from localized background levels, suggesting the presence of residual contamination, were marked for further investigation.

4.3 Surface Activity Measurement and Volumetric Sampling Locations

Confirmatory direct surface activity measurements were made at judgmental locations that were identified during the gamma radiation surface scanning phase. At each selected location, direct, quantitative alpha and beta static surface activity measurements were made using the Ludlum Model 43-92 and 44-142 scintillation detectors, respectively. Qualitative static gamma radiation measurements. were also made at each location using the Ludlum Model 44-10 NaI(Tl) scintillation detectors. All detectors were coupled to Ludlum Model 2221 ratemeter scalers. In substructures where no elevated gamma radiation anomalies were identified—the Control Complex and Turbine Building substructures—at least one representative location was selected for measurement.

The alpha, beta, and gamma radiation direct measurement gross counts were evaluated, and the count magnitudes and relative ratios of the three measurement type results were used to select locations for volumetric sampling.

Six of the direct measurement locations were selected for intrusive sampling. The volumetric concrete samples were collected from two concrete depth intervals at each location: initially from the 0 to 0.5 inch depth interval and then the 0.5 to 1.0 inch interval. ORISE personnel selected each location, and CR3 provided the equipment and personnel to operate a hammer drill that pulverized the concrete for collection while ORISE monitored the sampling. Table 4.2 lists the SU together with the number of direct measurements and volumetric concrete samples collected. Figures A.1 through A.6 show measurement and sampling locations.

Table 4.2. Confirmatory Direct Measurements and Volumetric Samples Collected

Building Substructure	SU ID	Direct Measurement (DM) Locations	Volumetric Sample ID	DM Location Sampled
Auxiliary Bldg. Wall	CRAB-01	DM8	--	--
Auxiliary Bldg. Floor	CRAB-02	DM3 – DM7, DM12	M0001 M0002	DM3
			M0007 M0008	DM12
Auxiliary Bldg/Reactor Wall	CRAB-06	DM9, DM10	--	--
Control Complex Bldg. Floor	CRCC-02	DM14	M0011 M0012	DM14
Intermediate Bldg. Floor	CRIB-01	DM1, DM2	M0003 M0004	DM1
Intermediate Bldg. and Intermediate/Reactor Bldg. Wall	CRIB-02	DM11	M0005 M0006	DM11
Turbine Bldg. Floor	CRTB-02	DM13	M0009 M0010	DM13

5. SAMPLE ANALYSIS AND DATA INTERPRETATION

Data collected on-site were transferred to the ORISE facility for analysis and interpretation. Sample custody was transferred under chain of custody to the NRC-directed laboratory, Southwest Research Institute. Sample analyses were performed in accordance with the laboratory's applicable procedures. Volumetric concrete samples were analyzed by gamma spectroscopy for the gamma emitting ROCs; then digested or otherwise prepared for analysis of HTDs via gas proportional counting for Sr-89/90; alpha spectroscopy for Am-241, Pu-238, Pu-239/240; and liquid scintillation spectroscopy for C-14, H-3, Fe-55, Ni-63, and Tc-99. Volumetric sample results are reported in units of picocuries per gram (pCi/g).

Direct alpha and beta surface activity measurement gross counts per minute (cpm) values—i.e., construction material specific or ambient backgrounds were not subtracted—were converted into units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2) by applying appropriate detector efficiencies and geometries.

The gamma radiation scan data files were downloaded from the GPS units for assessment. The data consisted of the count rates (cpm equivalent) and associated positioning coordinates that were generated each second during walkovers. Assessments were performed using ArcGIS and ProUCL software for generating descriptive statistics. The data were then binned for illustration based on statistical evaluation. The mean for all data combined with plus or minus multiples of the population standard deviation formed the binning basis. The data were plotted using ArcGIS. Scan results are presented as gross cpm.

6. FINDINGS AND RESULTS

The results of the CS activities ORISE conducted while at the CR3 site from January 5 through 8, are discussed in the following subsections.

6.1 SURFACE SCANS

Appendix A, Figures A.7 through A.10, provide illustrations of the gamma radiation scan results for the Auxiliary Building and Turbine Building floor survey units, survey units CRAB-02 and CRTB-02, where geo-referenced data were able to be collected. For the remaining survey units, excluding the Auxiliary Building sumps and Turbine Building Demin sump, the count rate data were collected without the geo-referencing due to the absence of adequate satellite signal in those areas. Surface scanning logistics in the sumps required the surveyor to wear a harness to climb into and out of the sumps and to utilize a detector mounted to a pole extender, making the use of the GPS as a data logger too cumbersome.

Figures A.11 through A.15 provide the scan data for all survey units investigated during the CS—excluding the sumps—in graphic format via Q-Q plots. The Q-Q plot is a graphical tool for assessing the distribution of a dataset. The Y-axis represents scan data in units of cpm, while the X-axis represents the data quantiles about the median value. Values less than the median are represented in the negative quantiles; the values greater than the median are represented in the positive quantiles. A normal distribution that is not skewed by outliers (i.e., a background population) will appear as a straight line, with the slope of the line subject to the degree of variability among the data population. More than one distribution, such as background plus contamination or other outliers, will appear as step functions.

Table 6.1 provides the gamma scan statistical summary information for each survey unit.

Table 6.1. Summary Scan Data Statistics

Scanning Electronic Data Capture File Name[a]/Survey Unit		Approximate Number of Data Points	General Statistics (cpm)				
			Minimum	Maximum	Mean	Standard Deviation	Median
AUXBLDG_FLOOR	CRAB-02	5,000	5,100	52,000	14,000	8,000	12,000
AUXBLDG_LW	CRAB-01	960	4,400	24,000	8,400	4,100	6,900
AUXBLDG_WALLS	CRAB-01	680	4,800	19,000	8,400	2,000	7,900
CONTCOMP	CRCC-02	3,200	5,200	11,000	8,300	1,000	8,500
CONTCOMP_LW	CRCC-01	230	4,700	7,600	6,000	570	6,000
INTBLDG_FL	CRIB-01	1,500	4,900	36,000	10,000	5,300	8,500
INTBLDG_LW	CRIB-02	1,000	4,300	37,000	11,000	5,800	9,500
IRB_WALLS	CRIB-02	1,400	3,600	23,000	8,000	2,000	7,900
RCTRBLDG_WALLS	CRAB-06	2,000	6,300	30,000	13,000	4,100	11724
TURBBLDG_FL	CRTB-01	8,500	4,100	12,000	6,900	1,300	6,700
TURBBLDG_LW	CRTB-02 and -06	1,400	3,300	8,300	5,600	880	5,500

[a]Electronic data capture file name corresponds with building (AUXBLDG = Auxiliary Building, CONTCOMP = Control Complex Building, INTBLDG = Intermediate Building, IRB = Intermediate/Reactor, RCTRBLDG = Reactor Building, and TURBBLDG = Turbine Building) and surface (floor or FL and wall or lower wall = LW) name and designation provided in captions of Figures A.5 through A.9

Elevated direct gamma radiation was identified and is evident based on the maximum scan data values in Table 6.1 and the Figure A.11, A.13, and A.15 Q-Q plot step functions. The elevated gamma anomalies were associated with the following:

- Auxiliary Building floor
- Auxiliary Building East wall section near a previously remediated area referred to as the "Window"
- Auxiliary Building/Reactor Building shared wall
- Intermediate Building floor
- Intermediate Building/Reactor Building shared wall

The gamma scans of the Control Complex and Turbine Buildings did not identify anomalous gamma radiation levels.

6.2 TOTAL SURFACE ACTIVITY LEVELS

Total surface activity levels for the 14 direct measurement locations are provided in Table B.1. The alpha activity ranges for all substructures provided evidence that there was limited, if any residual alpha emitter contamination. The direct measurement beta activity results indicated that the elevated gamma radiation anomalies identified during surface scans in the Auxiliary and Intermediate Buildings were the result of residual beta-gamma emitters. Activity ranges were as follows:

- Auxiliary and Intermediate Buildings, including the shared Reactor Building wall,

  Alpha – 0 to 30 dpm/100 cm^2

  Beta – 4,600 to 29,000 dpm/100 cm^2
- Turbine and Control Complex Buildings,

  Alpha – 10 to 50 dpm/100 cm^2

  Beta – 1,700 to 1,800 dpm/100 cm^2

All beta direct measurement results were at least an order of magnitude less than the 320,000 dpm/100 cm^2 gross activity DCGL for Cs-137. The post-sample direct measurement results that are provided in Table B.1 indicate that residual beta activity levels were reduced, and hence residual ROC activity within the concrete volume of the Auxiliary and Intermediate Buildings was collected into the samples. Post-sample measurements made at the Turbine and Control Complex Buildings sampling locations remained consistent with the pre-sample measurement indicating that that surface activity measured was consistent with the concrete construction material background rather than residual contamination.

