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November 26, 2019

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U.S. Nuclear Regulatory Commission
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Division of Decommissioning, Uranium Recovery, and Waste Programs
Reactor Decommissioning Branch
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SUBJECT: DOE Contract No. DE-SC0014664
INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE GENERAL ATOMICS TRIGA REACTOR FACILITY BUILDING G21 AND ASSOCIATED LAND AREA, SAN DIEGO, CALIFORNIA
DOCKET NOs. 05000089 AND 05000163; RFTA NO. 19-005;
DCN 5339-SR-01-0

Dear Ms. Doell:

The Oak Ridge Institute for Science and Education (ORISE) is pleased to provide the enclosed final report, which describes the procedures and results of the confirmatory survey of the General Atomics TRIGA Reactor Facility and associated land area. The U.S. Nuclear Regulatory Commission's (NRC's) comments on the draft report have been addressed in the final version.

You may contact me at 865.576.6659 or Kaitlin Engel at 865.574.7008 if you have any questions or require additional information.

Sincerely,

Erika N. Bailey

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Survey and Technical Projects Group Manager
ORISE

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K. M. Engel
ORISE

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

NOVEMBER 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 interagency agreement number 31310018N0014 with the U.S. Department of Energy (DOE). ORISE is managed by Oak Ridge Associated Universities (ORAU) under DOE contract number DE-SC0014664.

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE GENERAL ATOMICS TRIGA REACTOR FACILITY BUILDING G21 AND ASSOCIATED LAND AREA, SAN DIEGO, CALIFORNIA

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ACRONYMS

AA	alternative action
cm	centimeter
cpm	counts per minute
DCGL	derived concentration guideline level
dpm	disintegrations per minute
DQO	data quality objective
DS	decision statement
EMC	elevated measurement comparison
EPA	U.S. Environmental Protection Agency
FSS	final status survey
FSSP	final status survey plan
GA	General Atomics
GPS	global positioning system
HTD	hard-to-detect
m^2	square meter
MDC	minimum detectable concentration
NaI	sodium iodide
NIST	National Institute of Science and Technology
NORM	naturally occurring radioactive material
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocurie per gram
PSQ	principal study question
ROC	radionuclide of concern
SOF	sum of fraction
SU	survey unit
TAP	total absorption peak
TPU	total propagated uncertainty
TRF	TRIGA Reactor Facility
TRIGA	Training, Research, Isotopes, General Atomics
VSP	Visual Sample Plan

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE GENERAL ATOMICS TRIGA REACTOR FACILITY BUILDING G21 AND ASSOCIATED LAND AREA, SAN DIEGO, CALIFORNIA

EXECUTIVE SUMMARY

In 1997, General Atomics ceased all operations related to the Training, Research, Isotopes, General Atomics (TRIGA) reactor at its main site on the Torrey Pines mesa in San Diego, California. The facility is currently undergoing decontamination and decommissioning for unrestricted release of the reactor pools and remaining reactor rooms, as well as the related structures, systems, and components (GA 2019).

The Oak Ridge Institute for Science and Education (ORISE) performed independent assessment activities during the period of August 5–8, 2019. Confirmatory survey activities included gamma surface scans, gamma direct measurements, alpha-plus-beta scans, alpha-plus-beta direct measurements, smear sampling, and soil/volumetric sampling within Building G21 and associated land area, as applicable. The areas investigated included the following survey units (SUs): Mark I reactor pit, Mark F reactor pit and canal, Mark I reactor room (floor and lower walls), Mark F reactor room (floors and lower walls), the soil lab, mezzanine 1, mezzanine 2, TRIGA waste yard, TRIGA front yard, TRIGA back yard, and room 112, as well as a small section of the roof. The U.S. Nuclear Regulatory Commission (NRC) was aware that some SUs were still in the process of final status survey (FSS) during the time of the ORISE survey but requested that ORISE proceed with independent assessment.

Gamma and alpha-plus-beta scans identified multiple areas of elevated direct radiation both inside and outside of Building G21. Based on the results of scanning, 62 locations were selected for direct measurements and smear sampling. At 16 of these locations, smear samples were collected for hard-to-detect (HTD) radionuclide analysis. Two volumetric concrete samples were collected for analysis at locations identified by ORISE based on scan results. Additionally, the NRC selected two of the licensee’s archived FSS samples that were of interest to the NRC and directed ORISE to conduct independent laboratory analyses. Nine locations were selected for judgmental soil sampling and an additional 12 random soil samples were collected from the TRIGA back yard.

Eleven direct measurements exceeded the 5,000 disintegration per minute (dpm)/100 square centimeters (cm^2) screening value; however, all removable activity measurements were below the 1,000 dpm/100 cm^2 screening value. All smear samples collected for HTD analyses were below the analytical minimum detectable concentrations (MDCs). The two volumetric concrete samples collected from the ORISE-identified locations each had a sum-of-fraction (SOF) result greater than 1. Each sample had cesium-137 (Cs-137) concentrations that exceeded the screening value of 11 picocuries per gram (pCi/g). Additionally, strontium-90 (Sr-90) was identified in these two samples above the analytical MDC. No individual concentrations within the soil samples exceeded their respective screening levels for the site-identified radionuclides of concern (ROCs) and all SOFs were less than 1.

While results of the confirmatory survey support the licensee's classification of the SUs, as the locations that exceeded the screening values were in Class 1 areas, several key issues were identified: surface activity measurements exceeded the screening value in a SU that had already received FSS, several other locations on structural surfaces exhibited contamination that exceeded the applicable screening value, Sr-90 was identified in two volumetric samples when Sr-90 was not identified as a ROC, and elevated radiation levels were identified on the TRIGA building roof although it was not listed as a SU. Based on results of the August 5-8, 2019 confirmatory survey, certain survey units associated with Building G21 were found to contain residual radioactivity in excess of NRC-approved decommissioning criteria. As a result, the radiological conditions, at the time of the confirmatory survey, cannot be confirmed to meet the established decommissioning criteria. Survey results are provided herein for NRC's evaluation.

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE GENERAL ATOMICS TRIGA REACTOR FACILITY BUILDING G21 AND ASSOCIATED LAND AREA, SAN DIEGO, CALIFORNIA

1. INTRODUCTION

General Atomics (GA, the licensee) developed the Training, Research, Isotopes, General Atomics (TRIGA) class of research reactor, which was designed to be used in both industrial and academic applications. Between 1957 and 1966, three TRIGA reactors were constructed at the GA main site on the Torrey Pines mesa, also known as the TRIGA Reactor Facility (TRF), in San Diego, California. The three reactors were the TRIGA Mark I, Mark F, and Mark III. The Mark III reactor was shut down in 1972, decommissioned, and its license was terminated. Therefore, the Mark III was not within the scope of this confirmatory survey (GA 2019). The pool-type TRIGA Mark I reactor was operational from 1958 until 1997. It was licensed to operate at a power level of 250 kilowatts and could be rapidly pulsed to power levels of over 1,000 megawatts (NRC 2018a). The pool-type TRIGA Mark F reactor was operational from 1960 until 1995 and was designed to provide controlled instantaneous pulses of intense neutron and gamma radiation (NRC 2018b).

In 1997, GA ceased all remaining TRIGA reactor operations at the Torrey Pines facility. All irradiated fuel, activated and contaminated hardware, startup neutron sources, and balance of plant components associated with the Mark I and Mark F reactors have been removed from the facility. The two reactors were located in building G21 and currently are undergoing decontamination and decommissioning, with the goal of unrestricted release of the Mark I and Mark F reactor pools, reactor rooms, and other impacted areas within building G21 that supported reactor operations, as well as several exterior areas adjacent to the G21 building. This will allow termination of GA's U.S. Nuclear Regulatory Commission (NRC) reactor licenses R-38 and R-67 (GA 2019).

The NRC requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory survey activities within the impacted areas of the GA TRIGA site.

2. SITE DESCRIPTION

The TRIGA reactor facility is located 24 kilometers (15 miles) north of San Diego, California. Building G21 and the associated support facilities are located on a 120 hectare (295 acres) tract of land acquired by the General Dynamics Corporation (later General Atomics) from the city of San Diego in 1956. Figure 2.1 provides a site map of the TRF showing the delineation of the survey units (SUs) including the Mark I and Mark F reactor pits and impacted support rooms, as well as the associated land areas. There are 14 SUs associated with the TRF: nine Class 1 SUs, four Class 2 SUs, and one non-impacted SU (this area previously was approved for release) (GA 2019). Table 2.1 provides the SU descriptions and classifications.

Table 2.1. TRIGA Building (G21) Survey Units and Classification (GA 2019)[a]

Survey Unit #	Description	Initial Class	Area (m^2)
1	Mark I Reactor Pit	1	9.0
2	Mark F Reactor Pit and Canal	1	14.8
3	Mark I Reactor Room (floors and lower walls)	1	84.8
4	Mark I Reactor Room (upper walls and ceiling)	1	93.8
5	Mark F Reactor Room (floors and lower walls)	1	72.7
6	Mark F Reactor Room (upper walls and ceiling)	1	87.5
7	Soil Lab	1	27.4
8	Mezzanine 1	1	23.6
9	Mezzanine 2	2	30.6
10	TRIGA Waste Yard	1	117.7
11	TRIGA Front Yard (asphalt)	2	546
12	TRIGA Back Yard (soil)	2	886
13	Room 112	2	36.47
14	Non-Impacted Areas	Non-Impacted	283.6

[a] The roof was not listed as a SU in the *TRIGA Reactor Facility Final Status Survey Plan* (GA 2019).

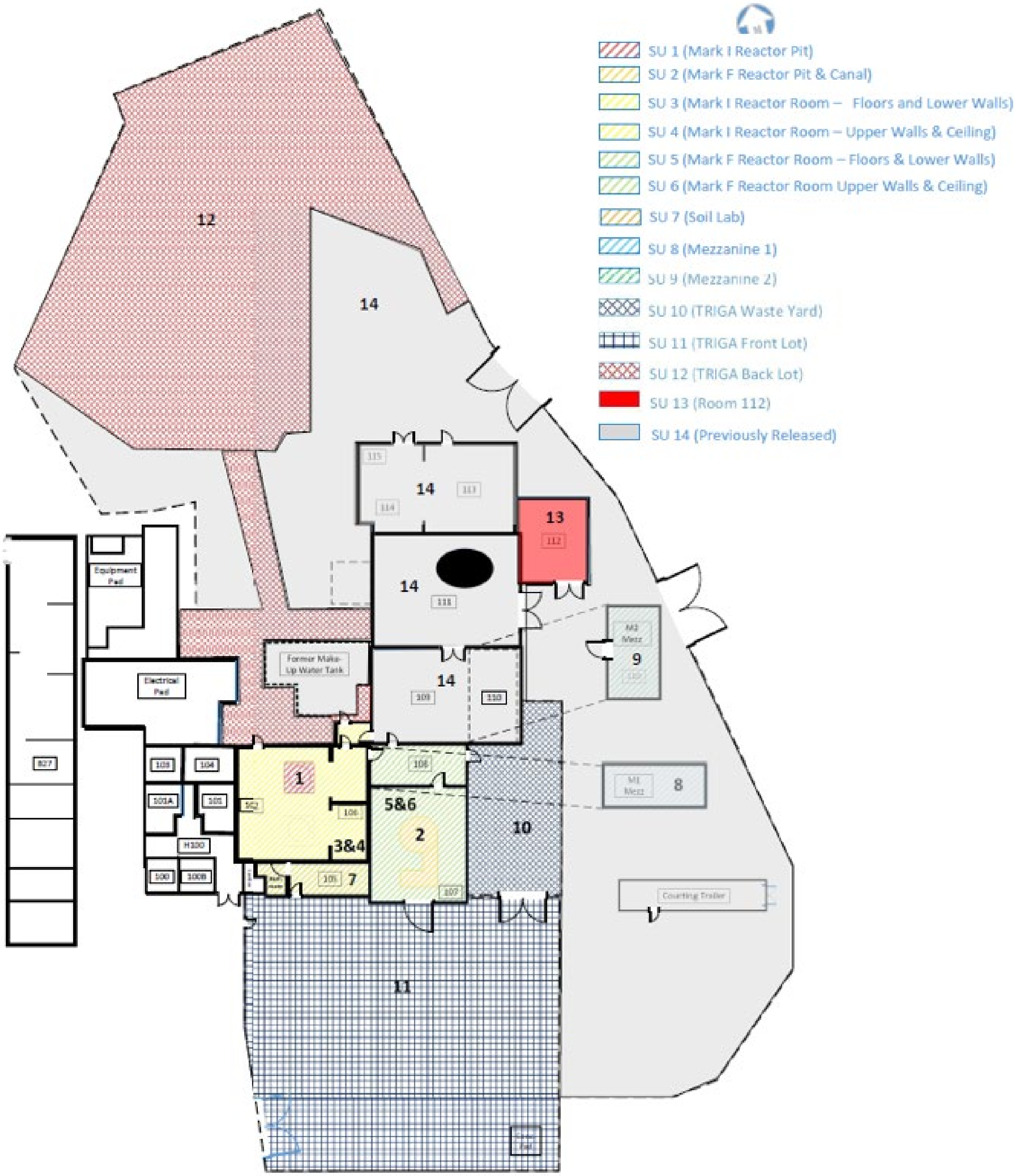

Figure 2.1. TRIGA Reactor Facility Site Map (GA 2019)

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 in the DQO process are as follows:

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

3.1 State the Problem

The first step in the DQO process defines the problem that necessitates the study. The licensee is requesting approval from the NRC for the unrestricted release of the TRIGA reactor pools, reactor rooms, and all impacted survey units noted in Table 2.1 that supported reactor operations. The NRC requested that ORISE perform confirmatory surveys to generate independent radiological data to assist the NRC in evaluating the licensee's final status survey (FSS) results. Therefore, the problem statement is as follows:

> Confirmatory surveys are necessary to generate independent radiological data to assist the NRC with their assessment and determination of the adequacy 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 identifies the principal study questions (PSQs) and alternative actions (AAs), develops decision statements (DSs), and organizes multiple decisions, as appropriate. This was done by specifying AAs that could result from a "Yes" response to the PSQs and combining the PSQs and AAs into DSs. Table 3.1 presents the PSQs, AAs, and DSs for this survey effort.

Table 3.1. Confirmatory Survey Decision Process	
Principal Study Questions	**Alternative Actions**
PSQ1: Are residual radioactivity concentrations associated with the TRIGA Reactor Facility below applicable limits?	**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, that quantitative field and laboratory data satisfied the NRC-approved decommissioing 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.
PSQ2: Do analytical results confirm the absence of fission and activation products not identified in the final status survey plan (FSSP), including hard-to-detect (HTD) radionuclides, within select confirmatory samples?	**Yes:** Provide analytical minimum detectable concentrations (MDCs) and the less-than-MDC results to the NRC. **No:** Provide analytical results to the NRC that include all identified radionuclides for their assessment and decision making. Provide independent interpreation of confirmatory sample analyses that identified anomalous results.
PSQ3: Do the confirmatory results support the classification of the FSS SUs?	**Yes:** Confirmatory results support the classification of the FSS SUs. 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 SUs. Summarize the discrepancies and provide technical comments to the NRC for their decision making.
Decision Statements	
Confirmatory survey results did/did not identify volumetric concentrations and/or surface activity levels that exceed the release criteria. Confirmatory survey results did/did not identify other fission and activation products and/or HTD radionuclides in select confirmatory samples. Confirmatory survey results do/do not agree with the FSS SU classification.	

