February 29, 2012

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**SUBJECT: DOE CONTRACT NO. DE-AC05-06OR23100
FINAL REPORT—INDEPENDENT VERIFICATION SURVEY
SUMMARY AND RESULTS FOR THE ARGONNE NATIONAL
LABORATORY BUILDING 330 PROJECT FOOTPRINT
ARGONNE, ILLINOIS DCN: 5121-SR-02-0**

Dear Ms. Heston:

Oak Ridge Associated Universities (ORAU) is pleased to provide the enclosed final report developed under the Oak Ridge Institute for Science and Education (ORISE) contract. The report details the verification activities that were performed for the Building 330 project footprint at the Argonne National Laboratory in Argonne, Illinois. These survey activities were requested and approved by the U.S. Department of Energy Argonne Site Office. The report summarizes ORISE's survey procedures and results of the verification survey activities performed onsite during the period of June 6 through June 7, 2011 and subsequent review of revised project documentation following additional remediation performed by the contractor.

Please contact me at 865.576.6659 or Tim Vitkus at 865.576.5073 should you have any questions.

Sincerely,

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Enclosure

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FINAL REPORT –INDEPENDENT VERIFICATION SURVEY SUMMARY AND RESULTS FOR THE ARGONNE NATIONAL LABORATORY BUILDING 330 PROJECT FOOTPRINT, ARGONNE, ILLINOIS

E. N. Bailey

Prepared for the
U.S. Department of Energy

ORISE

Approved for public release; further dissemination unlimited.

The Oak Ridge Institute for Science and Education (ORISE) is a U.S. Department of Energy institute focusing on scientific initiatives to research health risks from occupational hazards, assess environmental cleanup, respond to radiation medical emergencies, support national security and emergency preparedness, and educate the next generation of scientists. ORISE is managed by Oak Ridge Associated Universities.

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**FINAL REPORT—INDEPENDENT VERIFICATION SURVEY SUMMARY
AND RESULTS FOR THE ARGONNE NATIONAL LABORATORY
BUILDING 330 PROJECT FOOTPRINT
ARGONNE, ILLINOIS**

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Prepared for the
U.S. Department of Energy

FINAL REPORT

February 2012

This report is based on work performed by the Oak Ridge Institute for Science and Education under contract number DE-AC05-06OR23100 with the Department of Energy.

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**FINAL REPORT-INDEPENDENT VERIFICATION SURVEY SUMMARY
AND RESULTS FOR THE ARGONNE NATIONAL LABORATORY
BUILDING 330 PROJECT FOOTPRINT
ARGONNE, ILLINOIS**

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ACRONYMS

ANL	Argonne National Laboratory
CP-5	Chicago Pile 5
D&D	decontamination and decommissioning
DCGL _w	derived concentration guideline level
DOE	U.S. Department of Energy
FIDLER	Field Instrument for the Detection of Low Energy Radiation
FSS	final status survey
GM	Geiger-Muller
GPS	global positioning system
IEAV	Independent Environmental Assessment and Verification Program
ITP	Intercomparison Testing Program
IV	independent verification
MAPEP	Mixed Analyte Performance Evaluation Program
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
mrem	millirem
NAD	North American Datum
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRIP	NIST Radiochemistry Intercomparison Program
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
SPCS	State Plane Coordinate System
SU	survey unit

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**FINAL REPORT–INDEPENDENT VERIFICATION SURVEY SUMMARY
AND RESULTS FOR THE ARGONNE NATIONAL LABORATORY
BUILDING 330 PROJECT FOOTPRINT
ARGONNE, ILLINOIS**

1. INTRODUCTION

Argonne National Laboratory (ANL) is a direct descendant of the University of Chicago's Metallurgical Laboratory, part of the World War II Manhattan Project. Building 330, also known as the former Chicago Pile 5 (CP-5) research reactor facility, was constructed in 1951 and became operational in 1954. In 1979, the Building 330 reactor operations ceased; the fuel was removed and the facility remained idle until the early 1990s when decontamination and decommissioning (D&D) activities were initiated. The grounds surrounding Building 330 were used for the temporary storage of waste, to include potentially contaminated equipment and materials that primarily originated from Building 330. Characterization of the building and adjacent grounds began in the early 1990s. During that time period, contaminated systems, components, structures, and soils were removed or remediated in accordance with the U.S. Department of Energy (DOE) Order 5400.5 (DOE 1993).

Clauss Construction began demolition of the structure in early 2010. When building debris removal was nearing completion, elevated residual activity was detected in the soil associated with the building's former E-wing. Additional characterization and remediation of the E-wing soil area was performed. The major contaminant identified during demolition and final status survey (FSS) activities was Cs-137. FSS activities were completed for the building footprint, remaining A-wing concrete structure, and the piles of soil intended for excavation backfill. The portion of the A-wing west wall and pad were left in place to prevent the possibility of damage to the active steam line on the surface of the adjacent roadway just west of the wall.