6.3 RADIONUCLIDE CONCENTRATIONS IN VOLUMETRIC SAMPLES

Radionuclide concentrations in the volumetric concrete samples are provided individually in Table B.2. Table 6.2 lists the summary concentration ranges for all ROCs with DCGLs that are listed in Table 3.2 and the SOF range.

Table 6.2. Summary of Volumetric Sample Data

Radionuclide	Radionuclide Concentration Ranges (pCi/g) SOF (unitless)	DCGL (pCi/g) SOF (unitless)
H-3	1.05 to 5.59	423
C-14	-0.19 to 1.15	16.4
Fe-55	-0.36 to 5.21	33,600
Co-60	-0.10 to 6.65	3.4
Ni-59	170 to 285	2,685
Ni-63	2 to 103	980
Sr-90	-0.04 to 0.40	3.6
Tc-99	-0.36 to 0.94	61.8
Cs-137	0.01 to 569	9.6
Eu-152	-0.92 to 0.69	7
Eu-154	-0.10 to 0.59	7.2
Pu-239	0.00 to 0.04	59
Am-241	0.00 to 0.05	87.4
SOF	0.1 to 61.6	1

Two ROCs were detected in samples that exceeded their associated DCGLs, one sample with Co-60 and eight samples with Cs-137. Cs-137 was the primary ROC identified in samples where eight of the twelve samples—M0001 through M0008—exceeded the DCGL of 9.6, ranging from 32.6 to 569 pCi/g. These samples were collected from the floors and walls of both the Auxiliary Building and the Intermediate Building. The 6.65 pCi/g concentration of Co-60 in sample M0001, collected from the floor of the Auxiliary Building, exceeded the 3.4 pCi/g DCGL. This sample also contained the highest Cs-137 concentration. Most ROC concentrations were less than the analytical detection limit in samples collected from the Turbine and Control Complex Buildings—samples M0009 through M0012. The ROCs detected in these samples were Cs-137, and Eu-152 and -154 at concentrations below 10% or less of the respective DCGLs. Ni-59 concentrations were seemingly elevated, and in some cases slightly greater than 10% of the DCGL. However, the consistent activity observed in all samples indicated the activity present was naturally occurring due to cosmological origin within the materials used for the concrete. Ni-63 concentrations were also elevated with the maximum observed activity, sample M0008, at slightly over 10% of the DCGL.

6.4 Evaluation of PSQs

The surface scan, total and removable surface activity levels and radionuclide concentrations in volumetric samples provide the information necessary to answer the two verification PSQs.

6.4.1 PSQ1: Residual Contamination Levels

The CS design was based on judgmentally selecting locations where residual contamination was suspected to be present and then collecting the necessary data for assessing whether any residual contamination present satisfied the DCGLs. The CS was not designed to assess whether the mean residual activity concentrations within a survey unit might satisfy the DCGL, nor was the CS intended to include elevated measurement comparisons with hot spot limits, only whether observed residual surface activity and/or volumetric ROC concentrations were less than or greater than the Table 3.2 or Table 3.3 DCGLs, including SOFs for volumetric samples.

The CS gamma radiation surface scans were performed to identify locations where residual contamination was suspected on the surfaces of or within the concrete volume of the subsurface structures. The surface activity measurement made at locations identified by these scans and the volumetric sample analytical results provided the quantitative data necessary to address whether these independent results supported the site's conclusion that gross surface activity levels and/or volumetric ROC concentrations were less than the 25 mrem/yr free release criteria. All CS gross beta activity surface activity measurements were less than the gross activity DCGL, which corresponded with the Cs-137 surface activity DCGL for BOP subsurface structures. However, volumetric sample results did not support the condition that if the surface activity DCGL was met, then the volumetric DCGLs would be met. This determination is based on all eight judgmental samples collected from the Auxiliary and Intermediate Buildings having concentrations of Cs-137 above the corresponding volumetric DCGL.

6.4.2 PSQ2: ROC Mixture and Fractional Abundances

The CS volumetric sample radionuclide-specific laboratory analytical results were needed to independently assess whether identified radiological anomalies were due to site-related contamination and if so, whether the CR3 established ROC mixtures and associated fractional relationships used to calculate total activity results via either gross activity or surrogate measurement methods were appropriate. In the case of the BOP subsurface structures, the assumption was that Cs-137 was the dominant ROC, and application of the Cs-137 surface activity DCGL was adequate for demonstrating the 25 mrem/yr license termination dose limit rule.

The independent CS on-site investigations did identify radiological anomalies in both the Auxiliary and Intermediate Buildings. No anomalies were identified in the areas surveyed in the Control Complex and Turbine Building survey units. The radionuclide-specific laboratory analyses generally agreed with the licensee's determination that any residual contamination within the BOP substructures was dominated by Cs-137, although Co-60, Ni-59, and Ni-63 concentrations in some samples exceeded 10% of the respective ROC DCGLs. Additionally, the CS results indicate that surface activity measurements below the gross activity DCGL were non-conservative as a reliable indicator that the volumetric DCGLs would also be satisfied.

6.4.3 PSQ3: MARSSIM Classification

The CS surface activity measurement results and volumetric sample results also provided the data that allowed independent assessment of SU classifications. CR3 classified the BOP Auxiliary, Intermediate, and Control Complex Building substructures as Class 1. The CS results showed that Class 1 was particularly appropriate for the Auxiliary and Intermediate Buildings and was a potentially conservative classification for the Control Complex. The Turbine Building floors, walls, and Demin sump were classified as Class 2, and CS independent measurements supported the classification as all measurements were below 1% of gross activity DCGL.

7. SUMMARY AND CONCLUSIONS

The ORISE survey team performed independent confirmatory visual inspections, gamma radiation surface scans, gross alpha and beta direct measurements, and, with CR3 support, collected volumetric concrete samples for laboratory analyses. ORISE performed these confirmatory survey activities from January 5 through 8, 2026.

Gamma radiation scans were completed over most floor surface areas, and judgmental scans of select sumps and lower wall surfaces were performed within survey units of the Auxiliary Building, Intermediate Building, Control Complex Building, and the Turbine Building. Locations where anomalous gamma radiation levels were identified were marked for potential follow-up gross alpha and beta radiation measurements. In buildings where no distinct anomalies were identified, areas with the maximum observed gamma levels were noted. Gamma anomaly or maximally observed locations were judgmentally selected for static alpha, beta, and gamma direct measurements, and from these measurement locations, six were selected for intrusive, volumetric sampling and the samples submitted for laboratory analysis for the ROCs.

The gross alpha measurements were reviewed to assess whether there were suspect elevated alpha radiation levels. All data were comparable to expected alpha background counts for concrete surfaces. The gross beta surface activity levels were compared with the gross activity DCGL of 320,000 dpm/100 cm^2, which was based on the Cs-137 being the main ROC of concern. The maximum measurement was 29,000 dpm/100 cm^2.

Volumetric sample results showed concentrations of Cs-137 within the concrete matrix of the Auxiliary and Intermediate Building exceeded the DCGL. In each of these cases, the surface activity measurement did not correspond with an exceedance. Results did support that Cs-137 was the primary ROC, supporting the CR3 assumed condition with the BOP substructures.

The evaluation of the combined results indicated that CR3 had appropriately classified the survey units in accordance with MARSSIM guidance.