3.3 IDENTIFY INPUTS TO THE DECISION

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 to meet data requirements. For this effort, information inputs included the following:

- The site's FSSP (Note that, at the time of the confirmatory survey and ORISE reporting, the FSS was in progress; therefore, the FSS data/report were not available.)
- ORISE confirmatory survey results including:
  - Surface scans in the G21 building and SUs exterior to the building
  - Static direct measurements
  - Samples collected for determination of removable activity, including HTDs for select locations in the G21 building and on structural surfaces in SUs exterior to the building
  - ORISE volumetric sample analytical results
- Radionuclides of concern (ROCs) and their associated limits (discussed in Section 3.3.1)

3.3.1 Radionuclides of Concern and Release Criteria

The primary ROCs are beta-gamma emitters—fission and activation products resulting from reactor operation. Based on the historical site assessment, the licensee determined that the potential ROCs are cobalt-60 (Co-60), cesium-137 (Cs-137), europium-152 (Eu-152), and europium-154 (Eu-154) (GA 2019).

The radiological release criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation," Subpart E, "Radiological Criteria for License Termination," for unrestricted use will be used for the TRF. In order to demonstrate compliance with the 25 millirem per year release criterion, the NRC approved the use of default screening values instead of site-specific derived concentration guideline levels ($DCGL_{W}$s) for soil and concrete (NRC 2017). Default screening values used for this site are from NUREG-1757, *Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria,* Vol. 2 (NRC 2006). For structural surfaces, the release limits from NRC Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors* (NRC 1974), as noted in GA's decommissioning plan, will be implemented, hereafter also referred to as screening values for consistency (GA 1999). Based on the FSSP commitments, elevated measurement comparisons (i.e., $DCGL_{EMC}$ or screening value maximums) will not be implemented; therefore, the average screening values are treated as "not-to-exceed" threshold values (GA 2019).

Tables 3.2 and 3.3 provide the default screening values for the ROCs for surfaces and structures and concrete and soil, respectively. (Note that in Table 3.2 the maximum values are not applicable as individual FSS/confirmatory measurements will be compared to the average limit.) The SU is determined to meet the release criterion provided that all measurements are less than the screening values. Removable contamination measurements will be directly compared to the applicable screening values in table 3.2 (GA 2019).

Table 3.2. Surfaces and Structures Screening Values (GA 1999)

Radionuclides of Concern [a]	dpm/100 cm^2 [b]		
	Average [c]	Maximum [d]	Removable [e]
U-nat, U-235, U-238, and associated decay products	5,000	15,000	1,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-133, I-131	1,000	3,000	200
Beta/gamma emitters (nuclides with decay modes other than alpha), except Sr-90 and others noted above	5,000	15,000	1,000

[a] Where surface contamination by both alpha- and beta/gamma-emitting nuclides exists, the limits established for alpha- and beta/gamma-emitting nuclides should apply independently.
[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute (cpm) observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrument.
[c] Measurements of average contaminant should not be averaged over more than 1 square meter (m^2). For objects of less surface area, the average should be derived for each such object.
[d] The maximum contamination level applies to an area of not more than 100 square centimeters (cm^2).
[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with a dry filter or a soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

Table 3.3. Concrete and Soil Screening Values (GA 2019)	
Radionuclides of Concern	**Screening Level for Unrestricted Release (pCi/g)[b]**
Co-60	3.8
Cs-137	11.0
Eu-152	7.0[a]
Eu-154	8.0

[a] This value is lower than the value listed in NUREG-1757 (8.7 pCi/g) because of the memorandum of understanding between the Environmental Protection Agency (EPA) and the NRC.
[b] For informational purposes, the screening level for Sr-90 listed in NUREG-1757 is 1.7 pCi/g.

The values in Table 3.3 represent surficial surface soil concentrations of individual radionuclides that would be deemed in compliance with the 25 millirem per year (mrem/yr) unrestricted release dose limit in 10 CFR 20. Therefore, for sites with multiple radionuclides present, radionuclide-specific results (i.e., volumetric samples) on a per sample basis must be evaluated using the SOF approach to assess the total dose and demonstrate compliance with the dose limit. SOF calculations are performed as follows:

$$SOF_{TOTAL} = \sum_{j=0}^{n} SOF_j = \sum_{j=0}^{n} \frac{C_j}{DCGL_{W,j}}$$

Where C_j is the concentration of ROC "j" and $DCGL_{Wj}$ is the screening value for ROC "j". Note that gross concentrations are considered here for conservatism. A result that does not exceed 1 (unity) ensures compliance with the 25 millirem per year (mrem/yr) unrestricted release dose limit.

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 must be made.

Confirmatory survey activities took place in the G21 building and SUs exterior to the building listed in Table 2.1, except for the upper walls—above approximately 2.5 meters—and the ceilings in all structural SUs. The only SUs that did not receive some level of confirmatory surveys were SUs 4 and 6 because of time constraints with accessing the upper walls and ceilings along with the expectation of lower contamination potential. Additionally, per NRC's request, ORISE scanned a small section of the roof above the Mark F reactor room. Confirmatory survey activities were performed during the week of August 5, 2019, which constitutes the temporal boundary of the study.

3.5 DEVELOP A DECISION RULE

The fifth step in the DQO process specifies appropriate parameters (e.g., mean, median), confirms action levels were above detection limits, and develops an "if…then…" decision rule statement. The confirmatory survey focused on scanning to identify locations that could exceed the applicable screening values and/or analytical MDCs. With the exception of a limited random soil sample population generated for the TRIGA back yard (SU 12), all other locations selected for direct measurement or sampling were judgmentally selected.

Qualitative parameters of interest included the scan data collected across the accessible surfaces of SUs. For this study, the quantitative parameters of interest were individual sample concentrations (volumetric and removable), surface activity results (direct measurements), and the associated MDCs. The decision rules for each SU, as applicable, are stated below:

> If the individual concentrations of ROCs in the volumetric samples are below an SOF of 1 (unity) and if individual surface activity measurements are below their respective screening values, then recommend acceptance; otherwise, provide technical comments/recommendations to the NRC for their decision making.
>
> If other fission and activation products, including HTDs, are not positively identified in confirmatory samples, then provide the NRC with the analytical MDCs along with the less-than values; otherwise, summarize the results and present to the NRC along with technical recommendations, as necessary.
>
> Since ORISE focused on scanning in SUs to identify locations that could exceed the applicable screening values and/or analytical MDCs, if all measurements or sample results are below their respective screening values, then recommend acceptance of SU classification; otherwise, provide technical comments/recommendations to the NRC for their decision making.

3.6 SPECIFY LIMITS ON DECISION ERRORS

The sixth step in the DQO process specifies the decision maker's limits on decision errors, which are then used to establish performance goals for the survey. Decision errors are controlled both during the confirmatory investigations and during data quality assessment. The *a priori* detector scan

MDCs were below the screening values for the site-identified ROCs for hand-held detectors (see Appendix D). The survey level of performance matched the planned level of performance (i.e. slow scan speed and surveyors paused liberally); therefore, contamination levels near the scan MDC would have been identified. Any anomalies above background identified while performing the surveys were thoroughly investigated and discussed with the NRC staff.

The FSS data were unavailable at the time of the field activities, and FSS activities had only been completed in SUs 8 and 9. Therefore, with the exception of SU 12, ORISE proceeded with scanning and the collection of judgmental only measurements and samples. The number of random samples collected in the TRIGA back yard (SU 12) during the confirmatory survey was based on developing a non-parametric confidence interval and were collected to provide the NRC with an unbiased estimate of the mean/median ROC concentrations.

3.7 OPTIMIZE THE DESIGN FOR OBTAINING DATA

The seventh step in the DQO process reviews the DQO outputs, develops data collection design alternatives, formulates mathematical expressions for each design, selects the sample size to satisfy DQOs, decides on the most resource-effective design of agreed alternatives, and documents requisite details. The confirmatory survey followed a graded approach, such that survey efforts concentrated on the areas with the highest potential for contamination.

Specific survey procedures are presented in Section 4.

4. PROCEDURES

The ORISE survey team conducted independent confirmatory survey activities, including surface scans, surface activity measurements, and sampling activities, within the accessible portions of the SUs noted previously, or as specifically requested by the NRC, during the period of August 5-8, 2019. Survey activities were in accordance with the project-specific confirmatory survey plan, the *Oak Ridge Associated Universities (ORAU) Radiological and Environmental Survey Procedures Manual,* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORISE 2019, ORAU 2016a, 2019a). Appendices C and D provide additional information regarding survey instrumentation and related processes discussed within this section.

4.1 REFERENCE SYSTEM

ORISE referenced outdoor confirmatory measurement/sampling locations to global positioning system (GPS) coordinates, specifically NAD 1983 (CORS96) State Plane California, and also noted the locations on field forms. Indoor measurement/sampling locations were documented on detailed survey maps. Other prominent site features also were referenced.

4.2 SURFACE SCANS

Surface scans of land areas and structural surfaces were performed with Ludlum model 44-10 5 cm by 5 cm sodium iodide (NaI) scintillation detectors coupled to Ludlum model 2221 ratemeter-scalers with audible indicators. For the land area surveys, ratemeter-scalers also were coupled to GPS systems that enabled real-time gamma count rate and spatial data capture.

Surface scans of structural surfaces were also performed with either Ludlum model 43-68 gas proportional hand-held detectors or Ludlum model 43-37 gas proportional floor monitors operated in alpha-plus-beta mode. Both detector types were coupled to Ludlum model 2221 ratemeter-scalers with audible indicators. Scans with the floor monitor were qualitative (scan MDCs not calculated), but ORISE's experience is that floor monitors are effective at, and efficient for, identifying low levels of surface contamination that can be quantitatively investigated using other hand-held instruments.

The Class 1 SUs received high-density scan coverage (i.e., up to 100% coverage); Class 2 SUs received medium-density scan coverage (at least 50% coverage). Locations of elevated direct radiation, suggesting the presence of residual contamination, were marked for further investigation.

Limited gamma scans were performed in a small section of SU 14, a previously released area within the facility, per the NRC's request, although no conclusions can be drawn. Time constraints did not allow ORISE to collect follow-up, quantitative surface activity measurements.

4.3 SURFACE ACTIVITY MEASUREMENTS

Alpha-plus-beta surface activity measurements were performed with Ludlum model 2221 ratemeter-scalers with audible output paired with Ludlum model 43-68 gas proportional detectors with 0.8 milligram per square centimeter (mg/cm^2) thick Mylar windows for the beta-gamma emitting ROCs. For the purpose of converting counts per minute (cpm) to units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2), detector efficiencies used assumed that Co-60 was the predominant radionuclide for the SUs inside the building and Cs-137 was assumed for exterior SUs, based on discussions at the site between the NRC, ORISE, and the licensee.

Direct measurement locations were determined based on the results of scanning in all SUs. Table 4.1 provides a summary of the number of direct measurements and samples collected in each SU.

Table 4.1. Number and Type of Measurements

SU	Description	Number of Samples			
		Direct Measurements/ Smears	HTD Smears	Soil	Concrete
1	Mark I Reactor Pit	8	3	3	1
2	Mark F Reactor Pit and Canal	10	6	0	2
3	Mark I Reactor Room (floors and lower walls)	5	1	0	0
4	Mark I Reactor Room (upper walls and ceiling)	--[a]	--	--	--
5	Mark F Reactor Room (floors and lower walls)	9	3	0	0
6	Mark F Reactor Room (upper walls and ceiling)	--	--	--	--
7	Soil Lab	2	0	0	0
8	Mezzanine 1	5	0	0	0
9	Mezzanine 2	1	0	0	0
10	TRIGA Waste Yard	5	1	0	1
11	TRIGA Front Yard	5	0	1	0
12	TRIGA Back Yard	0	0	17	0
13	Room 112	9	2	0	0
14	Non-Impacted Areas	0	0	0	0
NA[b]	Roof	3	0	0	0
Total		62	16	21	4

[a] "--" Indicates measurements not performed in SU.
[b] The roof was not listed as a SU in the *TRIGA Reactor Facility Final Status Survey Plan* (GA 2019).

ORISE collected material-specific background measurements from the site-determined non-impacted reference area from materials that, to the extent possible, were of similar construction to the measurement surfaces. The background reference area was outdoors just outside the main entrance to Building 27. These background measurements were used for correcting gross measurement counts for conversion to surface activity levels.[1]

4.4 SURFACE REMOVABLE ACTIVITY MEASUREMENTS

Dry smear samples, for determining removable gross alpha/beta activity levels, were collected from each direct measurement location, as applicable. Wet smears for the determination of removable tritium (H-3) and carbon-14 (C-14) were collected at judgmentally selected direct measurement locations, in an area adjacent to the dry smear sample location. See Table 4.1 for a summary. Figures A.1 through A.15 in Appendix A show measurement locations as applicable for each SU.

4.5 VOLUMETRIC SAMPLING

Nine judgmental soil samples were collected. Judgmental soil sample locations were determined based on indications noted during gamma scans, with the exception of soil sample S0021 in the TRIGA front yard (SU 11). SU 11 is mostly asphalt and concrete with a small strip of soil on the south end of the SU, which appeared to be a drainage area and a good location for a judgmental sample (S0021). Three judgmental soil samples were collected from the bottom of the Mark I rector pit and five judgmental soil samples were collected from the TRIGA back yard (SU 12). Figures A.1 through A.15 show judgmental sample locations.

For the TRIGA back yard, Visual Sample Plan (VSP), version 7.10 was used to determine the number of samples required such that the estimated SU median was within 30 percentiles above or below the true SU median at the 95% confidence level (Figure 4.1). The 30 percentile interval provides a reasonable estimate of the median/mean, as data were expected to be less than the release limits given the Class 2 designation, and would generate an adequate sample size to minimize confirmatory survey decision errors. For this effort, based on the previously described planning inputs, 12 random samples were required; locations are depicted in Figure A.13. The target depth for soil samples was 0 to 15 cm. However, because of the extremely compacted and rocky soil, samples were collected as deep as possible, typically 0 to 8 cm using hand trowels. No additional depths were collected. Sampling equipment was rinsed and wiped after each sample to minimize the potential for cross-contamination.

[1] The construction material-specific backgrounds were considered appropriate for most confirmatory measurement surfaces with the exception of the reactor pits where the ambient background was lower because of subterranean shielding, resulting in mostly negative surface activity results.

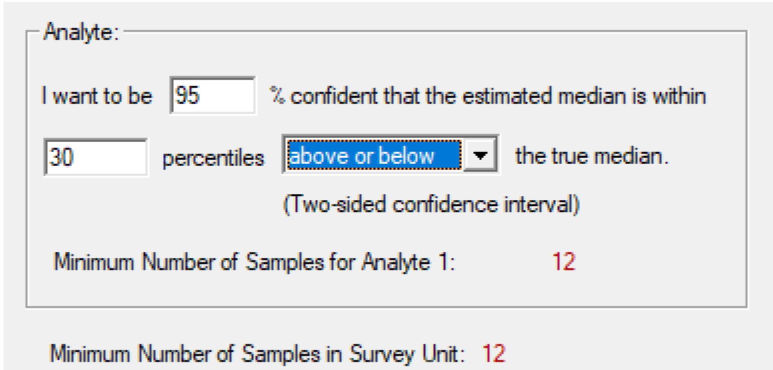

Figure 4.1 VSP Confirmatory Survey Planning Inputs

Two concrete samples were collected at ORISE judgmentally-selected locations based on the results of scans and direct measurements. One sample was collected from the Mark F reactor canal (SU 2) where fuel had been stored and the other was collected in the TRIGA waste yard (SU 10). The licensee collected both concrete samples per NRC's direction under general observation of ORISE; however, there was a delay in turning over sample custody to ORISE. Additionally, and prior to the confirmatory survey, the NRC had requested that the licensee save a split of all the FSS concrete samples that were collected in the Mark I and Mark F reactor pits. The NRC selected two samples: one each from the Mark I and Mark F reactor pits, and directed ORISE to conduct independent laboratory analysis.

