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Further. Together.

February 24, 2016

Ms. Kimberly Conway
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Division of Decommissioning, Uranium Recovery, and Waste Programs
Reactor Decommissioning Branch
U.S. Nuclear Regulatory Commission
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SUBJECT: INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY ACTIVITIES ASSOCIATED WITH THE ALAN J. BLOTCKY REACTOR FACILITY, OMAHA, NEBRASKA [RFTA No. 16-005, Docket No. 03004530] DCN: 5282-SR-01-0

Dear Ms. Conway:

ORAU, via the Oak Ridge Institute for Science and Education (ORISE) contract, is pleased to provide the enclosed final report that details the confirmatory survey activities that were performed on December 8–9, 2015, at the Alan J. Blotcky Reactor Facility in Omaha, Nebraska. The survey activities were conducted in accordance with the project-specific plan provided to and approved by the U.S. Nuclear Regulatory Commission (NRC).

Please contact me at 865.574.0685 or Erika Bailey at 865.576.6659 should you have any questions.

Sincerely,

David A. King, CHP, PMP
Sr. Health Physicist/Project Manager
ORAU

DAK:fs

Enclosure

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INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY ACTIVITIES ASSOCIATED WITH THE ALAN J. BLOTCKY REACTOR FACILITY OMAHA, NEBRASKA

David A. King

Prepared for the U.S. Nuclear Regulatory Commission

February 2016

ORAU provides innovative scientific and technical solutions to advance research and education, protect public health and the environment and strengthen national security. Through specialized teams of experts, unique laboratory capabilities and access to a consortium of more than 100 major Ph.D.-granting institutions, ORAU works with federal, state, local and commercial customers to advance national priorities and serve the public interest. A 501(c)(3) nonprofit corporation and federal contractor, ORAU manages the Oak Ridge Institute for Science and Education (ORISE) for the U.S. Department of Energy (DOE). Learn more about ORAU at www.orau.org.

INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY ACTIVITIES ASSOCIATED WITH THE ALAN J. BLOTCKY REACTOR FACILITY, OMAHA, NEBRASKA

Prepared by
D. A. King
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FEBRUARY 2016

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

Prepared by Oak Ridge Associated Universities under the Oak Ridge Institute for Science and Education contract, number DE-AC05-06OR23100, with the U.S. Department of Energy under interagency agreement (NRC FIN No. F-1244) between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy.

INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY ACTIVITIES ASSOCIATED WITH THE ALAN J. BLOTCKY REACTOR FACILITY, OMAHA, NEBRASKA

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FINAL REPORT

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ACRONYMS

AJBRF	Alan J. Blotcky Reactor Facility
CFR	Code of Federal Regulations
DCGL	derived concentration guideline level
DQO	data quality objective
FSSP	final status survey plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
NRC	U.S. Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
PSP	project-specific plan
ROC	radionuclide of concern
SU	survey unit
TRIGA	Training, Research, Isotopes, General Atomics
VA	Veterans Affairs

INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY ACTIVITIES ASSOCIATED WITH THE ALAN J. BLOTCKY REACTOR FACILITY, OMAHA, NEBRASKA

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) requested that ORAU, via the Oak Ridge Institute for Science and Education (ORISE) contract, perform confirmatory survey activities of the Alan J. Blotcky Reactor Facility (AJBRF) in Omaha, Nebraska. The AJBRF is approximately 2,500-ft^2 and was used to support nuclear medicine and research programs conducted at the Omaha Veteran Affairs (VA) medical center. Confirmatory activities are intended to ensure, if supported by the data, that AJBRF complies with the 25 mrem/year criteria in 10 Code of Federal Regulations Part 20, Subpart E. ORAU reviewed the final status survey plan and discovered no information or data that questions the quality or validity of reported results or suggests a significant risk of false negative decisions. The confirmatory survey, performed on December 8–9, 2015, included cursory gamma scans (100% of the facility floor), judgmental scans of multiple surfaces, direct measurements, and smear collection (wet and dry). ORAU and VA site data overwhelmingly support the conclusion that the AJBRF meets the conditions for unconditional release.

INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY ACTIVITIES ASSOCIATED WITH THE ALAN J. BLOTCKY REACTOR FACILITY, OMAHA, NEBRASKA

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) requested that ORAU, via the Oak Ridge Institute for Science and Education (ORISE) contract, perform confirmatory survey activities of the Alan J. Blotcky Reactor Facility (AJBRF) in Omaha, Nebraska. The AJBRF is a U.S. Department of Veterans Affairs (VA) facility operated under NRC Facility Operating License R-57. The pool-type reactor facility was previously fueled with standard Training, Research, Isotopes, General Atomics (TRIGA) fuel elements enriched to less than 20% uranium-235 zirconium hydride. Fuel elements were removed in June 2002 and shipped off-site. Radionuclides of concern (ROCs) are beta- and beta-gamma-emitting fission and activation products: H-3, C-14, Fe-55, Co-60, Ni-63, and Cs-137. Release criteria have been established per NUREG-1757 Appendix B (NRC 2006).

The VA contractor, NorthStar, performed final status surveys based on guidance provided in the *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (NUREG-1575) to facilitate license termination (NRC 2000 and NorthStar 2015). ORAU reviewed the NorthStar final status survey plan (FSSP), performed confirmatory survey activities on December 8 and 9, 2015, and then reviewed raw survey data tables, collected per the FSSP. Confirmatory activities are intended to ensure, if supported by the data, that AJBRF complies with the 25 mrem/yr criteria in 10 Code of Federal Regulations (CFR) Part 20, Subpart E. The project-specific plan (PSP) presents the approach for performing confirmatory activities for this decommissioning effort (ORAU 2015a).

2. SITE DESCRIPTION

The approximately 2,500-ft^2 AJBRF, illustrated in Figure 2.1, is constructed of brick and reinforced concrete, including the floors, walls, and ceiling. Table 2.1 lists and describes the rooms and areas within the Radioisotope Reactor Research Laboratory (B526). The low-power TRIGA nuclear reactor was operated as a source for neutron activation analysis of biological samples and for hot atom chemistry research. Additionally, from 1989 to 2001, the reactor was used for training Fort Calhoun Station nuclear power reactor operators. Historically, samples to be irradiated in the reactor

were typically prepared in rooms B537, B535, and B533A. Irradiated samples were then processed in room B540 and stored in the isotope storage area B540A (AECOM 2011).

