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June 28, 2012

Mr. John Hickman
U.S. Nuclear Regulatory Commission
Materials Decommissioning Branch
Division of Waste Management and Environmental Protection
Mail Stop T8F5
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SUBJECT: FINAL REPORT—INDEPENDENT CONFIRMATORY SURVEY OF THE NUCLEAR RESEARCH LABORATORY AT THE UNIVERSITY OF ILLINOIS, URBANA-CHAMPAIGN, ILLINOIS; (DOCKET NO. 50-151; RFTA NO. 12-006)
DCN 5173-SR-01-0

Dear Mr. Hickman:

Oak Ridge Associated Universities (ORAU), under the Oak Ridge Institute for Science and Education (ORISE) contract, is pleased to provide the enclosed final report. The report details the confirmatory survey activities—performed during the week of May 7, 2012—for the Nuclear Research Laboratory at the University of Illinois located in Urbana-Champaign, Illinois. Comments on the draft report submitted on June 26 have been incorporated.

You may contact me via my information below or Erika Bailey at 865.576.6659 if you require additional information.

Sincerely,

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Enclosure

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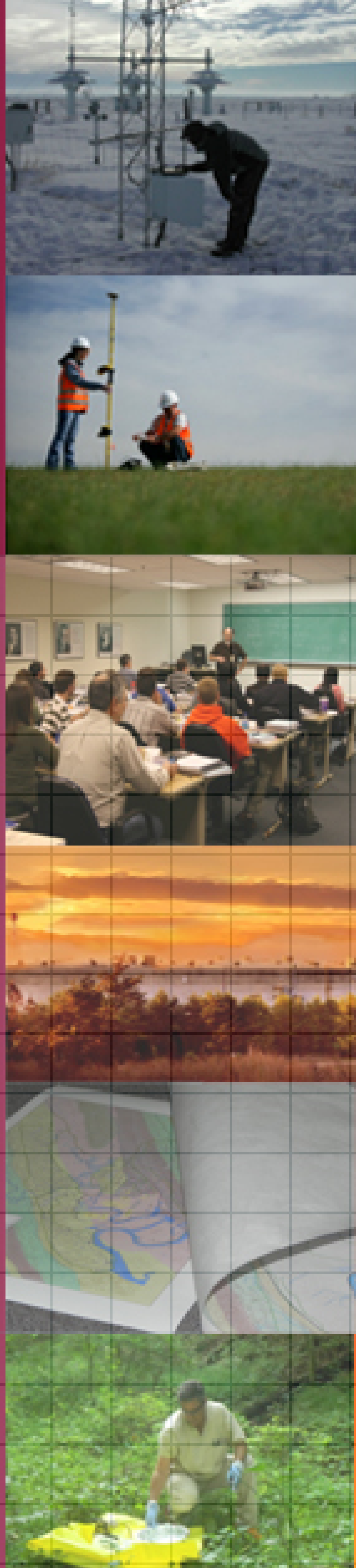


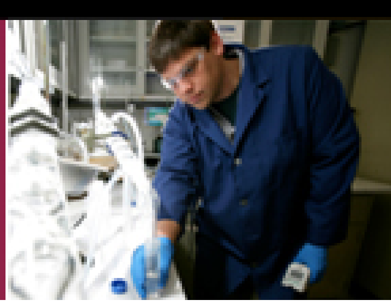
INDEPENDENT CONFIRMATORY SURVEY OF THE NUCLEAR RESEARCH LABORATORY AT THE UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN, ILLINOIS

E. M. Harpenau

Prepared for the
U.S. Nuclear Regulatory Commission

Final Report

ORISE



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INDEPENDENT CONFIRMATORY SURVEY OF THE NUCLEAR RESEARCH LABORATORY AT THE UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN, ILLINOIS

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Prepared for the
U.S. Nuclear Regulatory Commission

FINAL REPORT

JUNE 2012

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INDEPENDENT CONFIRMATORY SURVEY OF THE NUCLEAR RESEARCH LABORATORY AT THE UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN, ILLINOIS

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FINAL REPORT

June 2012

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ACRONYMS

cpm	counts per minute
DCGL	derived concentration guideline level
DP	decommissioning plan
dpm/100cm^2	disintegrations per minute per 100 square centimeters
ES	EnergySolutions
FSS	final status survey
FSSP	final status survey plan
GAB	gross alpha and gross beta
LOPRA	Low Power Reactor Assembly
LVI	LVI Environmental Services, Inc.
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
MDCR	minimum detectable count rate
MeV	million electron volts
NAD	no analytical data reported
NaI	sodium iodide
NRC	U.S. Nuclear Regulatory Commission
NRL	Nuclear Research Laboratory
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
Q	quantile
ROC	radionuclide of concern
SU	survey unit
TRIGA	Teaching Research Isotope General Atomic
University	University of Illinois

INDEPENDENT CONFIRMATORY SURVEY OF THE NUCLEAR RESEARCH LABORATORY AT THE UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN, ILLINOIS

1. INTRODUCTION

The University of Illinois (University) Nuclear Research Laboratory (NRL) contained an Advanced Teaching Research Isotope General Atomic (TRIGA) Mark II swimming pool-type reactor designed by the General Atomic Division of General Dynamics Corporation. Construction of the NRL began in the summer of 1959; construction was completed the following summer and the reactor achieved initial criticality on August 16, 1960. The NRL was operated under U.S. Nuclear Regulatory Commission (NRC) license R-115 and was operational until its permanent shutdown on August 6, 1998. A subcritical assembly, known as the Low Power Reactor Assembly (LOPRA), was added to the Bulk Shielding Tank on the south side of the reactor in 1971. The LOPRA used TRIGA fuel, and operated under its own NRC license (No. R-117) until 1995 when the LOPRA license was transferred to license R-115. NRC license R-117 was then terminated, and all reactor operations were conducted under license R-115 until reactor shutdown (ES 2006).

The historical site assessment and initial site characterization activities were conducted in 2005 by Scientech, LLC to assess and detail the radiological status of the NRL facility. The characterization activities determined that many reactor components and systems—including the soil under the building and some sub-surface structural components—were either radiologically activated, contaminated, or had a potential to contain residual contamination. EnergySolutions (ES) prepared a facility decommissioning plan (DP) that detailed the methodology that would be used to achieve the unrestricted release of the NRL facility (ES 2006). The steps to achieve unrestricted release included: additional characterization surveys of the building and reactor process components, removal of activated/contaminated materials, and surveys for release of the remaining reactor components and building materials. The material release program is intended to demonstrate that building materials/debris that will result from facility demolition do not have detectable radioactivity levels due to site operations. Final status surveys (FSS) will be conducted on the remaining structural components and soils after demolition of the NRL.

At NRC's request, Oak Ridge Associated Universities (ORAU), operating under the Oak Ridge Institute for Science and Education (ORISE) contract, conducted in-process surveys/inspections and confirmatory survey activities of the NRL.

2. SITE DESCRIPTION

The University is approximately 110 miles southwest of Lake Michigan and 35 miles from the Illinois-Indiana border. The campus is centered on the dividing line for the adjoining cities of Urbana and Champaign. The approximate 5,000 square foot NRL facility is located just south of the Engineering Sciences Building on the University's campus between Green and Springfield Streets. The NRL facility is divided into three levels: the lower level (reactor room), mezzanine level (offices, former control room, and restrooms), and storage level (located above the mezzanine with one office) (ES 2006).

3. OBJECTIVES

The objectives of the confirmatory activities were to provide independent contractor field data reviews, evaluate the licensee's survey process, and generate independent radiological data for use by the NRC in evaluating the adequacy and accuracy of the licensee's release survey results.

4. DOCUMENT REVIEW

Prior to on-site activities, ORAU staff reviewed the licensee's DP, final status survey plan (FSSP), guidance for pre-demolition surveys, and survey data sheets (ES 2006, ES 2007, and LVI 2012a and 2012b). The DP was specifically reviewed for historical information, as well as to identify the radionuclides of concern (ROCs) and the applicable release criteria. The purpose of these reviews was to ensure that the regulatory requirements for release of the building materials were being met by the licensee and to develop the ORAU confirmatory survey plan. ORAU also reviewed *Guidance for Pre-Demolition Surveys of Nuclear Research Laboratory Building, University of Illinois Urbana-Champaign* (LVI 2012a) to gather information detailing the criteria used for survey unit (SU) breakdown. The licensee provided the survey data sheets to explain how their instrumentation detection capabilities were calculated. ORAU was to ensure that the release survey activities within the NRL facility were adequate and appropriate, taking into account any supporting documentation and applicable

Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) guidance (NRC 2000). The University's FSSP was reviewed for information pertaining to derived concentration guideline levels (DCGLs) in soil in the event of soils becoming exposed during structural remediation. Final status surveys will only be implemented to address the building footprint and immediate surrounding areas that remain after demolition (ES 2007).

During the site visit, ORAU was provided two additional documents detailing the sampling guidance for concrete floor samples and guidance for release of the bioshield (LVI 2012c and 2012d).

5. APPLICABLE SITE GUIDELINES

The primary ROCs for the NRL are beta-gamma emitters—fission and activation products—resulting from reactor operation. Alpha contamination was not identified during characterization nor decommissioning activities (LVI 2012a). During remediation of the reactor, additional concrete samples (cores) were collected from the area around the NRL tank wall and floor. These samples indicated that the only ROCs were europium-152 (Eu-152), cobalt-60 (Co-60), and tritium (H-3) (ES 2006). In addition to these ROCs, Table 2-4 in the DP, which was derived from NUREG-1640, identified iron-55 (Fe-55) and nickel-63 (Ni-63) as other potential concrete rubble contaminants. Fe-55 was identified during the site characterization and was considered a possible soil contaminant. Furthermore, carbon-14 (C-14) is mentioned as a primary ROC for soil in Table 2-3 of the DP. Like H-3, Fe-55, and Ni-63, C-14 is considered a hard-to-detect ROC associated with the operation of a TRIGA reactor.

5.1 RELEASE CRITERIA FOR BUILDING SURFACES

The radioactivity values presented in Table 1 were used to evaluate materials to be released for reuse, recycle, or disposal as clean waste. The release criterion approved for use at the NRL states that the materials will be free of detectable surface contamination in accordance with guidance provided by the NRC in IE Circular 81-07. This circular is specific to removal from radiologically restricted areas of items (tools and equipment) and materials (e.g., scrap material, paper products, and trash) that could potentially be contaminated. The radioactivity limits shown in the Circular and provided in Table 1 represent the upper bound on the required detection capability of the survey procedures and instrumentation used. The licensee used standard survey instrumentation and smear samples to survey the NRL materials. A liquid scintillation counter with a minimum detectable concentration (MDC) of 1,000 dpm was used to determine activity on tritium smears (LVI 2012a). Furthermore, the licensee used an additional standard for volumetric radioactivity limits. These limits are provided in Table 1 as well. The total and removable surface activity data and volumetric sample data were compared with the respective limits.

Table 1. Release Criteria for Building Surface and Material Contamination			
Surface Activity Measurements (IE Circular 81-07)			
	Average	**Maximum**	**Removable**
Net beta-gamma activity (dpm/100cm^2)	5,000	15,000	1,000
Net alpha activity (dpm/100cm^2)	100	300	20
Recycling and Disposal of Concrete Rubble (NUREG-1640)			
Radionuclide	**95th Percentile Dose (mrem/yr per pCi/g)**	**Associated 1 mrem/yr Release Criteria (pCi/g)**	**Critical Scenario**
Tritium	1.1E-03	9.1E+02	Leachate-industrial
Iron-55	1.5E-05	6.7E+04	Processing concrete
Cobalt-60	2.0E+00	5.0E-01	Road building
Nickel-63	1.5E-05	6.7E+04	Processing concrete
Europium-152	8.8E-01	1.1E+00	Road building

5.2 RELEASE CRITERIA FOR VOLUMETRIC CONCRETE AND SOILS

The concrete floor of the reactor room and remaining concrete bioshield were sampled for volumetric contamination levels prior to initiating pre-demolition surveys to ensure that volumetric contamination, if present, was less than the criteria provided in Table 2-4 of the NRL DP. The criteria in the NRL DP correspond to the values listed under "Recycling and Disposal of Concrete Rubble" and reproduced in Table 1.

Determination of soil compliance will be demonstrated after the NRL demolition in accordance with the approved FSSP (LVI 2012a). The NRC default screening DCGLs and U.S. Environmental Protection Agency Memorandum of Understanding consultation triggers will be utilized for soil release (ES 2006). Though soil samples were not collected during remediation and release survey activities, radionuclide-specific screening levels are listed in Table 2.

Table 2. DCGLs for Primary Radionuclides of Concern in Soil

Radionuclide	Screening Level for Unrestricted Release (pCi/g)
Cobalt-60	3.8
Europium-152	6.9
Tritium (H-3)	110
Carbon-14	12
Iron-55	10,000
Nickel-63	2,100
Cesium-137	11
Europium-154	8.0

6. SURVEY PROCEDURES

At NRC's request, the ORAU survey team visited the University during the time period of May 8 through 10, 2012 to perform in-process and confirmatory survey activities. The in-process and confirmatory survey activities included evaluation of the licensee's implementation of the methodologies as written in their guidance documents, visual inspections, surface scans, surface activity measurements, and sample collection. The confirmatory survey activities were conducted in accordance with the approved project-specific plan and the ORAU Survey Procedures and Quality Program Manuals (ORAU 2012a, 2012b, and 2011a). Questions and concerns were brought to the immediate attention of the NRC and are also noted in the Findings and Results section of this report.

