Prepared for
VHA's Veterans Service Center
4141 Rockslide Road, Suite 110
Severn Hills, Ohio 44131
GSA No. GS-10F-0115K

Prepared by
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AJ Blotcky Reactor Facility Characterization Report

VA Nebraska-Western Iowa Health Care System

Omaha, Nebraska
Order No. VA701-BP-004/VA-101-G05020
July 2011

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1.0 Introduction

The Alan J. Blotcky reactor facility located within the United States (U.S.) Department of Veterans Affairs (VA) Medical Center (VAMC) in the city of Omaha, Douglas County, Nebraska maintains U.S. Nuclear Regulatory Commission (NRC) Facility Operating License R-57. The reactor is housed in the basement of the southwest wing of the medical center building. The Omaha VAMC is part of the VA Nebraska-Western Iowa Health Care System.

The reactor is a Training, Research, Isotopes, and General Atomics (TRIGA) MARK I Reactor, owned by the VA. It is a pool-type facility that was previously fueled with standard TRIGA fuel elements enriched to less than 20% uranium-235 zirconium hydride. Fuel elements were removed in June 2002 and shipped to the U.S. Geological Survey TRIGA reactor in Denver, Colorado.

The VAMC building is constructed of brick and reinforced concrete, including the floors, walls, and ceiling. Entrance to the reactor laboratory is normally through the secured door marked B526 on Figure 1.

Table 1 lists the rooms and areas within the Radioisotope Reactor Research Laboratory (B526). 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.

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. Since 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.

1.1 Site Characterization

A previous characterization of the reactor facility was performed in 2003 (Duratek 2003). This characterization included rooms, ventilation systems, drainage systems, cooling systems, storage areas, the reactor structures, and outside areas. The purpose of the characterization survey was to collect sufficient survey data to allow VA to develop a detailed Decommissioning Plan (DP, VA 2004). Upon its review of the draft decommissioning plan, NRC requested that additional characterization be performed to support conclusions and objectives presented in the plan (NRC 2008). The specific items noted in the Request for Additional Information (RAI) were the subject of discussion during an on-site meeting between VA and NRC in October 2010, as well as subsequent teleconference discussions.

In April 2011 AECOM prepared a Characterization Work Plan (AECOM 2011) to direct the additional characterization effort and provide the procedures to be used in addressing the additional items. The work plan was reviewed by both the VA and the NRC prior to mobilization.

Section 4 of the work plan contained the Sampling and Analysis Plan which described the survey and sampling protocols.

AECOM mobilized to the Omaha facility in May 2011 to perform the additional characterization activities as described in the work plan. Characterization activities included collecting surface and subsurface soil samples, surveying wastewater drain-lines and lab hood exhaust ducts, and performing additional surveys in the general reactor area. Additionally, samples from the reactor tank water demineralizer resins were collected to address questions about the isotope mix to be used in evaluating residual contamination. Results of these characterization activities are provided in the following sections of this report.