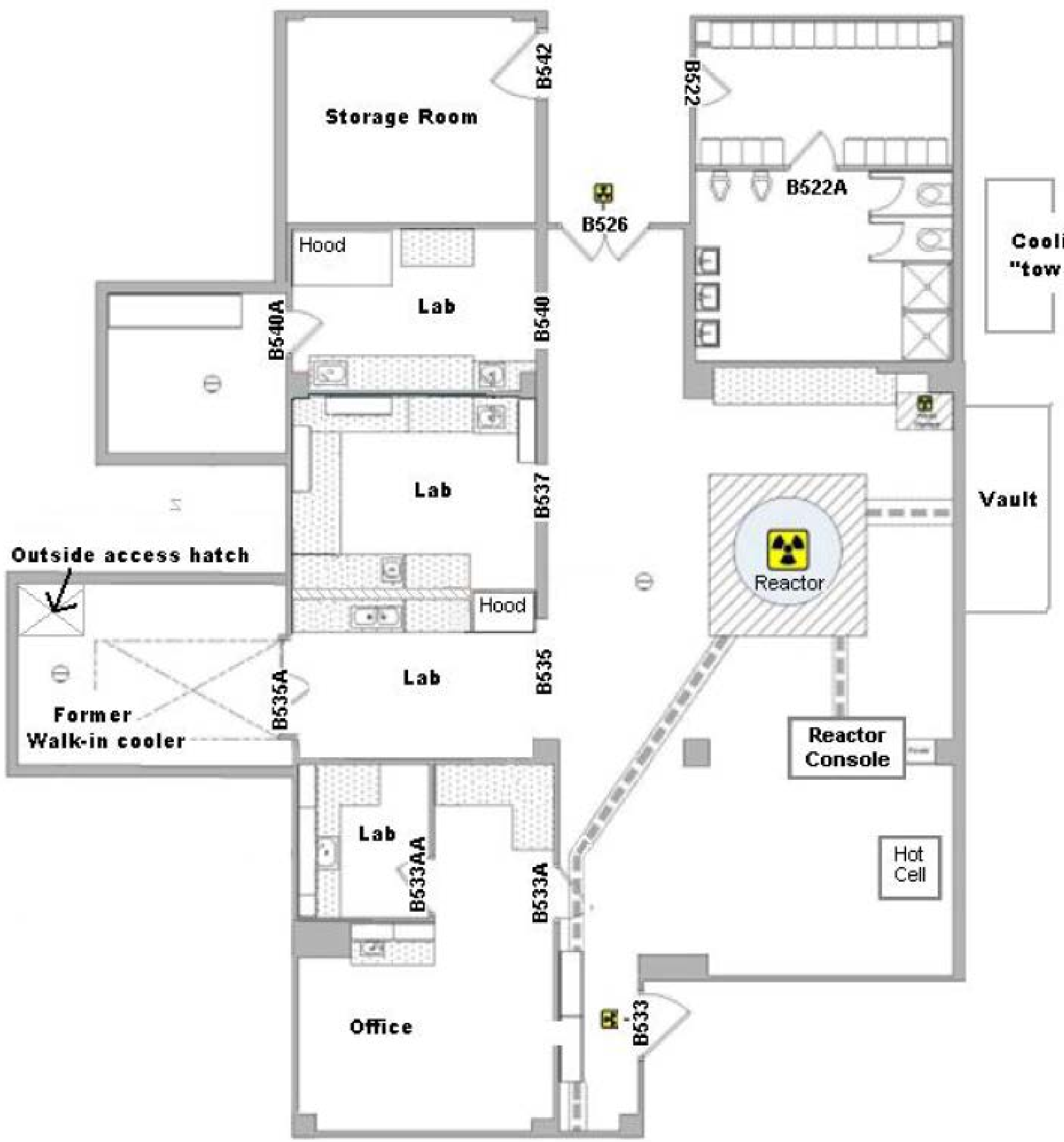

Figure 2.1. The Alan J. Blotcky Reactor Facility (AECOM 2011)

Table 2.1. List and Description of AJBRF Rooms and Areas			
New (Old) Rm No.	Description	Former Use	Current Use
B522 (SW 1)	Locker room	Storage of personal items by hospital staff	Storage of personal items by hospital staff
B522A (SW 1A)	Restroom and shower	Restroom and shower for hospital staff	Restroom and shower for hospital staff
B526 (SW 2)	Radioisotope reactor research laboratory	Research activities and storage; contains one of two fume hoods	None
B533A (SW 2A)	Nuclear research lab and office	Sample preparation	None
B533AA (SW 2B)	Office/darkroom	Darkroom, office, and storage space	None
B537 (SW 2C)	Nuclear research lab and office	Sample preparation	None
B535A (SW 2D)	Walk-in cooler	Cold storage	None
B540 (SW 2E)	Nuclear research lab and office	Sample processing; contains one of two fume hoods	None
B540A (SW 2F)	Isotope and general storage	Storage of irradiated samples	None

Reactor construction began on January 8, 1959, and was completed on June 24, 1959, in accordance with the construction permit CPRR-36, the provisions of the Atomic Energy Act of 1954, and the regulations of the Atomic Energy Commission. An operating license was issued on June 26, 1959, two days after construction was finished. The initial license set operating parameters for the reactor, including a maximum steady state operating power of 18 kW thermal. There have been a total of 11 amendments to the license so far, with the most recent amendment in 2002. Most notable of the amendments was Amendment 9, issued on April 12, 1991, which increased the maximum allowed steady state power output of the reactor to 20 kW thermal, and also provided for the installation of a microprocessor-based neutron monitoring system. The reactor operated at a maximum of 20 kW thermal from then until final shutdown on November 5, 2001. The total integrated power generated during the operation of the AJBRF was 515,058 kW-hours (AECOM 2014).

The AJBRF was used to support nuclear medicine and research programs conducted at the Omaha VA medical center. Between 1959 and 1965, the facility was funded as a national laboratory and

employed approximately 30 people. The principal use of the reactor was for neutron activation of biological samples. Typical irradiation times were up to 60 minutes in duration. Sample vials were opened in fume hoods to allow argon-41 gas (Ar-41; half-life of 1.8 hours) to vent to the atmosphere. Additionally, from 1989 until shutdown in 2001, and as noted earlier, the reactor was used for training Fort Calhoun Station nuclear power reactor operators (AECOM 2014).