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 2019b). Volumetric samples were crushed and homogenized, and a portion was set aside before drying for future analyses before being analyzed by gamma spectrometry for gamma-emitting fission and activation products. For all soil samples, analytical results are reported for the site ROCs and other gamma-emitting naturally occurring radioactive material (NORM) (e.g., uranium, radium), for completeness. Volumetric sample results are reported in units of gross pCi/g. Smears were analyzed for gross alpha/beta activity, H-3, and C-14, and were reported in units of dpm/100 cm^2 and pCi/smear respectively. At the request of the NRC, Sr-90 analyses also was performed on the four concrete samples as well as four judgmental soil samples.

All scan results are presented as gross cpm. Total surface activity measurements are presented as net dpm/100 cm^2, with the appropriate material-specific background subtracted from the gross measurement.

6. FINDINGS AND RESULTS

The results of the confirmatory survey are discussed in the following subsections.

Appendix A provides the gamma walkover and sample location figures and Appendix B provides data tables. Appendices C and D provide additional details regarding field and laboratory instrumentation, as well as additional information on calibration, quality assurance, survey and analytical procedures, and detection sensitivities.

6.1 SURFACE SCANS

Tables 6.1 provides a summary of the scanning survey data for each SU investigated.

Table 6.1. Summary of Scan Results

SU	Description	Scan Ranges (cpm)		
		NaI	Hand-Held Alpha-Plus-Beta	Floor Monitor Alpha-Plus-Beta
1	Mark I Reactor Pit	19,000 to 30,000	350 to 800	--[a]
2	Mark F Reactor Pit and Canal	9,000 to 25,000	200 to 55,000	--
3	Mark I Reactor Room (floors and lower walls)	13,000 to 20,000	200 to 1,800	1,400 to 2,800
5	Mark F Reactor Room (floors and lower walls)	13,000 to 65,000	200 to 1,800	1,200 to 3,500
7	Soil Lab	11,000 to 17,000	300 to 750	1,200 to 2,000
8	Mezzanine 1	--	300 to 640	1,600 to 2,400
9	Mezzanine 2	--	200 to 600	800 to 1,200
10	TRIGA Waste Yard	10,000 to 31,000	--	1,200 to 6,600
11	TRIGA Front Yard	7,600 to 31,000	--	1,300 to 4,100
12	TRIGA Back Yard	9,200 to 25,000	--	--
13	Room 112	14,000 to 22,000	250 to 700	1,400 to 3,000
14	Non-Impacted Areas	15,000 to 23,000	--	--
NA[b]	Roof	7,900 to 43,000	250 to 1,300	--

[a] "—" Indicates scan data not collected.
[b] The roof was not listed as an SU in the *TRIGA Reactor Facility Final Status Survey Plan* (GA 2019).

Multiple locations were marked in the SUs for further investigation based on the observed surface scan results. Inside the G21 building, there were several areas of elevated direct alpha-plus-beta radiation identified, while there was minimal gamma radiation response at these locations. This was most notable in the Mark F reactor canal on both concrete surfaces and along a "track" near the entrance to the canal, which still contained a "seal" of unknown material. Similarly, one area on the roof had elevated alpha-plus-beta radiation levels on the side of a capped ventilation trunk, but no elevated gamma levels were noted for this location. One drain inside the building was identified as having elevated gamma radiation levels. The drain was located to the east of the Mark F reactor pit in a trench about one meter below a metal covering. The beta detector could not access the internal drain surfaces; therefore, elevated beta radiation levels inside of the drain could not be confirmed. Debris was noted in the elbow of the drain, but could not be reached. Outside of the G21 building, areas exhibiting elevated direct radiation were identified during both gamma and alpha-plus-beta scans. Figures A.9 through A.13 show exterior gamma scan results. It should be noted in Figure A.9 that an elevated gamma response is shown in the north section of the SU. This area was marked and further investigated; however, the elevated response could not be reproduced during follow-up investigations. Figure A.13 shows elevated gamma responses in an excavated area in the south section of the SU. These elevated responses occurred near concrete and rocks and may have been due to NORM in the rocks and concrete.

6.2 TOTAL AND REMOVABLE SURFACE ACTIVITY LEVELS

Table 6.2 provides a summary of the surface activity measurements collected both inside and outside the G21 building, and Tables B.1 through B.11, and B.15 in Appendix B provide individual measurement results for each SU. Table B.18 provides the construction material-specific background data for each detector type.

Five of the SUs investigated had direct measurements that exceeded the screening value for surfaces and structures, including SU 8 (mezzanine 1), which had already received FSS.

In the Mark F reactor canal (SU 2), multiple areas were identified as having elevated radiation levels during scanning; however, only a few were selected for direct measurements because of time constraints. The areas selected for the quantitative direct measurements represent a range of the observed detector responses (i.e., low, medium, high scanning count rates).

All removable activity was below the screening value. All removable H-3 and C-14 concentrations were below the respective analytical MDCs.

Table 6.2. Summary of Surface Activity Results								
SU	Description	No. of Locations that Exceed Screening Values	Gamma (cpm)	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
				dpm/100 cm²			pCi/sample	
1	Mark I Reactor Pit	0	15,000 to 27,000	-1,200 to -450	0 to 2	0 to 5	-4.1 to 0.6	-2.7 to 0.8
2	Mark F Reactor Pit and Canal	5	8,300 to 25,000	-1,800 to 440,000	0 to 2	0 to 47	-6.8 to 1.9	-2.0 to 7.3
3	Mark I Reactor Room (floors and lower walls)	1	11,000 to 19,000	-950 to 11,000	0 to 4	0 to 4	-8.2	-1.2
5	Mark F Reactor Room (floors and lower walls)	2	11,000 to 320,000	-770 to 11,000	0 to 2	0 to 25	-4.5 to -3.1	0.1 to 1.6
7	Soil Lab	0	16,000 to 17,000	570 to 650	0 to 2	0	--	--
8	Mezzanine 1	2	--[a]	2,700 to 16,000	0 to 2	0 to 13	--	--
9	Mezzanine 2	0	--	-290	0	3	--	--
10	TRIGA Waste Yard	1	14,000 to 31,000	680 to 11,000	0 to 2	2 to 6	-6.5	0.8
11	TRIGA Front Yard	0	17,000 to 35,000	-330 to 1,900	0	0 to 3	--	--
12	TRIGA Back Yard	--	13,000 to 20,000	--	--	--	--	--
13	Room 112	0	15,000 to 23,000	-2,100 to 3,200	0 to 4	0 to 5	-5.5 to 3.7	0.3 to 0.4
NA[b]	Roof	0	8,800 to 43,000	1,800 to 3,500	--	--	--	--

[a] "--" Indicates measurement not collected.

[b] The roof was not listed as an SU in the *TRIGA Reactor Facility Final Status Survey Plan* (GA 2019).

6.3 RADIONUCLIDE CONCENTRATIONS IN VOLUMETRIC SAMPLES

Tables 6.3 and 6.4 provide summaries of the results of the volumetric soil and concrete samples, respectively. Pre- and post-sampling measurement data and individual sample results for the various matrices are provided in Tables B.12 through B.17.

All ROC concentrations in soil samples were less than the respective screening levels and all SOFs were less than 1. For completeness, NORM concentrations also are reported for the soil samples. Sr-90—a pure beta emitter—was not identified by the licensee as a ROC; however, because the ORISE confirmatory survey alpha-plus-beta scans identified elevated beta radiation during scans without a corresponding increase in the gamma response in those same areas, some samples were selected for Sr-90 analyses.

The two volumetric concrete samples from ORISE-identified locations each had SOFs greater than 1 because of the Cs-137 concentration, which exceeded the screening level for both samples. The sample from outside of the G21 building (in SU 10, TRIGA waste yard, sample 5339M0002) had 114.0 pCi/g of Cs-137 and 6.40 pCi/g of Sr-90. The Co-60 concentration was less than 1 pCi/g. This sample agreed with the licensee's assumption that Cs-137 is the primary ROC in the SUs outside the G21 building. The sample from inside of the building (in SU 2, Mark F reactor canal, 5229M0001) had 26.5 pCi/g of Cs-137 and 0.49 pCi/g of Sr-90. The Co-60 concentration was less than 1 pCi/g. However, based on discussions at the site and for conservatism in calculating surface activity values, Co-60 is assumed as the predominant ROC inside the G21 building for ORISE calculations. Specifically, the gas proportional detector total efficiency associated with Co-60 is lower than that of Cs-137, resulting in a conservatively higher surface activity.

Table 6.3. Summary of Soil Sample Concentrations							
SU	Description	No. of Locations that Exceed Screening Values	Cs-137	Co-60	Eu-152	Eu-154	SOF
			pCi/g				
1	Mark I Reactor Pit	0	-0.022 to 0.012	0.042 to 0.130	0.394 to 3.19	-0.083 to 0.14	0.07 to 0.50
11	TRIGA Front Yard	0	0.083	0.002	-0.036	-0.59	0.01
12	TRIGA Back Yard	0	-0.014 to 0.933	-0.033 to 0.021	-0.069 to 0.062	-0.54 to -0.045	0.00 to 0.08

Table 6.4. Summary of Concrete Concentrations								
SU	Description	No. of Locations that Exceed Screening Values	Cs-137	Co-60	Eu-152	Eu-154	SOF	Sr-90
			pCi/g					pCi/g
1	Mark I Reactor Pit	0	0.025	-0.039	0.06	-0.56	0.01	0.07
2	Mark F Reactor Pit & Canal	1	0.127 to 26.5	-0.011 to 0.211	-0.03 to 0.043	-0.36 to -0.12	0.02 to 2	0.04 to 0.49
10	TRIGA Waste Yard	1	114.0	-0.041	0.07	-0.33	10	6.40

7. SUMMARY

At the NRC's request, ORISE conducted confirmatory survey activities at the GA TRIGA Reactor Facility near San Diego, California, during the period of August 5–8, 2019. Confirmatory survey activities included gamma surface scans, gamma direct measurements, alpha-plus-beta scans, alpha-plus-beta direct measurements, smear sampling, and soil/volumetric sampling within the facility and associated land area, as applicable. The areas investigated included the Mark I reactor pit, Mark F reactor pit and canal, Mark I reactor room (floor and lower walls), Mark F reactor room (floors and lower walls), the soil lab, mezzanine 1, mezzanine 2, TRIGA waste yard, TRIGA front yard, TRIGA back yard, and room 112. Limited gamma scanning was performed in areas previously released within the facility. The NRC was aware that some SUs were still in the process of FSS during the time of the ORISE confirmatory survey activities but requested that ORISE proceed with independent assessment.

Gamma and alpha-plus-beta scans identified multiple areas of elevated direct radiation both inside and outside of the facility. Based on the results of scanning, 62 locations were selected for direct measurements and smear sampling. At 16 of these locations, smear samples were collected for HTD radionuclide analysis. Four volumetric samples were obtained for analysis. ORISE identified two locations for sampling based on scan results. The NRC selected the other two samples from the licensee's archived FSS samples that were of interest to the NRC and directed the ORISE confirmatory analysis. ORISE collected judgmental soil samples from nine locations and an additional 12 random soil samples were collected from the TRIGA back yard.

Eleven direct measurements exceeded the 5,000 dpm/100 cm^2 screening value; however, all removable activity was below the 1,000 dpm/100 cm^2 screening value. All removable H-3 and C-14 concentrations were below the analytical MDCs. The two volumetric concrete samples from ORISE-identified locations each had SOFs greater than 1. Each sample had Cs-137 concentrations that exceeded the screening value of 11 pCi/g. Sr-90 also was identified in these samples. No individual concentrations within the soil samples exceeded their respective screening levels and all SOFs were less than 1.

While results of the confirmatory survey support the overall classification of the SUs because locations that exceeded the screening values were in Class 1 areas, several key issues were identified:

- Surface activity measurements exceeded the screening value in a SU that had already received FSS;
- Several other locations on structural surfaces exhibited contamination greater than the applicable screening value;
- Sr-90 was identified in two volumetric concrete samples when Sr-90 was not included as an ROC, and
- Elevated radiation levels were identified on the roof although it was not listed as a SU.

ORISE was notified that additional remediation would be performed at the site based on these confirmatory findings. Based on the August 5 through 8, 2019 confirmatory survey and the REAL analytical results, all Building G21 and associated land areas investigated could not be confirmed to satisfy the applicable NRC decommissioning criteria for unrestricted use.

8. REFERENCES

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NRC 2018b. General Atomics – TRIGA MARK F. https://www.nrc.gov/info-finder/decommissioning/research-test/general-atomics-triga-mark-f.html. Accessed July 26, 2019.

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ORISE 2019. *Project-Specific Plan for Confirmatory Survey Activities for the General Atomics TRIGA Reactor Facility Building G21 and Associated Land Area, San Diego, California.* Oak Ridge Institute for Science and Education. Oak Ridge, Tennessee. August.