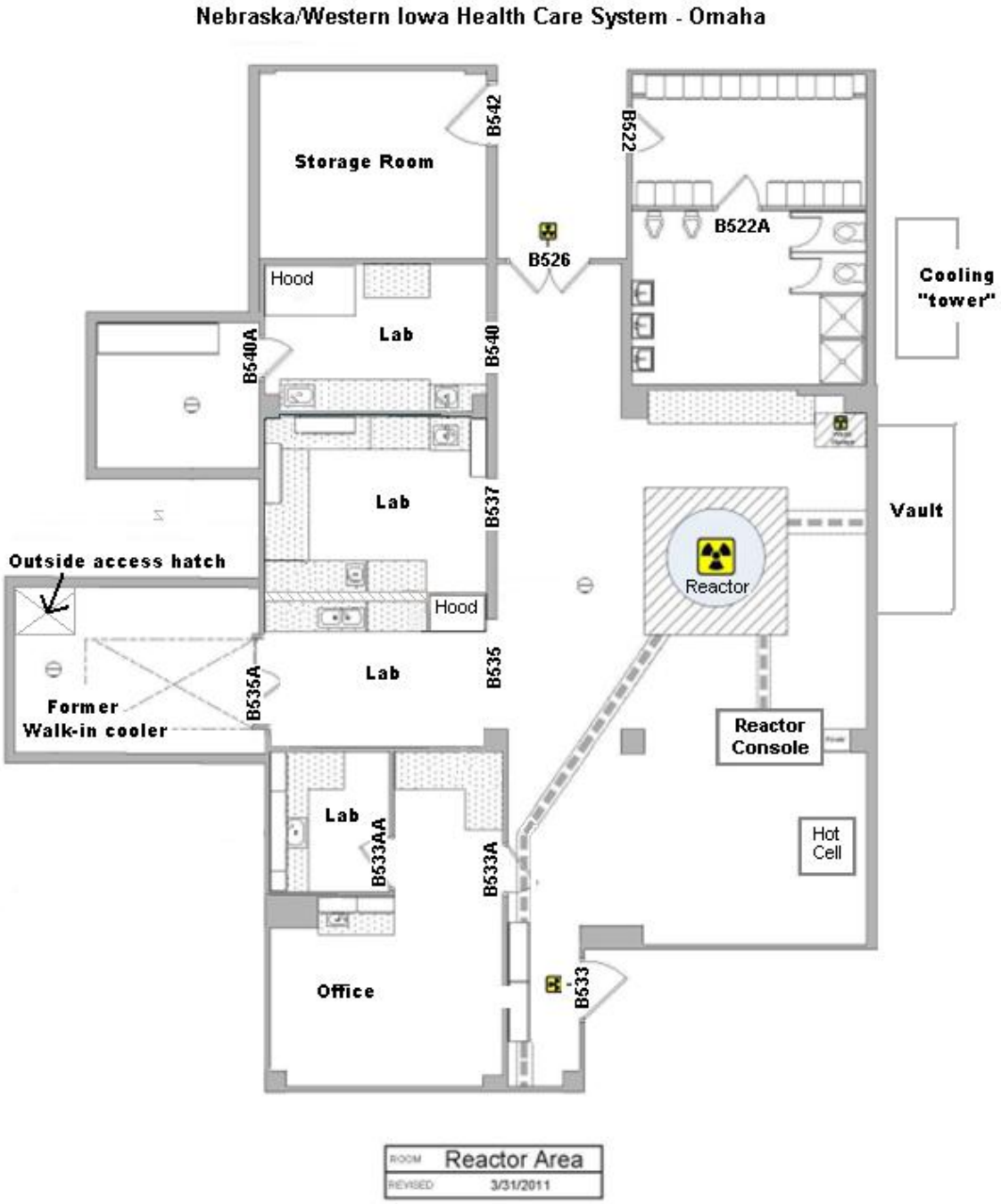
1.2 Additional Site Activities

In addition to the characterization activities, AECOM removed a significant amount of non-operational loose materials (e.g., desks, chairs, bookcases, etc.) from the reactor area after surveying the materials and demonstrating that they met free-release criteria. Along with the loose materials free-released, the walk-in cooler was disassembled and free-released to allow direct access to the outside of the facility (through roof of the cooler alcove) during decontamination efforts. A description of these activities, a listing of all free-released items, and the accompanying survey data are documented in the AJ Blotcky Reactor Facility Free Release Materials Report provided to the VA.

Table 1. Reactor Area Rooms

Room Numbers New (old)	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 1 of 2 fume hoods	None
B533A (SW 2A)	Nuclear research lab and office	Sample Preperation	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 1 of 2 fume hoods	None
B540A (SW 2F)	Isotope and general storage	Storage of irradiated samples	None

Figure 1. Reactor Area Map



2.0 Characterization Activities

2.1 Resin Bed Sampling

The reactor water demineralizer tank, located in the water cooling system "vault" or "pit" outside the building foundation east of the reactor area was sampled to determine the mix of isotopes that may be applicable to the contamination in and around the reactor. Previous resin sample analysis reported identified several isotopes of concern (Duratek 2003) that were not expected and the analysis or reporting of the analysis was called into question by the NRC (NRC 2008). Three resin samples were collected from the tank in a layered approach from the top, middle, and bottom of the tank to ensure representative sampling.

Resin samples were analyzed for ten isotopes using the technique and equivalent methods as listed in Table 2. Analysis included hard-to-detect (HTD) isotopes as well as gamma spectroscopy for activated metals. Analysis was performed by TestAmerica Laboratories (St. Louis, Missouri). Analytical methods were selected based on potential contaminants and to confirm or dismiss uncertain results from the previous sampling effort. Table 3 provides a summary of the results with the complete results are provided in Appendix A.

Table 2. Resin Sample Analytical Requirements

Isotope or Isotope Series	Analysis Technique	Analysis Method
Tritium (H-3)	liquid scintillation counting	EML HASL 300
Carbon-14 (C-14)	liquid scintillation counting	EERRF – C01
Iron-55 (Fe-55)	liquid scintillation counting	Column Separation
Nickel-59/63 (Ni-59/63)	liquid scintillation counting	EML HASL 300
Polonium-210 (Po-210)	alpha spectroscopy	EML HASL 300
Plutonium-241 (Pu-241)	liquid scintillation counting	EML HASL 300
Plutonium isotopes	alpha spectroscopy	EML HASL 300
Uranium isotopes	alpha spectroscopy	EML HASL 300
Thorium isotopes	alpha spectroscopy	EML HASL 300
Gamma emitting isotopes	Gamma spectroscopy	EML HASL 300

Notes:

EML – Environmental Measurements Laboratory
HASL – Health and Safety Laboratory

Table 3. Resin Sample Results Summary

Isotope	Test Method	Top Layer (pCi/g)	Middle Layer (pCi/g)	Bottom Layer (pCi/g)
H-3	LSC	2.99 ± 0.48	2.75 ± 0.45	2.28 ± 0.56
C-14	LSC	15.2 ± 1.8	<MDC	3.34 ± 0.85
Fe-55	LSC	<MDC	13.9 ± 7.8	16.8 ± 8.1
Ni-59 Ni-63	LSC	<MDC 2.8 ± 1.8	<MDC 16.6 ± 3.7	<MDC 18.0 ± 4.0
Po-210	Alpha spectroscopy	<MDC	<MDC	<MDC
Pu-241	LSC	<MDC	<MDC	<MDC
Plutonium Isotopes	Alpha spectroscopy	<MDC	<MDC	<MDC
Uranium Isotopes	Alpha spectroscopy	<MDC	<MDC	<MDC
Thorium Isotopes	Alpha spectroscopy	<MDC	<MDC	<MDC
Gamma Emitting Isotopes Cs-137 Co-60 Eu-152 K-40	Gamma spectroscopy	 <MDC 3.65 ± 0.33 1.38 ± 0.31 <MDC	 2.87 ± 0.46 49.6 ± 3.3 12.3 ± 1.1 13.1 ± 2.3	 3.88 ± 0.52 57.8 ± 3.8 14.2 ± 1.4 17.0 ± 2.6

Notes:

LSC – Liquid Scintillation Counter
pCi/g – picoCuries per gram
MDC – Minimum Detectable Concentration

2.2 Activated Metal Analysis

There were several pieces of activated metal outside of the reactor pool available for sampling. These included stainless steel cables and aluminum cable clamps which had been used to suspend items in the reactor pool. Both the stainless steel and aluminum have elements and/or impurities capable of activation with the reactor. One sample of each of the cables and cable clamps were collected and analyzed to identify nuclides in the activated materials. The identification of the activated metals contained within samples of stainless steel and aluminum and their ratios will benefit future waste profiling needs and provide estimates of the activity of internal reactor components. Levels of activation isotopes such as cobalt-57 (Co-57), cobalt-60 (Co-60), and cesium-137 (Cs-137) can facilitate the calculation of other isotopes which may also be present.

TestAmerica analyzed the activated metal samples using gamma spectroscopy for identification of gamma-emitting isotopes including common activation products. Results are only approximate as the laboratory did not have a calibration standard for the specific geometry of each sample (approximated with air filter geometry). An accurate geometry assists the calculation of the specific gamma wavelength allowing a more complete isotope and activity

characterization. The laboratory was requested to maintain the samples in their inventory for further analysis as necessary. A summary of results are presented in Table 4; the complete results are provided in Appendix B.

Table 4. Activated Metal Analysis Results

Aluminum	
Isotope	**Activity (pCi/g)**
Cs-137	3.7 ± 1.2
Stainless Steel	
Isotope	**Activity (pCi/g)**
Cs-137	< MDC
Co-60	39.2 ± 3.3

Notes:
pCi/g – picoCuries per gram
MDC – Minimum Detectable Concentration

2.3 Surface Contamination Surveys

Another goal of the characterization effort was to determine the extent, if any, of total and removable surface contamination throughout the reactor facility area. Some data was available from the previous characterization, but during that event there were few removable contamination swipes analyzed for tritium (10%) and other HTD isotopes (0). During the current characterization, direct alpha/beta measurements were made and removable contamination swipes were collected across the floors of Rooms B526, B540, and B540A in 1-meter square grids. Room B535A (the walk-in cooler) was also surveyed after the removal of the cooler wall liner. This area was not previously characterized.

Swipes were also taken in other locations likely to accumulate contamination and in areas where the previous characterization efforts identified alpha or beta contamination. These locations included the following:

- Reactor pool covers
- Reactor bridge and bridge components
- Pneumatic transfer trench
- Trench surrounding top of reactor
- Cooling system “vault” floor

Total surface contamination measurements were made using 100 square centimeter (cm^2) alpha/beta phoswich scintillation detectors. Direct measurements were collected using a 1-minute count time.

Floors swipe samples were collected from a 100 cm^2 area in low-dust areas. Dust can collect radon daughter products and dusty samples can result in cloudy scintillation fluid which can impact the analytical results. These swipes were analyzed for removable alpha and beta

activity using Ludlum Model 2929 scaler counters with 43-10-1 sample counting heads. Swipe samples were then analyzed for H-3, C-14, and other HTD isotopes using a Packard TriCabb 2100 TR liquid scintillation counter (LSC). This LSC is owned by the Omaha VAMC but was operated by an AECOM technician. Prior to use, a multi-point calibration with applicable quench standards was applied to the LSC.