Based on guidance in a DOE Environmental Management Memorandum, independent verification (IV) shall be performed at all DOE cleanup sites (DOE 2011a). At the request of the DOE Argonne Site Office, Oak Ridge Associated Universities' (ORAU) Independent Environmental Assessment and Verification (IEAV) Program conducted onsite verification activities of the Building 330 project footprint during the period of June 6 through June 7, 2011 under the Oak Ridge Institute for Science and Education (ORISE) contract.

2. SITE DESCRIPTION

ANL is located in the city of Argonne, Illinois, approximately 25 miles southwest of Chicago. ANL occupies 1,500 wooded acres, and the land area surrounding the site consists of residential, commercial, and industrial properties (Fig. A-1). The Building 330 site is located in the southwest portion of the industrialized area of ANL (Fig. A-2).

3. OBJECTIVES

The objectives of the verification activities were to provide independent contractor document and field data reviews in evaluating the adequacy and accuracy of the contractor's procedures and FSS documentation and results. The onsite verification survey was performed in order to generate independent radiological scan and sample data to verify that all remedial actions were effective and that the final conditions meet the applicable release criteria.

4. DOCUMENT REVIEW

Prior to onsite activities, ORISE was tasked with reviewing the site's FSS plans, Radiological Sample and Analysis Plan, and derived concentration guideline level (DCGL_w) development documents for the Building 330 project. In January 2012, following the ORISE site visit, the revised FSS plan for the Building 330 D&D project was provided for review following additional remediation in survey units (SUs) 2, 3, and 9 (DCSI 2011c). Refer to Section 8.0, Findings and Results, for additional details.

5. PROCEDURES

Following the completion of the initial FSS activities, the ORISE survey team visited ANL from June 6 through 7, 2011 to perform visual inspections, verification measurements, and sample collection at the Building 330 project. The verification survey activities were conducted in accordance with a project-specific plan, the ORISE/IEAV *Survey Procedures Manual* and *Quality Program Manual* (ORISE 2011a, ORISE 2008, and ORAU 2011a). Questions and concerns were brought to the immediate attention of the DOE and/or other project representatives and are also noted in the Findings and Results section of this report.

The SUs were classified by the contractor, based on contamination potential, as either Class 1, 2, or 3 in accordance with the *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (NRC 2000). A description of each class designation follows:

Class 1: Buildings or land areas that have a significant potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) that exceeds the expected $DCGL_{\text{W}}$ value.

Class 2: Buildings or land areas, often contiguous to Class 1 areas that have a potential for radioactive contamination, but at levels less than the expected $DCGL_{\text{W}}$.

Class 3: Remaining impacted buildings and land areas that are not expected to contain residual contamination or are expected to contain levels of residual contamination at a small fraction of the $DCGL_{\text{W}}$.

Verification survey activities were conducted within each of the nine FSS units. Additionally, structural verification surveys were also performed in SU 7, the remaining A-wing concrete structure.

5.1 REFERENCE SYSTEM

Survey scan data were referenced to global positioning system (GPS) coordinates. The coordinate reference system used for the verification survey was State Plane Coordinate System (SPCS) Illinois East Feet NAD 1983, with units represented in meters. Positional accuracy was within 0.5 meters at the 95th percentile.

5.2 SURFACE SCANS

The FSS results and the project's remediation history were considered when determining the level of effort required to adequately represent each SU. Additionally, the surface scan coverage of each SU varied based on the SU classification, size, and the accessibility of the area. Very limited scans were performed on the excavated sidewalls due to the difficulty in traversing the steep inclines. Although the contractor was actively pumping water, the deepest excavations were inaccessible due to rainfall and/or subsurface seeps. Surface scans for gamma radiation were performed using 2 inch $\times$ 2 inch sodium iodide, thallium-activated [NaI(Tl)], and Field Instrument for the Detection of Low Energy

Radiation (FIDLER) detectors coupled to ratemeter-scalers with audible indicators. ORISE field personnel relied on the audio output to identify any locations of elevated direct gamma radiation that might suggest the presence of residual contamination. Gamma detectors were also coupled to GPS systems that enable real-time gamma count rate and position data capture. Locations of elevated direct gamma radiation were marked with flags for further investigation during the sampling phase. Scan coverage is shown on Figs. A-3 and A-4.

Additionally, a hand-held gas proportional detector coupled to a ratemeter-scaler with audible indicator was used to scan the remaining A-wing concrete structure for alpha-plus-beta radiation.

5.3 SOIL SAMPLING

Approximately 0.5 kilogram of surface (0 to 15 cm) soil material was collected at each sample location. At three of the eighteen sample locations, a second sample was collected because the gamma count rate increased significantly after collecting the initial surface sample. Collected samples were placed in plastic sample bags, sealed, and labeled in accordance with ORISE survey procedures. Fig. A-5 shows the sampled locations.