APPENDIX A: FIGURES

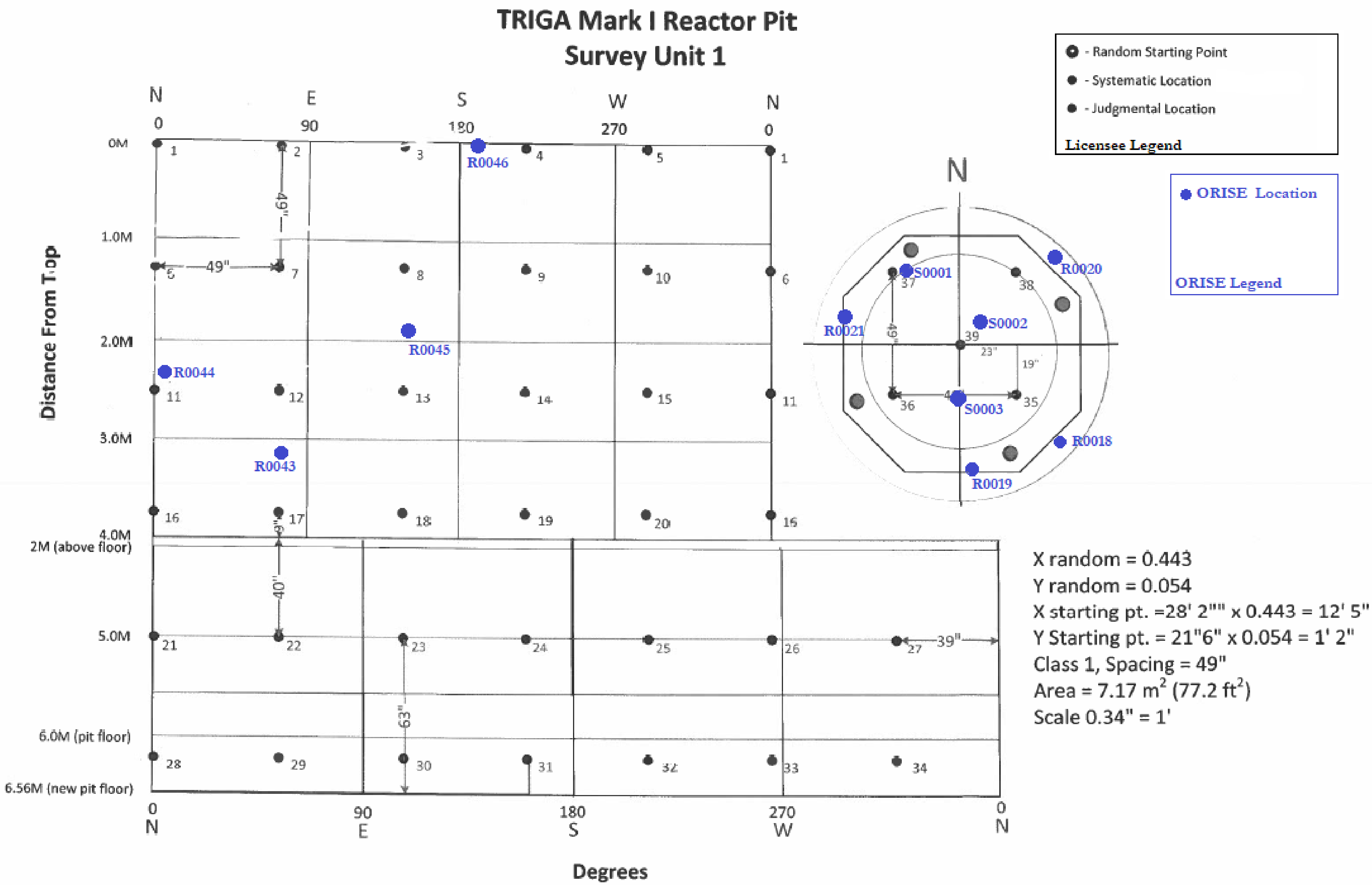

Figure A.1. SU 1 Mark I Reactor Pit Survey Locations

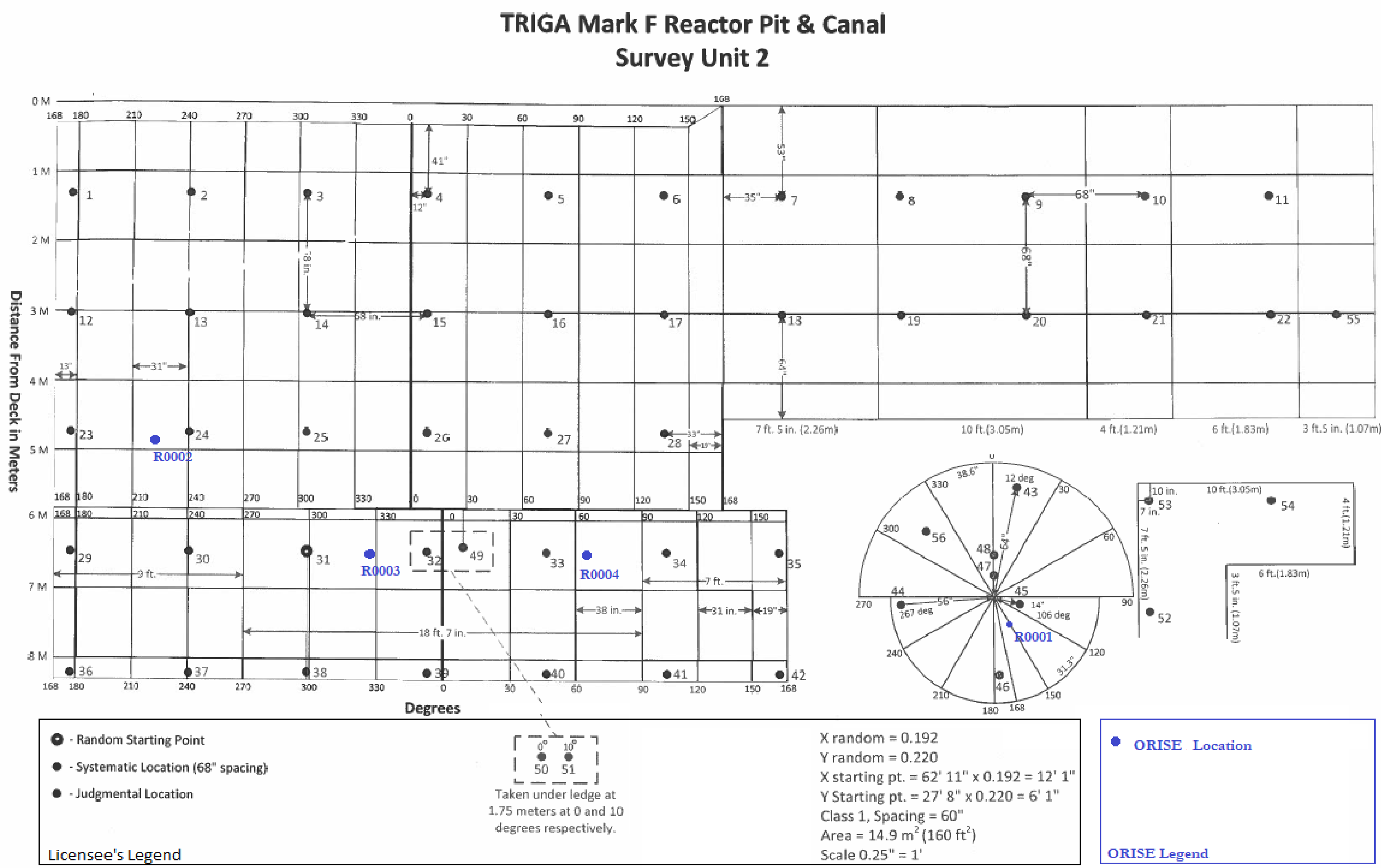

Figure A.2. SU 2 Mark F Reactor Pit Survey Locations

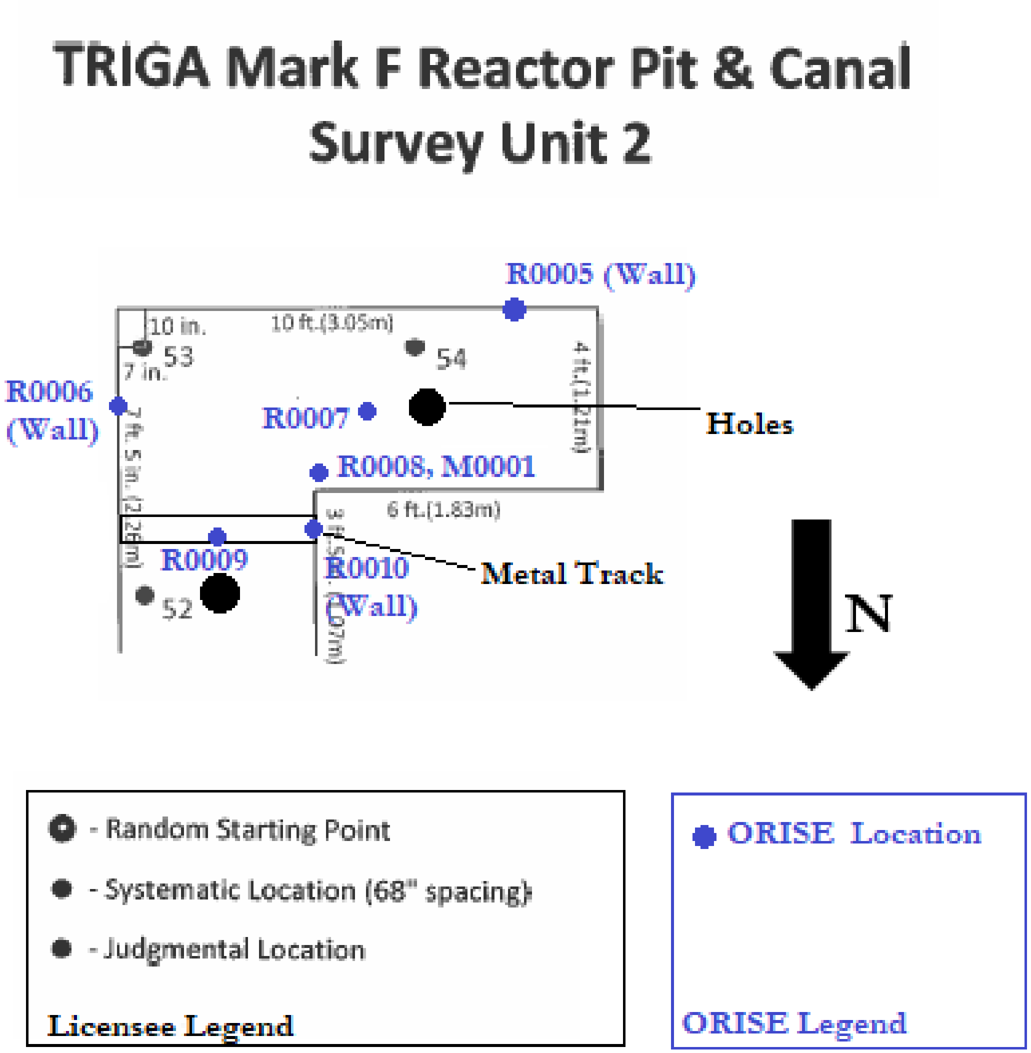

Figure A.3. SU 2 Mark F Reactor Pit Canal Survey Locations

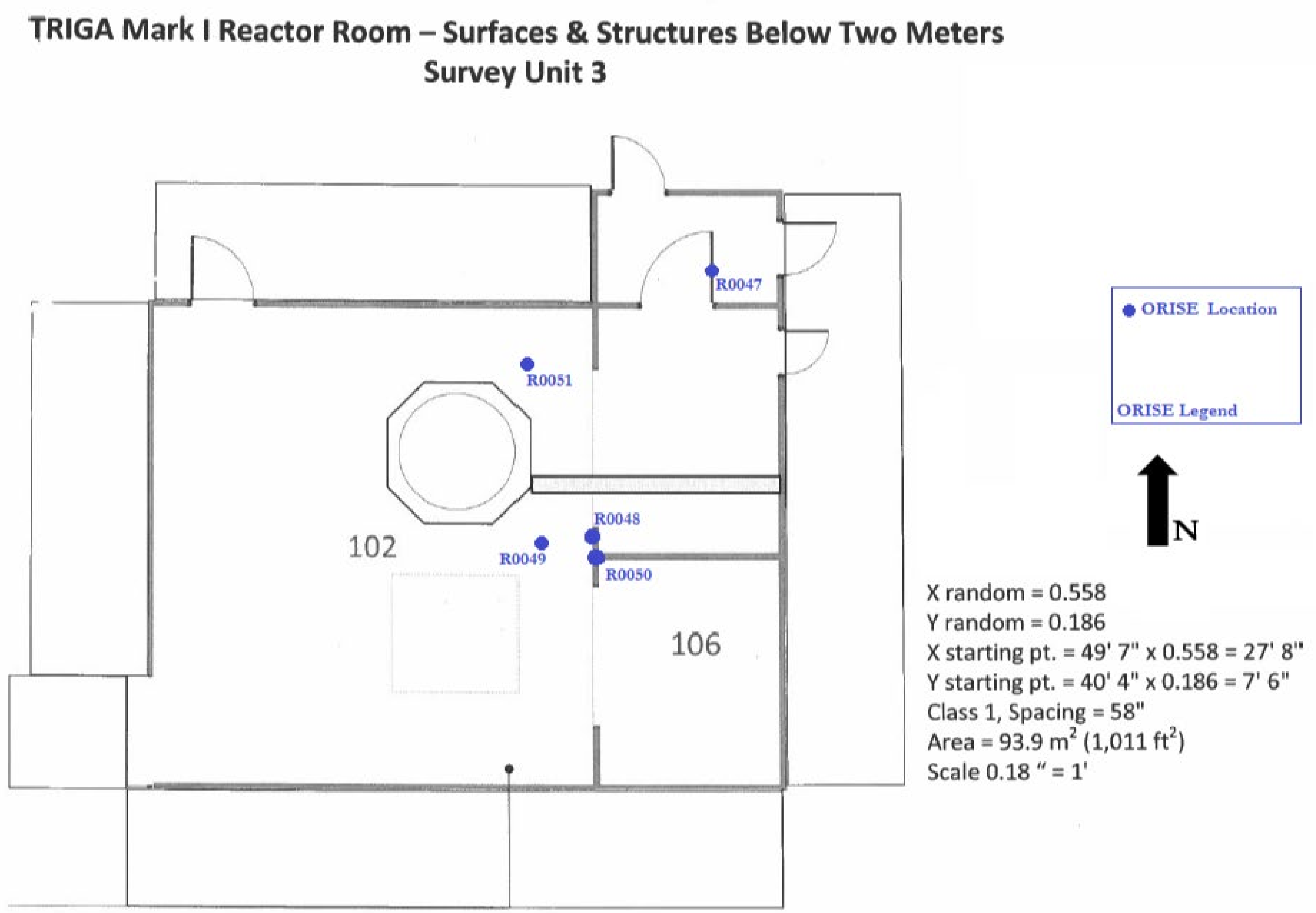

Figure A.4. SU 3 Mark I Reactor Room (floors and lower walls) Survey Locations

Figure A.5. SU 5 Mark F Reactor Room (floors and lower walls) Survey Locations

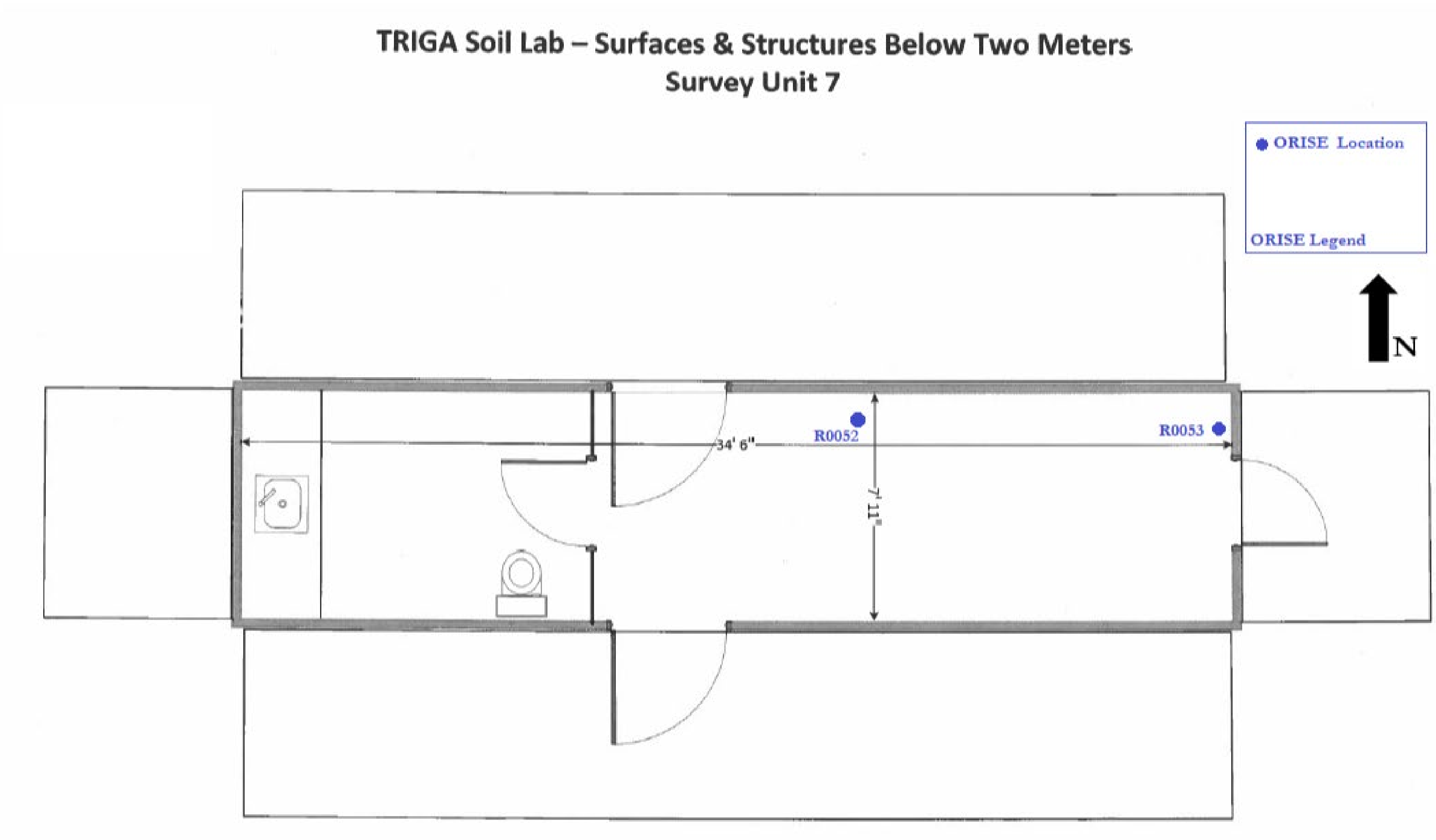

Figure A.6. SU 7 Soil Lab Survey Locations

TRIGA Mezzanine 1 – Surfaces & Structures Below Two Meters