8. REFERENCES

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APPENDIX A: FIGURES

5390 - Crystal River
Picture log of DM locations

DM #1 Intermediate Bldg, sample ID 5390M0003 and M0004

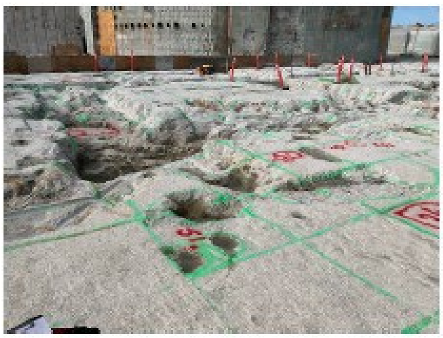
DM #2 Intermediate Bldg

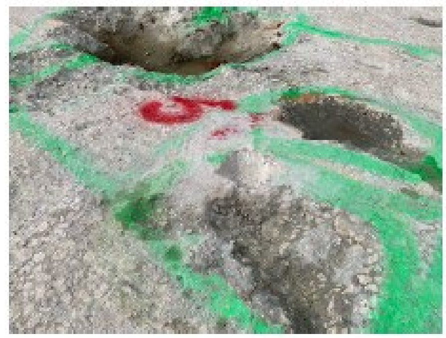
DM #1 and #2 Intermediate Bldg

Figure A.1. Intermediate Building Direct Measurement Locations and Volumetric Sample Locations

DM #3 Auxiliary Bldg
(At the 51 marked location)

DM #3 Auxiliary Bldg
(At the 51 marked location), sample IDs 5390M0001 and M0002





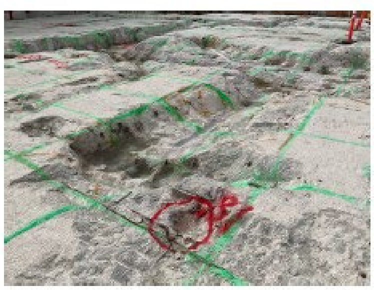
DM #4 Auxiliary Bldg
(At the 38 marked location)

DM #4 Auxiliary Bldg
(At the 38 marked location)

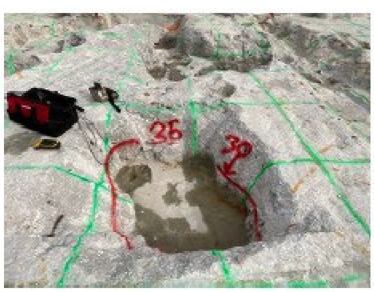
Figure A.2. Auxiliary Building Direct Measurement and Volumetric Sample Locations

DM #5 Auxiliary Bldg
(At the 41 marked location)

DM #5 Auxiliary Bldg
(At the 41 marked location)



DM #6 Auxiliary Bldg
(At the 31 marked location)



DM #7 Auxiliary Bldg
(At the 35 marked location)

Figure A.3. Auxiliary Building Direct Measurement Locations

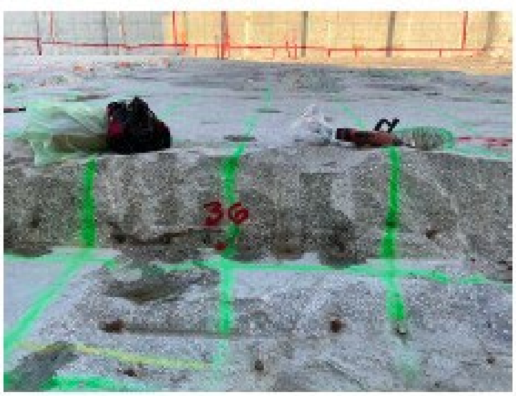
DM #8 Auxiliary Bldg
(Lower wall location marked by white box in photo)

DM #9 and #10 Reactor/Auxiliary Bldg wall
(Locations marked in photo)

Figure A.4. Auxiliary Building/Reactor Wall Direct Measurement Locations

DM #11 Intermediate/Reactor Bldg wall, sample IDs M0005 and M0006
(Location marked in photo with 35)

DM #12 Auxiliary Building floor, sample IDs M0007 and M0008
(Location marked in photo with 36)



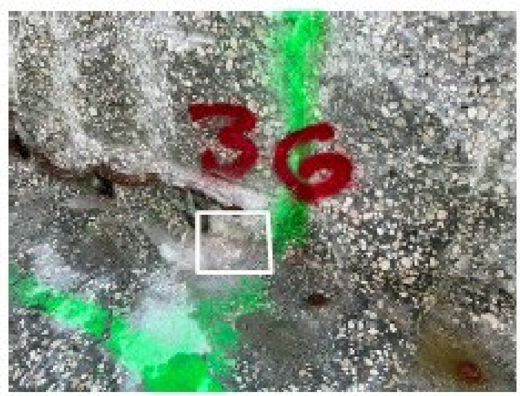
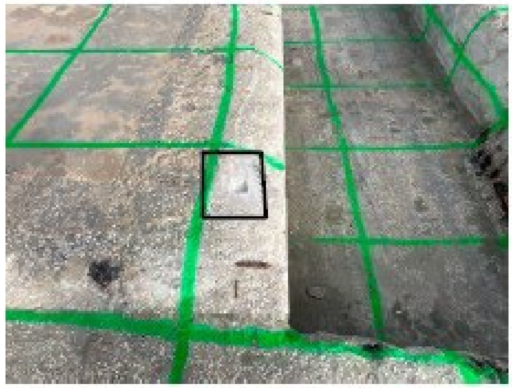


Figure A.5. Intermediate/Reactor Building Wall, Auxiliary Building Direct Measurement and Volumetric Sample Locations

DM #13 Turbine Building floor, sample IDs M0009 and M0010
(Location marked by box in photos)



DM #14 Control Complex Building floor, sample IDs M0011 and M0012
(Location marked by box in photos)

Figure A.6. Turbine Building and Control Complex Buildings Direct Measurement and Volumetric Sample Locations