6.1 Reference System

Measurements and sampling locations were referenced to prominent site features and documented in the field logbook. Table 3 details the SU breakdown for the NRL during the survey for release phase.

Table 3. NRL Survey Units and Associated Classifications			
Survey Unit	**Description**		**MARSSIM Classification**
SU1	Lower Level Floor	Including pit and tunnel surfaces; loading area floor	2
SU2	Bioshield	Exterior surface of bioshield	2
SU3	Reactor Room Walls	Flooring to ceiling (excluding mezzanine level walls); heater/air units; pipes and ducts	3
SU4	Reactor Room Ceiling	Ceiling and horizontal surfaces (including overhead mezzanine); ventilation ducts; lights	3
SU5	Mezzanine Level	Floor and wall surfaces; ventilation system components; furniture/fixtures	3
SU6	Main Level	Floor and wall surfaces; ventilation system components; furniture/fixtures; ceiling	3
SU7	Roof	Roof surface; ventilation components	3
SU8	Exterior Walls	All exterior walls; cooling tower	3
SU9	Paved Areas	Asphalt and concrete walkways	3

6.2 Surface Scans

Gamma scans were performed using sodium iodide (NaI) scintillation detectors coupled to ratemeter-scalers with audible indicators. Alpha-plus-beta surface scans were performed using hand-held gas proportional detectors coupled to ratemeter-scalers with audible indicators. Both NaI and gas proportional detector/instrument combinations were connected to hand-held electronic data collectors equipped with real-time data-logging software to record instrument response during scans.

Confirmatory scan coverage during the confirmatory survey was based on the SU MARSSIM classifications presented in Table 3. Low-density scans were performed on the accessible surfaces in SUs 5 and 6. SUs 1 and 3 initially received medium-density scans of the floor and lower walls. The scans were increased to high density where elevated activity was detected. High-density scans of all accessible areas were also performed in SU 2. Though the licensee's pre-demolition release survey approach did not designate any class 1 SUs, ORAU performed high-density scans of the class 2 SUs where significant remediation had been performed or when elevated activity was observed during initial confirmatory scans. Confirmatory surveys were planned for SUs 4, 7, 8, and 9. However, due to the amount of investigative surveys performed in SUs 1, 2, 3, 5, and 6, ORAU was unable to survey the remaining four SUs before the end of the site visit.

6.3 SURFACE ACTIVITY MEASUREMENTS

Based on alpha-plus-beta scan results, direct surface measurements for total and removable alpha-plus-beta activity were performed at 39 judgmentally selected locations within the NRL. The locations were selected based on the elevated radiation levels identified by the ORAU survey team. In addition to alpha-plus-beta, alpha-only measurements were collected at 37 of the judgmental locations. Direct measurements were performed by using hand-held gas proportional detectors coupled to ratemeter-scalers. Due to the approved release criterion for beta-gamma contamination, ORAU collected material-specific background measurements within the large mechanical room where radiation levels were indicative of typical operating background for the gas proportional detectors.

Smear samples for gross alpha and gross beta (GAB) activity levels, and H-3/C-14 were collected primarily from non-remediated surfaces at the direct measurement locations. The decision to collect either a GAB or H-3/C-14 smear was also based on material surface characteristics and/or correlation to reactor processes.

6.4 CONCRETE SAMPLING

ORAU collected four concrete samples during confirmatory survey activities; three within the bioshield and one in the N16 coolant tunnel (Fig. 1). Samples were obtained from these areas due to increased activity measurements or at the request of the NRC. An additional five volumetric concrete samples collected and previously analyzed by the licensee were selected by the NRC and provided to ORAU for an inter-lab comparison.

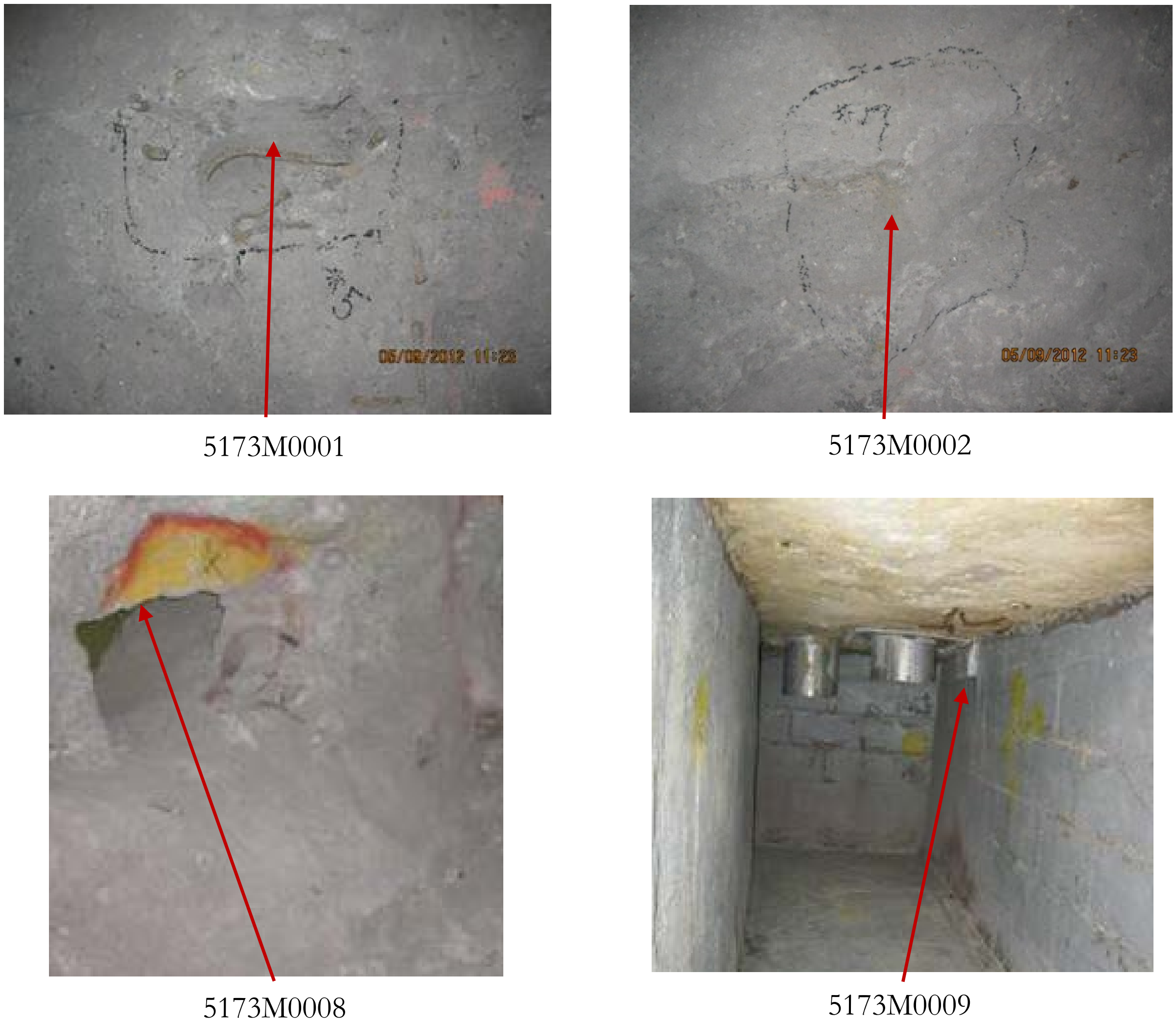

Fig. 1. Confirmatory Sample Locations for the Bioshield and N16 Coolant Tunnel

7. SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data collected were returned to the ORAU/ORISE facility in Oak Ridge, Tennessee for analysis and interpretation. All sample analyses were performed in accordance with the ORAU/ORISE Laboratory Procedures Manual (ORISE 2012). The concrete samples collected were analyzed by solid-state gamma spectroscopy for gamma-emitting ROCs. Fe-55 and Ni-63 concentrations were determined by radiochemical separation and then counted via liquid scintillation. Samples 5173M0008 and 5173M0009 were also analyzed by alpha spectroscopy at NRC's request. Smears for GAB were analyzed with a gas flow proportional counter. Smears for H-3 and C-14 were analyzed via a liquid scintillation counter. Analytical results for the volumetric samples are reported in units of picocuries per gram (pCi/g) and smear results in pCi/sample. The data generated were compared with the approved NRC screening values as presented in Table 2. Appendices C and D provide further details on the survey and laboratory instrumentation and procedures used.

8. FINDINGS AND RESULTS

The results for each of the in-process and confirmatory survey activities are discussed in the following subsections.

8.1 DOCUMENT REVIEW

Reviews of the documents provided prior to the site visit revealed multiple items of concern. First, the licensee divided the NRL facility into nine SUs with different MARSSIM classifications (LVI 2012a). However, none of the SUs were designated as Class 1 areas, even with extensive remediation performed on the bioshield, coolant tunnel piping, and N16 tanks. Section 4.4 of MARSSIM designates that Class 1 areas are those that have (or had prior to remediation) a potential for radioactive or known contamination above site release criteria. This includes site areas subject to remedial actions, locations where leaks or spills have occurred, waste storage sites, and areas with contaminants in discrete solid pieces of material with high specific activity (NRC 2000).

The next item of concern was the reference inputs on the licensee's survey data sheets. ORAU noticed that the scan MDC for total direct beta-gamma measurements in a completed survey data sheet exceeded 5,000 disintegrations per minute per 100 square centimeters (dpm/100 cm^2) (LVI 2012b). This resulted in conflicts between the licensee's two guidance documents. The pre-demolition document said portable survey instruments would be capable of detecting total beta-gamma contamination at the release limit and set the minimum detectable count rate (MDCR) at approximately 75% of the release criteria (representative of approximately 3,750 dpm/100 cm^2 beta-gamma), which the licensee committed to (LVI 2012a). Conversely, the licensee stated that the measureable limit criteria for the bioshield equated to approximately 2,500 dpm/100cm^2 (beta/gamma) for concrete surface surveys, which is the limit set in the bioshield guidance document (LVI 2012d).

Further review of the same completed survey data sheet revealed an uncharacteristically high number of negative beta-gamma surface activity results. This suggests that the reference background data inputs used may not have been appropriate for this SU. This sheet also included the application of the instrument efficiency for technetium-99 (Tc-99) to surface activity calculations even though C-14, a radionuclide with a lower detection efficiency, was one of the ROCs. This conflicts directly with calibration requirements of both Circular 81-07 and MARSSIM. Using the Tc-99 instrument efficiency versus the more restrictive C-14 efficiency could result in an overestimation of the instrument's detection capability; this would lead to an underestimation of surface activity. To mitigate the possibility of underestimating surface activity, ORAU instrumentation was calibrated to the C-14 efficiency using a 0.4mg/cm^2 face on the gas proportional detectors.

8.2 Surface Scans

The gross count rates for alpha-plus-beta and gamma radiation surface scan data for each SU were prepared for report presentation using quantile (Q) plots. The Q-plots are presented in Appendix A. They are a graphical technique for determining if there is a common distribution in data sets. The advantage of the Q-plot is that population distributional aspects can be evaluated simultaneously. The detectable aspects include:

- Shifts in scale
- Changes in symmetry (skewness of the data)
- The presence of outliers

Q-plots were generated by uploading the scan data files into the U.S. Environmental Protection Agency's ProUCL software. In the Q-plots provided in Appendix A, the Y-axis represents observed count rates in counts per minute (cpm). The X-axis represents the data quantiles about the mean value. A normal distribution that is not skewed by outliers will appear as a straight line with the slope of the line subject to the degree of variability among the data population (i.e., a background radiation population). Values less than the mean are represented in the negative quantiles, and values greater than the mean are represented in the positive quantiles. The presence of more than one population—e.g., background radiation population and contamination—would display on a Q-plot as a step function. Small areas of localized contamination will appear on the Q-plot as outlier points in the upper right quadrant.

The ORAU survey team detected residual radioactivity in SUs 1, 2, 3, 5, and 6 while performing surface scans with hand-held gas proportional and NaI detectors. Instrument response for alpha-plus-beta and gamma scans ranged from approximately 60 to 3,600 gross cpm and 2,800 to 24,000 cpm, respectively, over all SUs addressed during confirmatory surveys. The N16 Tank Vault had the highest alpha-plus-beta instrument response for walls due to the detection of localized contamination on the south wall and on the remaining pipe flanges leading into the coolant tunnel. The highest average instrument response for the floors occurred in the Sealed Source Cage. All other areas of elevated surface activity identified during confirmatory surveys were bounded, quantified, and reported to the onsite NRC inspector. The Q-plots clearly show those SUs where contamination was detected during surface scans.

8.3 Surface Activity Measurements

Total surface activity and removable activity levels for the 39 judgmentally determined measurement/sample locations are presented in Table B-1. Table 4 presents a summary of remaining activity levels.