The reactor room ventilation supply provides 100% outside air, heated or cooled, to the reactor laboratory through six ceiling ducts. The exhaust exits the reactor room to the outside air through either an exhaust fan installed in the outside wall of the building or one of two continuously operated laboratory fume hoods. The exhaust suction fans are located in a small penthouse on the roof above the 12th story of the medical center. Because the hood exhaust is operated as a suction system, the entire ductwork is under negative pressure; therefore any air leakage would be into the duct rather than out, eliminating the potential for exposure within the medical center (AECOM 2011).

The survey units (SUs) were classified (by the contractor) based on contamination potential, as either Class 1, 2, or 3 in accordance with MARSSIM (NRC 2000). A description of each class designation is as 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 are expected to exceed site derived concentration guideline level (DCGL) values.

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 DCGLs.

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 DCGLs.

The survey areas and classifications based on the MARSSIM are listed in Table 2.2.

Table 2.2. AJBRF Survey Units and MARSSIM Classification		
Survey Area		**MARSSIM Class**
Reactor tank wall	Portions not removed	1
Reactor tank pit	Exposed concrete and/or soil	1
Reactor water cooling system vault	Floors and walls	1
Rooms B526, B535, B535A, B537, B540, and B540A	Floors	1
	Walls <2 meters	2
	Walls >2 meter and ceiling	3
Rooms B533A and B533AA	Floors and walls <2 meters	2
	Walls >2 meter and ceiling	3
Rooms B522 and B522A	Floors only	3
Hall outside Room B526	Floors only	2
Stairs on south side of Room B526	Floors only	2

3. OBJECTIVES

The project objective is to provide an authoritative and unbiased assessment of the licensee's radiological release process, including procedures and methodologies, and to generate survey data that may be used to independently evaluate the suitability of building materials for unconditional release. ORAU performed these tasks through the application of a formal data quality objectives (DQOs) process for planning confirmatory investigations, performing independent radiation surveys, and ensuring that the type, quality, and quantity of data collected are adequate for the intended decision applications (ORAU 2015a).

4. DOCUMENT REVIEW

Prior to on-site activities, ORAU reviewed the licensee's FSSP (NorthStar 2015) for consistency with the industry-accepted radiological survey practices described in MARSSIM and related documents such as ISO-7503 and NUREG-1507 (NRC 2000, ISO 1988, and NRC 1997). ORAU also reviewed NorthStar's preliminary data reports and evaluated results relative to the ORAU results and site DCGLs. Finally, ORAU reviewed provided instrumentation paperwork (e.g., calibration certificates) to assess the quality of NorthStar measurement data.

5. APPLICABLE SITE GUIDELINES

The FSSP presents release criteria for the AJBRF ROCs, taken from NUREG-1757 Appendix B. The default criteria have been approved as DCGLs for the project. The following is a list of ROCs and respective DCGLs for the AJBRF. It is noted that the FSSP presents DCGLs for underlying soils, though this confirmatory effort is limited to structural materials.

ROC	DCGL (dpm/100 cm^2)
H-3	1.2E+08
C-14	3.7E+06
Fe-55	4.5E+06
Co-60	7.1E+03
Ni-63	1.8E+06
Cs-137	2.8E+04

DCGLs are applicable to each SU per standard MARSSIM guidance, including overall average, statistical testing, and small area (i.e., "hot spot") requirements. Nickel-63 and Fe-55 were confirmed as ROCs in the reactor water filter resins. Therefore, these hard-to-detect isotopes are potentially present in activated materials and contaminated items. However, the easy-to-detect Co-60 was also present along with Ni-63 and Fe-55 at consistent ratios of about 3.5:1 and 3:1, respectively. Cesium-137 was also detected in the resin sample at a Co-to-Cs ratio of approximately 15:1 (AECOM 2011, Table 3). Characterization data from subsurface piping and pneumatic transfer lines indicate that, while there are detectable levels of H-3, C-14, and total beta activity, the activity in the embedded drain lines and pneumatic tubing is well below screening criteria (AECOM 2011). To simplify, each of the ROCs and respective DCGLs are listed again below, relative to the primary contaminant, Co-60 (AECOM 2011, Table 3):

ROC	DCGL (dpm/100 cm^2)	Co-60:ROC Ratio	Relative Activity
H-3	1.2E+08	20 : 1	0.05
C-14	3.7E+06	20 : 1	0.05
Fe-55	4.5E+06	3 : 1	0.33
Co-60	7.1E+03	1 : 1	1.0
Ni-63	1.8E+06	3.5 : 1	0.29
Cs-137	2.8E+04	15 : 1	0.067

Using these activity ratios, the adjusted gross DCGL is 12,500 dpm/100 cm^2, to three significant digits. Beta instruments were calibrated considering these relative activities and using a multi-point calibration procedure and sources of C-14, Tc-99, Tl-204, and Sr-90. Other procedures are generally described in the following section.