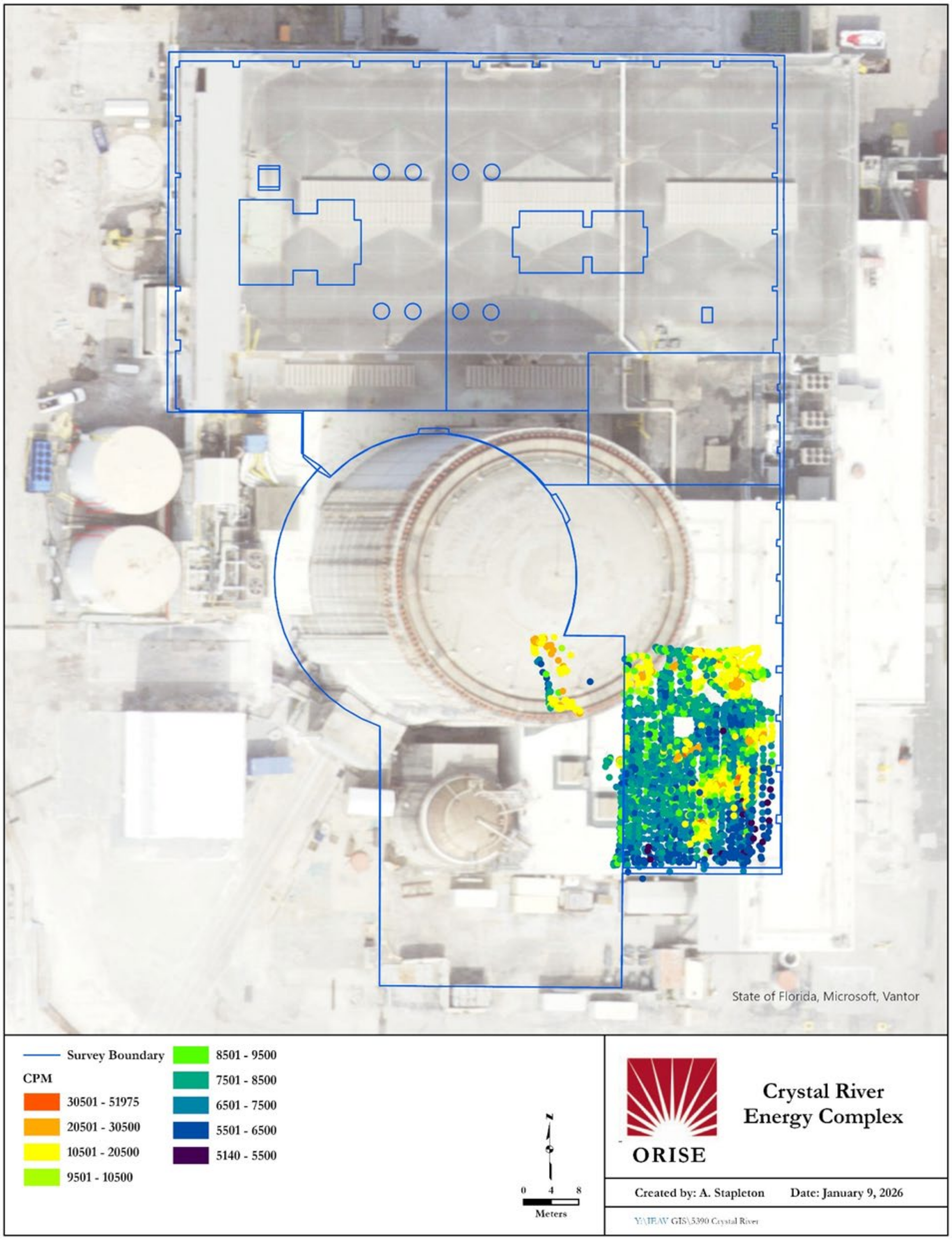


Figure A.7. Auxiliary Building South Floor Gamma Scans

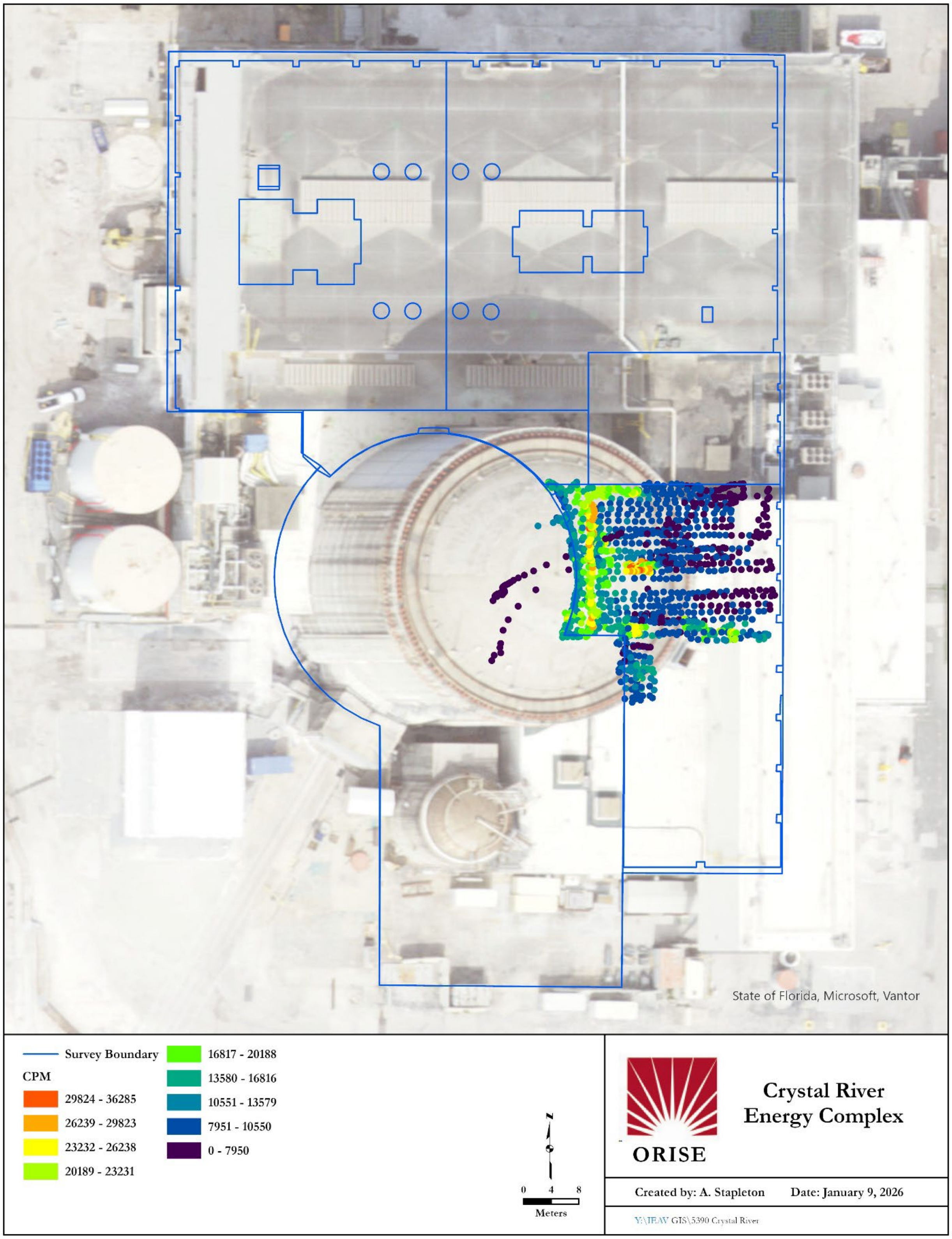


Figure A.8. Auxiliary Building North Floor Gamma Scans

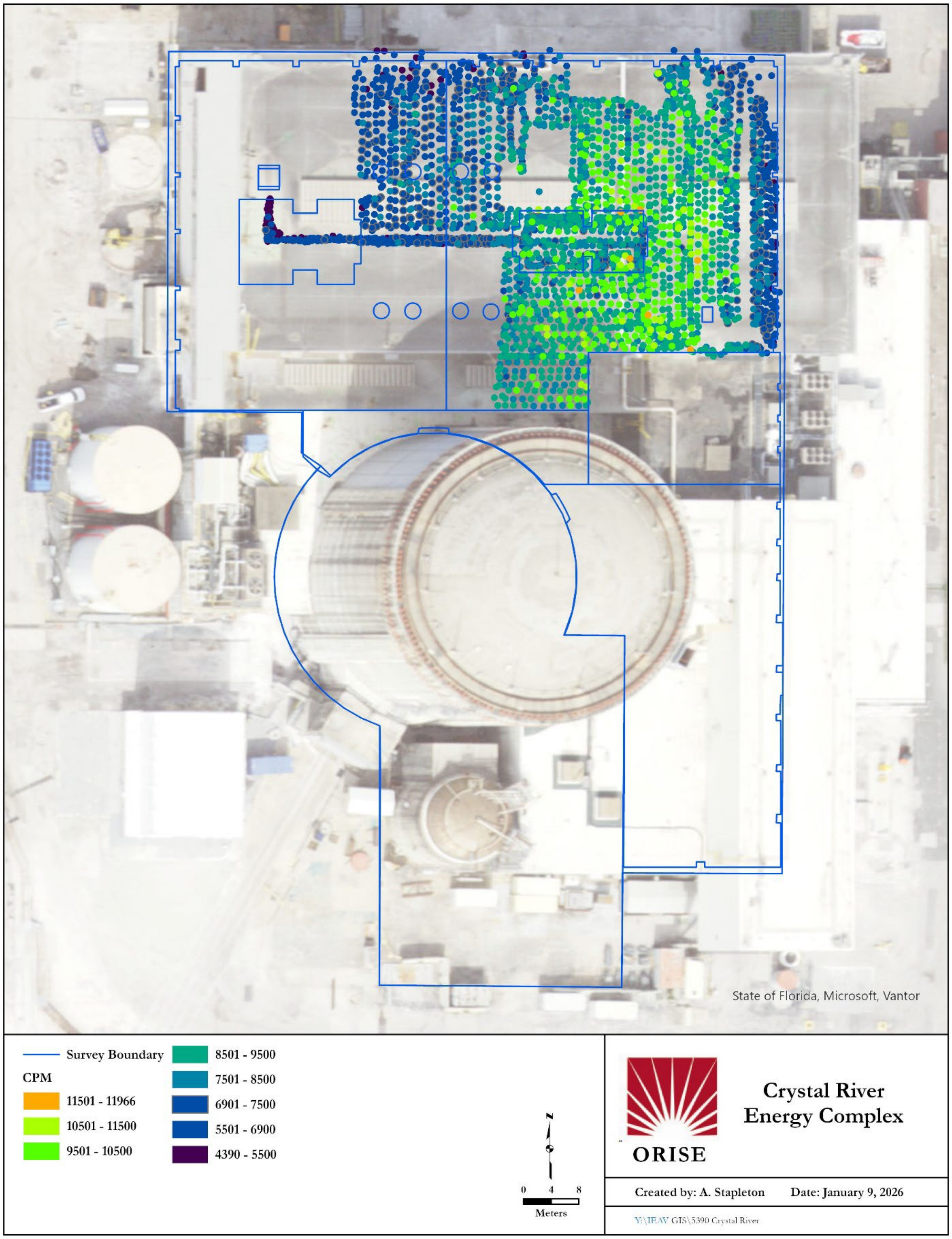


Figure A.9. Turbine Building East Gamma Scans