Survey Unit 8

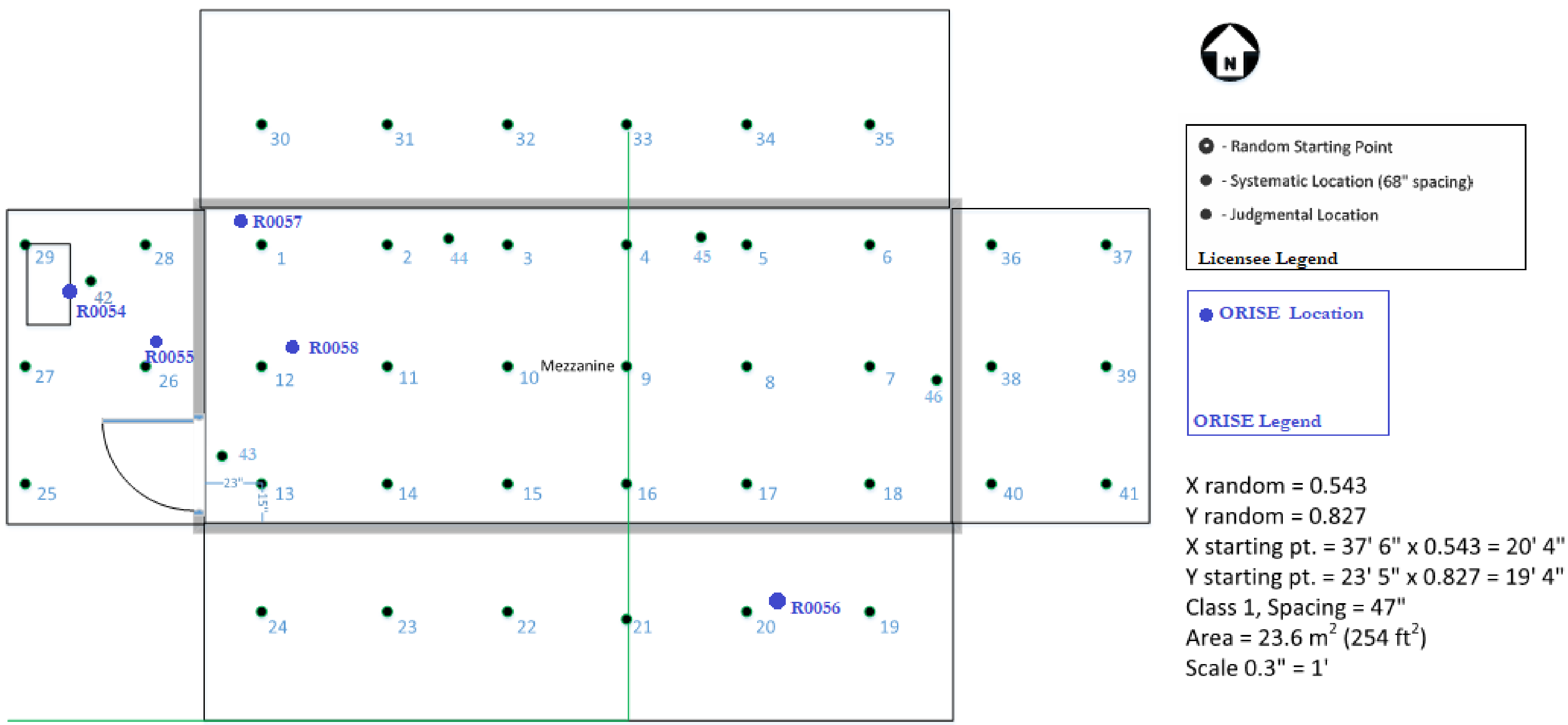

Figure A.7. SU 8 Mezzanine 1 Survey Locations

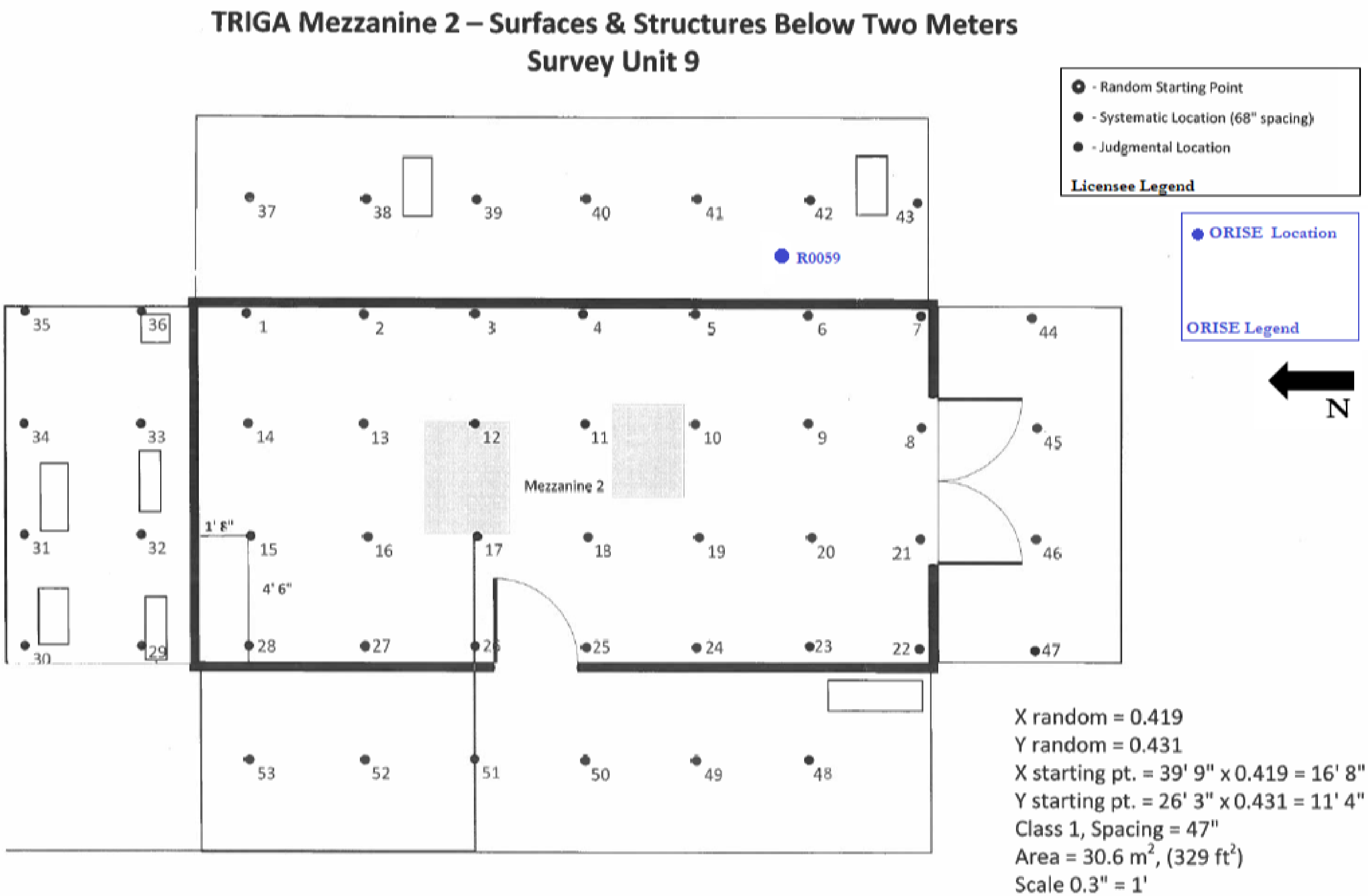

Figure A.8. SU 9 Mezzanine 2 Survey Locations

Figure A.9. SU 10 TRIGA Waste Yard Gamma Walkover Map with Survey Locations

Figure A.10. SU 10 TRIGA Waste Yard Alpha-Plus-Beta Walkover Map with Survey Locations

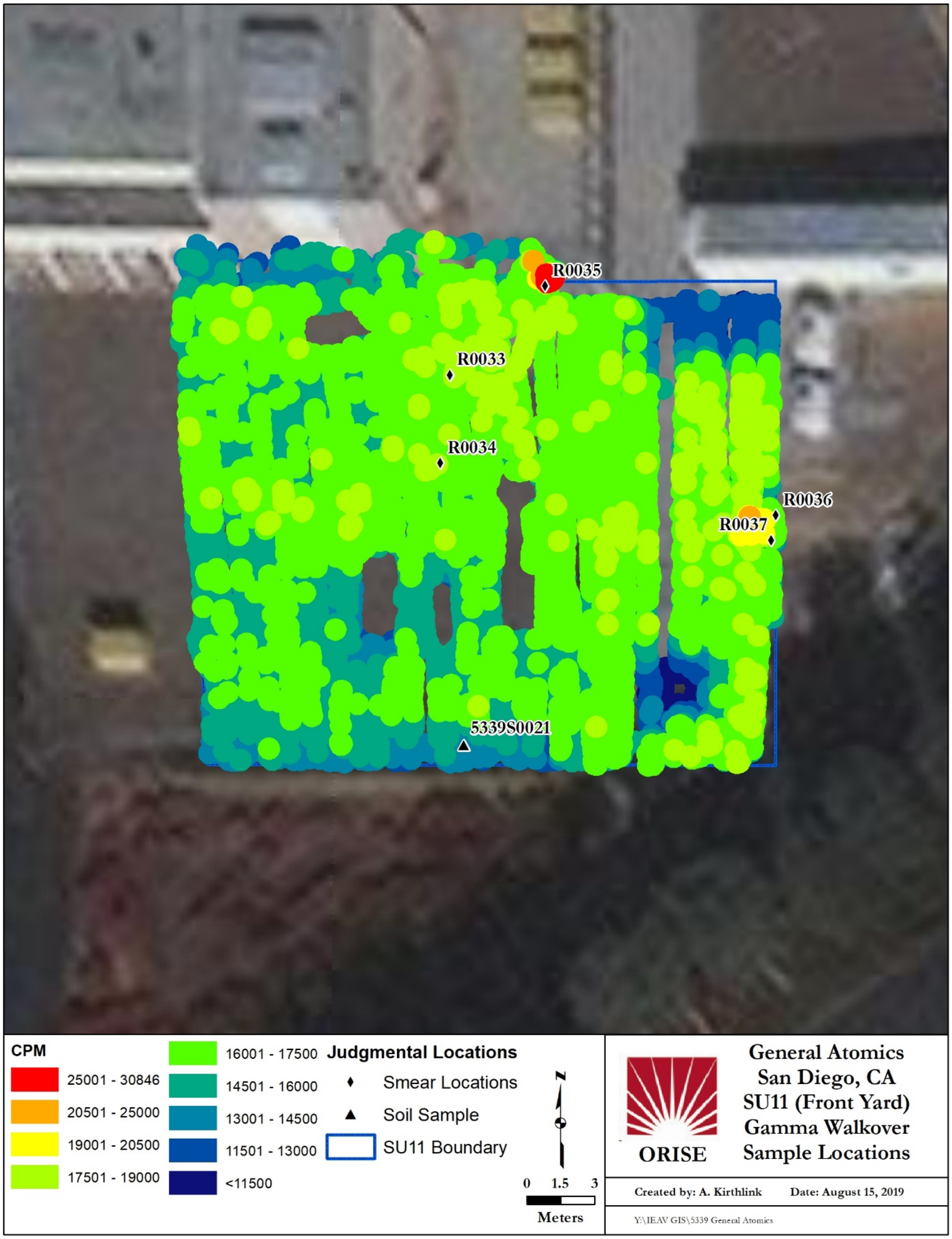

Figure A.11. SU 11 TRIGA Front Yard Gamma Walkover Map with Survey Locations

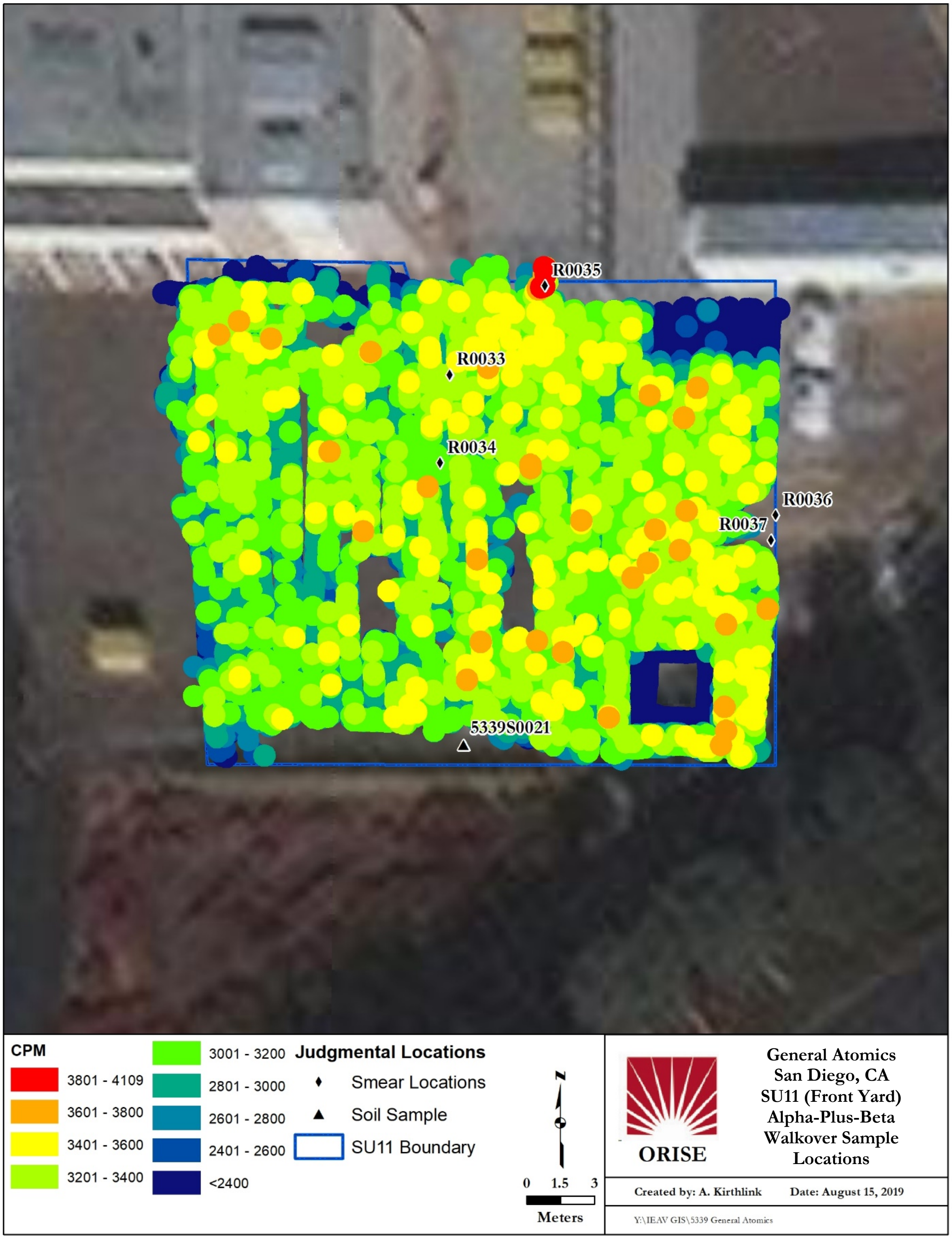

Figure A.12. SU 11 TRIGA Front Yard Alpha-Plus-Beta Walkover Map with Survey Locations