6. SURVEY PROCEDURES

At NRC's request, ORAU performed confirmatory survey activities at AJBRF. Confirmatory survey activities included visual inspection, surface scans, surface activity measurements, and sample (smear) collection. The confirmatory survey activities were conducted in accordance with the PSP, the *ORAU Radiological and Environmental Survey Procedures Manual,* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2015a, ORAU 2015b, and ORAU 2015c). Appendix A provides a detailed outline of the survey procedures and analytical methods used. 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. ORAU instrumentation is presented in Appendix B.

6.2 SURFACE SCANS

Surface scans for gamma and beta radiation were performed on areas with the highest potential for contamination, primarily including floors and other horizontal surfaces. Scans were performed using Ludlum Model 44-10 sodium iodide (NaI[Tl]) scintillation detectors for gamma radiation, large-area Ludlum Model 43-37 gas proportional detectors for floors, and hand-held Ludlum Model 43-68 gas proportional detectors for beta radiation scans of walls and accessible overhead structures. All detectors were coupled to ratemeter-scalers with audible output. Measurement ranges were recorded on scaled drawings.

Surveyors initiated activities by using the NaI[Tl] detector to perform general area scans to identify gross sources of gamma radiation that could, if located, indicate the presence of contamination. This was followed by the systematic, 100% survey of floors with the Model 43-37 floor monitor and investigation of potentially anomalous readings with the Model 43-68 hand-held gas proportional detector. The 43-68 was also used to scan horizontal surfaces within the facility. The Model 43-68 was fitted with a 0.4 mg/cm^2 Mylar to optimize the measurement of hard-to-detect beta emitters.

6.3 Surface Activity Measurements

Direct measurement data, using the Model 43-68, were used to quantify total beta activity as necessary for decision making, primarily using judgmental locations. Material-specific background measurements were collected as necessary from non-impacted structures or surfaces of, to the extent possible, similar construction to the SU construction materials. These background measurements were used for correcting gross SU measurement results when converting the data to surface activity levels. Ten reference area measurements were collected from terra cotta, poured concrete walls, poured concrete floors, and engineered tile. Table 6.1 presents reference area data and the resulting minimum detectable concentration (MDC) calculations for each medium. Note that a reference gunite material was not available so a conservative average of 300 counts per minute (cpm) is assumed.

Table 6.1. Reference Material Averages and MDC Calculations

Meas. No.	12×12 Painted Terra Cotta[a]	Poured Painted Concrete Wall[b]	Bare/Stripped Concrete Floor[c]	"Marble" Engineered Tile[d]	Gunite[e]
Medium specific reference data (cpm)					
1	504	371	430	525	—
2	506	396	414	559	—
3	498	355	458	551	—
4	518	398	365	498	—
5	535	369	369	560	—
6	495	420	349	612	—
7	428	346	370	546	—
8	416	413	405	535	—
9	516	383	381	568	—
10	526	371	413	569	—
Averages	494	382	395	552	300
Medium-Specific MDCs (dpm/100 cm^2)					
MDCs	1,170	1,040	1,050	1,240	920

[a] Rm B542 adjacent to study area
[b] Rm B542 adjacent to study area
[c] Rm. B589 Plumbing Shop
[d] Rm 3754
[e] Reference material not found in VA facility; 300 cpm conservatively assumed

Direct measurement (1-minute-count) data were collected at 35 judgmentally selected locations. In some cases the locations correspond to a VA-assigned random coordinate, in some cases the location was selected to represent a feature of interest (e.g., a drainage trench), and in the balance of cases the location was selected based on floor monitoring results. For the latter, the general area of potentially elevated activity was scanned and the specific spot with the highest values was subject to direct measurement. Smears were also collected from these locations.