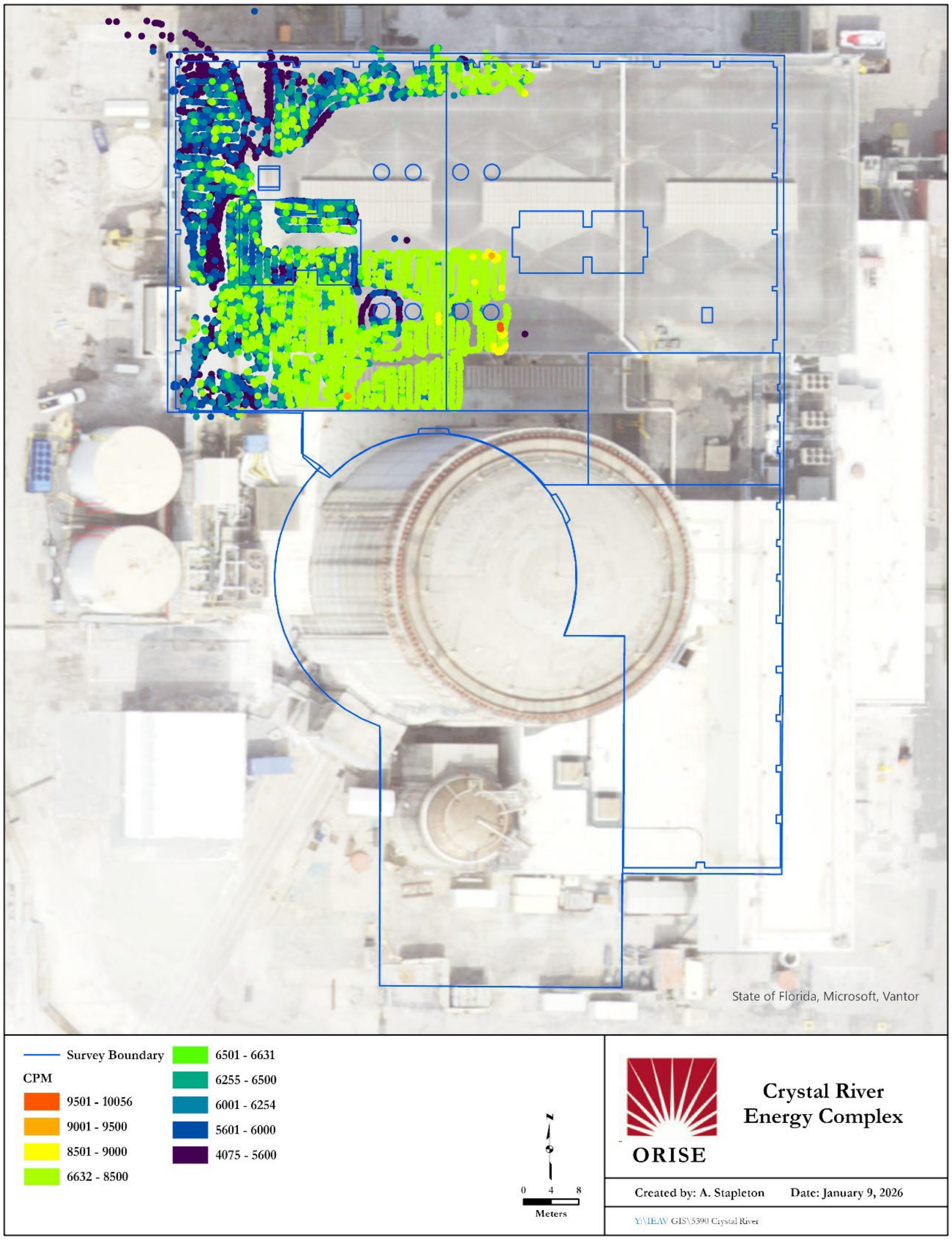


Figure A.10. Turbine Building West Gamma Scans

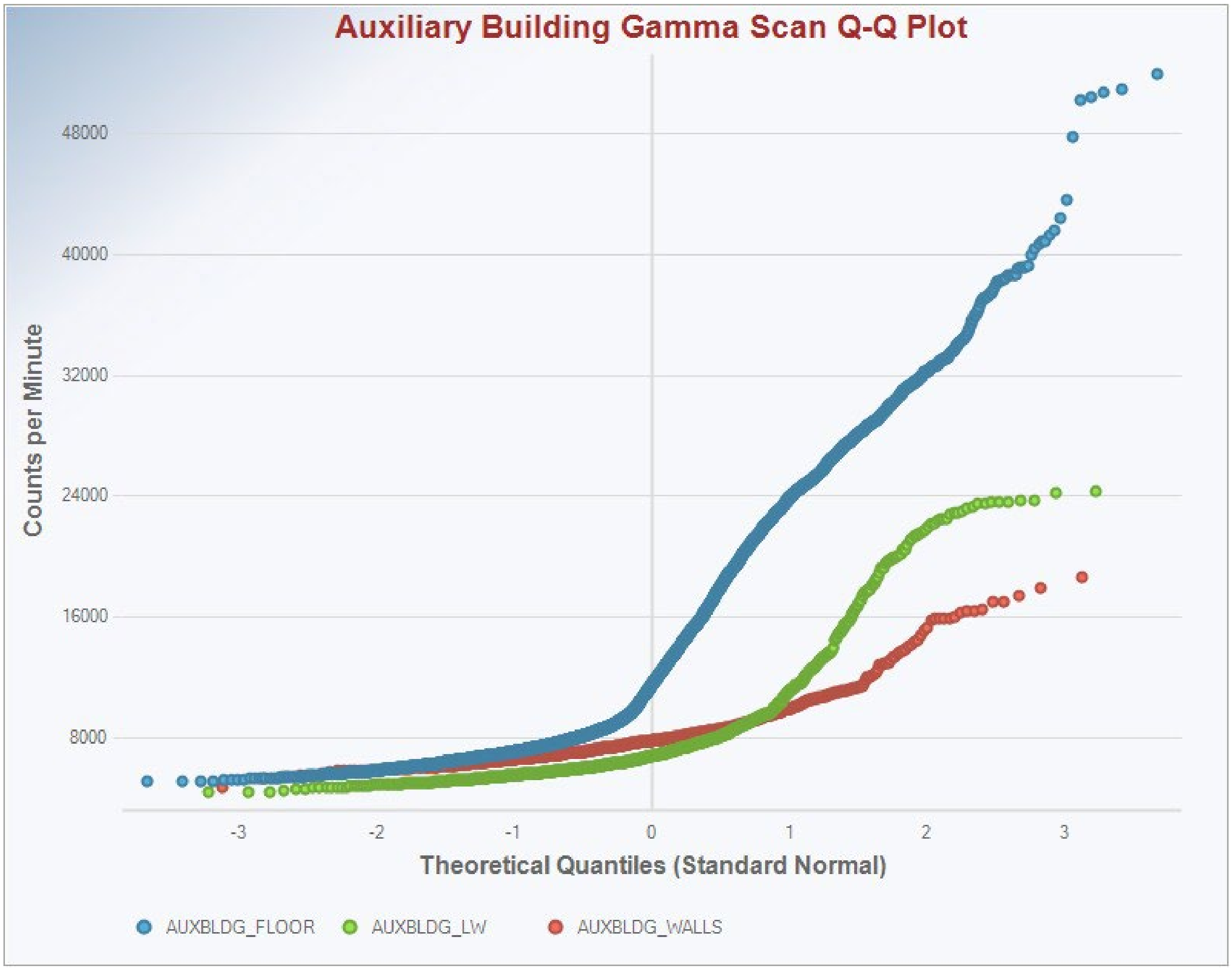


Figure A.11. Auxiliary Building Floor (AUXBLDG_FLOOR) and Walls (AUXBLDG_LW and _WALLS) Gamma Scan Q-Q Plots