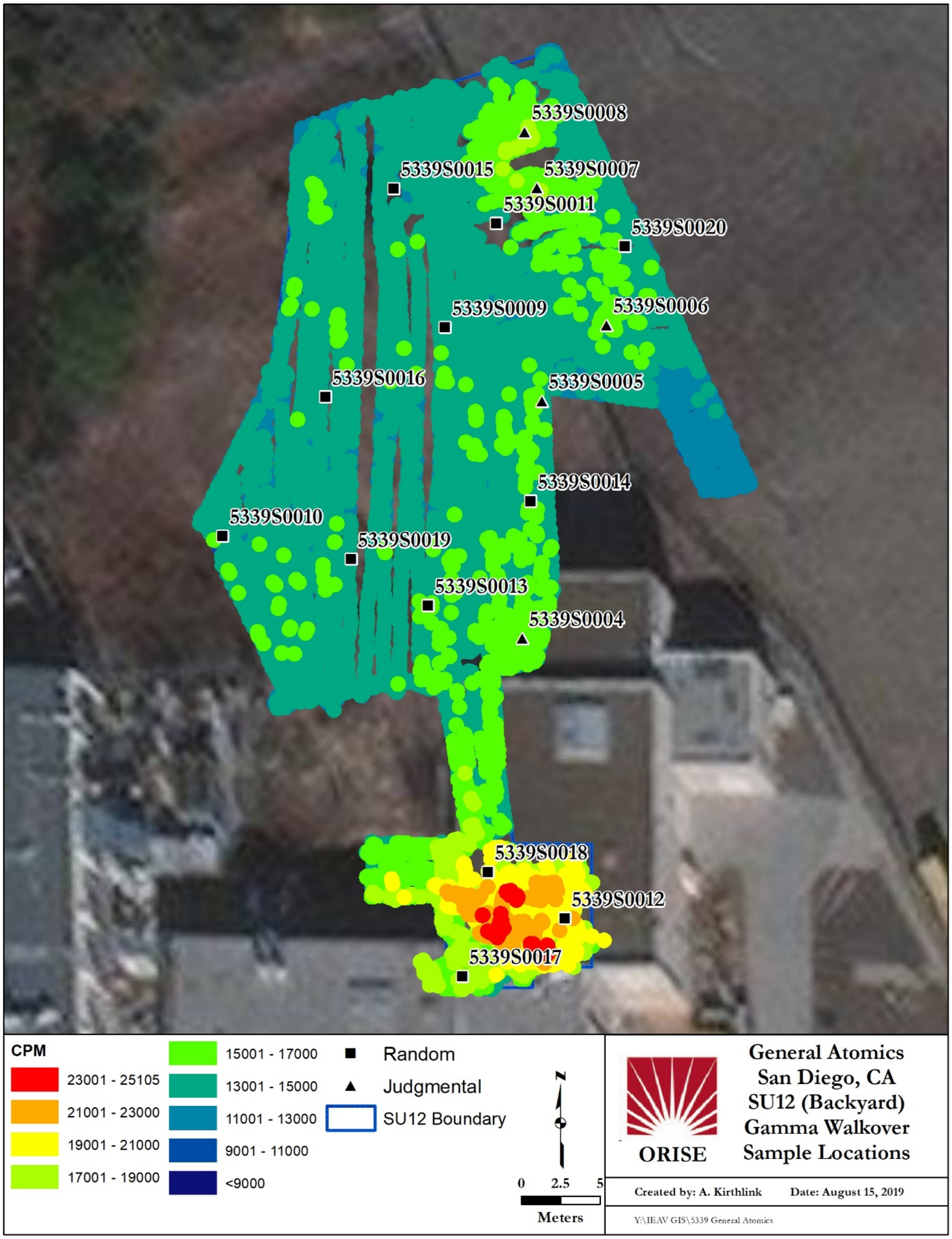

Figure A.13. SU 12 TRIGA Back Yard Gamma Survey Map with Survey Locations

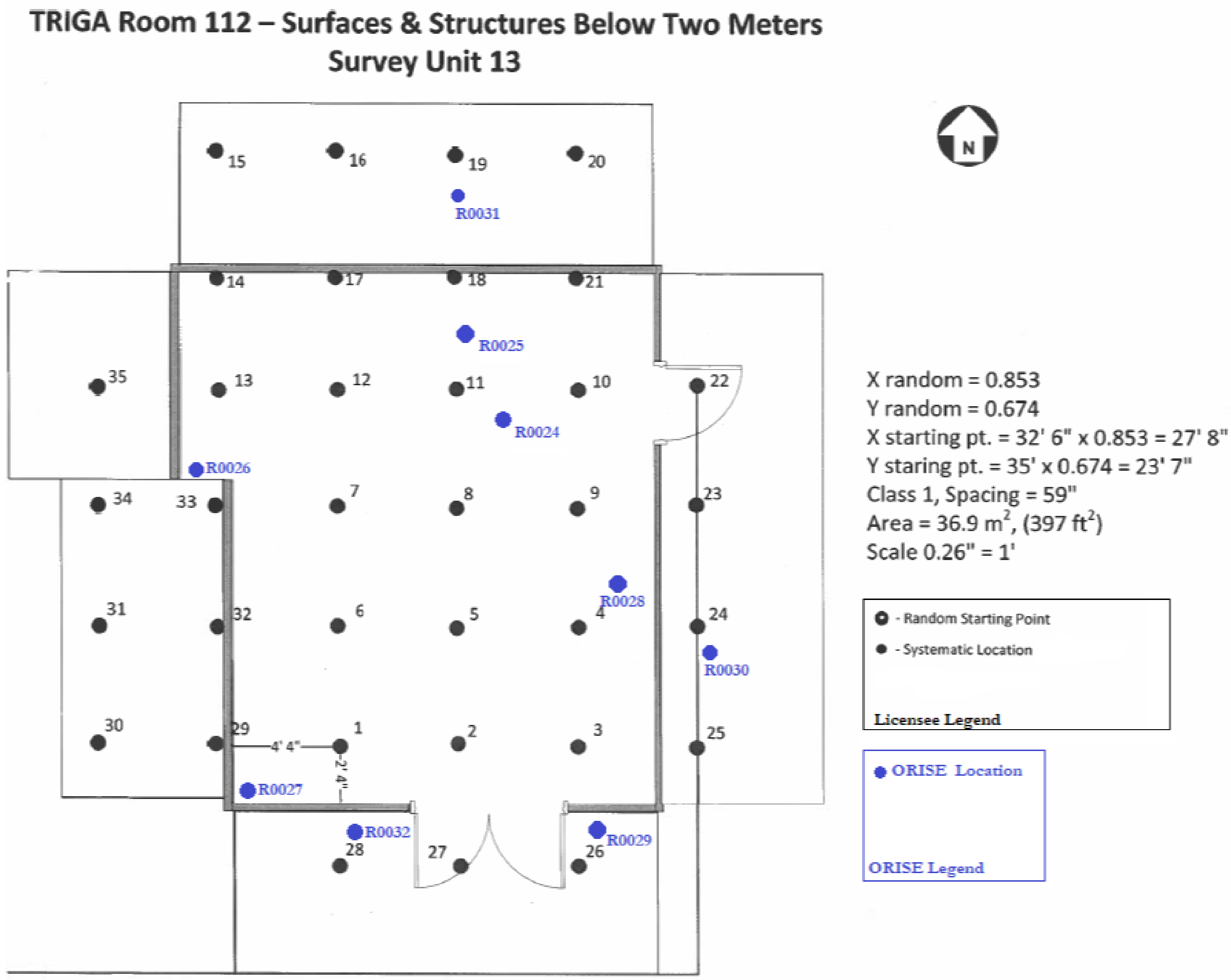

Figure A.14. SU 13 Room 112 Survey Locations

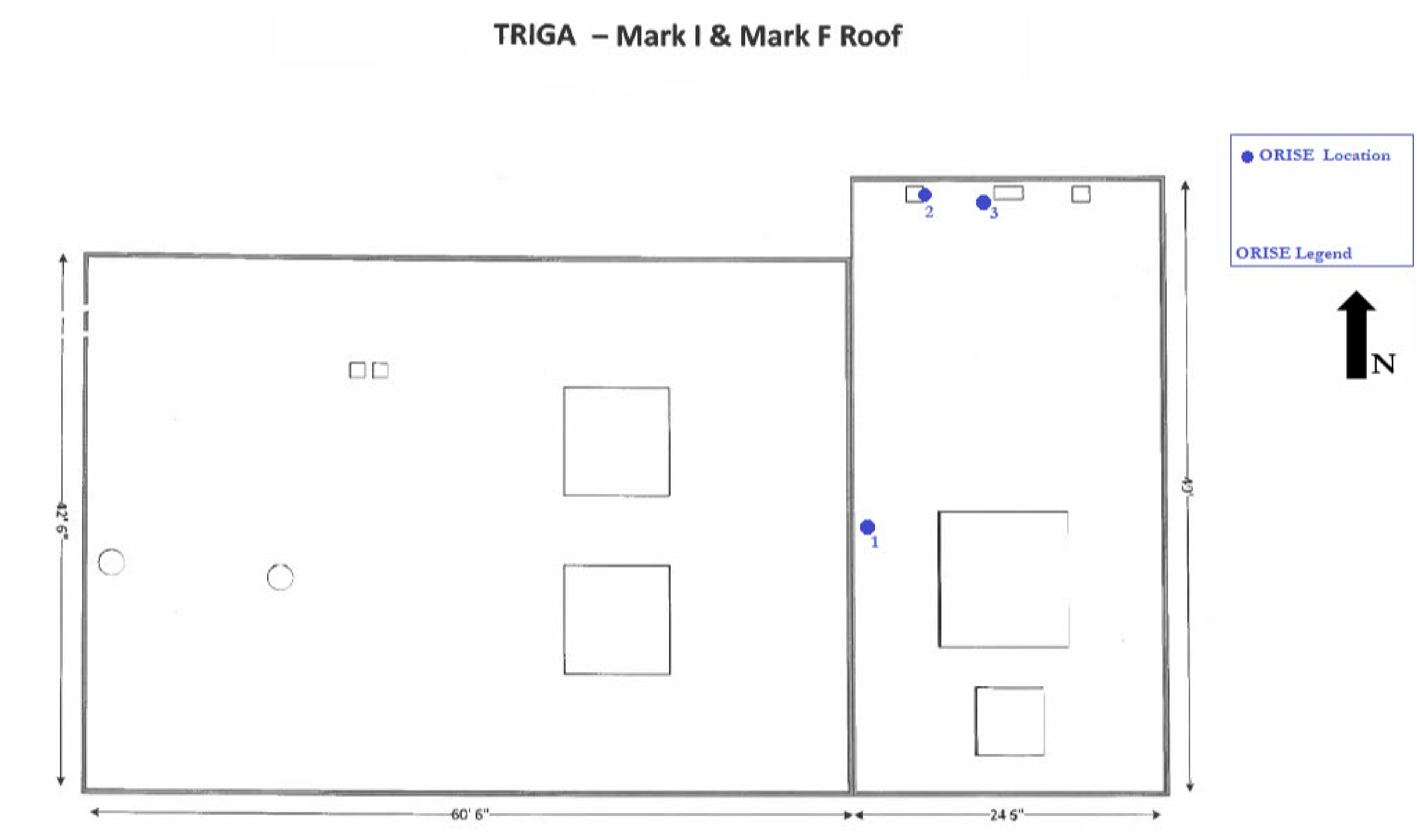

Figure A.15. Mark I and Mark F Roof (only a small area of Mark F was scanned)

APPENDIX B: DATA TABLES

Table B.1. SU 1 Mark I Reactor Pit Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Location	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0018	Concrete	South/SW Sidewall	26,000	-560	0	3	-4.1	-2.7
5339R0018a	Concrete	South/SW Sidewall	--	--	--	--	-4.1	-2.7
5339R0019	Concrete	South Sidewall	27,000	-590	0	5	0.6	-2.2
5339R0020	Concrete	NE Sidewall	26,000	-450	0	2	--	--
5339R0021	Concrete	West Sidewall	25,000	-740	0	0	--	--
5339R0043	Concrete	NE Cylinder, Between 12 & 17	22,000	-1,100	2	0	-2.7	0.8
5339R0044	Concrete	North Cylinder, Near 11	22,000	-1,200	2	0	--	--
5339R0045	Concrete	SE Cylinder, Between 8 & 13	21,000	-1,100	2	0	--	--
5339R0046	Concrete	South Cylinder, Between 3 & 4	15,000	-1,200	0	0	--	--
Minimum			15,000	-1,200	0	0	-4.1	-2.7
Maximum			27,000	-450	2	5	0.6	0.8

Table B.2. SU 2 Mark F Reactor Pit and Canal Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	**Material**	**Location**	**Gamma**	**Total Alpha-Plus-Beta Activity**	**Removable Activity**			
					Alpha	**Beta**	**H-3**	**C-14**
			cpm	**dpm/100 cm²**	**dpm/100 cm²**		**pCi/sample**	
5339R0001	Concrete	Floor	25,000	360	0	0	-2.2	-0.9
5339R0002	Concrete	SE Wall	20,000	-1,800	2	3	-1.4	1.5
5339R0003	Concrete	NW Wall	23,000	-690	0	0	0.0	1.5
5339R0004	Concrete	N Wall	25,000	-110	0	2	0.0	-2.0
5339R0005	Concrete	Canal S Wall	10,000	2,700	0	5	-6.8	-1.0
5339R0006	Concrete	Canal E Wall	11,000	12,000	0	3	--	--
5339R0007	Concrete	Canal Floor	14,000	23,000	0	0	--	--
5339R0008	Concrete	Canal Floor	14,000	440,000	0	**15**[a]	1.9	7.3
5339R0009	Metal	Canal Track Floor	13,000	410,000	0	**47**	--	--
5339R0010	Metal	Canal Track West Side	8,300	35,000	0	0	--	--
Minimum			8,300	-1,800	0	0	-6.8	-2.0
Maximum			25,000	440,000	2	47	1.9	7.3

[a] Bolded values are greater than the analytical MDC.