6.4 REMOVABLE ACTIVITY SAMPLING

Smear samples were collected to quantify the removable gross alpha and beta (dry smear) and hard-to-detect (wet smear) activity. Dry smears were collected at all 35 judgmentally selected locations and subject to gross alpha and gross beta measurement. Wet smears were also collected from the five trenches leading to/from the pit and from two concrete vault locations. These wet smears were subject to tritium and C-14 analysis.

7. SAMPLE ANALYSIS AND DATA INTERPRETATION

Smears were returned to the Radiological and Environmental Analytical Laboratory (REAL) in Oak Ridge, Tennessee for analysis and interpretation. Sample analyses were performed in accordance with the *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2015d). Dry smear samples were analyzed for gross alpha and gross beta activity using a low-background gas proportional counter. The analytical results and (dry) surface activity measurement data were reported in units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2). REAL analyzed wet smears using a liquid scintillation analyzer and reported results in dpm/smear (which translates to dpm/100 cm^2).

8. FINDINGS AND RESULTS

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

8.1 Document Review

ORAU reviewed the FSSP titled *Final Status Survey Plan –AJ Blotcky Reactor Facility Decommissioning Project* (NorthStar 2015) and submitted minor comments on December 3, 2015. Reviewers were also provided with instrumentation paperwork and measurement results from randomly-selected locations. ORAU discovered no information or data that suggests a significant risk of false negative decisions. However, a few issues are noted for the record:

- Beta measurement efficiencies were calculated using a single beta source and do not account for the relative activities discussed in Section 5 of this report. A similar comment was offered by ORAU during the FSSP review.

- Direct measurement count times of six seconds were also used, which is not ideal for estimating cpm and dpm measurement data. A similar comment was offered by ORAU during the FSSP review.

- All direct measurement results (total is the most relevant here), presented in site-provided files AJBFF-001 through -005, are listed as "<MDA." This appears to be in error given several results are above the listed background activity. Also, there is a significant disconnect between the results on these sheets and the "Instrument Reference Check Determination" for the Model 43-93.

- The same background value of 120 (counts every six seconds, or 1,200 cpm) is used for all media. This seems to be non-conservative, though measurement data across the facility are on the same order of magnitude and all results are still below the gross adjusted DCGL of 12,500 dpm/100 cm^2.

Fortunately, the facility is clean relative to DCGLs, and the associated measurement data are adequate for decision-making.

8.2 Surface Scans

The floor monitor survey located a small area (~1 in^2) on the floor west of the reactor pit containing significantly elevated beta activity. The VA used a hammer to spot-remediate the area, lowering the

original 5,960 gross cpm activity to 502 cpm. The latter is higher than the average background value of 395 cpm but is within the range of measurements collected across facility floors. Besides this single location, there were no significantly elevated results.

8.3 SURFACE ACTIVITY MEASUREMENTS

Table 8.1 presents results from the 35 discrete measurement locations. The table includes the ORAU and VA location numbers, when applicable, the gross activity measurement (via the Model 43-68), the associated reference values, wet and dry smear results, and a short description of the target location. As shown, no direct measurement, dry smear, or wet smear result approaches the most restrictive DCGL.

The maximum direct measurement was collected near the bottom of the reactor pit, producing a Model 43-68 response of 970 cpm for approximately 6,346 dpm/100 cm^2. Very small amounts of removable activity were "detected" with a maximum of ~4 dpm/100 cm^2 gross alpha and less than 9 dpm/100 cm^2 gross beta. The maximum tritium and C-14 results are ~5 dpm/100 cm^2 and less than 10 dpm/100 cm^2, respectively. The review of liquid scintillation spectra identified no additional peaks that would suggest the presence of other hard-to-detect contaminants. All values are below the adjusted gross DCGL of 12,500 dpm/100 cm^2.

ORAU data were not compared directly to VA data because of significant differences in detector efficiency calculation methods, reference values used, and other factors as previously discussed. However, both the VA and ORAU data demonstrate surface activities consistent with background or otherwise significantly lower than the most restrictive DCGLs.