Table 4. Surface Activity Measurement Summary

Variable	Number of Observations	Range
Total Alpha (dpm/100cm²)	37	-14 to 820
Total Beta (dpm/100cm²)	38	-1,200 to 15,000
Removable Alpha (dpm/100cm²)	13	-1 to 6
Removable Beta (dpm/100cm²)	13	0 to 184
Removable Tritium (pCi/sample)	21	-5.5 to 773.0
Removable Carbon-14 (pCi/sample)	21	-1.6 to 207.0

ORAU identified 31 pre-remediation locations above the conservative static measurement MDC of 690 dpm/100cm^2 for the alpha-plus-beta gas proportional detectors, 7 of which were above 5,000 dpm/100cm^2. Spot remediation was performed by the licensee, resulting in 29 locations still above the alpha-plus-beta MDC, and 4 still above 5,000 dpm/100cm^2 at the conclusion of the confirmatory survey. Material-specific backgrounds were used to correct gross counts prior to calculating the alpha-plus-beta surface activity. Total remaining confirmatory survey alpha-plus-beta surface activity levels ranged from approximately 1,200 to 15,000 dpm/100 cm^2.

The highest reported surface activity of approximately 15,000 dpm/100 cm^2 was identified on the floor of the source cage. The second highest reported surface activity of approximately 10,000 dpm/100 cm^2 was observed on a conduit pipe on the exterior wall of the bioshield. Additional surveys of this conduit did not reveal any alpha or gamma contamination. The licensee committed to sealing the conduit and disposing of it in the proper waste stream when the remaining bioshield components were demolished. The third highest activity (approximately 9,300 dpm/100 cm^2) was located on the south wall of the reactor room. Remediation was being performed on the third location, but the ORAU team departed the site before remediation of it or other locations could be confirmed. The fourth and final location above 5,000 dpm/100cm^2 was on the north wall of the decay tank with a surface activity of approximately 6,200 dpm/100 cm^2.

Alpha surface activity measurements were collected at several elevated activity locations identified during the alpha-plus-beta scans to ensure that the licensee's decision not to collect alpha measurements was justified. Elevated alpha surface activity measurements were observed primarily in the concrete-lined coolant tunnel. Smears and a volumetric sample were collected from the coolant tunnel to determine if the observed alpha surface activity was due to contamination or radon gas buildup. Review of the analytical results indicated that the observed activity was due to radon in the tunnel.

Laboratory analysis of the 13 smears collected in association with direct surface activity measurements identified a maximum gross beta contamination of 180 dpm/100 cm^2 on the conduit pipe of the bioshield. All remaining GAB removable activity results ranged from -1 to 6 dpm/100 cm^2 and 0 to 9 dpm/100 cm^2, respectively. Removable activity for the hard-to-detect radionuclides across all SUs ranged from -6 to 770 pCi/sample for H-3, and -2 to 210 pCi/sample for C-14. The removable activity levels were below the release criteria. The surface activity results are summarized in Table 4.

8.4 RADIONUCLIDE CONCENTRATION IN CONCRETE

Additional concerns were identified in the radiological analysis of the four confirmatory concrete samples collected from the bioshield and coolant tunnel. Review of the analytical results revealed concentrations of Co-60 in excess of the 0.5 pCi/g release limit for all samples, and two samples had concentrations in excess of the 1.1 pCi/g Eu-152 release limit (Table B-2).

At NRC's request, ORAU Samples 5173M0008 and 5173M0009 were also analyzed via alpha spectroscopy (Table B-3). Alpha spectroscopy results did not identify any concentrations of the ROCs above the respective release limits.

The NRC also selected five of the licensee's split concrete samples and provided them to ORAU for an inter-lab comparison. A full inter-lab comparison could not be performed on the samples provided due to insufficient reporting of the licensee's analytical results. Only a percentage of the licensee's samples were analyzed for the hard-to-detect ROCs, and a different percentage received gamma spectroscopy analysis. The licensee's results were then reported as detects with known concentrations and errors, less than MDC (<MDC), or no analytical data reported (NAD). Before reporting the inter-lab comparison results, ORAU did observe that the licensee did not include C-14, cesium-137 (Cs-137), or Eu-154 as ROCs for the recycling and disposal of concrete rubble per NUREG-1640. However, C-14, Cs-137, and Eu-154 were added to the initial ROC list in Table 3 which lists the primary ROCs in soils. Analytical results for each of the ROCs listed in Tables 2 and 3 have been included in the ORAU columns, even if the corresponding analysis had not been requested by the licensee (Tables B-4 and B-5).

ORAU is of the opinion that the ROC list associated with the release of underlying soil should be consistent with the ROC list associated with the release criteria for the recycle and disposal of building rubble. This opinion was reached on the basis that the source of contamination would originate from within the building, and then would have to migrate through the construction materials in order to impact the soil.

9. SUMMARY

At NRC's request, ORAU conducted confirmatory survey activities within the NRL at the University during the week of May 7, 2012. The survey activities included visual inspections/ assessments, surface activity measurements, and volumetric concrete sampling activities.

During the course of the confirmatory activities, ORAU noted several issues with the survey-for-release activities performed at the University. Issues included inconsistencies with: survey unit classifications were not designated according to MARSSIM guidance; survey instrument calibrations were not representative of the ROCs; calculations for instrumentation detection capabilities did not align with the release criteria discussed in the licensee's survey guidance documents; total surface activity measurements were in excess of the release criteria; and Co-60 and Eu-152 concentrations in the confirmatory concrete samples were above their respective guidelines.

Based on the significant programmatic issues identified, ORAU cannot independently conclude that the NRL satisfied the requirements and limits for release of materials without radiological restrictions.

10. REFERENCES

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APPENDIX A
SURVEY UNIT SUMMARIES

Alpha-plus-Beta Scan Summary for Reactor Room						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	2,705	453	3,127	1,001	900	405
Walls	6,064	69	1,657	292	278	116

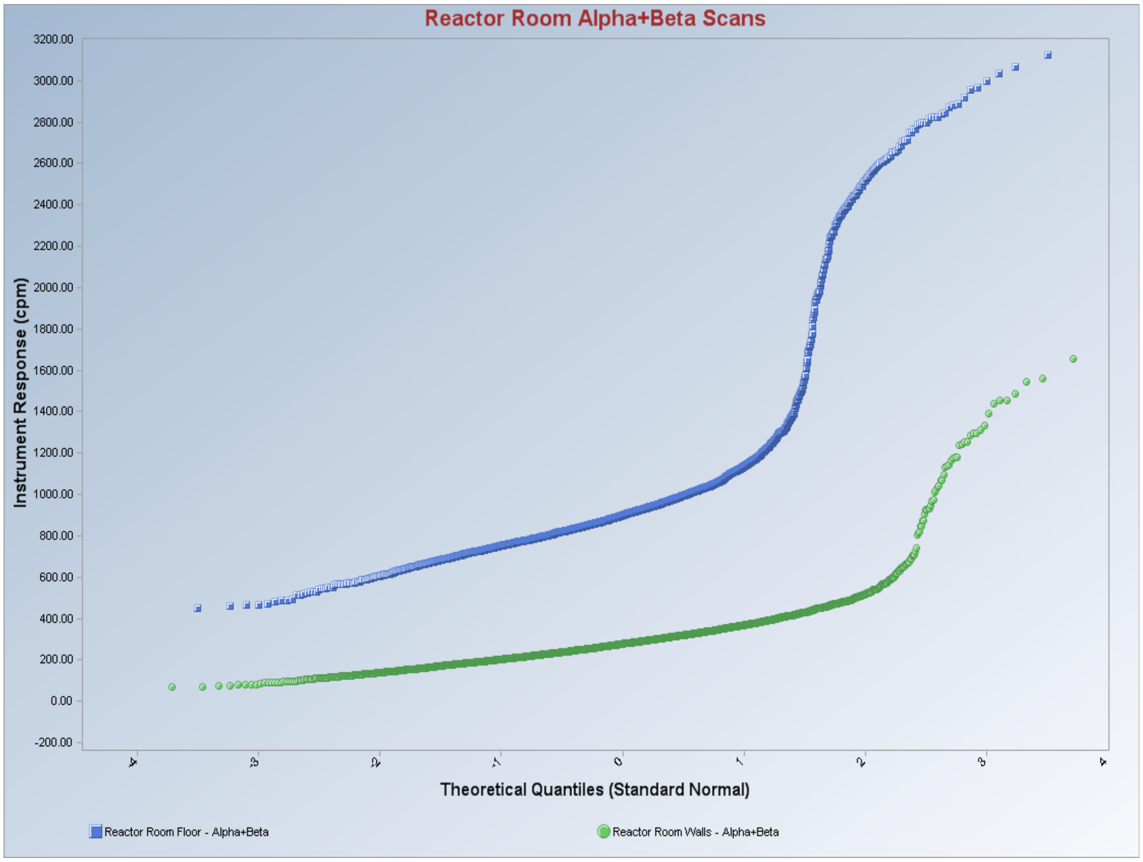

Fig. A-1. Alpha-plus-Beta data package for the Reactor Room

Gamma Scan Summary for Reactor Room						
Structure	n	Min	Max	Mean	Median	SD
Floor	1,506	4,265	23,864	6,524	6,317	1204
Walls	1,154	4,811	10,355	6,642	6,357	1031

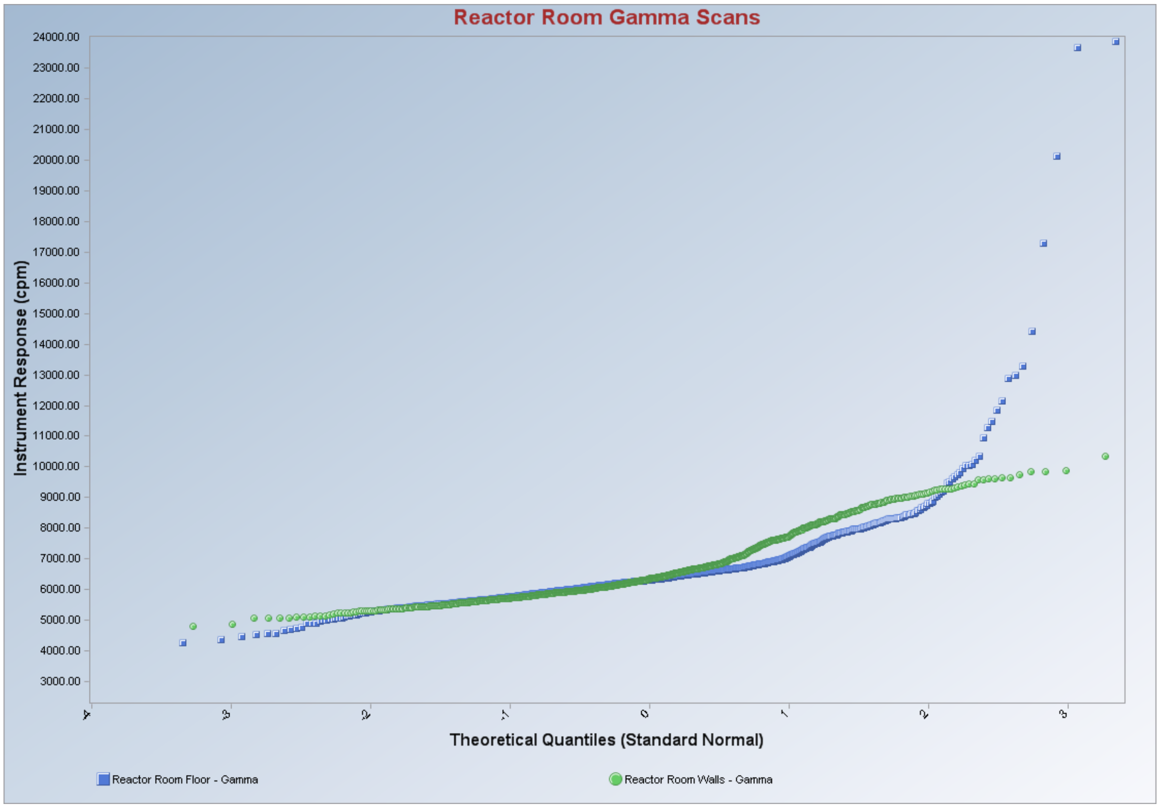

Fig. A-2. Gamma data package for the Reactor Room

Alpha-plus-Beta Scan Summary for Bioshield Exterior						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
BioShield Exterior Top	673	121	621	350	345	77
BioShield Exterior Walls	3,163	56	1,361	257	247	97