Table B.3. SU 3 Mark I Reactor Room Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	**Material**	**Location**	**Gamma**	**Total Alpha-Plus-Beta Activity**	**Removable Activity**			
					Alpha	**Beta**	**H-3**	**C-14**
			cpm	**dpm/100 cm²**	**dpm/100 cm²**		**pCi/sample**	
5339R0047	Metal	NE Door	11,000	2,900	4	4	--	--
5339R0048	Concrete	Wall Between 102 & 106	19,000	11,000	0	3	-8.2	-1.2
5339R0049	Concrete	Floor	16,000	-300	0	0	--	--
5339R0050	Concrete	Wall Between 102 & 106	16,000	-580	0	0	--	--
5339R0051	Concrete	Floor	16,000	-950	0	2	--	--
Minimum			11,000	-950	0	0	--	--
Maximum			19,000	11,000	4	4	-8.2	-1.2

Table B.4. SU 5 Mark F Reactor Room Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	**Material**	**Location**	**Gamma**	**Total Alpha-Plus-Beta Activity**	**Removable Activity**			
					Alpha	**Beta**	**H-3**	**C-14**
			cpm	**dpm/100 cm²**	**dpm/100 cm²**		**pCi/sample**	
5339R0011	Concrete	Floor SE Corner	20,000	3,100	0	2	-3.6	1.6
5339R0012	Concrete	South Wall	17,000	2,000	0	5	--	--
5339R0013	Metal	South Door	11,000	11,000	0	**25**[a]	--	--
5339R0014	Concrete	West Wall	14,000	-770	0	0	--	--
5339R0015	Concrete	West Wall	13,000	-120	0	3	--	--
5339R0016	Concrete	East Wall	14,000	1,600	0	2	--	--
5339R0017	Concrete/Tile	Floor SW Corner	14,000	3,800	0	0	--	--
5339R0022	Concrete	East Trench	130,000	7,100	2	0	-3.1	1.3
5339R0023	Metal Drain	East Trench	320,000	--	0	0	-4.5	0.1
Minimum			11,000	-770	0	0	-4.5	0.1
Maximum			320,000	11,000	2	5	-3.1	1.6

[a] Bolded values are greater than the analytical MDC.

Table B.5. SU 7 Soil Lab Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Location	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0052	Laminate Over Tile	Floor	16,000	650	0	0	--	--
5339R0053	Laminate Over Tile	Floor	17,000	570	2	0	--	--
Minimum			16,000	570	0	0	--	--
Maximum			17,000	650	2	0	--	--

Table B.6. SU 8 M1 Mezzanine Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Location	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0054	Cinder Block	West Window Ledge	--	6,200	0	0	--	--
5339R0055	Cinder Block	West Wall	--	2,900	2	9	--	--
5339R0056	Cinder Block	South Window Ledge	--	2,700	0	13	--	--
5339R0057	Concrete	NW Corner of Floor	--	3,600	0	3	--	--
5339R0058	Concrete	Floor	--	16,000	2	8	--	--
Minimum			--	2,700	0	0	--	--
Maximum			--	16,000	2	13	--	--

Table B.7. SU 9 M2 Mezzanine Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Location	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0059	Cinder Block	East Wall	--	-290	0	3	--	--
Minimum			--	--	--	--	--	--
Maximum			--	--	--	--	--	--

Table B.8. SU 10 TRIGA Waste Yard Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Location (E, N) (ft)	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0038	Concrete	6259749, 1907389	31,000	11,000	0	2	-6.5	0.8
5339R0039	Concrete	6259749, 1907393	19,000	680	0	6	--	--
5339R0040	Concrete	6259747, 1907396	17,000	3,800	2	4	--	--
5339R0041	Concrete	6259749, 1907392	16,000	4,200	0	2	--	--
5339R0042	Concrete	6259749, 1907387	14,000	4,100	0	5	--	--
Minimum			14,000	680	0	2	--	--
Maximum			31,000	11,000	2	6	-6.5	0.8

Table B.9. SU 11 TRIGA Front Yard Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Coordinates (E, N) (ft)	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0033	Asphalt	6259719, 1907371	17,000	-330	0	3	--	--
5339R0034	Asphalt	6259718, 1907358	17,000	1	0	0	--	--
5339R0035	Asphalt	6259736, 1907382	35,000	1,900	0	0	--	--
5339R0036	Asphalt	6259767, 1907351	20,000	-170	0	0	--	--
5339R0037	Asphalt	6259766, 1907347	19,000	-290	0	0	--	--
Minimum			17,000	-330	0	0	--	--
Maximum			35,000	1,900	0	3	--	--

Table B.10. SU 13 Room 112 Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample ID	Material	Location	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
5339R0024	Concrete	Floor	16,000	1,500	0	5	-5.5	0.4
5339R0025	Concrete	Floor	16,000	-2,100	4	0	--	--
5339R0026	Concrete	Floor	23,000	-1,100	0	0	--	--
5339R0027	Concrete	Floor	20,000	-1,200	0	0	--	--
5339R0028	Concrete	Floor	16,000	3,200	0	0	3.7	0.3
5339R0029	Cinder Block	South Wall	16,000	100	2	0	--	--
5339R0030	Cinder Block	East Wall	15,000	-2	0	0	--	--
5339R0031	Cinder Block	North Wall	15,000	-210	0	3	--	--
5339R0032	Cinder Block	South Wall	18,000	100	0	5	--	--
Minimum			15,000	-2,100	0	0	-5.5	0.3
Maximum			23,000	3,200	4	5	3.7	0.4

Table B.11. Roof Measurement Locations, Gamma Measurements, and Surface Activity Levels								
Sample Loc.	Material	Location	Gamma	Total Alpha-Plus-Beta Activity	Removable Activity			
					Alpha	Beta	H-3	C-14
			cpm	dpm/100 cm²	dpm/100 cm²		pCi/sample	
1	Rocks/ Roof	Near West Edge over Mark F Reactor Room	43,000	3,500	--	--	--	--
2	Metal	East Side of West Trunk over Mark F Reactor Room	8,800	1,800	--	--	--	--
3	Rocks/ Roof	To the West of the Middle Trunk over Mark F Reactor Room	17,000	2,600	--	--	--	--
Minimum			8,800	1,800	--	--	--	--
Maximum			43,000	3,500	--	--	--	--

Table B.12. Mark I Pit (SU 1), Front Yard (SU 11), and Back Yard (SU 12) Soil Sample Locations and Gamma Measurements					
Sample ID	**Coordinates (ft)**		**Area**	**Gamma (cpm)**	
	Easting	**Northing**		**Pre**	**Post**
Random Locations					
5339S0009	6259682	1907572	Back Yard	14,000	16,000
5339S0010	6259636	1907529	Back Yard	14,000	17,000
5339S0011	6259693	1907594	Back Yard	14,000	17,000
5339S0012	6259707	1907449	Back Yard	20,000	20,000
5339S0013	6259679	1907514	Back Yard	14,000	17,000
5339S0014	6259700	1907536	Back Yard	15,000	19,000
5339S0015	6259672	1907601	Back Yard	13,000	15,000
5339S0016	6259657	1907558	Back Yard	13,000	15,000
5339S0017	6259686	1907437	Back Yard	18,000	20,000
5339S0018	6259677	1907454	Back Yard	16,000	18,000
5339S0019	6259663	1907524	Back Yard	15,000	16,000
5339S0020	6259720	1907589	Back Yard	14,000	17,000
Mean				15,000	17,000
Minimum				13,000	15,000
Maximum				20,000	20,000
Judgmental Locations					
5339S0001	--	--	Mark I Reactor Pit	29,000	29,000
5339S0002	--	--	Mark I Reactor Pit	29,000	29,000
5339S0003	--	--	Mark I Reactor Pit	27,000	27,000
5339S0004	6259695	1907507	Back Yard	15,000	19,000
5339S0005	6259698	1907559	Back Yard	14,000	18,000
5339S0006	6259712	1907575	Back Yard	16,000	18,000
5339S0007	6259698	1907604	Back Yard	17,000	20,000
5339S0008	6259696	1907615	Back Yard	17,000	19,000
5339S0021	6259719	1907321	Front Yard	14,000	18,000
Minimum				14,000	18,000
Maximum				29,000	29,000

Table B.13. Mark I Pit (SU 1), Front Yard (SU 11), and Back Yard (SU 12) Soil Sample Concentrations																
Sample ID	**Cs-137 (pCi/g)**			**Co-60 (pCi/g)**			**Eu-152 (pCi/g)**			**Eu-154 (pCi/g)**			**SOF**	**Sr-90 (pCi/g)**		
	Conc.	**TPU**	**MDC**	**Conc.**	**TPU[a]**	**MDC[b]**	**Conc.**	**TPU**	**MDC**	**Conc.**	**TPU**	**MDC**		**Conc.**	**TPU**	**MDC**
Random Locations																
5339S0009	**0.933**[c]	0.079	0.046	-0.005	0.028	0.058	-0.037	0.062	0.147	-0.36	0.15	0.27	0.08	--	--	--
5339S0010	-0.012	0.022	0.046	0.010	0.024	0.052	0.062	0.052	0.129	-0.50	0.16	0.21	0.01	--	--	--
5339S0011	0.041	0.020	0.043	-0.033	0.036	0.065	-0.012	0.065	0.143	-0.46	0.16	0.25	0.00	--	--	--
5339S0012	**0.088**	0.021	0.034	-0.004	0.023	0.046	-0.006	0.053	0.126	-0.215	0.098	0.192	0.01	--	--	--
5339S0013	-0.004	0.033	0.069	-0.019	0.037	0.071	-0.049	0.070	0.151	-0.46	0.18	0.30	0.00	--	--	--
5339S0014	**0.066**	0.017	0.030	-0.005	0.016	0.047	0.012	0.050	0.118	-0.29	0.10	0.19	0.01	--	--	--
5339S0015	**0.131**	0.023	0.035	0.007	0.024	0.051	-0.016	0.051	0.118	-0.42	0.13	0.21	0.01	--	--	--
5339S0016	**0.216**	0.031	0.043	0.009	0.021	0.054	-0.041	0.063	0.136	-0.38	0.15	0.24	0.02	--	--	--
5339S0017	**0.069**	0.017	0.031	0.019	0.023	0.051	-0.041	0.050	0.114	-0.28	0.10	0.19	0.01	--	--	--
5339S0018	**0.077**	0.025	0.048	0.019	0.033	0.072	-0.069	0.068	0.142	-0.24	0.13	0.29	0.01	--	--	--
5339S0019	0.017	0.015	0.035	0.001	0.027	0.055	-0.013	0.056	0.130	-0.31	0.11	0.22	0.00	--	--	--
5339S0020	0.016	0.026	0.065	0.021	0.032	0.071	0.003	0.067	0.149	-0.54	0.18	0.28	0.01	--	--	--
Mean	0.1	--	--	0.002	--	--	-0.02	--	--	-0.37	--	--	0.02	--	--	--
Minimum	-0.012	--	--	-0.033	--	--	-0.069	--	--	-0.54	--	--	0.00	--	--	--
Maximum	0.216	--	--	0.021	--	--	0.062	--	--	-0.215	--	--	0.08	--	--	--

Table B.13. Mark I Pit (SU 1), Front Yard (SU 11), and Back Yard (SU 12) Soil Sample Concentrations																
Sample ID	Cs-137 (pCi/g)			Co-60 (pCi/g)			Eu-152 (pCi/g)			Eu-154 (pCi/g)			SOF	Sr-90 (pCi/g)		
	Conc.	TPU	MDC	Conc.	TPU[a]	MDC[b]	Conc.	TPU	MDC	Conc.	TPU	MDC		Conc.	TPU	MDC
Judgmental Locations																
5339S0001	-0.003	0.028	0.059	0.042	0.030	0.072	**0.394**	0.057	0.083	-0.38	0.15	0.28	0.07	--	--	--
5339S0002	0.012	0.024	0.058	**0.130**	0.030	0.052	**3.19**	0.21	0.14	0.100	0.066	0.279	0.50	--	--	--
5339S0003	-0.022	0.034	0.070	**0.085**	0.030	0.057	**1.81**	0.14	0.12	0.04	0.12	0.34	0.29	--	--	--
5339S0004	0.019	0.012	0.044	0.011	0.029	0.065	-0.028	0.070	0.152	-0.045	0.095	0.305	0.00	0.04	0.16	0.28
5339S0005	-0.014	0.023	0.050	0.007	0.025	0.054	0.021	0.054	0.129	-0.28	0.12	0.22	0.00	--	--	--
5339S0006	**0.264**	0.021	0.047	0.004	0.028	0.056	0.029	0.056	0.128	-0.26	0.11	0.24	0.03	0.15	0.15	0.26
5339S0007	**0.171**	0.025	0.032	-0.013	0.026	0.050	-0.032	0.052	0.120	-0.28	0.10	0.20	0.02	0.27	0.18	0.28
5339S0008	**0.235**	0.032	0.038	-0.021	0.016	0.062	0.034	0.061	0.147	-0.44	0.16	0.24	0.03	0.16	0.16	0.27
5339S0021	**0.083**	0.018	0.041	0.002	0.027	0.057	-0.036	0.057	0.130	-0.59	0.16	0.23	0.01	--	--	--
Minimum	-0.022	--	--	-0.021	--	--	-0.036	--	--	-0.59	--	--	0.00	0.04	--	--
Maximum	0.264	--	--	0.13	--	--	3.19	--	--	0.1	--	--	0.50	0.27	--	--

[a] Uncertainties are based on total propagated uncertainties at the 95% confidence level.
[b] MDC = minimum detectable concentration.
[c] Bolded values are greater than the analytical MDC.