Table 8.1 Confirmatory Direct Measurement and Smear Results for the AJBRF										
ORAU	VA		Gross	Ref.	Totals Result	Removable (dpm/100 cm^2)				
Loc. No.	Loc. No.	Smear	cpm[a]	cpm[b]	(dpm/ 100 cm^2)	H-3	C-14	GA	GB	Medium and location description
1	24	Dry	340	300	442	—	—	-0.37	1.52	Gunite wall in reactor pit
2	25	Dry	287	300	0	—	—	-0.37	-0.93	Gunite wall in reactor pit
3	19	Dry	310	300	110	—	—	-0.37	-0.93	Gunite wall in reactor pit
4	13	Dry	316	300	177	—	—	-0.37	5.2	Gunite wall in reactor pit
5	7	Dry	371	300	784	—	—	-0.37	1.52	Gunite wall in reactor pit
6	10	Dry	521	395	1,387	—	—	-0.37	1.52	Concrete after gunite removed in reactor pit
7	30	Dry	552	395	1,729	—	—	-0.37	0.29	Concrete after gunite removed in reactor pit
8	31	Dry	576	395	1,994	—	—	4.21	1.52	Concrete after gunite removed in reactor pit
9	33	Dry	586	395	2,105	—	—	-0.37	1.52	Concrete after gunite removed in reactor pit
10	Judge.	Dry	970	395	6,346	—	—	-0.37	2.75	Concrete after gunite removed in reactor pit, 3 ft above base, due north
11	31	Wet	339	395	0	-3.3	8.2	—	—	Concrete adjacent to reactor pit
12	31	Dry	339	395	0	—	—	1.92	0.29	Concrete adjacent to reactor pit
13	Judge.	Wet	478	395	912	-10.4	4.4	—	—	Concrete southwest trench
14	Judge.	Dry	478	395	912	—	—	-0.37	2.75	Concrete southwest trench
15	Judge.	Wet	522	395	1,398	-15.8	-1.6	—	—	Concrete trench near B533A
16	Judge.	Dry	522	395	1,398	—	—	4.21	2.75	Concrete trench near B533A
17	Judge.	Wet	493	395	1,078	-2.2	0.9	—	—	Concrete south trench
18	Judge.	Dry	493	395	1,078	—	—	-0.37	-2.16	Concrete south trench

Table 8.1 Confirmatory Direct Measurement and Smear Results for the AJBRF										
ORAU	VA		Gross	Ref.	Totals Result	Removable (dpm/100 cm²)				
Loc. No.	Loc. No.	Smear	cpm[a]	cpm[b]	(dpm/ 100 cm²)	H-3	C-14	GA	GB	Medium and location description
19	*Vial Dropped/Broken—Resampled (see ORAU No. 21)*									
20	Judge.	Dry	547	395	1,674	—	—	1.92	-2.16	Concrete east trench
21	Judge.	Wet	547	395	1,674	5.1	-3.8	—	—	Concrete east trench
22	Judge.	Dry	664	395	2,966	—	—	-0.37	-2.16	Vault sump
23	Judge.	Wet	664	395	2,966	-6.4	-0.2	—	—	Vault sump
24	Judge.[d]	Dry	578	395	2,017	—	—	1.92	6.42	Concrete vault wall
25	1	Wet	604	395	2,304	-6.4	6.7	—	—	Concrete vault floor
26[c]	1	Dry	604	395	2,304	—	—	1.92	0.29	Concrete vault floor
27	Judge.	Dry	620	395	2,480	—	—	-0.37	8.87	Concrete NW floor in B526
28	Judge.	Dry	502	395	1,177	—	—	-0.37	1.52	Concrete floor west of pit after hot spot removed
29	Judge.	Dry	554	395	1,751	—	—	-0.37	0.29	Concrete floor B535 doorway
30	Judge.	Dry	587	395	2,116	—	—	-0.37	1.52	Concrete floor B537 doorway
31	12	Dry	628	395	2,569	—	—	-0.37	1.52	Concrete B535A floor
32	14	Dry	614	395	2,414	—	—	-0.37	1.52	Concrete B540A floor
33	15	Dry	548	395	1,685	—	—	-0.37	-2.16	Concrete B540 floor
34	23	Dry	420	382	417	—	—	1.92	5.2	Concrete B526 wall
35	20	Dry	525	494	340	—	—	-0.37	1.52	12×12 painted terra cotta tile, B526 wall

[a] 43-68 in alpha plus beta mode, 0.4 mg/cm² Mylar
[b] 43-68 in alpha plus beta mode, 0.4 mg/cm² Mylar, see reference dataset
[c] Entry correction: logbook lists smear 26 as wet but it was actually a dry smear
[d] Was not determined (possibly vault Location 1), but location has vault floor coordinate (4,3)

9. SUMMARY

At the NRC's request, ORAU conducted confirmatory survey activities within the AJBRF, a VA facility operated under NRC Facility Operating License R-57. The survey activities, performed on December 8–9, 2015, included cursory gamma scans, 100% of the facility floor, judgmental scans of multiple surfaces, direct measurements, and smear collection (wet and dry).