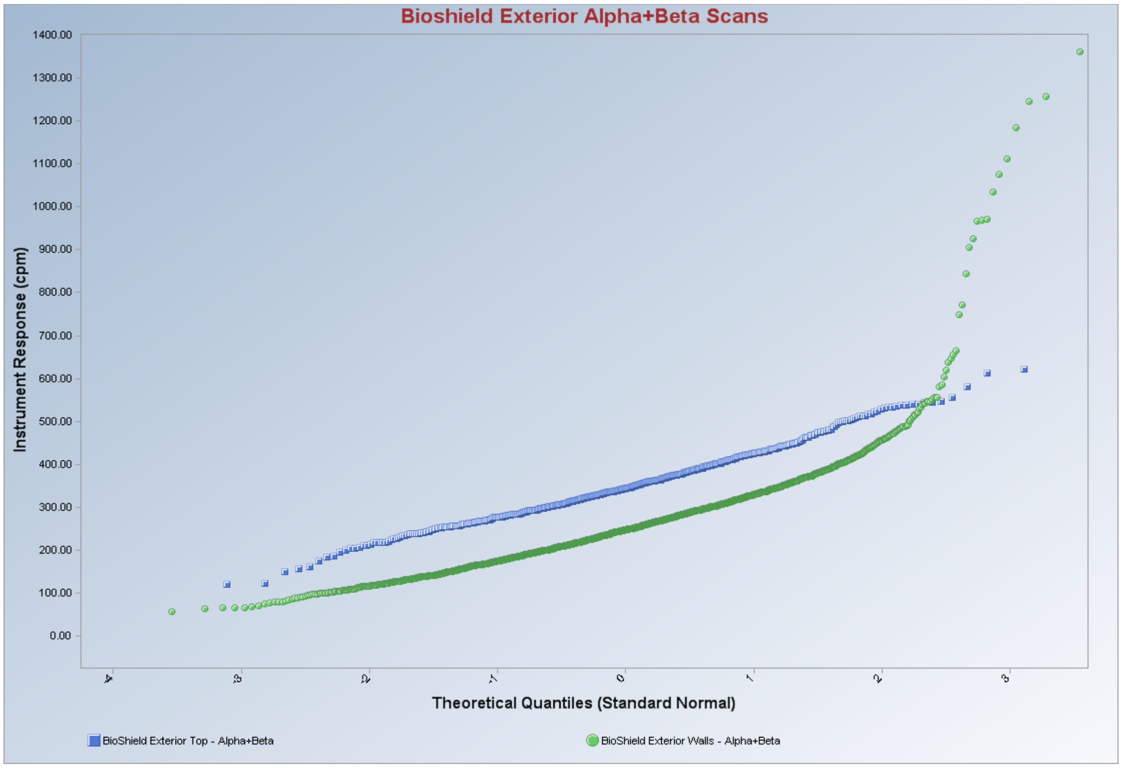

Fig. A-3. Alpha-plus-Beta data package for the Bioshield Exterior

Gamma Scan Summary for Bioshield Exterior						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
BioShield Exterior	518	4,183	7,058	5,573	5,564	383

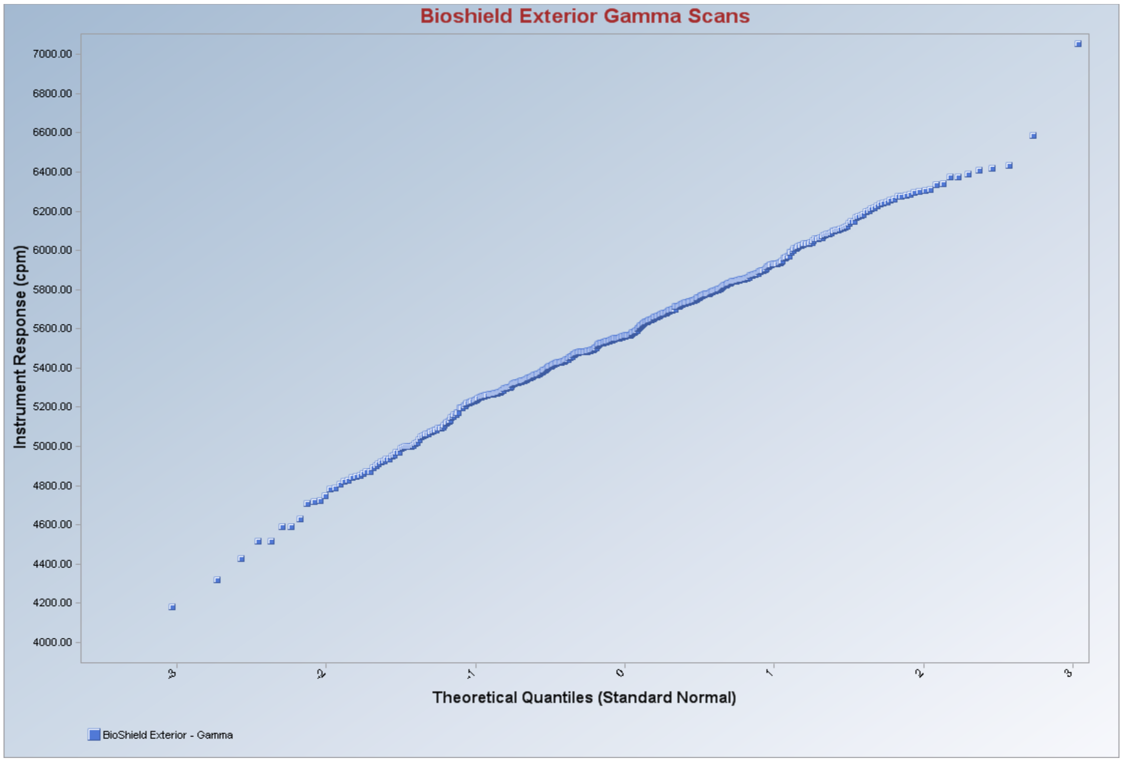

Fig. A-4. Gamma data package for the Bioshield Exterior

Alpha-plus-Beta Scan Summary for Bioshield Interior						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Upper Walls	515	103	470	256	245	68
Lower Walls	3,123	51	1,273	324	313	103

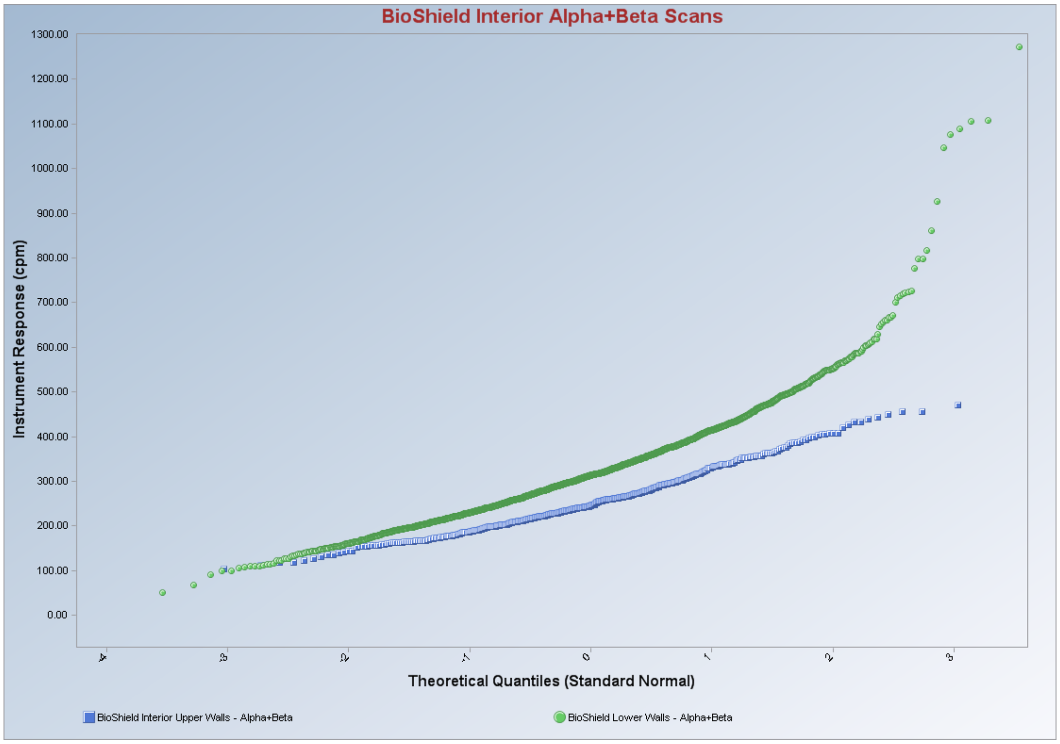

Fig. A-5. Alpha-plus-Beta data package for the Bioshield Interior

Gamma Scan Summary for Bioshield Interior						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
All Surfaces	859	5,340	21,108	8,058	7,288	2205

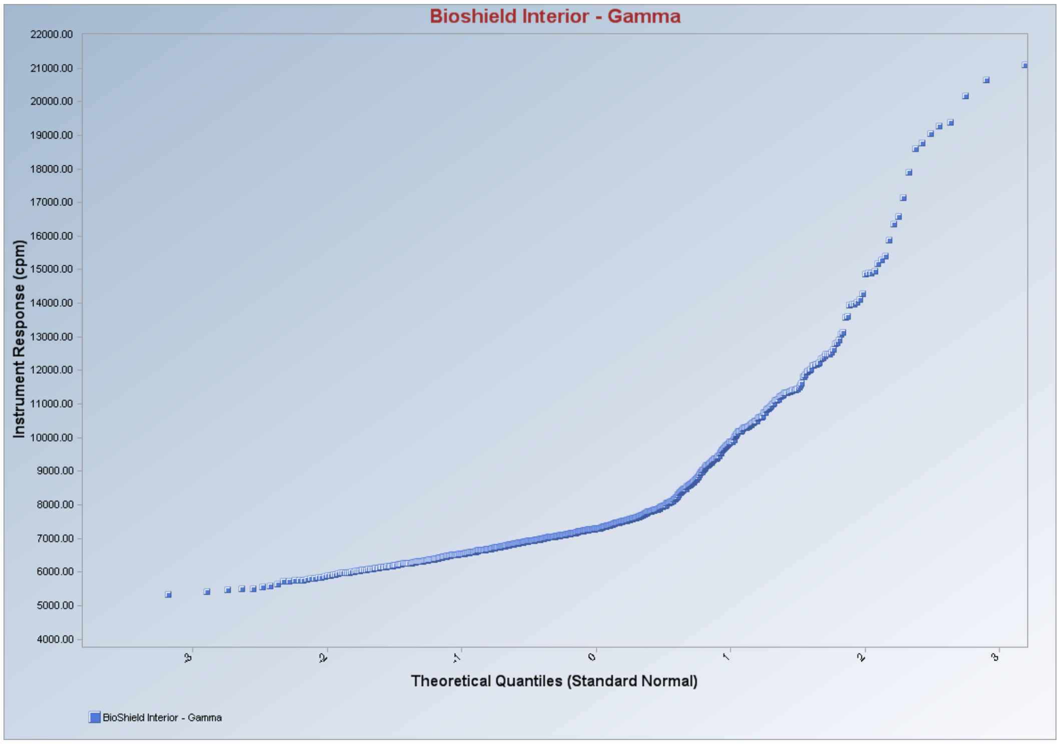

Fig. A-6. Gamma data package for the Bioshield Interior

Alpha-plus-Beta Scan Summary for Coolant Tunnel						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
All Surfaces	1,856	118	928	422	408	126

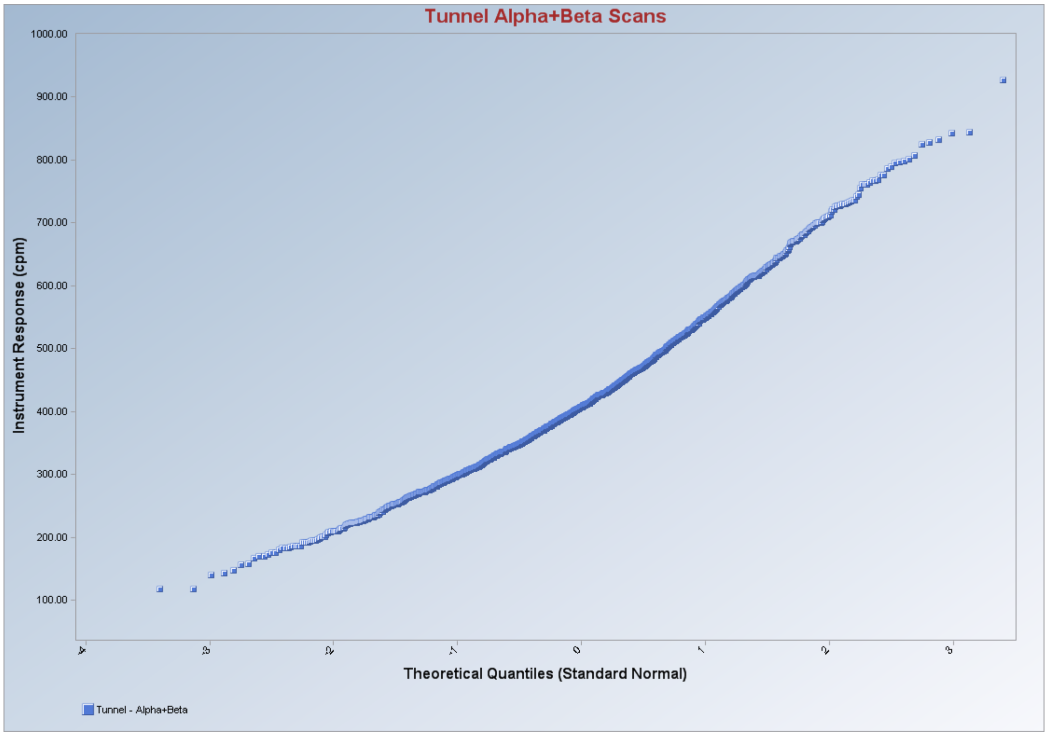

Fig. A-7. Alpha-plus-Beta data package for the Coolant Tunnel

Gamma Scan Summary for Coolant Tunnel						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
All Surfaces	444	4,109	8,391	6,422	6,693	904