6. SAMPLE ANALYSIS AND DATA INTERPRETATION

Soil samples and data were returned to the ORISE facility in Oak Ridge, Tennessee for analysis and interpretation. All sample analyses were performed in accordance with the ORISE Laboratory Procedures Manual (ORISE 2011b). Many of the soil samples contained large pieces of material, possibly natural rocks or demolition-type debris. The ORISE Project Manager requested the laboratory staff to sieve the samples and analyze the $< \frac{1}{4}$ inch sample material via gamma spectroscopy in accordance with the standard laboratory procedure, but to also retain the $> \frac{1}{4}$ inch sample material and then screen it for elevated radiation levels using a Geiger-Muller (GM) detector. Five of the 18 samples contained $> \frac{1}{4}$ inch material exhibiting elevated radiation levels. This material was isolated and counted separately via gamma spectroscopy using the laboratory's "AF10" geometry (i.e., counting the sample 10 cm from the detector).

Additionally, 10 of the 18 samples were selected for full-suite analysis based on the project-specific soil DCGL_{ws} (Table1). The selection process for full-suite analysis was based on observations made in the field while sampling, as well as Cs-137 being the only contaminant of note in the FSS data

provided to ORISE. Pu-239/240 and Am-241 concentrations were determined by alpha spectroscopy using total sample dissolution and extraction chromatography. Sr-90 was quantified by radiochemical separation and counted on a low-background proportional counter. Tc-99 was quantified by radiochemical separation using extraction chromatography and counted by liquid scintillation. C-14 analysis was performed using a material oxidizer and counted by liquid scintillation. Radionuclide concentrations are reported in Tables B-1 and B-2.

7. APPLICABLE SITE GUIDELINES

The primary contaminants of concern for the Building 330 footprint are beta-gamma emitters and fission and activation products resulting from reactor operations and other research activities. Alpha contamination, specifically Pu-239, has also been identified. DCGL_ws for soil were developed for the significant radionuclide contributors listed in Table 1. The soil contaminants of concern were selected based on soil characterization following the demolition and removal of Building 330. Soil guidelines were calculated using a 25 millirem/year limit for all scenarios (DOE 2011b).

Table 1. Applicable Derived Concentration Guideline Levels (DCGL_ws) for Soils	
Contaminants of Concern	DCGL_w (pCi/g)
Am-241	2.4E+02
Ba-133	6.5E+01
C-14	5.4E+02
Cs-137	3.6E+01
Pu-238	4.3E+02
Pu-239	1.8E+02
Sr-90	6.0E+01
Tc-99	4.6E+02

The soil DCGL_ws represent the average levels that can remain in a SU. If multiple radionuclide contaminants are present, the sum of the ratios of the contaminant concentrations to their respective DCGL_ws must be < 1 (DCSI 2011a).

During the initial planning phase of the Building 330 demolition project, the surface activity criteria from DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, was applicable to the remaining A-wing wall structure (DOE 1993 and 1995). Although DOE Order 5400.5 has been

superseded by DOE Order 458.1, the surface activity guidelines from DOE Order 5400.5 remain applicable until they are replaced or revised by DOE (DOE 2011a). The most conservative of the applicable DOE residual surface activity guidelines were selected based on the potential contaminants from several of the guideline categories found in DOE Order 5400.5.

The applicable alpha surface activity guidelines (based on plutonium as the contaminant) are:

Total Activity

100 dpm/100 cm², averaged over a 1 m² area

300 dpm/100 cm², maximum in a 100 cm² area

Removable Activity

20 dpm/100 cm²

The applicable beta surface activity guidelines (based on Sr-90 as the contaminant) are:

Total Activity

1,000 dpm/100 cm², averaged over a 1 m² area

3,000 dpm/100 cm², maximum in a 100 cm² area

Removable Activity

200 dpm/100 cm²

8. FINDINGS AND RESULTS

The results for each of the verification activities are discussed in the following subsections.

8.1 DOCUMENT REVIEW

DOE initially requested that ORISE review the following documents in preparation for a site visit that was originally planned for December 2010:

Radiological Final Status Survey Plan Excavation of Building 330 Footprint Argonne National Laboratory, Lemont Illinois, Revision 0. Dated November 5, 2010 as final although cover page is not signed.

Radiological Sample & Analysis Plan Building 330 Footprint, prepared by DCSI; November 5, 2010.

Radiological Criteria for Building 330 Excavation - The development of Derived Concentration Guidelines (DCGL) using RESRAD analysis for the BLD-330 Demolition Project, Argonne, Illinois, Revision 1, Draft E. prepared by DCSI; November 8, 2010.

ORISE submitted a document review letter to DOE on November 12, 2010 (ORISE 2010) which included several significant comments regarding the development of the DCGL_ws and the characterization and FSS approach. This, combined with the contractor's discovery of soil contamination associated with the E-wing, resulted in the development of new and/or revised documents. DOE then provided ORISE with a revised *Radiological Final Status Survey Plan* (DCSI 2011a), *Characterization Sample & Analysis Plan for the E-wing* (DCSI 2011b), and the new DCGL_w development document (ANL 2011) for review. The only additional comment ORISE provided was to note the Sr-90 DCGL_w was incorrect in the revised FSS plan.