Selected results are reported in Table 5 in units of disintegrations per minute (dpm) per 100 cm^2. These samples were the most active as well as those showing only background, allowing comparision along the entire range sampled. Of the estimated 800 swipes examined less than 1% had any results above background. These results can be compared to the free-release removable contamination criteria applied to the release of materials from the reactor facility (200 dpm for total beta and 1,000 dpm for tritium) (AECOM 2011). The complete survey results are included in Appendix C. Additionally, eight removable contamination sample vials were sent to the independent laboratory (New World Environmental) for quality verification. The duplicate results were within +/-10% of the field results obtained from the LSC at the VA facility.

Table 5. Surface Contamination Results of Interest (Floors)

Location	Direct		M2929	LSC		
	α* (dpm)	β/γ* (dpm)	β/γ* (dpm)	Total β (dpm)	H-3 (dpm)	C-14 (dpm)
Blank	**-**	**-**	**<MDC**	**42**	**7**	**13**
Vault Floor 6	**47**	**2157**	<MDC	49	17	24
Vault Floor 1	**47**	**2142**	<MDC	51	6	22
B526 Floor C8	<MDC	<MDC	<MDC	**337**	**326**	31
B526 Floor D8	<MDC	<MDC	<MDC	**463**	**464**	11
B526 Floor D9	<MDC	<MDC	<MDC	**813**	**816**	26
B526 Floor B2	<MDC	709	<MDC	51	19	20
B526 Floor F4	<MDC	<MDC	<MDC	**78**	34	24
B540 Floor C2	**36**	<MDC	<MDC	**63**	38	12
B540 Floor B2	<MDC	<MDC	<MDC	50	15	27
B540A Floor C1	**28**	**1984**	<MDC	**66**	13	17
B540A Floor B2	<MDC	2543	<MDC	53	4	33
B535A Floor A1	**69**	**1488**	<MDC	42	10	14
B535A Floor D1	**55**	**1843**	<MDC	45	6	16
B535A Floor D4	**40**	**1449**	<MDC	**71**	1	22

Note:

* - MDCs were 8 dpm for direct α, approximately 450 dpm for direct β/γ , and 151 dpm for removable β/γ

The results provided in Table 5 demonstrate the following:

- There was no detectable removable contamination in the “vault” or “pit” area containing the reactor cooling system (swipes consistent with blank)
- Direct/total alpha and beta/gamma activity in the “vault” is above background levels. However, this activity could be a result of radon and radon daughter products as the vault is below grade and has limited ventilation.

- There is some measureable removable H-3 contamination on the floor in the reactor area. However, the removable contamination levels suggest that the contamination levels are well below the 1.2 x 10^8 dpm/100cm^2 screening level for H-3 in NUREG-1757, Volume 1, Appendix B (NRC 2006).
- There is some detectable direct/total alpha and beta/gamma activity on the floor in the reactor area (B526). However, these levels are below the general decommissioning release criteria (100 dpm/100cm^2 α and 5,000 dpm/100cm^2 β/γ). This activity could be a result of radon and radon daughter products as the basement floor is below grade.
- There may be some removable beta/gamma activity other than H-3 and C-14 that is detectable in the LSC that is not detectable with the 2929 scintillation detector. This could be indicative of low levels of low-energy beta emitters such as Ni-63 or Fe-55.

The LSC was unable to be calibrated to quantify Ni-63 activity as originally planned as the Ni-63 quench standards were not available. However, Ni-63 activity is included in the total beta (β) activity. It should be noted that the sum of the H-3 and C-14 samples collected at locations B526 Floor C8, D8, D9, and F4 are approximately equal to the total beta/gamma activity. For samples collected at B540A Floor C1 and B535A Floor D4, there may be other beta activity present.

2.4 Ductwork, Drain Lines, Pneumatic Tubing

Removable contamination swipes were collected from inside the air exhaust system that originates with the two vent hoods in the reactor area. The two fume hoods were discovered to have low levels of contamination in the 2003 characterization (Duratek 2003). It was recommended in the NRC RAI (NRC 2008) that the extent of the contaminants be quantified. In response, swipes were taken at multiple locations inside the hoods, at multiple locations inside the ventilation ducts prior to the point where they go vertical from the basement level (in room B526), and at access points in the roof-top penthouse immediately prior to discharge. All sampling access openings in the ductwork were closed following collection. Ductwork sample locations are shown in Figure 2 below. Complete sample results are included in Appendix C, Survey number ABJ-050.

Sanitary sewer drain lines in the reactor facility were surveyed and sampled through sink and floor drains and other access points both inside and outside the facility. Figure 3 shows the location of the approximate drain lines and sampling point inside the building. Access points outside the building including two locations to the west side of the building. Complete sample results for drain lines are included in Appendix C and D, Survey numbers ABJ-047, 048, and 058.

Pneumatic tubing formerly used for sample transport into the reactor vessel was likewise sampled at the open ends in room B533A. Pneumatic tubing sample results are shown in Appendix D, Survey number ABJ-033.