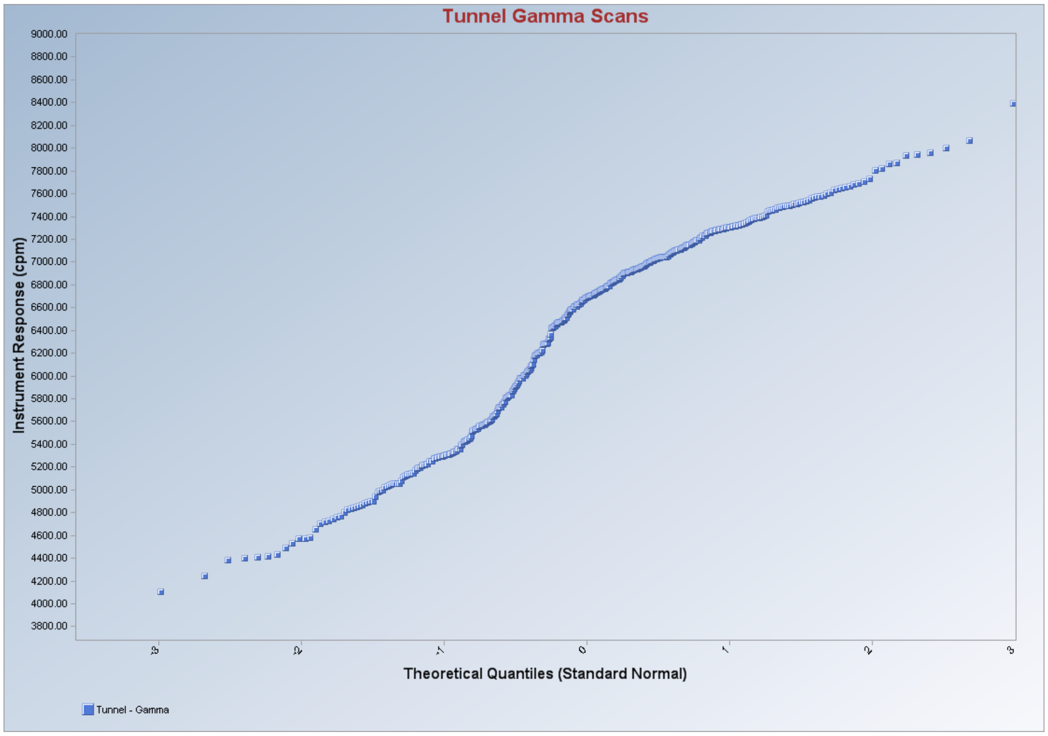

Fig. A-8. Gamma data package for the Coolant Tunnel

Alpha-plus-Beta Scan Summary for N16 Tank Vault						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	27	117	385	226	226	69
Walls	2,103	79	3,622	310	270	284
Ceiling	376	76	458	231	225	65

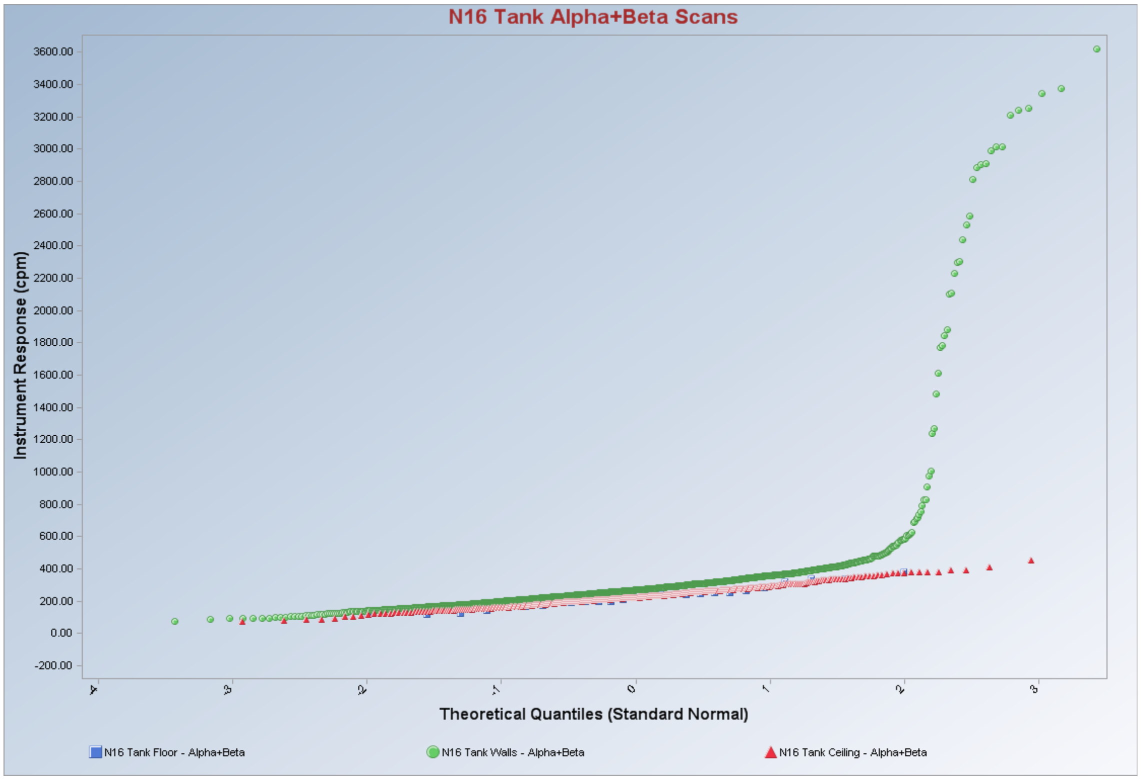

Fig. A-9. Alpha-plus-Beta Data Package for the N16 TankVault

Gamma Scan Summary for N16 Tank Vault						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	274	4,436	6,287	5,085	5,024	358
Walls	830	3,430	6,306	4,691	4,701	480
Ceiling	101	2,842	4,968	3,616	3,547	421

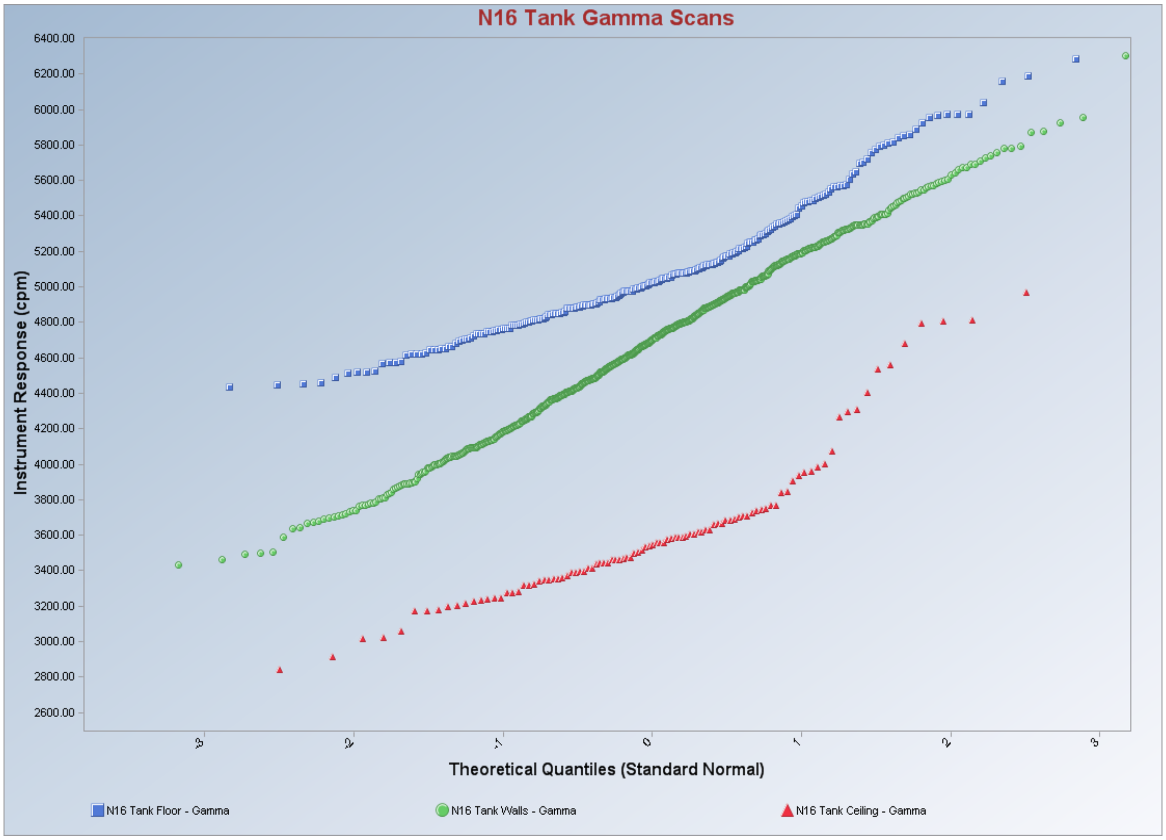

Fig. A-10. Gamma Data Package for the N16 Tank Vault

Alpha-plus-Beta Scan Summary for Decay Tank						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	226	200	677	392	386	88
Walls	702	104	1,235	372	349	135

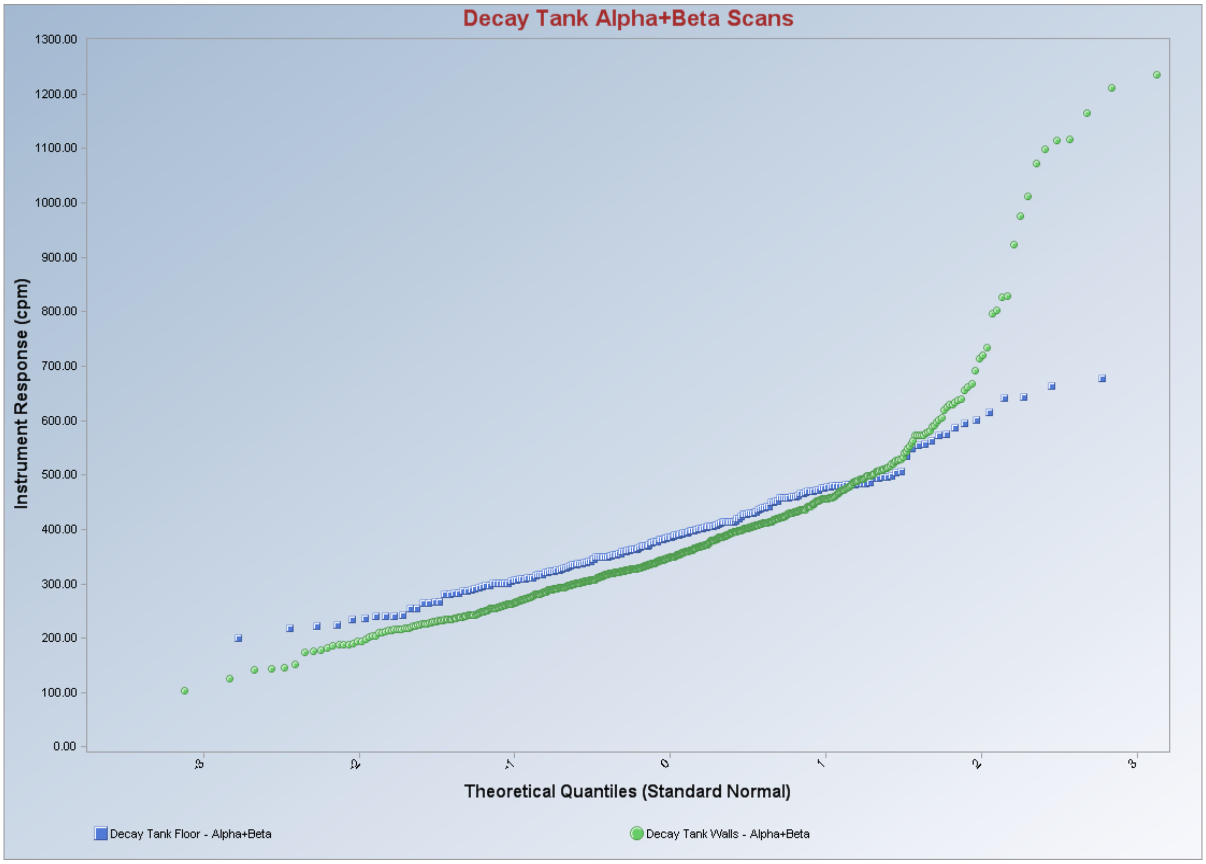

Fig. A-11. Alpha-plus-Beta Data Package for the Decay Tank

Gamma Scan Summary for Decay Tank						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
All Surfaces	392	5,613	8,576	6,894	6,851	461