Following the site visit and issuance of the ORISE draft report, ANL determined that additional remediation would be performed in SUs 2, 3, and 9. The contractor subsequently issued a revised FSS plan for review which contained an addendum from the FSS Report describing the evaluation of areas exhibiting elevated activity in SUs 2, 3, and 9 (DCSI 2011c). No formal comment response was generated for the document.

8.2 SURFACE SCANS

Surface scan results for the structural and soil SUs are discussed below.

8.2.1 Structural Survey Unit Scans

The only structure that remains is the A-wing concrete structure. The major contaminant identified during demolition and FSS activities was Cs-137. ORISE opted to only perform alpha-plus-beta scans on vertical wall surfaces to identify locations of elevated surface activity based on a review of the FSS data collected by the contractor for SU 7. Investigation levels for structural scans were based on the applicable alpha and beta surface activity guidelines as stated in Section 7. No locations exhibiting elevated radiation levels were detected during the scans; therefore, no direct measurements were collected. Medium-density gamma scans were performed over the remaining concrete pad.

8.2.2 Soil Survey Unit Scans

Numerous locations exhibiting elevated direct gamma radiation were identified during the scanning phase and marked with flags for further investigation during the sampling phase. Gamma radiation levels ranged from less than 1,700 cpm to 12,000 cpm using FIDLER detectors, and from less than 6,500 to 43,000 cpm using 2×2 NaI detectors (Figs. A-3 and A-4). The elevated activity identified with the 2×2 NaI detectors corresponded to the same locations identified by the FIDLER detectors.

8.3 SOIL SAMPLING

The numerous locations identified during the scanning phase were evaluated. However, due to the number of locations flagged and time constraints, ORISE opted to sample only the highest locations based on the gamma surface scan results. Eighteen judgmental soil samples were collected from fifteen locations, as described previously under the “Procedures” section and as shown in Fig. A-5. All samples collected fell within the boundaries of either SU 2, 3, or 9. SUs 2 and 3 included the deepest part of the excavation. SU 9 is a soil pile that consists of excavated material from the building footprint either to be used as backfill or packaged and disposed of.

Gamma spectroscopy sample results for Cs-137 and Ba-133 are provided in Table B-1. The gamma spectra were also reviewed for other identifiable photopeaks. Concentrations of Cs-137, the primary contaminant identified, range from 1.34 to 101 pCi/g for the < ¼ inch material. The > ¼ inch material that was isolated and counted separately for five samples are also provided in Table B-1 and range from 4,600 to 57,400 pCi/g. It is important to note the Cs-137 values for the > ¼ inch material are only being reported as estimated values due to geometry issues associated with counting this material; however, it is evident this material exceeds the Cs-137 DCGI_w by an order of magnitude or greater. The shading in Table B-1 indicates all the values in excess of the 36.0 pCi/g Cs-137 DCGI_w when factoring in the uncertainty associated with the sample count. The darker borders around the three pairs of samples represent the three locations in which a consecutive second sample was collected. This occurred after ORISE observed the count rate increase significantly following the collection of the first sample. In each instance, the second sample contained Cs-137 concentrations in excess of the DCGI_w. A total of eight samples from the fifteen locations sampled exceed the Cs-137 DCGI_w.

Following the collection of sample number 5121S0008, ORISE uncovered a large rock exhibiting extremely elevated gamma radiation levels approaching 800,000 cpm using a 2×2 NaI detector. Clauss was notified and assumed custody of the rock by removing it from the excavation. ORISE collected a second sample (5121S0009) following the removal of the rock; the second sample measured in excess of the Cs-137 DCGL_w at 101 pCi/g.

In the site closeout meeting, the contractor asked ORISE if our laboratory would determine if the sample material collected was volumetrically contaminated or surface contaminated. The ORISE laboratory staff performed limited tests that indicated the > 1/4 in material was volumetrically contaminated. The laboratory staff size-reduced some of the larger pieces of material and the resulting smaller pieces exhibited elevated activity using a GM detector. They also rinsed a few of these larger pieces with acid solution trying to remove any surface contamination and although a few counts were observed in the solution, the larger pieces of material that had been rinsed still showed the majority of the activity.

9. SUMMARY

At the request of the DOE Argonne Site Office, ORISE conducted onsite verification activities of the Building 330 project footprint during the period of June 6 through June 7, 2011. The verification activities included technical reviews of project documents, visual inspections, radiation surface scans, and sampling and analysis. The draft verification report was issued in July 2011 with findings and recommendations. The contractor performed additional evaluations and remediation, and the revised FSS plan was provided to ORISE for review.