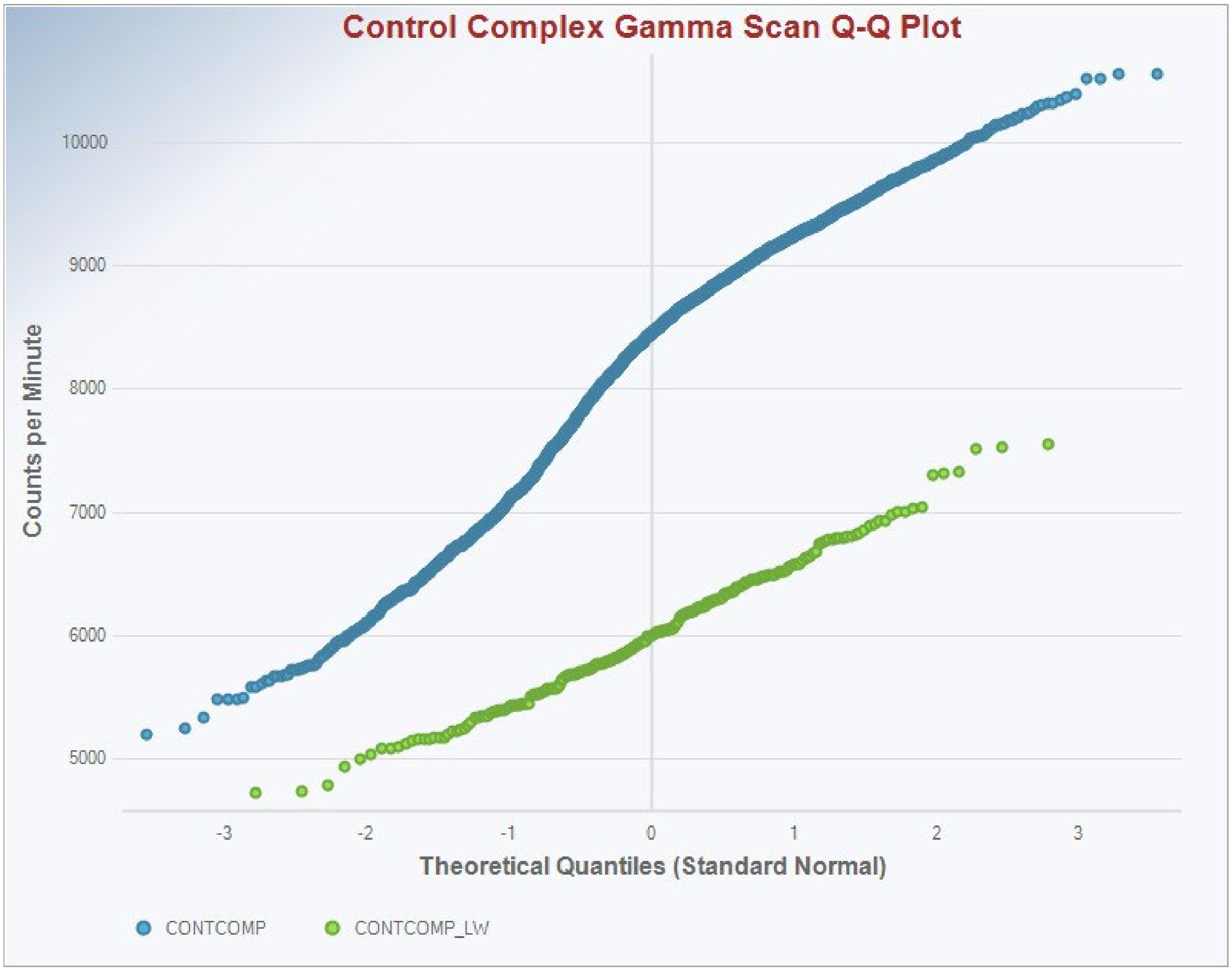


Figure A.12. Control Building Floor (CONTCOMP) and Walls (CONTCOMP_LW) Gamma Scan Q-Q Plots

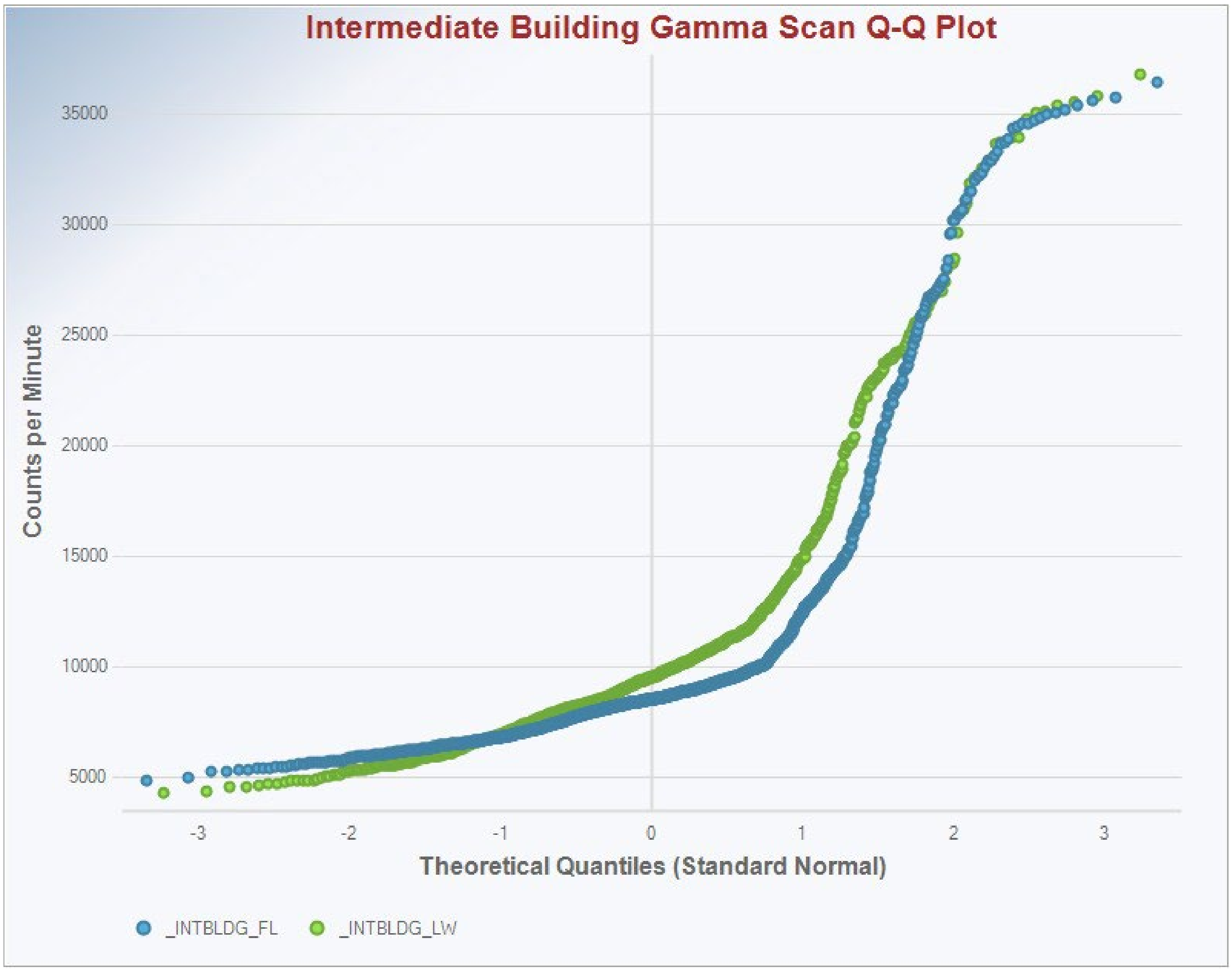


Figure A.13. Intermediate Building Floor (_INTBLDG_FL) and Walls (_INTBLDG_LW) Gamma Scan Q-Q Plots