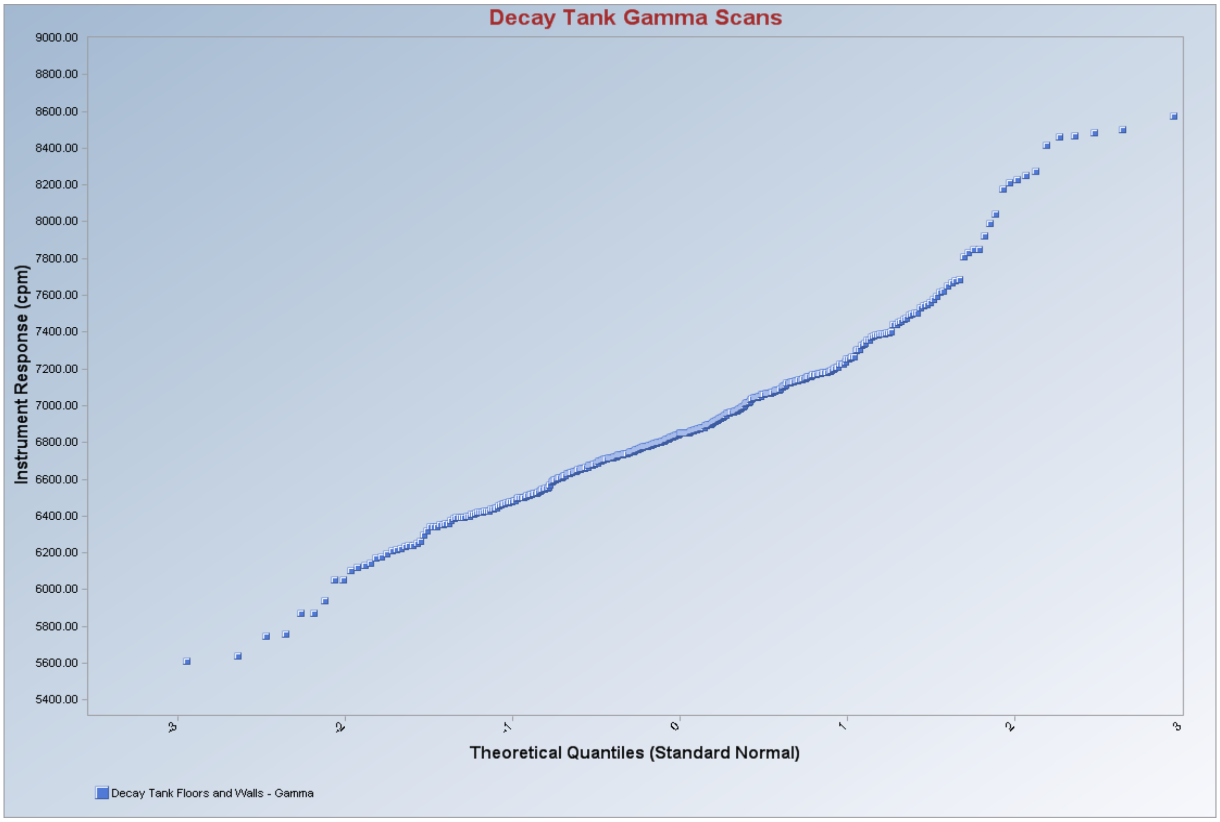

Fig. A-12. Gamma Data Package for the Decay Tank

Alpha-plus-Beta Scan Summary for Mechanical Room						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	368	530	1,269	891	885	130
Walls	1,733	61	502	250	247	65

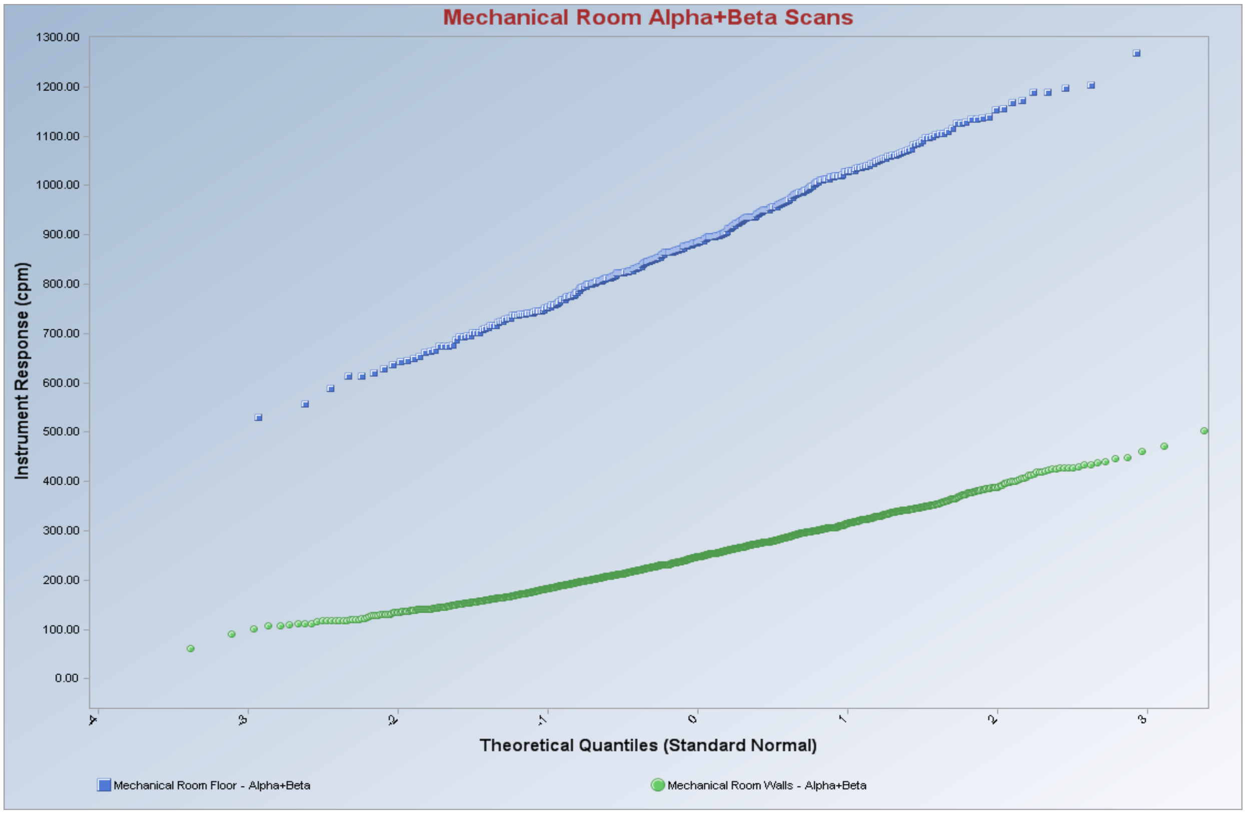

Fig. A-13. Alpha-plus-Beta Data Package for the Mechanical Room

Gamma Scan Summary for Mechanical Room						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	162	4,501	6,557	5,591	5,639	379.3
Walls	297	4,742	6,946	5,761	5,766	357.1
Small Mech. Room All Surfaces	194	4207	7232	5767	5873	696

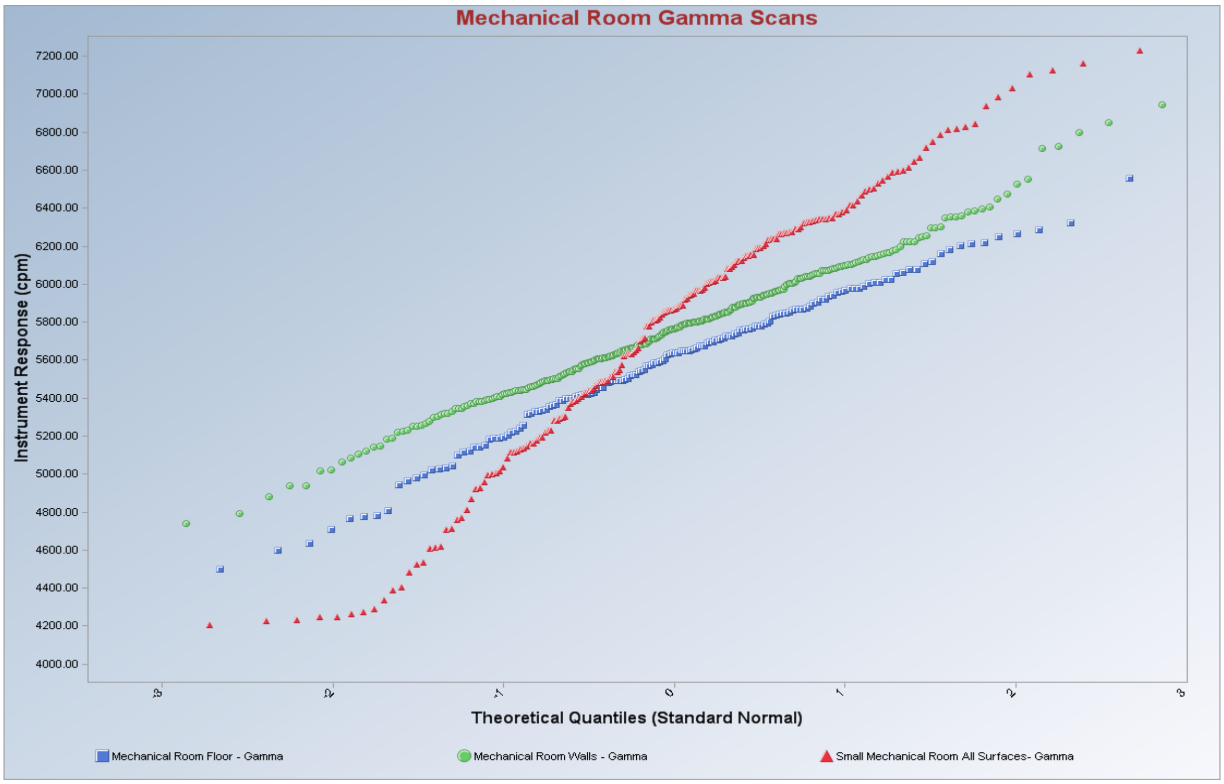

Fig. A-14. Gamma Data Package for the Mechanical Room

Alpha-plus-Beta Scan Summary for Source Areas						
Structure	n	Min	Max	Mean	Median	SD
Walls	958	100	620	281	276	77

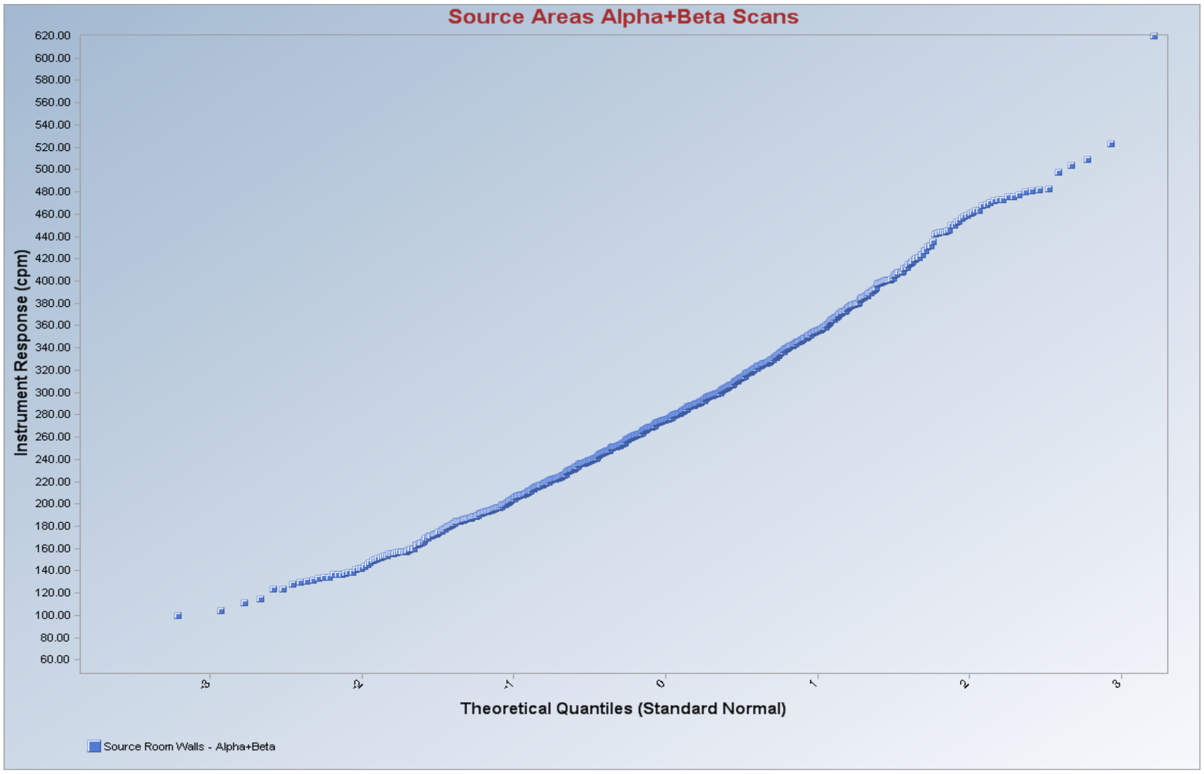

Fig. A-15. Alpha-plus-Beta Data Package for the Source Areas

Gamma Scan Summary for Source Areas						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
NE Well	53	3,536	6,655	4,531	4,102	926
NW Well	67	3,620	6,657	4,429	4,247	605
SE Well	78	3,821	7,021	4,568	4,407	704
SW Well	83	3,946	6,122	4,435	4,349	364
Lower Level Cage	135	5,484	7,954	6,610	6,598	511