Sample technicians collected swipes of the pipe/tube interiors and analyzed the samples in a LSC as detailed in the previous section. Samples were also screened with hand-held instruments or using the Ludlum Model 2929 sample counter prior to placing them in the scintillation vials. No sufficient volume of sediment/sludge could be collected for testing from

either drain lines or sewer lines. Table 6 provides a range of the sample results that are indicative of the ductwork, drain line and pneumatic tubing survey. The size of all the piping sampled was documented along with the surveys and is presented in Appendix D.

Figure 2. Duct Sample Locations

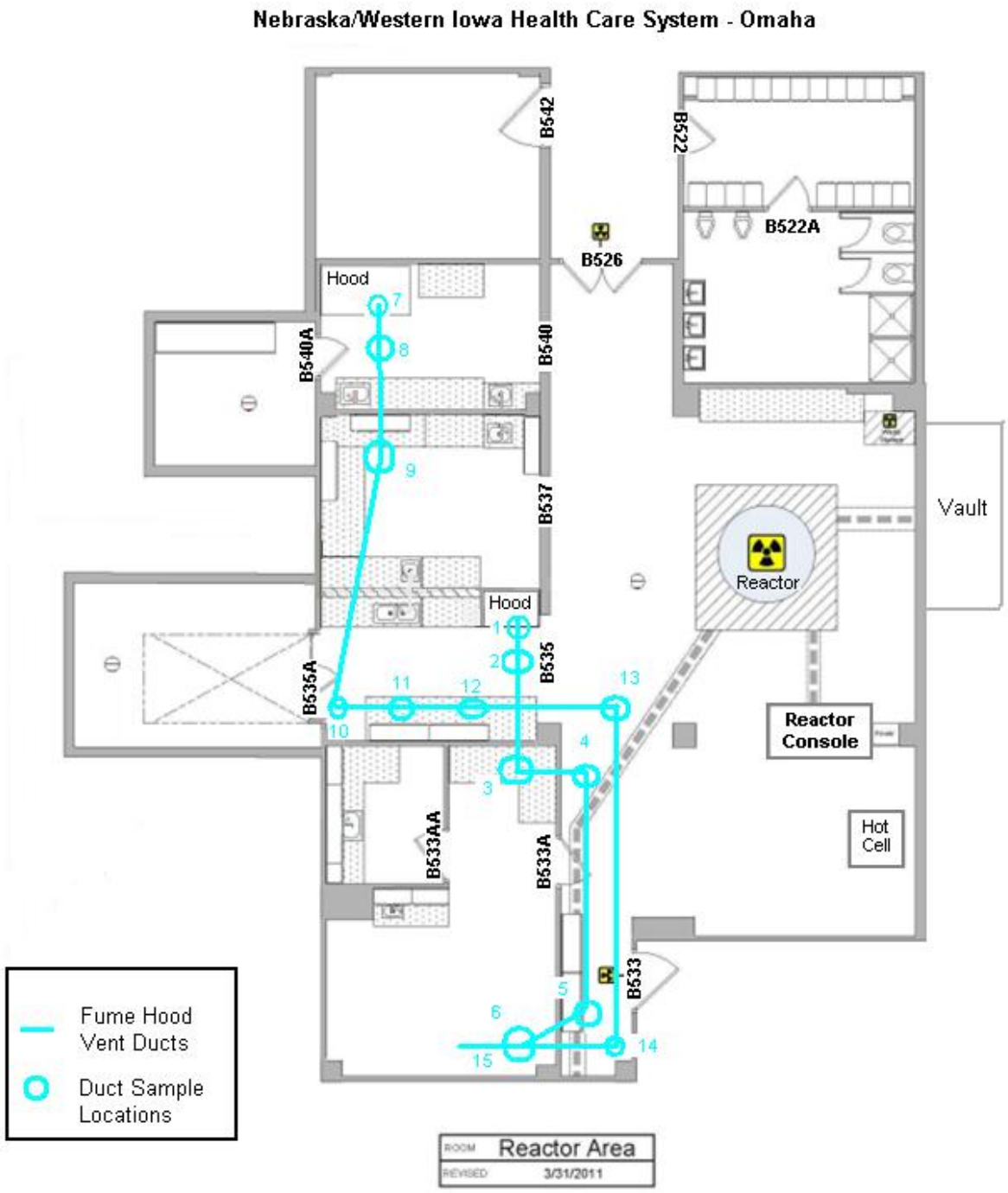


Figure 3. Drain Locations

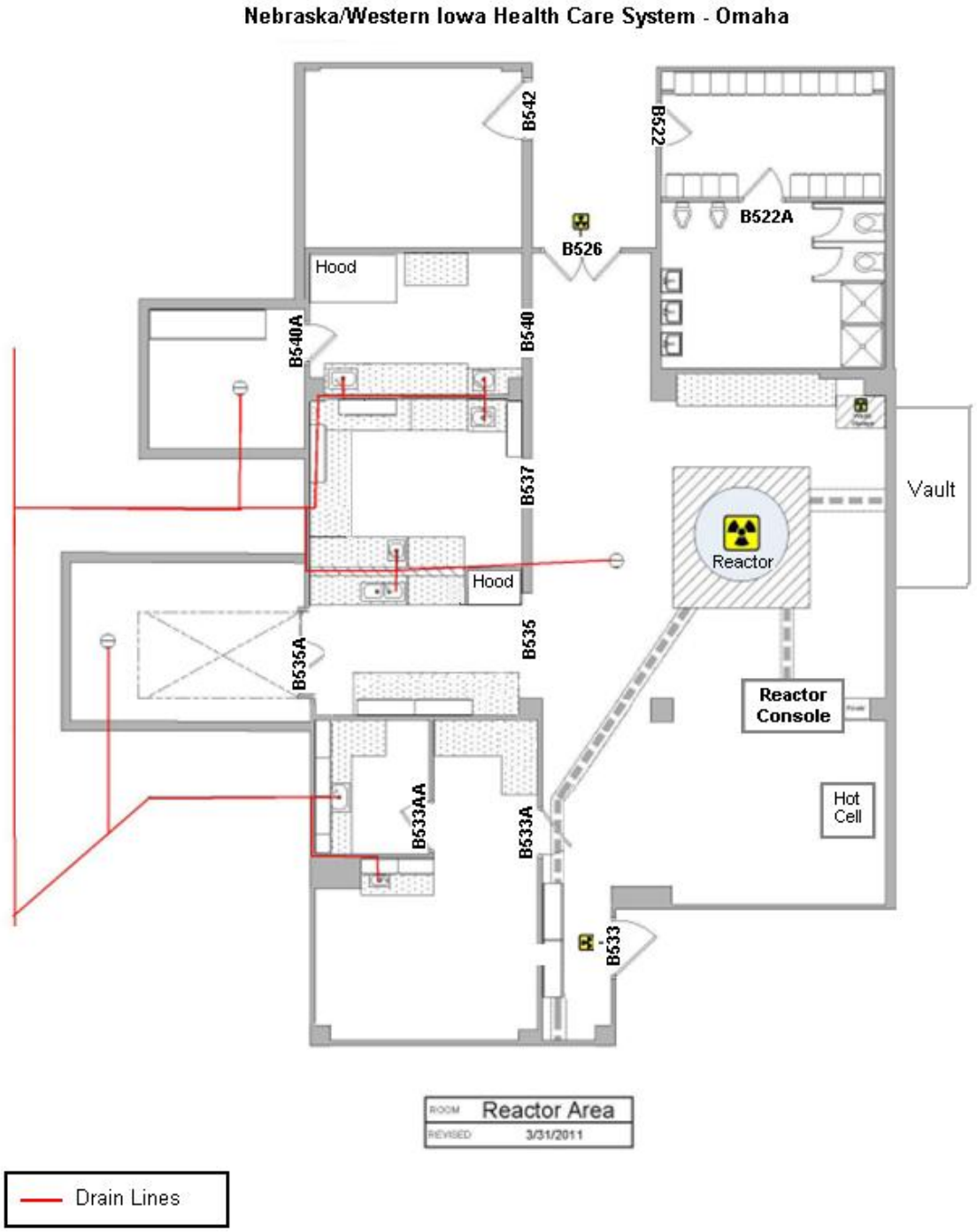


Table 6. Utility Removable Contamination Survey Results of Interest

Location	M2929		LSC		
	α* (dpm)	β/γ* (dpm)	Total β (dpm)	H-3 (dpm)	C-14 (dpm)
Blank	**<MDC**	**<MDC**	**42**	**7**	**13**
B535 Fume Hood Drain	<MDC	227	1179	0	1162
B540A Floor Drain	<MDC	<MDC	56	19	20
4" Sewer Inlet from ABJ Reactor	<MDC	<MDC	56	14	23
8" Pipe Inlet	<MDC	<MDC	62	0	19
Pneumatic Tube Interior	<MDC	<MDC	54	15	16
B535 Fume Hood Roof Duct	<MDC	<MDC	51	22	19
Fume Hood Duct B535	<MDC	<MDC	78	29	30

Note:

* - MDCs for M2929 were 151 dpm β/γ and 7 dpm α.

2.5 Soil

Previous characterization activities collected subsurface soil samples inside the reactor facility adjacent to and below the level of the reactor tank bottom by drilling through the concrete floor. While no radioactive contamination was identified in any of these samples (Duratek 2003), these samples were analyzed using only gamma spectroscopy which would not identify hard to detect beta-emitting isotopes such as H-3 and Ni-63. The additional characterization in 2011 collected subsurface soil samples outside the facility to determine if activity, including H-3 and HTD isotopes, had a significant impact of the facility. Seven soil samples were collected near the two "hatches" that provide access to the basement area. Analysis included isotopes and the isotope series provided in Table 7. No additional samples were collected adjacent to the reactor tanks.