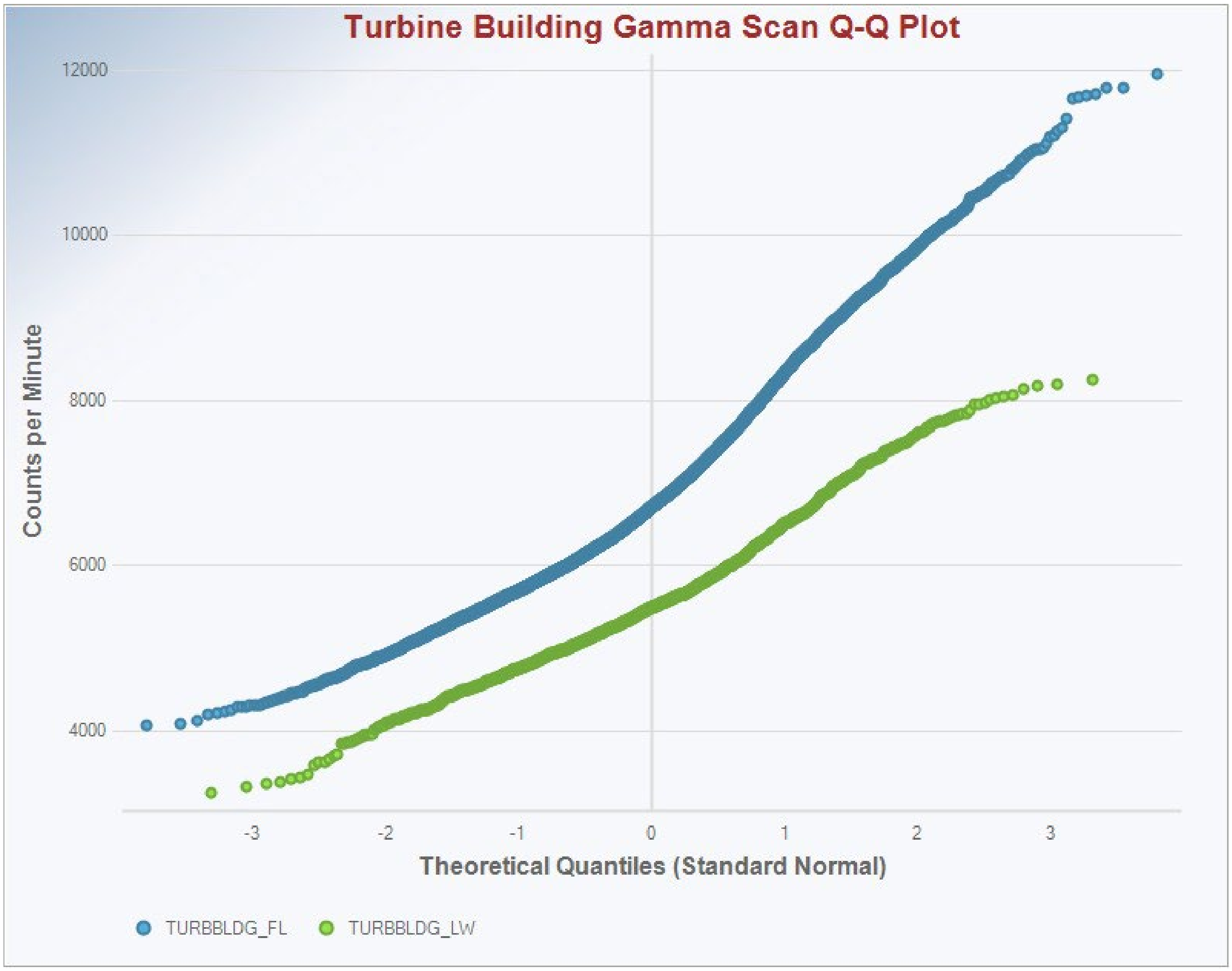


Figure A.14. Turbine Building Floor (TURBBLDG_FL) and Wall (TURBBLDG_LW) Gamma Scan Q-Q Plots

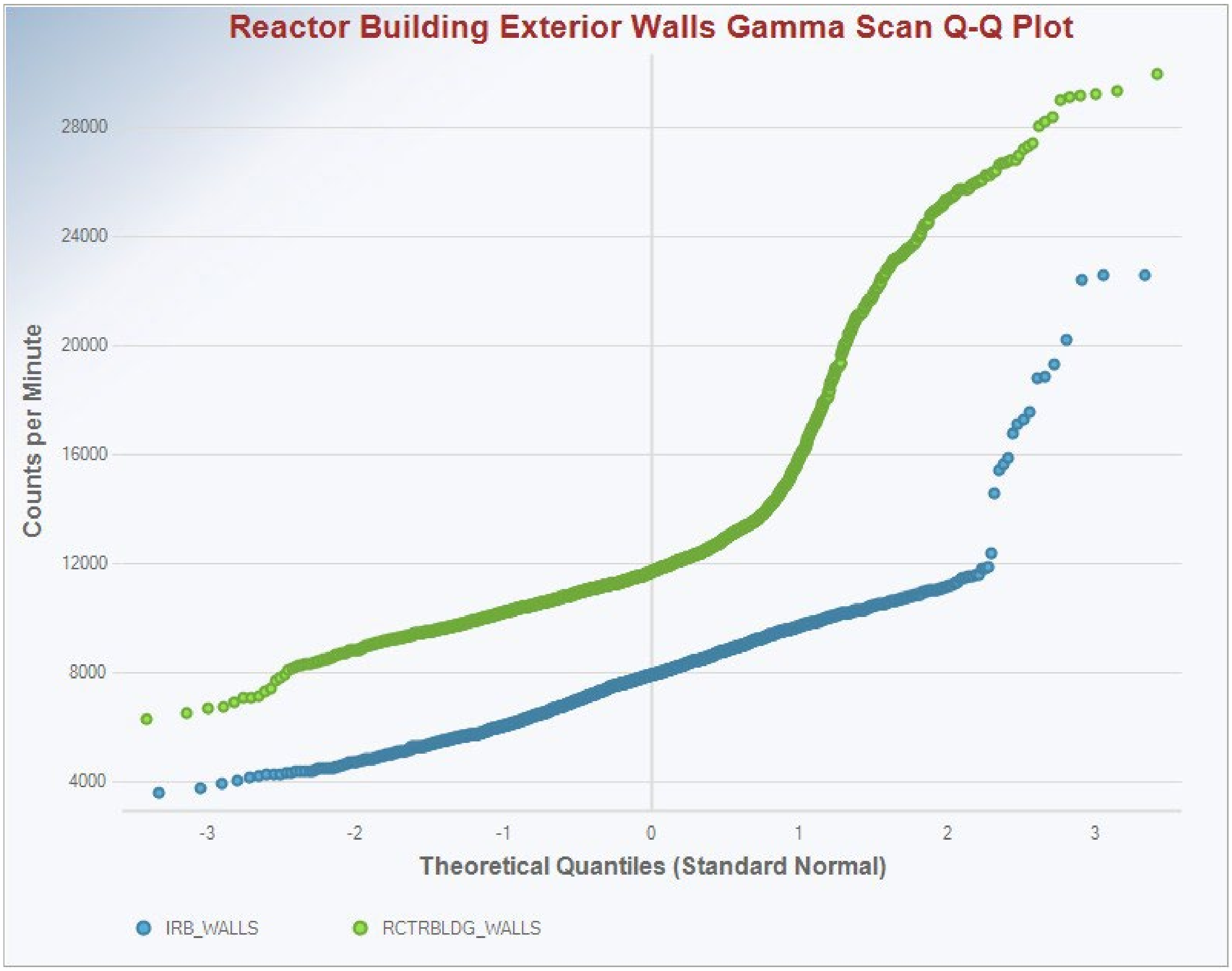


Figure A.15. Reactor/Intermediate Building Wall (IRB_WALLS) and Reactor/Auxiliary Building Wall (RCTRBLDG_WALLS) Gamma Scan Q-Q Plots

APPENDIX B: DATA TABLES

Table B.1. Alpha and Beta Surface Activity Levels			
Direct Measurement ID[a]	Surface	Surface Activity dpm/100 cm^2	
		Alpha	Beta
Intermediate Building			
DM1	Floor	10	13,000
DM1 post sample	Floor	10	5,900
DM2	Floor	10	4,600
Auxiliary Building			
DM3	Floor	10	20,000
DM3 post sample	Floor	10	11,000
DM4	Floor	20	29,000
DM5	Floor	0	15,000
DM6	Floor	20	7,800
DM7	Floor	20	7,700
DM8	E. Lower Wall	30	7,300
Auxiliary/Rx Bldg E. Wall			
DM9	E. Lower Wall	30	14,000
DM10	E. Lower Wall	10	17,000
Intermediate and Auxiliary Buildings			
DM11	Int. Rx Wall	20	22,000
DM11 Post Samples	0.5 to 1.0	10	11,000
DM12	Floor	10	6,600
DM12 Post Sample	0.5 to 1.0	30	5,100
Turbine and Control Complex Buildings			
DM13	Floor	50	1,700
DM13 Post Sample	Floor	20	1,700
DM14	Floor	30	1,700
DM14 Post Sample	Floor	10	1,800