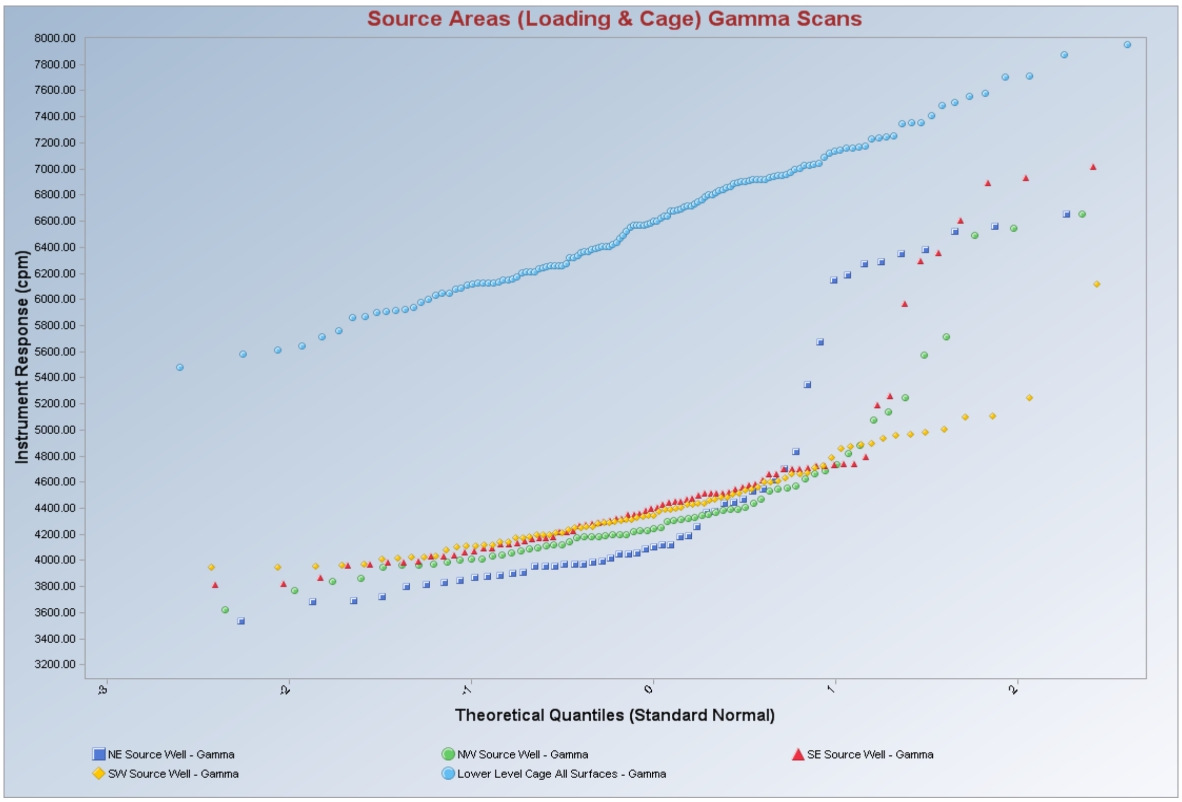

Fig. A-16. Gamma Data Package for the Source Areas

Gamma Scan Summary for Main Level						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	265	5,437	8,701	7,175	7,228	721
Walls	214	6,259	9,728	8,151	8,273	776

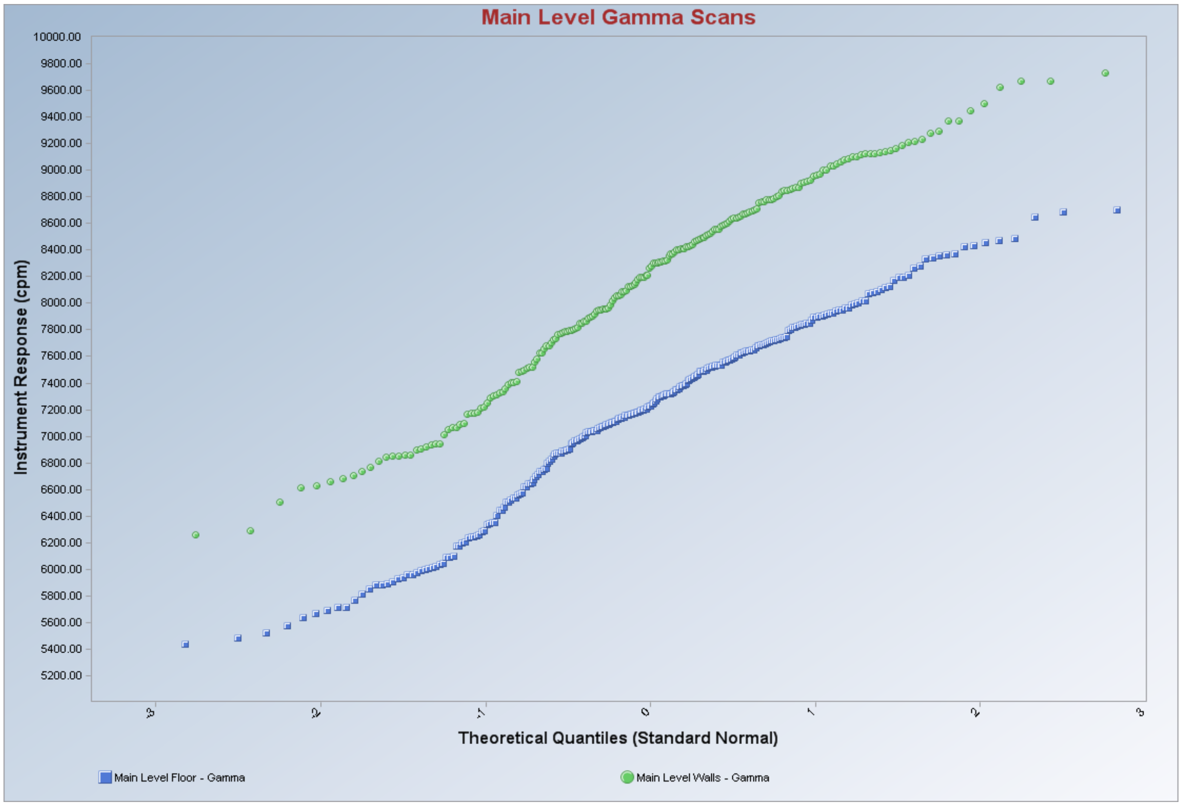

Fig. A-17. Gamma Data Package for the Main Level

Gamma Scan Summary for Mezzanine Level						
Structure	**n**	**Min**	**Max**	**Mean**	**Median**	**SD**
Floor	239	4,603	6,863	5,700	5,633	474
Walls	292	4687	7428	5825	5793	473

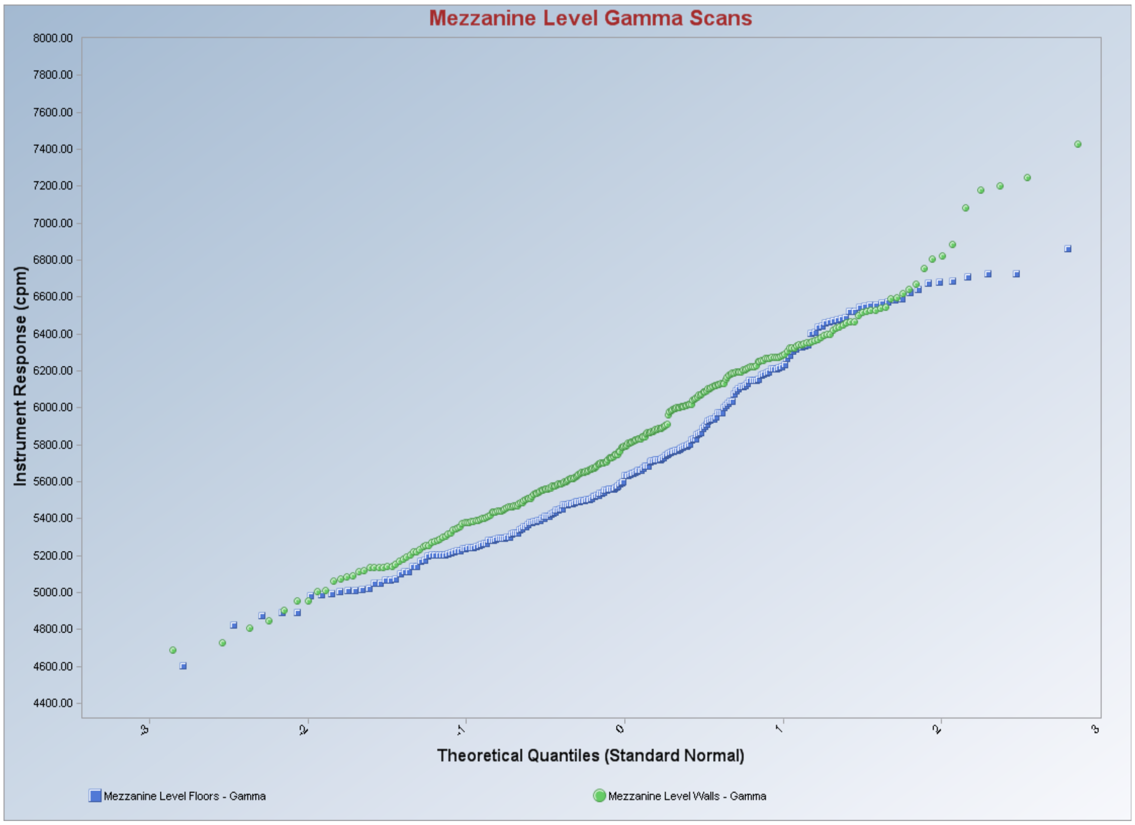

Fig. A-18. Gamma Data Package for the Mezzanine Level

APPENDIX B
TABLES

Table B-1. Surface Activity Results for the Nuclear Research Laboratory									
ORAU Location ID	Description	Total Activity				Removable Activity (dpm 100 cm^{-2})			
		Alpha Pre-Remediation (dpm/100cm^2)	Alpha Post-Remediation (dpm/100cm^2)	Alpha-plus-Beta Pre-Remediation (dpm/100cm^2)	Alpha-plus-Beta Post-Remediation (dpm/100cm^2)	Gross Alpha (dpm/100cm^2)	Gross Beta (dpm/100cm^2)	Tritium (pCi/sample)	Carbon-14 (pCi/sample)
5173R0001	Reactor Room-Floor	14		3,300		n/a	n/a	n/a	n/a
5173R0002	Reactor Room-Floor	7		3,700		n/a	n/a	n/a	n/a
5173R0003	Reactor Room-Floor	7		4,200		n/a	n/a	n/a	n/a
5173R0004	Reactor Room-Floor	43		2,400		n/a	n/a	n/a	n/a
5173R0005	Bioshield-Floor	14		4,000	310	-1	1	0.1	3.0
5173R0006	Bioshield-Floor	14		1,800		n/a	n/a	n/a	n/a
5173R0007	Bioshield-Wall	7		9,800	2,100	-1	8	261.0	207.0
5173R0008	Bioshield-Wall	0		1,800		n/a	n/a	n/a	n/a
5173R0009	Bioshield-Wall	-7		610		n/a	n/a	n/a	n/a
5173R0010	Bioshield-Wall	0		1,200		n/a	n/a	n/a	n/a
5173R0011	Bioshield-Wall	36		1,300		n/a	n/a	n/a	n/a
5173R0012	Bioshield-Wall	7		540		n/a	n/a	n/a	n/a
5173R0013	Bioshield-Wall	14		1,400		n/a	n/a	n/a	n/a
5173R0014	Bioshield-Wall	0		1,400	770	n/a	n/a	n/a	n/a
5173R0015	Bioshield-Wall	0		1,700		n/a	n/a	n/a	n/a
5173R0016	Bioshield-Wall	7		4,000	-150	-1	2	10.7	15.0
5173R0017	Conduit Bioshield-Wall	n/a		10,000		6	184	773.0	187.0
5173R0018	Tunnel-Wall	950	640	3,400	2,200	4	9	1.2	1.3
5173R0019	Tunnel-Wall	640	560	3,300	1,900	4	9	-1.3	1.5
5173R0020	Tunnel-Wall	760	820	2,700	1,800	4	1	4.2	8.7
5173R0021	Tunnel-Wall	65	120	2,700	280	-1	6	-3.3	-0.8
5173R0022	Tunnel-Wall	760	520	2,000	1,000	6	0	11.3	9.1