9.1 DRAFT REPORT CONCLUSIONS

ORISE is of the opinion, based on the independent measurement and sample results obtained during verification activities of the Building 330 project footprint, that SU 1 and SUs 4 through 8 meet the applicable DCGL_ws for the site to be released without radiological restrictions. However, the ORISE analytical results of samples obtained within SUs 2, 3, and 9, demonstrate that multiple locations do not meet the Cs-137 DCGL_w. As mentioned previously, numerous locations were flagged by ORISE but not sampled due to time constraints. Based on the ORISE gamma scan results, analytical results, and the history and origin of the SU 9 soil pile, ORISE does not recommend this material be considered further for use as backfill material. Additionally, three

samples were collected in SU 2 and two of these exceeded the Cs-137 $DCGL_w$. Eight samples were collected within SU 3 from six locations and four of these samples exceeded the Cs-137 $DCGL_w$. After discovering two locations with residual Cs-137 contamination in excess of the $DCGL_w$ at depth within SU 3, the surface gamma scans, and the Cs-137 sample results for SUs 2 and 3, ORISE opined that SUs 2 and 3 were not ready for FSS activities. Instead, these SUs either required additional remediation or should be released with radiological restrictions after further investigation and documentation.

It is also important to note the ORISE sampling process was questioned by contractor employees during the verification activities as to why a second sample was being collected (essentially “chasing” the contamination) as this procedure varied from the contractor’s FSS procedure. The ORISE Project Manager was informed the procedure for collecting FSS samples (even judgmental samples identified by scans) was to only sample the surface material without any further investigation. However, based on the ORISE findings it is very probable that some of the judgmental locations identified by the contractor during FSS scans in SUs, 2, 3, and 9 contained residual contamination at depth in excess of the Cs-137 $DCGL_w$ —all three locations in which ORISE collected a second sample at depth exceeded the Cs-137 $DCGL_w$. ORISE suggested that the contractor adapt a screening process for judgmental locations similar to the ORISE process whereby further investigation/sampling commences when the count rate increases significantly after the first sample collection. The ORISE PM explained that verification activities are clearly different from FSS activities and the purpose of verification is to identify any residual contamination in excess of the guidelines that can be detected by scans (even at depth) and verified by laboratory analysis. These are common procedures implemented at decommissioning projects that ensure all contaminated media are identified and addressed.

9.2 FINAL REPORT CONCLUSIONS

When the project DCGLs were initially developed by ANL, only average $DCGL_w$ values were generated for data comparison; $DCGL_{EMCs}$ were not generated to enable the evaluation of smaller areas of elevated activity “hotspots” with respect to the 25 millirem/year limit. Following the issuance of the draft verification report, the contractor generated $DCGL_{EMC}$ values for Cs-137 and also performed additional remediation in SUs 2, 3, and 9. The remediation consisted of removing all the $> \frac{1}{4}$ inch material (consisting of some contaminated building debris) from SU 9. This was done

using an excavator and commercial rock screening device followed by walkovers of the remaining < 1/4 inch material and performing additional remediation as necessary based on the walkover results. Additional building debris was also removed from SUs 2 and 3 on a smaller scale. The remaining < 1/4 inch material from SU 9 was used as backfill in the bottom of the excavation; this eliminated SU 9 altogether. Following the additional “hotspot” evaluations, remediation by the contractor, and subsequent review of the revised project documentation, ORISE is of the opinion the appropriate actions were taken in SUs 2, 3, and 9 to ensure the applicable DCGLs were met for the site to be released without radiological restrictions.

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APPENDIX A

FIGURES

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Fig. A-1. Location of the Argonne National Laboratory

Fig. A-2. Argonne National Laboratory, Building 330 Footprint Plot Plan

Fig. A-3. FIDLER Gamma Scan Results for the Building 330 Project Footprint

Fig. A-4. Sodium Iodide (2×2) Gamma Scan Results for the Building 330 Project Footprint

Fig. A-5. Soil Sample Locations for the Building 330 Project Footprint

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APPENDIX B

TABLES

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Table B-1. Radionuclide Concentrations in Soil via Gamma Spectroscopy			
ORISE Sample ID ^a	Radionuclide Concentrations, TPU ^b , and MDC ^c (pCi/g)		
	< 1/4 inch material		> 1/4 inch material ^d
	Ba-133	Cs-137	Cs-137
5121S0001	-0.44 ± 0.10, 0.16	17.1 ± 1.2, 0.1	n/a ^e
5121S0002	-0.41 ± 0.12, 0.20	17.3 ± 1.5, 0.1	n/a
5121S0003	-0.16 ± 0.10, 0.19	62.6 ± 5.8, 0.2	n/a
5121S0004	0.02 ± 0.08, 0.13	20.7 ± 1.9, 0.1	n/a
5121S0005	-0.48 ± 0.11, 0.17	23.5 ± 1.6, 0.1	n/a
5121S0006	-1.04 ± 0.18, 0.25	35.4 ± 3.1, 0.1	57,400 ± 4,400, 150
5121S0007	-0.05 ± 0.06, 0.12	14.4 ± 1.3, 0.1	n/a
5121S0008	0.02 ± 0.04, 0.09	1.34 ± 0.14, 0.04	n/a
5121S0009	-0.65 ± 0.17, 0.29	101.2 ± 6.9, 0.2	n/a
5121S0010	-0.49 ± 0.13, 0.22	24.8 ± 2.2, 0.1	n/a
5121S0011	-0.04 ± 0.07, 0.15	35.8 ± 3.3, 0.1	n/a
5121S0012	0.00 ^f ± 0.08, 0.13	22.8 ± 2.1, 0.1	10,420 ± 820, 83
5121S0013	-0.57 ± 0.13, 0.18	22.6 ± 1.6, 0.1	9,610 ± 740, 49
5121S0014	-0.20 ± 0.09, 0.21	15.9 ± 1.4, 0.1	n/a
5121S0015	-0.16 ± 0.07, 0.14	22.5 ± 2.1, 0.1	n/a
5121S0016	0.02 ± 0.07, 0.12	17.3 ± 1.6, 0.1	11,500 ± 880, 41
5121S0017	-0.50 ± 0.16, 0.28	80.2 ± 5.4, 0.2	4,650 ± 360, 17
5121S0018	-0.41 ± 0.14, 0.24	31.5 ± 2.7, 0.2	n/a