Only one location was identified with significantly elevated activity, and that location was satisfactorily remediated soon after discovery. The VA FSSP, instrumentation paperwork, and measurement data were reviewed and ORAU found them satisfactory. While ORAU does not agree with every VA method and specific comments have been offered for consideration, ORAU and VA site data overwhelmingly support the conclusion that residual activity levels satisfy the DCGLs.

10. REFERENCES

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ORAU 2015a. *Project-Specific Plan for the Confirmatory Survey of the Alan J. Blotcky Reactor Facility, Omaha, Nebraska*. DCN 5282-PL-01-0. Oak Ridge Associated Universities. Oak Ridge, Tennessee. December 2.

ORAU 2015b. *ORAU Radiological and Environmental Survey Procedures Manual*. Oak Ridge Associated Universities. Oak Ridge, Tennessee. May 7.

ORAU 2015c. *ORAU Environmental Services and Radiation Training Quality Program Manual*. Oak Ridge Associated Universities. Oak Ridge, Tennessee. August 7.

ORAU 2015d. *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual*. Oak Ridge Associated Universities. Oak Ridge, Tennessee. August 8.

ORAU 2015e. *ORAU Health and Safety Manual*. Oak Ridge Associated Universities. Oak Ridge, Tennessee. June.

APPENDIX A
SURVEY AND ANALYTICAL PROCEDURES

A.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 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. ORAU also had a site escort at all times due to the various alarms and notifications associated with an active industrial facility. All survey and laboratory activities were conducted in accordance with ORAU health and safety and radiation protection procedures (ORAU 2015e and 2014).

A.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:

- *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2015b)
- *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2015c)
- *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2015d)

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

Quality control procedures include:

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

A.3. SURVEY PROCEDURES

A.3.1 SURFACE SCANS

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.

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 were not determined as the instrument was used solely as a qualitative means to identify elevated gamma radiation levels in excess of background. Beta scans were performed using small, hand-held gas proportional detectors with a 0.4 mg cm^{-2} window.. Beta surface scan MDCs were estimated using the approach described in NUREG-1507 (NRC 1997). 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 weighted beta total efficiency of 0.072 was calculated using a multi-point calibration procedure and sources of C-14, Tc-99, Tl-204, and Sr-90. The minimum detectable count rate (MDCR) and scan MDC were calculated, for example, considering a background count rate of 390 cpm:

$B_i = (390)\ (2\ s)\ (1\ min/60\ s) = 13\ counts$
$MDCR = (2.32)\ (13\ counts)^{1/2}\ [(60\ s/min)/2s] = 251\ cpm$
$MDCR_{surveyor} = 251/(0.5)^{1/2} = 355\ cpm$
$Scan\ MDC = (355)\ /\ (0.072 \times 1.26) \approx 3{,}900\ dpm/100\ cm^2$

A.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 determined construction material-specific background for each surface type encountered for determining net count rates. 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 1,100 dpm/100 cm^2 for the AJBRF source described in the main document, assuming an average background of about 390 cpom.

A.3.3 REMOVABLE ACTIVITY MEASUREMENTS

Removable gross alpha and gross beta activity levels were collected 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 envelopes 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.

A.4. RADIOLOGICAL ANALYSIS

A.4.1 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.

A.4.2 Tritium and C-14 Analysis

Analyses for tritium and C-14 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 14.1 pCi/smear for tritium and 9.1 pCi/smear for C-14.

A.4.3 Detection Limits

Detection limits, referred to as MDCs, were based on 95% confidence level via the NUREG-1507 method (NRC 1997). 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.

APPENDIX B
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.

B.1 SCANNING AND MEASUREMENT INSTRUMENT/DETECTOR COMBINATIONS

B.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)

B.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)

Ludlum Gas Proportional Floor Monitor Model 43-37, Physical area: 584 cm^2
(Ludlum Measurements, Inc., Sweetwater, TX)
Coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, TX)

B.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)

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)

Low-Background Gas Proportional Counter
Model LB-5100-W
(Tennelec/Caberra, Meriden, CT)

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