[a]Refer to Figures A.1 through A.6

Table B.2. Volumetric Concrete Sample Concentrations (pCi/g)																					
Sample ID	H-3			C-14			Fe-55			Co-60			Ni-59			Ni-63			Sr-90		
	Conc.	TPU[b]	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC
5390M0001	2.3	3.7	5.9	**1.15**	0.36	0.52	-0.2	4.3	3.7	**6.65**	0.51	0.18	**194**	93	70	**98**	13	2	0.06	0.17	0.32
5390M0002	**5.6**	3.7	5.5	0.43	0.33	0.53	1.5	5.2	4.4	**0.34**	0.15	0.26	**172**	57	52	**38.6**	6.8	1.9	-0.04	0.12	0.27
5390M0003	3.2	3.4	5.3	0.21	0.32	0.52	1.0	4.7	4.0	**0.88**	0.17	0.09	**170**	87	65	**103**	14	2	0.15	0.19	0.30
5390M0004	2.1	3.3	5.2	-0.19	0.30	0.52	-0.4	5.0	4.3	**0.33**	0.11	0.08	**207**	50	54	**63.4**	9.6	1.8	0.22	0.21	0.30
5390M0005	3.5	3.4	5.3	-0.10	0.31	0.53	**5.2**	5.3	4.3	-0.02	0.13	0.16	**208**	65	61	**5.9**	3.3	1.7	0.40	0.32	0.44
5390M0006	4.3	3.6	5.6	-0.08	0.31	0.53	1.7	5.7	4.8	-0.04	0.12	0.12	**178**	59	53	**4.2**	3.4	1.8	0.01	0.13	0.26
5390M0007	3.3	3.8	6.0	0.23	0.32	0.52	1.5	5.5	4.8	**1.07**	0.20	0.07	**208**	83	68	**78**	12	2	0.13	0.18	0.29
5390M0008	1.4	3.2	5.1	-0.04	0.30	0.51	2.6	4.5	3.6	**1.71**	0.24	0.11	**285**	59	70	**101**	14	2	0.09	0.14	0.24
5390M0009	2.3	3.4	5.3	0.05	0.30	0.51	1.7	6.4	5.9	-0.10	0.16	0.14	**220**	41	53	**3.5**	3.1	1.8	0.17	0.22	0.35
5390M0010	4.3	3.8	5.8	-0.08	0.29	0.49	1.3	4.7	4.0	0.046	0.090	0.137	**208**	72	65	**2.5**	3.3	1.9	0.03	0.13	0.24
5390M0011	1.1	3.3	5.3	-0.13	0.30	0.52	1.4	4.5	3.8	0.06	0.12	0.12	**219**	51	56	**2.6**	3.2	1.8	0.06	0.16	0.28
5390M0012	4.3	3.6	5.6	-0.01	0.29	0.50	2.2	5.4	4.8	0.03	0.09	0.14	**230**	49	57	**2.4**	3.0	1.7	0.13	0.19	0.31

Table B.2. Volumetric Concrete Sample Concentrations (pCi/g) (continued)																			
Sample ID	Tc-99			Cs-137			Eu-152			Eu-154			Pu-239			Am-241			SOF[a]
	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC	
5390M0001	0.79	0.55	0.87	**569**	19	1	0.03	0.69	0.84	0.4	1.4	0.5	0.035	0.040	0.068	0.009	0.029	0.080	61.6
5390M0002	0.21	0.52	0.86	**164.0**	5.7	0.3	-0.9	1.0	0.6	**0.59**	0.49	0.41	0.028	0.038	0.072	0.041	0.042	0.064	17.3
5390M0003	**0.94**	0.56	0.88	**314.0**	10.7	0.4	0.24	0.49	0.78	-0.10	0.89	0.39	0.005	0.021	0.066	0.054	0.049	0.068	33.2
5390M0004	-0.36	0.52	0.88	**46.4**	1.8	0.2	-0.15	0.44	0.35	0.14	0.59	0.25	0.010	0.028	0.076	0.019	0.032	0.071	5.1
5390M0005	-0.14	0.54	0.90	**353**	12	0	0.15	0.49	0.74	0.28	0.71	0.45	0.010	0.027	0.073	0.026	0.035	0.067	37.0
5390M0006	0.27	0.53	0.88	**32.8**	1.3	0.2	0.02	0.58	0.33	**0.40**	0.54	0.23	0.009	0.027	0.072	0.035	0.041	0.076	3.5
5390M0007	0.21	0.52	0.86	**47.6**	1.8	0.2	0.01	0.60	0.37	**0.43**	0.69	0.26	0.026	0.035	0.067	0.014	0.027	0.063	5.6
5390M0008	0.20	0.53	0.88	**32.6**	1.3	0.2	**0.69**	0.50	0.33	0.07	0.87	0.23	0.021	0.036	0.078	0.000	0.025	0.090	4.2
5390M0009	0.38	0.53	0.87	**0.126**	0.075	0.107	-0.28	0.62	0.16	**0.27**	0.64	0.12	0.020	0.034	0.075	0.000	0.027	0.073	0.1
5390M0010	0.20	0.53	0.88	0.01	0.11	0.14	**0.64**	0.32	0.16	-0.03	0.68	0.12	0.024	0.033	0.062	0.012	0.025	0.057	0.2
5390M0011	0.05	0.52	0.86	**0.233**	0.087	0.109	0.08	0.51	0.16	**0.18**	0.65	0.11	0.014	0.028	0.066	0.026	0.037	0.076	0.2
5390M0012	0.39	0.53	0.86	**0.095**	0.055	0.073	**0.34**	0.23	0.15	**0.51**	0.46	0.11	0.035	0.040	0.067	0.032	0.037	0.062	0.3

[a]SOF calculations include concentrations

[b]Uncertainties presented are total propagated uncertainties (TPU) at the 95% confidence level.

APPENDIX C: MAJOR INSTRUMENTATION

C.1. SCANNING AND MEASUREMENT INSTRUMENT/ DETECTOR COMBINATIONS

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

C.1.1 GAMMA

Ludlum NaI[Tl] Scintillation Detector Model 44-10, Crystal: 5.1 cm × 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Trimble Geo 7X
(Trimble Navigation Limited, Sunnyvale, CA)
or
Coupled to: Trimble Nomad
(Trimble Navigation Limited, Sunnyvale, CA)

C.1.2 ALPHA

Ludlum Scintillation Detector Model 43-92, Physical Area 100 cm^2
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)
43-92

C.1.3 BETA

Ludlum Scintillation Detector Model 44-142, Physical Area 100 cm^2
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

APPENDIX D: SURVEY AND ANALYTICAL PROCEDURES

D.1. PROJECT HEALTH AND SAFETY

The Oak Ridge Institute for Science and Education (ORISE) performed all survey activities in accordance with the *Oak Ridge Associated Universities (ORAU) Radiation Protection Manual*, the *ORAU Radiological and Environmental Survey Procedures Manual,* and the *ORAU Health and Safety Manual* (ORAU 2020b, ORAU 2016, and ORAU 2020a). 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 walk down 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 ORAU 2016 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.

D.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 2016)
- ORAU Environmental Services and Radiation Training Quality Program Manual (ORAU 2024)

The procedures contained in these manuals were developed to meet the requirements of U.S. Department of Energy (DOE) Order 414.1D and U.S. Nuclear Regulatory Commission's (NRC's) *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.

- Training and certification of all individuals performing procedures.
- Periodic assessments.

D.3. SURVEY PROCEDURES

D.3.1 SURFACE SCANS

Gamma scans were performed using Ludlum model 44-10 2-inch by 2-inch thallium-doped sodium iodide (NaI[Tl]) detectors. Scans to identify elevated radiation levels were performed by passing the detector slowly over the ground surface as the surveyor walked at a pace of approximately 1 meter per second. The distance between the detector and surface was maintained at a minimum to maximize measurement potential, and surveyors paused to investigate when the detector's audible response was judged to be above the local ambient background response. The NaI(Tl) gamma detectors were used solely as a qualitative means to identify elevated radiation levels in excess of background. However, for reference, NUREG-1507, Table 6-6, provides NaI scintillation detector scan minimum detectable concentrations (MDCs) for Common Radiological Contaminants (NRC 2020). For Cs-137, the scan MDC[1] is 5.5 pCi/g. For Th-230 and Ra-226, the scan MDCs are on the order of 2,000 and 3 picocuries per gram (pCi).

D.3.2 VOLUMETRIC SAMPLES

Concrete samples were collected by CR3 personnel under ORISE personnel observation using a rotary hammer drill. Concrete fines were collected into a clean container. Each container was then labeled and security sealed in accordance with ORISE procedures. ORISE shipped samples under chain-of-custody to the Southwest Research Institute for analysis.

D.4. RADIOLOGICAL ANALYSIS

The soil samples were transferred under chain-of-custody to a contracted laboratory for analysis in accordance with the laboratory's applicable procedures or other special instructions from NRC/ORISE. The analytical suite for all volumetric samples included gamma spectroscopy for the gamma emitting ROCs; then digested or otherwise prepared for analysis of HTDs via gas proportional counting for Sr-89/90; alpha spectroscopy for Am-241, Pu-238, Pu-239/240; and liquid scintillation spectroscopy for C-14, H-3, Fe-55, Ni-63, and Tc-99. Details of the radiological

[1] Scan MDC values can be calculated independently assuming inputs such as an index of sensitivity of 1.96, observation interval of 1 second, survey efficiency of 0.75, and background count rate of 9,500 counts per minute (https://oriseapps.orau.gov/hpcalculator).

analysis performed are presented in the report narrative section of the SwRI analytical laboratory reports. Detection limits, referred to as MDCs, were based on a 95% confidence level. Due to variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument.

APPENDIX E:
ANALYTICAL DATA PACKAGE