The shading indicates all the values in excess of the 36.0 pCi/g Cs-137 DCGL_W when factoring in the uncertainty associated with the sample count. The darker borders around the three pairs of samples represent the three locations in which a consecutive second sample was collected.

^aRefer to Fig. A-5.

^bUncertainties are at the 95% confidence level based on total propagated uncertainties (TPUs).

^cThe MDCs are after the comma.

^dThe Cs-137 values are considered estimated values for the > ¼ inch material due to geometry issues associated in counting the sample material.

^eNo > ¼ inch material was identified exhibiting elevated radiation levels; therefore analysis was not performed.

^fZero values are due to rounding or the sample and background counts being equal.

Table B-2. Radionuclide Concentrations in Soil

ORISE Sample ID ^a	Radionuclide Concentrations, TPUs ^b , and MDCs ^c (pCi/g)					
	Am-241	C-14	Pu-238	Pu-239/240	Sr-90	Tc-99
5121S0001	-0.01 ± 0.02, 0.06	0.9 ± 1.3, 2.1	-0.02 ± 0.03, 0.07	0.00 ^d ± 0.01, 0.03	0.00 ± 0.38, 0.69	0.14 ± 0.14, 0.24
5121S0003	0.01 ± 0.02, 0.05	3.5 ± 1.2, 1.9	-0.01 ± 0.02, 0.04	0.00 ± 0.01, 0.03	0.48 ± 0.42, 0.69	0.22 ± 0.16, 0.25
5121S0005	-0.03 ± 0.02, 0.07	0.2 ± 1.0, 1.7	0.00 ± 0.02, 0.05	-0.01 ± 0.01, 0.04	0.33 ± 0.43, 0.73	0.20 ± 0.15, 0.25
5121S0007	-0.01 ± 0.02, 0.06	-0.4 ± 1.2, 2.0	0.01 ± 0.02, 0.04	0.00 ± 0.01, 0.02	0.50 ± 0.45, 0.74	0.20 ± 0.16, 0.25
5121S0009	0.01 ± 0.03, 0.07	-0.7 ± 1.2, 2.0	0.01 ± 0.01, 0.03	0.00 ± 0.01, 0.01	0.49 ± 0.44, 0.72	0.15 ± 0.15, 0.25
5121S0011	0.00 ± 0.03, 0.07	-0.5 ± 1.1, 2.0	0.00 ± 0.01, 0.04	0.00 ± 0.01, 0.02	0.12 ± 0.43, 0.75	0.17 ± 0.15, 0.24
5121S0012	0.00 ± 0.03, 0.08	-1.0 ± 1.2, 2.1	0.02 ± 0.02, 0.05	0.01 ± 0.01, 0.03	0.12 ± 0.38, 0.66	0.15 ± 0.15, 0.25
5121S0013	0.03 ± 0.02, 0.05	0.2 ± 1.1, 1.8	0.00 ± 0.01, 0.03	0.01 ± 0.01, 0.01	0.22 ± 0.41, 0.71	0.23 ± 0.16, 0.25
5121S0015	-0.02 ± 0.03, 0.08	-0.3 ± 1.1, 1.9	0.01 ± 0.01, 0.02	0.00 ± 0.01, 0.03	-0.05 ± 0.40, 0.73	0.17 ± 0.15, 0.25
5121S0017	0.00 ± 0.03, 0.07	-0.5 ± 1.1, 2.0	0.00 ± 0.01, 0.04	0.00 ± 0.00, 0.02	1.46 ± 0.47, 0.67	0.34 ± 0.16, 0.25

^aRefer to Fig. A-5.^bUncertainties are at the 95% confidence level based on total propagated uncertainties.^cThe MDCs are after the comma.^dZero values are due to rounding or the sample and background counts being equal.