Table B.14. Mark I Pit (SU 1), Front Yard (SU 11), and Back Yard (SU 12) Soil Sample NORM Concentrations												
Sample ID	K-40 (pCi/g)			Ra-226 by Pb-214 (pCi/g)			U-238 by Th-234 (pCi/g)			Th-232 by Ac-228 (pCi/g)		
	Conc.	TPU[a]	MDC[b]	Conc.[c]	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC
Random Locations												
5339S0009	**19.4**[d]	1.6	0.5	**0.877**	0.079	0.079	**1.42**	0.51	0.89	**1.16**	0.15	0.16
5339S0010	**21.1**	1.6	0.7	**0.826**	0.072	0.078	**1.86**	0.83	1.60	**1.41**	0.15	0.15
5339S0011	**19.9**	1.6	0.6	**0.847**	0.078	0.074	**1.30**	0.47	0.81	**1.10**	0.16	0.18
5339S0012	**16.3**	1.3	0.7	**0.606**	0.062	0.070	1.14	0.74	1.59	**1.08**	0.14	0.15
5339S0013	**25.1**	1.9	0.7	**1.008**	0.090	0.088	**2.16**	0.77	1.38	**1.59**	0.19	0.19
5339S0014	**21.3**	1.6	0.6	**0.963**	0.078	0.076	0.79	0.63	1.42	**1.43**	0.15	0.13
5339S0015	**20.0**	1.5	0.7	**0.754**	0.068	0.072	1.26	0.83	1.79	**1.03**	0.13	0.14
5339S0016	**19.4**	1.6	0.6	**0.750**	0.071	0.073	**1.16**	0.43	0.75	**1.00**	0.14	0.16
5339S0017	**20.5**	1.5	0.6	**1.211**	0.087	0.082	**1.40**	0.69	1.39	**1.44**	0.15	0.13
5339S0018	**21.6**	1.7	0.6	**1.023**	0.088	0.084	**1.15**	0.47	0.88	**1.62**	0.19	0.18
5339S0019	**23.8**	1.8	0.8	**0.985**	0.083	0.086	1.47	0.87	1.85	**1.70**	0.18	0.16
5339S0020	**20.9**	1.6	0.6	**0.971**	0.083	0.077	**1.06**	0.43	0.81	**1.42**	0.17	0.17
Mean	20.9	--	--	0.902	--	--	1.35	--	--	1.33	--	--
Minimum	16.3	--	--	0.606	--	--	0.79	--	--	1.00	--	--
Maximum	25.1	--	--	1.211	--	--	2.16	--	--	1.70	--	--
Judgmental Locations												
5339S0001	**18.0**	1.4	0.6	**0.838**	0.073	0.074	**1.10**	0.41	0.73	**1.30**	0.16	0.16
5339S0002	**20.8**	1.6	0.9	**0.874**	0.085	0.118	1.52	0.91	1.97	**1.35**	0.17	0.24
5339S0003	**18.1**	1.5	0.6	**0.850**	0.081	0.097	**1.28**	0.51	0.97	**1.22**	0.18	0.25
5339S0004	**23.8**	1.9	0.6	**0.997**	0.092	0.095	**1.55**	0.59	1.09	**1.44**	0.20	0.25
5339S0005	**23.6**	1.7	0.7	**0.917**	0.076	0.077	**1.93**	0.89	1.74	**1.69**	0.17	0.17
5339S0006	**18.5**	1.5	0.6	**0.769**	0.067	0.067	**1.06**	0.38	0.67	**1.09**	0.14	0.15
5339S0007	**18.4**	1.4	0.7	**0.734**	0.066	0.081	1.09	0.74	1.61	**1.25**	0.14	0.15
5339S0008	**19.0**	1.5	0.8	**0.997**	0.084	0.079	1.35	0.85	1.83	**1.69**	0.18	0.17
5339S0021	**23.0**	1.7	0.8	**0.889**	0.077	0.093	**2.51**	0.99	1.79	**1.53**	0.17	0.15
Minimum	18	--	--	0.734	--	--	1.06	--	--	1.09	--	--
Maximum	23.8	--	--	0.997	--	--	2.51	--	--	1.69	--	--

[a] Uncertainties are based on total propagated uncertainties at the 95% confidence level.
[b] MDC = minimum detectable concentration.
[c] Ra-226 values are estimated only. The Pb-214 progeny did not have sufficient ingrowth to achieve equilibrium.
[d] Bolded values are greater than the analytical MDC.

Table B.15. Concrete Measurement Locations, Gamma Measurements, and Surface Activity Levels							
Sample ID	Area	Material	Location (E, N) (ft)	Gamma		Total Alpha-Plus-Beta Activity	
				Pre-Sample	Post-Sample	Pre-Sample	Post-Sample
				cpm	cpm	dpm/100 cm²	dpm/100 cm²
5339M0001	Mark F Reactor Pit Canal	Concrete	Floor	14,000	--	440,000[a]	-430
--	Mark F Reactor Pit Canal	Concrete	N of M0001	--	--	--	180,000
--	Mark F Reactor Pit Canal	Concrete	S of M0001	--	--	--	200,000
5339M0002	Waste Yard	Concrete	6259749, 1907389	31,000	17,000	11,000	1,900
--	Waste Yard	Concrete	Beside M0002	--	21,000	--	3,800
5339M0003	Mark I Reactor Pit	Concrete	Site Loc. 19	--	--	--	--
5339M0004	Mark F Reactor Pit	Concrete	Site Loc. 25	--	--	--	--

[a] For completeness an alpha-only measurement was collected prior to sample collection and the result was 13 counts per minute (gross).

Table B.16. Concrete Sample Concentrations															
Sample ID	Cs-137 (pCi/g)			Co-60 (pCi/g)			Eu-152 (pCi/g)			EU-154 (pCi/g)			SOF	Sr-90 (pCi/g)		
	Conc.	TPU[a]	MDC[b]	Conc.	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC		Conc.	TPU	MDC
5339M0001	**26.5**[c]	1.5	0.1	**0.211**	0.053	0.081	-0.03	0.20	0.44	-0.36	0.21	0.44	2	**0.49**	0.22	0.32
5339M0002	**114.0**	6.5	0.3	-0.041	0.061	0.111	0.07	0.39	0.85	-0.33	0.24	0.47	10	**6.40**	0.54	0.33
5339M0003	0.025	0.031	0.074	-0.039	0.055	0.103	0.06	0.11	0.27	-0.56	0.21	0.42	0.01	0.07	0.16	0.28
5339M0004	**0.127**	0.032	0.055	-0.011	0.046	0.094	0.043	0.097	0.233	-0.12	0.16	0.39	0.02	0.04	0.17	0.29

[a] Uncertainties are based on total propagated uncertainties at the 95% confidence level.
[b] MDC = minimum detectable concentration.
[c] Bolded values are greater than the analytical MDC.

Table B.17. Concrete Sample NORM Concentrations												
Sample ID	K-40 (pCi/g)			Ra-226 by Pb-214 (pCi/g)			U-238 by Th-234 (pCi/g)			Th-232 by Ac-228 (pCi/g)		
	Conc.	TPU[a]	MDC[b]	Conc.[c]	TPU	MDC	Conc.	TPU	MDC	Conc.	TPU	MDC
5339M0001	**18.8**[d]	1.8	1.3	**0.859**	0.083	0.294	0.87	0.66	1.47	**1.08**	0.21	0.31
5339M0002	**16.2**	1.7	1.5	**0.77**	0.21	0.46	0.97	0.75	1.69	**0.90**	0.19	0.30
5339M0003	**19.3**	1.9	2.4	**1.12**	0.14	0.23	1.39	0.80	1.68	**1.47**	0.25	0.39
5339M0004	**7.0**	1.2	2.1	0.193	0.089	0.199	0.68	0.58	1.29	**0.37**	0.16	0.33

[a] Uncertainties are based on total propagated uncertainties at the 95% confidence level.
[b] MDC = minimum detectable concentration.
[c] Ra-226 values are estimated only. The Pb-214 progeny did not have sufficient ingrowth to achieve equilibrium.
[d] Bolded values are greater than the analytical MDC.

Table B.18. Material Specific Backgrounds[a]					
Ludlum Model 43-68 Gas Proportional Detector					
#	**Poured Unpainted Concrete**	**Unpainted Cinder Block**	**Metal**	**Asphalt**	**Ambient Air**
	Measurement (cpm)	**Measurement (cpm)**	**Measurement (cpm)**	**Measurement (cpm)**	**Measurement (cpm)**
1	688	489	293	911	334
2	649	510	276	885	351
3	749	507	290	894	331
4	731	491	307	877	299
5	704	519	281	887	322
6	756	539	261	928	317
7	732	462	282	880	329
8	712	461	251	949	327
9	725	519	283	951	335
10	729	456	286	936	322
Mean	720	500	280	910	330
Ludlum Model 44-10 Scintillation Detector					
#	**Poured Unpainted Concrete**	**Unpainted Cinder Block**	**Metal**	**Asphalt**	**Ambient Air**
	Measurement (cpm)	**Measurement (cpm)**	**Measurement (cpm)**	**Measurement (cpm)**	**Measurement (cpm)**
1	13,937	11,954	8,581	16,058	--
2	13,575	12,756	8,248	16,206	--
3	14,246	12,142	8,342	15,957	--
4	14,408	12,433	8,407	15,999	--
5	13,883	11,889	8,733	16,249	--
6	14,635	11,425	8,618	16,222	--
7	14,059	11,556	8,925	16,113	--
8	14,348	11,534	8,501	15,657	--
9	14,707	11,476	8,578	15,642	--
10	14,021	11,721	8,928	16,144	--
Mean	14,000	12,000	8,600	16,000	--

[a] The site determined background reference area was outdoors just outside the main entrance to Building 27.

APPENDIX C: 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.

C.1. SCANNING AND MEASUREMENT INSTRUMENT/ DETECTOR COMBINATIONS

C.1.1 GAMMA

Ludlum NaI 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)

C.1.2 ALPHA-PLUS-BETA

Ludlum Gas-flow Proportional Detector Model 43-68, 126 cm^2 physical area, 0.8 mg/cm^2 Mylar window
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to: Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

Ludlum Gas-flow Proportional Detector Model 43-37, 584 cm^2 physical area, 0.8 mg/cm^2 Mylar window
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to: Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)

C.2. LABORATORY ANALYTICAL INSTRUMENTATION

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

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)

Liquid Scintillation Counter
Perkin Elmer Tricarb 5110TR
(Perkin Elmer, Waltham, Massachusetts)

APPENDIX D: SURVEY AND ANALYTICAL PROCEDURES

D.1. PROJECT HEALTH AND SAFETY

ORISE performed all survey activities in accordance with the *Oak Ridge Associated Universities (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 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 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.

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 2016a)
- *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2019b)
- *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2019a)

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

D.3. SURVEY PROCEDURES

D.3.1 Surface Scans

Gamma scans were performed using a hand-held sodium iodide (NaI) scintillation detector with a 5 centimeter (cm) by 5 cm crystal. Alpha-plus-beta scans were performed using either the large-area (floor monitor) or hand-held gas proportional detector with a 0.8 mg/cm^2 window with a physical detector area of 584 cm^2 or 126 cm^2, respectively. Scans for elevated radiation were performed by passing the detector slowly over the surface. The distance between the detector and surface was maintained at a minimum. NUREG-1507, Table 6.4, provides NaI Scintillation Detector Scan MDCs for Common Radiological Contaminants (NRC 1998). For cesium-137 (Cs-137), the scan minimum detectable concentration (MDC) is 6.4 picocuries per gram (pCi/g) and, for cobalt-60 (Co-60), the MDC is 3.4 pCi/g. A specific scan MDC for the floor monitor was not determined as the instrument was used solely as a qualitative means to identify elevated radiation levels in excess of background. Identifications of elevated radiation levels that could exceed the background were determined based on an increase in the audible signal from the indicating instrument and quantitatively investigated using other hand-held instruments.

Surface scan MDCs for the hand-held gas proportional detectors were estimated using the approach described in NUREG-1507 (NRC 1998). The scan MDC is a function of many variables, including a 2-second observation interval; a specified level of performance at the first scanning state of 95% true positive and 25% false positive rate, which yields a d' value of 2.32 (NUREG-1507, Table 6.1); and a surveyor efficiency of 0.5. For survey units inside the building, the predominant radionuclide is assumed to be Co-60 and outside of the building, Cs-137. As such the total weighted efficiencies for beta are 0.10 for inside the building and 0.22 for outside the building. The scan MDC was calculated using the following equation:

$$Scan\ MDC = \frac{d' \times \sqrt{C_b \times (i/60)} \times (60/i)}{\sqrt{p} \times \mathcal{E}_t \times \frac{Probe\ Area}{100\ cm^2}}$$

Where:

d' = index of sensitivity

C_b = background (cpm)

i = observation interval (sec)

p = surveyor efficiency

$\mathcal{E}_t$ = total efficiency

The scan MDC for surveys inside the building was 2,700 disintegrations per minute (dpm)/100 cm^2 and 1,200 dpm/100 cm^2 outside the building, both based on a nominal instrument background of 330 cpm for the hand-held gas proportional detector.

D.3.2 Surface Activity Measurements

Measurements of gross alpha-plus-beta surface activity levels were performed using hand-held gas proportional detectors coupled to portable ratemeter-scalers. Count rates, which were integrated over 1 minute with the detector held in a static position, were converted to activity levels by dividing the count rate by the total static efficiency and correcting for the physical area of the detector plus background. The MDC for static surface activity measurements was calculated using the following equation:

$$MDC = \frac{3 + (4.65\sqrt{B})}{TG\mathcal{E}_{tot}}$$

Where:

B = background in time interval, T (1 min)

T = count time (min) used for field instruments

$\mathcal{E}_{tot}$ = total efficiency = $\varepsilon_i \times \varepsilon_s$ (instrument efficiency × source efficiency)

G = geometry correction factor (1.26)

The static MDC for inside the building was 720 dpm/100 cm^2 and 320 dpm/100 cm^2 outside the building, both based on a nominal instrument background of 330 cpm.

D.3.3 REMOVABLE ACTIVITY SAMPLING

Smear sampling for removable gross alpha and gross beta contamination as well as for hard-to-detect (HTD) radionuclides were obtained from independent confirmatory measurement locations. Removable activity samples were collected using numbered filter paper disks. Moderate pressure was applied to the smear and approximately 100 cm^2 of the surface was wiped. Dry smears for gross alpha and beta analysis were placed in labeled envelopes. Smears for HTD analysis were first wetted with deionized water before the surface was wiped. Wet smears were placed in glass vials with deionized water. Locations and other pertinent data were recorded and all samples were transferred under chain-of-custody.

D.3.4 SOIL SAMPLING

Soil samples (approximately 0.5 kilogram each) were collected by ORISE personnel using a clean garden trowel to transfer soil into a new sample container. The container was then labeled and security sealed in accordance with ORISE procedures. ORISE shipped samples under chain-of-custody to the ORISE laboratory for analysis.

D.4. RADIOLOGICAL ANALYSIS

D.4.1 GROSS ALPHA/BETA

Smears were counted on a low-background proportional counter for gross alpha and beta activity. The minimum detectable activity of the procedures is approximately 11 dpm/100 cm^2 for alpha and 14 dpm/100 cm^2 for beta.

D.4.2 GAMMA SPECTROSCOPY

Samples were analyzed as received, mixed, crushed, and/or homogenized as necessary, and a portion sealed in an appropriately sized Marinelli beaker. The quantity placed in the beaker was chosen to reproduce a calibrated counting geometry. Net material weights were determined, and the samples were 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 computer capabilities inherent in the analyzer system. All total absorption peaks (TAPs) associated with the radionuclides of interest were reviewed for consistency of activity. Spectra also were reviewed for other identifiable TAPs.

Table D.1. Typical MDCs Total Absorption Peak		
Radionuclide	**TAP (MeV)[a]**	**MDC (pCi/g)**
Co-60	1.332	0.06
Cs-137	0.662	0.05
Eu-152	0.344	0.10
Eu-154	0.723	0.15

D.4.3 H-3 AND C-14 ANALYSIS

Smear samples were placed into a scintillation cocktail without sample preparation and counted in a liquid scintillation analyzer. Tritium (H-3) and carbon-14 (C-14) values were calculated using the known efficiency in the appropriate energy region.

D.4.4 DETECTION LIMITS

Detection limits, referred to as MDCs, were based on 95% confidence level. Because of 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.