Table B-1. Surface Activity Results for the Nuclear Research Laboratory

ORAU Location ID	Description	Total Activity				Removable Activity (dpm 100 cm^{-2})			
		Alpha Pre-Remediation (dpm/100cm^2)	Alpha Post-Remediation (dpm/100cm^2)	Alpha-plus-Beta Pre-Remediation (dpm/100cm^2)	Alpha-plus-Beta Post-Remediation (dpm/100cm^2)	Gross Alpha (dpm/100cm^2)	Gross Beta (dpm/100cm^2)	Tritium (pCi/sample)	Carbon-14 (pCi/sample)
5173R0023	Tunnel-Ceiling	700	330	640	1,300	n/a	n/a	3.8	12.4
5173R0024	Tunnel-Ceiling	820	460	1,600	2,000	n/a	n/a	5.1	14.7
5173R0025	Tunnel-Ceiling	120		-820		n/a	n/a	-5.4	3.6
5173R0026	Tunnel-Ceiling	43		-1,200		n/a	n/a	-3.0	3.0
5173R0027	Tunnel-Ceiling	58		580		n/a	n/a	-5.5	2.5
5173R0028	Decay Tank-Wall	36		6,200		-1	3	-4.1	2.1
5173R0029	Reactor Room-Wall	0		9,300		n/a	n/a	n/a	n/a
5173R0030	W. Beam Port-Wall	22		3,000		n/a	n/a	n/a	n/a
5173R0031	NE Beam Port-Wall	-7		3,600		n/a	n/a	n/a	n/a
5173R0032	E Beam Port-Wall	36		2,600		n/a	n/a	n/a	n/a
5173R0033	N16 Pipe Flange	-14		6,400	730	1	4	4.0	4.1
5173R0034	N16 Tank Room-Wall	280	100	24,000	-540	-1	1	-2.8	-0.3
5173R0035	N16 Tank Room-Ceiling	220		530		n/a	n/a	-2.5	1.5
5173R0036	N16 Tank Room-Floor	43		870		n/a	n/a	-2.5	-1.6
5173R0037	N16 Tank Room-Drainage Trench	n/a		n/a		n/a	n/a	0.4	1.8
5173R0038	Source Cage-Floor	22		15,000		-1	4	-3.8	2.0
5173R0039	Reactor Room-Floor	-14		3,600		n/a	n/a	n/a	n/a

Table B-2. Radionuclide Concentration in Concrete

ORAU Sample ID	Concentration (pCi/g)							
	Eu-152	Eu-154	Co-60	Cs-137	Fe-55	Ni-63	H-3	C-14
5173M0001[a]	6.96 ± 0.45[b]	0.32 ± 0.20	3.52 ± 0.26	0.04 ± 0.09	11.06 ± 3.02	0.85 ± 0.66	35.3 ± 3.8	1.5 ± 1.9
5173M0002[c]	0.33 ± 0.09	-0.11 ± 0.25	3.18 ± 0.23	0.09 ± 0.07	-1.06 ± 2.45	2.96 ± 0.76	17.5 ± 2.9	33.2 ± 3.4
5173M0003[d]	-0.02 ± 0.07	0.03 ± 0.04	-0.01 ± 0.05	0.01 ± 0.03	-3.32 ± 2.34	0.72 ± 0.65	0.1 ± 2.5	0.7 ± 1.8
5173M0004[e]	-0.10 ± 0.10	-0.10 ± 0.21	0.02 ± 0.06	0.00 ± 0.05	-4.13 ± 2.36	1.05 ± 0.67	0.5 ± 2.5	-0.4 ± 1.7
5173M0005[f]	-0.03 ± 0.07	-0.11 ± 0.12	0.00 ± 0.04	-0.01 ± 0.03	0.39 ± 2.51	1.46 ± 0.68	0.2 ± 2.6	0.2 ± 1.8
5173M0006[g]	0.07 ± 0.11	0.07 ± 0.18	0.05 ± 0.06	0.04 ± 0.04	-3.42 ± 2.39	1.15 ± 0.67	2.1 ± 2.6	-0.1 ± 1.8
5173M0007[h]	-0.02 ± 0.05	-0.01 ± 0.06	0.01 ± 0.03	0.01 ± 0.02	-2.91 ± 2.38	0.79 ± 0.66	0.9 ± 2.6	0.3 ± 1.8
5173M0008i	7.82 ± 0.46	0.28 ± 0.18	2.57 ± 0.18	0.11 ± 0.04	10.76 ± 2.99	0.79 ± 0.65	128.6 ± 7.2	8.2 ± 2.0
5173M0009j	0.07 ± 0.07	-0.17 ± 0.25	0.54 ± 0.07	0.00 ± 0.05	2.96 ± 2.6	1.43 ± 0.69	4.3 ± 2.7	1.7 ± 1.8

[a]Sample collected from south floor inside the bioshield
[b]Uncertainties are reported at the 95% confidence interval
[c]Sample collected from west floor/wall interface inside the bioshield
[d]ORAU/ORISE Laboratory analysis of licensee sample SS-RRF-006 (2)
[e]ORAU/ORISE Laboratory analysis of licensee sample SS-RRF-013 (2)
[f]ORAU/ORISE Laboratory analysis of licensee sample SS-LAF-001 (2)
[g]ORAU/ORISE Laboratory analysis of licensee sample UI-NTF2-2 (2)
[h]ORAU/ORISE Laboratory analysis of licensee sample SS-BCT-NW (2)
[i]Sample collected from west beam port of the bioshield
[j]Sample collected from the N16 coolant tunnel upper wall under the bioshield

Table B-3. Alpha Spectroscopy Results of Volumetric Concrete Samples								
Sample ID	**Radionuclide Concentration (pCi/g)**							
	Am-241	**Pu-238**	**Pu-239**	**Th-228**	**Th-232**	**U-234**	**U-235**	**U-238**
5173M0008	0.00 ± 0.02[a]	0.01 ± 0.02	0.01 ± 0.01	0.51 ± 0.07	0.33 ± 0.05	0.29 ± 0.06	0.01 ± 0.01	0.31 ± 0.06
5173M0009	0.00 ± 0.02	0.02 ± 0.03	0.01 ± 0.01	0.29 ± 0.05	0.23 ± 0.05	0.36 ± 0.07	0.02 ± 0.01	0.41 ± 0.07

[a]Uncertainties are reported at the 95% confidence interval.

Table B-4. Inter-lab Comparison of Hard-to-Detect Sample Analysis

Sample ID		Radionuclide of Concern Concentration (pCi/g)					
		H-3		Fe-55		Ni-63	
ORAU	LVI	ORAU	LVI	ORAU	LVI	ORAU	LVI
5173M0003	SS-RRF-006	0.1 ± 2.5[a]	<MDC (2.73)[b]	-3.3 ± 2.3	<MDC (32.7)	0.72 ± 0.65	<MDC (0.45)
5173M0004	SS-RRF-013	0.5 ± 2.5	<MDC (2.57)	-4.1 ± 2.4	<MDC (24.6)	1.05 ± 0.67	<MDC (0.46)
5173M0005	SS-LAF-001	0.2 ± 2.6	<MDC (2.00)	0.4 ± 2.5	NAD[c]	1.46 ± 0.68	NAD
5173M0006	UI-NPF-2-2	2.1 ± 2.6	<MDC (2.52)	-3.4 ± 2.4	NAD	1.15 ± 0.67	NAD
5173M0007	SS-BCT-NW	0.9 ± 2.6	<MDC (2.26)	-2.9 ± 2.4	NAD	0.79 ± 0.66	NAD

[a]Uncertainties are reported at the 95% confidence interval.
[b]Parenthetical value represents the minimum detectable concentration of the respective contaminant in each sample.
[c]NAD=No analytical data was provided in the licensee’s analysis report.

Table B-5. Inter-lab Comparison of Gamma Spectroscopy Sample Analysis							
Sample ID		Radionuclide of Concern Concentration (pCi/g)					
		Co-60		Cs-137[a]		Eu-152	
ORISE	LVI	ORAU	LVI	ORAU	LVI	ORAU	LVI
5173M0003	SS-RRF-006	-0.01 ± 0.05[b]	NAD[c]	0.01 ± 0.03	NAD	-0.02 ± 0.07	NAD
5173M0004	SS-RRF-013	0.02 ± 0.06	NAD	0.00[d] ± 0.05	NAD	-0.10 ± 0.10	NAD
5173M0005	SS-LAF-001	0.00 ± 0.04	<MDC (0.04)[e]	-0.01 ± 0.03	<MDC (0.04)	-0.03 ± 0.07	<MDC (0.09)
5173M0006	UI-NPF-2-2	0.05 ± 0.06	<MDC (0.08)	0.04 ± 0.04	<MDC (0.07)	0.07 ± 0.11	<MDC (0.19)
5173M0007	SS-BCT-NW	0.01 ± 0.03	<MDC (0.05)	0.01 ± 0.02	<MDC (0.05)	-0.02 ± 0.05	<MDC (0.11)

[a]Laboratory analysis was not requested by the licensee.
[b]Uncertainties are reported at the 95% confidence interval.
[c]NAD=No analytical data was provided in the licensee's analysis report.
[d]Zero values for sample results are due to rounding.
[e]Parenthetical value represents the minimum detectable concentration of the respective contaminant in each sample.

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 x 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, TX)
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, TX)

C.1.2 ALPHA PLUS BETA

Ludlum Gas Proportional Detector Model 43-68, Physical area: 126 cm^2
(Ludlum Measurements, Inc., Sweetwater, TX)
Coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, TX)
Coupled to GeoXH Receiver and Data Logger
(Trimble Navigation Limited, Sunnyvale, CA)

Ludlum Pancake Probe Model 44-9, Physical area: 126 cm^2
(Ludlum Measurements, Inc., Sweetwater, TX)
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, two units: 26.3% and 27.7% efficiencies
(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)

High-Purity, Extended Range Intrinsic Detector
Model No. GMX-45200-5, 45% efficiency
(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.
(EG&G, 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/Caberra, 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)

APPENDIX D
SURVEY AND ANALYTICAL PROCEDURES

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, contractor representatives provided Oak Ridge Associated Universities (ORAU)with general safety information within the project area. 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 ORAU health and safety and radiation protection procedures (ORAU 2012c and 2011b).

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 (IEAV) Program:

- Survey Procedures Manual (ORAU 2012b)
- Laboratory Procedures Manual (ORISE 2012)
- Quality Program Manual (ORAU 2011a)

The procedures contained in these manuals were developed to meet the requirements of U.S. Department of Energy (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. Specific scan MDCs for the NaI detector was not determined as the instrument 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.

Beta scans were performed using small, hand-held gas proportional detectors with a 0.4 mg cm^{-2} window. Identification of elevated radiation levels was based on increases in the audible signal from the indicating instrument. Beta surface scan minimum detectable concentrations (MDCs) were estimated using the approach described in NUREG-1507. The scan MDC is a function of many variables, including the background level. Additional parameters selected for the calculation of scan MDCs included a two-second observation interval, a specified level of performance at the first scanning stage 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. The beta total efficiency was 0.10 for C-14. The detector used had a background of 305 cpm for concrete. The minimum detectable count rate (MDCR) and scan MDC was calculated as:

B_i = (305)(2 s)(1 min/60 s) = 10 counts

MDCR = (2.32)(10 counts)$^{1/2}$[(60 s/min)/2s] = 220 cpm

$MDCR_{surveyor}$ = 220/(0.5)$^{1/2}$ = 311 cpm

Scan MDC = (311)/[(.10)(1.26)] = 2,468 dpm/100 cm ^{2}

D.3.2 SURFACE ACTIVITY MEASUREMENTS

Measurements of total beta surface activity levels were performed using hand-held gas proportional detectors coupled to portable ratemeter-scalers. Count rates (cpm), which were integrated over one minute with the detector held in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the count rate by the total static efficiency ($\varepsilon_i \times \varepsilon_s$) and correcting for the physical area of the detector. ORAU collected material-specific backgrounds for each surface type encountered. The respective material-specific background was then subtracted from the direct gross count when determining surface activity. The *a priori* MDC for beta activity is given by:

$$MDC = \frac{3 + \left(4.65\sqrt{B}\right)}{G\ \varepsilon_{tot}}$$

Where:

B = background
ε_{tot} = total efficiency
G = geometry correction factor (1.26)

The *a priori* beta static MDC was approximately 688 dpm/100 cm^2 for C-14.

D.3.3 REMOVABLE ACTIVITY MEASUREMENTS

Removable gross alpha and gross beta activity levels were determined using numbered filter paper disks, 47 mm in diameter. Moderate pressure was applied to the smear and approximately 100 cm^2 of the surface was wiped. Smears were placed in labeled envelops with the location and other pertinent information recorded.

For tritium and C-14 determinations, a second smear was moistened with deionized water and an adjacent 100 cm^2 was wiped. The smear was then sealed in a labeled liquid scintillation vial with the location and pertinent information recorded.

D.3.4 CONCRETE SAMPLING

Approximately 0.5 kilogram of concrete was collected for samples 5173M0001, 5173M0002, and 5173M0008. ORAU personnel were only able to collect approximately 0.2 kilograms for sample 5173M0009. Each sample was placed in a plastic bag, sealed, and labeled in accordance with ORAU survey procedures.

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 are displayed in Table D-1.

Table D-1. MDC Derived from Total Absorption Peak		
Radionuclide	**TAP (MeV)**	**MDC (pCi/g)**
Eu-152	0.344	0.13
Eu-154	0.723	0.28
Co-60	0.661	0.06
Cs-137	1.173	0.08

Spectra were also reviewed for other identifiable TAPs.

D.4.2 GROSS ALPHA/GROSS BETA ANALYSIS

Smears were counted on a low-background gas proportional system for gross alpha and beta activity. The minimum detectable activities (MDA) of the procedure were 11 dpm and 14 dpm for alpha and beta, respectively.

D.4.3 TRITIUM ANALYSIS

Analyses for tritium were performed by placing a smear or a representative portion of the samples into a scintillation cocktail and counting on a liquid scintillation analyzer. Samples were then spiked with a known amount of the appropriate standard and recounted. The MDA of the procedure was 16.8 pCi.

D.4. DETECTION LIMITS

Detection limits, referred to as minimum detectable concentrations (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 from instrument to instrument.