APPENDIX C

MAJOR INSTRUMENTATION

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

C.1 SCANNING AND MEASUREMENT INSTRUMENT/DETECTOR COMBINATIONS

C.1.1 GAMMA

Ludlum NaI Scintillation Detector Model 44-10, Crystal: 5.1 cm x 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, TX)

coupled to:

Ludlum Ratemeter-scaler Model 2221

(Ludlum Measurements, Inc., Sweetwater, TX)

coupled to:

Trimble GeoXH Receiver and Data Logger (Trimble Navigation Limited, Sunnyvale, CA)

Bicron Field Instrument Detector for Low Energy Radiation (FIDLER) Model G5

(Bicron Corporation, Newburg, OH)

coupled to

Ludlum Ratemeter-scaler Model 2221

(Ludlum Measurements, Inc., Sweetwater, TX)

coupled to

Trimble GeoXH Receiver and Data Logger (Trimble Navigation Limited, Sunnyvale, CA)

C.1.2 ALPHA PLUS BETA

Ludlum Gas Proportional Detector Model 43-68, 126cm² physical area

coupled to:

Ludlum Ratemeter-scaler Model 2221

(Ludlum Measurements, Inc., Sweetwater, TX)

C.2 LABORATORY ANALYTICAL INSTRUMENTATION

High-Purity, Extended Range Intrinsic Detector

CANBERRA/Tennelec Model No: ERVDS30-25195

(Canberra, Meriden, CT)

Used in conjunction with:

Lead Shield Model G-11

(Nuclear Lead, Oak Ridge, TN) and

Multichannel Analyzer

Canberra's Apex Gamma Software

Dell Workstation

(Canberra, Meriden, CT)

LABORATORY ANALYTICAL INSTRUMENTATION (CONTINUED)

High-Purity, Extended Range Intrinsic Detector

Model No. GMX-45200-5

(AMETEK/ORTEC, Oak Ridge, TN)

used in conjunction with:

Lead Shield Model SPG-16-K8

(Nuclear Data)

Multichannel Analyzer

Canberra's Apex Gamma Software

Dell Workstation

(Canberra, Meriden, CT)

High-Purity Germanium Detector

Model GMX-30-P4, 30% Eff.

(AMETEK/ORTEC, Oak Ridge, TN)

Used in conjunction with:

Lead Shield Model G-16

(Gamma Products, Palos Hills, IL) and

Multichannel Analyzer

Canberra's Apex Gamma Software

Dell Workstation

(Canberra, Meriden, CT)

Alpha Spectrometry System

Tennelec Model 256

(Canberra, Meriden, CT)

Used in conjunction with:

Ion Implanted Detectors and

Multichannel Analyzer

Canberra Apex Alpha Software

Dell Workstation

(Canberra, Meriden, CT)

Alpha Spectrometry System

Canberra Model 7401VR

(Canberra, Meriden, CT)

Used in conjunction with:

Ion Implanted Detectors and

Multichannel Analyzer

Canberra Apex Alpha Software

Dell Workstation

(Canberra, Meriden, CT)

Low-Background Gas Proportional Counter

Model LB-5100-W

(Tennelec/Canberra, Meriden, CT)

Tri-Carb Liquid Scintillation Analyzer

Model 3100

(Packard Instrument Co., Meriden, CT)

Tri-Carb Liquid Scintillation Analyzer
Model 3100
(Packard Instrument Co., Meriden, CT)

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APPENDIX D
SURVEY AND ANALYTICAL PROCEDURES

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D.1 PROJECT HEALTH AND SAFETY

The proposed survey and sampling procedures were evaluated to ensure that any hazards inherent to the procedures themselves were addressed in current job hazard analyses. Prior to on-site activities, a pre-job integrated safety management checklist was completed and discussed with field personnel. Additionally, upon arrival at the site, representatives from ANL provided site-specific safety awareness training. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. All survey and laboratory activities were conducted in accordance with Oak Ridge Institute for Science and Education (ORISE) health and safety and radiation protection procedures (ORAU 2011b and c).

D.2 CALIBRATION AND QUALITY ASSURANCE

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

Analytical and field survey activities were conducted in accordance with procedures from the following documents of the Independent Environmental Assessment and Verification Program:

- Survey Procedures Manual (ORISE 2008)
- Laboratory Procedures Manual (ORISE 2011b)
- Quality Program Manual (ORAU 2011a)

The procedures contained in these manuals were developed to meet the requirements of DOE Order 414.1C and the U.S. Nuclear Regulatory Commission (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 (MAPEP), NIST Radiochemistry Intercomparison Program (NRIP), and Intercomparison Testing Program (ITP) 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

Scans for elevated gamma radiation were performed by passing the detector slowly over the surface. The distance between the detector and surface was maintained at a minimum. The 2x2 NaI scintillation detector and a low-energy photon FIDLER detectors (detector area 127 cm²) were coupled to global positioning system (GPS) units with data loggers enabling real-time recording in one-second intervals of both geographic position and the gamma count rate. Position and gamma count rate data files were transferred to a computer system, positions differentially corrected, and the results plotted on geo-referenced aerial photographs or plot plans of the facility. Positional accuracy was within 0.5 meters at the 95th percentile.

Specific scan minimum detectable concentrations (MDCs) for the detectors were not determined as the instruments were used solely as a qualitative means to identify elevated gamma radiation levels in excess of background. The identification of elevated radiation levels that could exceed the site criteria were determined based on an increase in the audible signal from the indicating instrument. However, the scan MDC for the 2x2 NaI scintillation detectors is approximately 6.4 pCi/g for cesium-137, as provided in NUREG-1507.