The soil sample locations are provided in Figure 4. Actual sampling locations were finalized following utility clearance by the VA facilities personnel. Using direct-push sampling technology, two soil samples were collected from each of two locations on the west side of the building (marked on Figure 4 as soil sample locations 1 and 2). At each location, one sample consisted of a composite of the first 6 inches of soil. The second sample at each location was collected in the 2-foot interval that is at the level of either drain lines leaving the reactor area or just below the basement floor level (about 13 to 15 feet below ground surface). Groundwater was not encountered at these depths.

The remaining three samples were collected east of the building in the vicinity of the circulation equipment vault and cooling tower (marked on Figure 4 as soil sample locations 3 - 5). Each sample consisted of a composite of the first 6 inches of soil beneath the cement walkway. Deeper samples were not collected as there are no drains below the basement floor level on east side of the building.

A background sample was collected from surface soil at a nearby location (see Figure 4). This was analyzed in the same manner as the other seven soil samples.

Samples were analyzed for the isotopes listed in Table 2 using the stated analysis and equivalent methods. Analytical methods are based on potential contaminants and to confirm or dismiss uncertain results from the previous sampling effort. None of the soil sample results show any significant amount of contamination above background levels. Table 7 provides a summary of the soil sampling results showing the primary isotopes of interest. The complete soil sample results are provided in Appendix E.

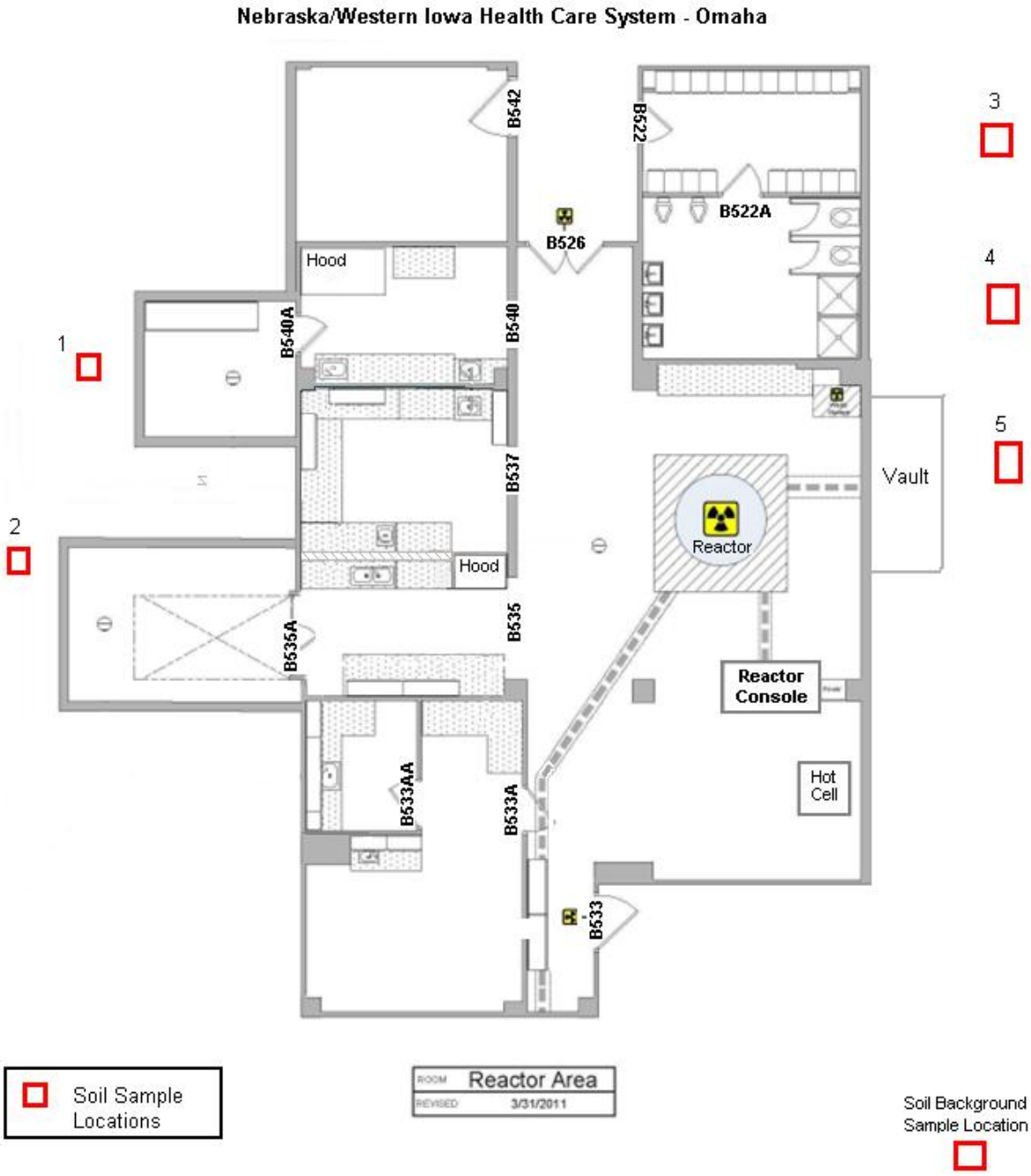
Table 7. Summary of Soil Sampling Results