Surface scans of the remaining A-wing concrete structure were performed by passing the detector slowly over the surface while maintaining the distance between the detector and the surface at a minimum. Small area (126 cm²) gas proportional hand-held detectors (with a 0.8 mg/cm² window) were used. Identification of elevated radiation levels is based on increases in the audible signal from the recording and/or indicating instrument.

D.3.2 SOIL SAMPLING

Approximately 0.5 kilogram of soil was collected at each sample location. At three of the eighteen sample locations a second sample was collected because the count rate increased significantly after collecting the first sample. ORISE noted there was a significant amount of rock and demolition-type debris in many of the samples. Samples were placed in a plastic bag, sealed, and labeled in accordance with ORISE survey procedures.

D.3.3 SURFACE ACTIVITY MEASUREMENTS

No locations exhibiting elevated radiation levels were detected during surface scans of the remaining A-wing concrete structure; therefore, no surface activity measurements were collected.

D.3.4 REMOVABLE ACTIVITY MEASUREMENTS

No locations exhibiting elevated radiation levels were detected during surface scans of the remaining A-wing concrete structure; therefore, no removable activity measurements (smears) were collected.

D.4 RADIOLOGICAL ANALYSIS

D.4.1 GAMMA SPECTROSCOPY

Samples were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in a 0.5-liter Marinelli beaker or other appropriate container. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry. Net material weights were determined and the samples counted using intrinsic germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. All Total Absorption Peaks (TAP) associated with the ROCs were reviewed for consistency of activity. TAPs used for determining the activities of ROCs and the typical associated MDCs for a one-hour count time were:

Table D-1. MDC Derived from Total Absorption Peak		
Radionuclide	TAP (MeV)	MDC (pCi/g)
Ba-133	0.356	0.20
Cs-137	0.662	0.12

Spectra were also reviewed for other identifiable TAPs.

D.4.2 ALPHA SPECTROSCOPY

Samples were homogenized and dissolved by a combination of potassium hydrogen fluoride and pyrosulfate fusions. The fusion cakes were dissolved and all alpha emitters are co-precipitated on barium sulfate. The barium sulfate is re-dissolved and the contaminant of concern was separated from the other actinides by extraction chromatography utilizing Eichrom Technologies resins, co-precipitated with cerium fluoride, and analyzed using ion implanted detectors, alpha spectrometers, and multichannel analyzers.

The alpha spectroscopy detector system calculates a MDC for each individual isotope per sample based on the detector background, counting efficiency, chemical yield, and sample quantity. TAPs used for determining the activities of ROCs and the typical associated MDCs for a one-hour count time were:

Table D-2. MDC Derived from Total Absorption Peak		
Radionuclide	TAP (MeV)	MDC (pCi/g)
Am-241	5.275	0.05
Pu-238	5.499	0.04
Pu-239/240	5.155	0.02

D.4.3 Sr-90 ANALYSIS

Sr-90 concentrations were quantified by total sample dissolution followed by radiochemical separation and counted on a low background proportional counter.

Samples were homogenized and dissolved by a combination of potassium hydrogen fluoride and pyrosulfate fusions. The fusion cakes were dissolved and strontium is coprecipitated on lead sulfate. The strontium was separated from residual calcium and lead by reprecipitating strontium sulfate from EDTA at a pH of 4.0. Strontium was separated from barium by complexing the strontium in DTPA while precipitating barium as barium chromate. The strontium was ultimately converted to strontium carbonate and counted on a low-background gas proportional counter. The typical MDC for a one gram sample and a 60-minute count time was 0.75 pCi/g.

D.4.4 Tc-99 ANALYSIS

Tc-99 was quantified by radiochemical separation using extraction chromatography and counted by liquid scintillation. Samples were homogenized and leached with dilute nitric acid. The leachates were passed through an extraction chromatographic column containing a resin (TEVA resin) which is highly specific for technetium in the pertechnetate form. The technetium is absorbed onto the extraction resin. The resin is added to a scintillation vial containing an appropriate cocktail and counted using a liquid scintillation analyzer. All interfering beta emitting radionuclides are effectively removed (including C-14, P-32, S-35, Sr-90, Y-90, and Th-234) using TEVA resin under the conditions in this procedure. The typical MDC for a five gram sample and a 60-minute count time was 0.25 pCi/g.

D.4.5 C-14 ANALYSIS

C-14 analysis was performed using a material oxidizer and counted by liquid scintillation.

The Material Oxidizer combusts soil samples in a stream of oxygen gas, passes the products through a series of catalysts, then traps the carbon-14 into a carbon dioxide trapping scintillation cocktail.

The typical MDC for a 0.75 gram sample and a 60-minute count time was 2 pCi/g.

D.4.6 DETECTION LIMITS

Detection limits, referred to as MDCs, were based on 95% confidence level via NUREG 1507 method. 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.

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