	Back-ground	Hole 1		Hole 2		Hole 3	Hole 4	Hole 5
Isotope	**6in Depth**	**6in Depth**	**13-15ft Depth**	**6in Depth**	**13-15ft Depth**	**6in Depth**	**6in Depth**	**6in Depth**
H-3	0.13 ± 0.11*	0.18 ± 0.12	0.26 ± 0.12	0.39 ± 0.16	0.12 ± 0.11*	0.13 ± 0.11*	-0.01 ± 0.07*	0.24 ± 0.12
C-14	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC
Fe-55	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC
Ni-59/63	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC
Cs-137	< MDC	0.27 ± 0.20	0.27 ± 0.20	< MDC	< MDC	< MDC	< MDC	< MDC
Po-210	1.15 ± 0.36	1.03 ± 0.43	1.16 ± 0.48	0.91 ± 0.37	1.18 ± 0.39	0.88 ± 0.35	1.12 ± 0.39	0.88 ± 0.30
U-238	1.10 ± 0.27	0.77 ± 0.21	0.85 ± 0.24	0.91 ± 0.24	0.70 ± 0.20	0.67 ± 0.20	0.89 ± 0.24	0.85 ± 0.23
Th-232	1.24 ± 0.26	1.17 ± 0.25	1.11 ± 0.25	1.19 ± 0.26	1.08 ± 0.24	1.27 ± 0.26	1.22 ± 0.26	1.03 ± 0.24
Pu-241	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC	< MDC

Notes:

* - Less than MDC
pCi/g – picoCuries per gram
MDC – Minimum Detectable Concentration

Figure 4. Soil Sample Locations



2.6 Instrumentation

Radiation survey instruments capable of detecting the alpha, beta, and gamma radiations to the standards set forth by the NRC Guide 1.86 were used during the operations and surveys. These instruments included gas proportional detectors, Geiger-Müller (GM) detectors, alpha beta (phoswich) detectors, gamma scintillation (NaI) detectors, and swipe/sample counters. All instruments used on-site were calibrated and managed according to AECOM standard operating procedures for radiological services SOP01 and SOP02. Records of calibration and instrument control logs were maintained. These daily logs confirm instrument accuracy relative to an acceptable response range. All instrument check and control logs and calibration records are in Appendix F.

3.0 Radioactive Material Storage

During pre-decommissioning activities, small amounts of radioactive materials were encountered. These materials include activated metal, items that could not be adequately surveyed (such as small diameter conduit), and items that did not meet free release criteria for removal from site. These materials were placed in a standard 96-cubic foot radioactive waste box (a B-25 box) dedicated to radioactive materials storage. The box was secured within the reactor area and a detailed inventory of items placed in the box was kept and attached to the box (Table 8). Disposition of this box will ultimately be managed by the reactor decommissioning contractor along with other materials destined for removal from site.

Contaminated lead was inventoried and stored separate from the clean lead and other radioactive materials. Contaminated lead, listed in Table 9, was placed in a 55-gallon drum. The lid was placed on the drum prior to demobilization and a radioactive materials tag was attached to the drum lid ring.

Table 8. Material in Radioactive Materials Storage Box

Quantity	Item
~50 feet	Conduit piping
9	55-gallon bags of investigation derived waste (in yellow “RAD” bags)
~20	Ceiling tiles
4	Plastic containers containing solidified concrete mix
3	Radioactive waste containers (yellow trash cans)
~20 feet	Miscellaneous wire
1	Source holder rod
1	Polonium-Beryllium (Po-Be) source (in its own separate retrievable container)

Table 9. Contaminated Lead

Quantity	Item
3	Small lead wall plugs from Room B540A
2	Large lead floor plugs from Room 540A
2	Lead discs from detector collimator from Room 526
1	Small lead pig from Room 526

4.0 Conclusion

All characterization activities outlined in the scope of work for the Blotcky Reactor were completed as planned except for quantification of Ni-63 in removable contamination samples. The following statements supported by the characterization data directly address the NRC's requests for additional information on the decommissioning plan (NRC 2008).

- Surface and subsurface soils collected near the reactor facility do not indicate the presence of reactor-related isotopes, including the hard-to-detect isotopes Ni-63 and Fe-55. While H-3 was detected in four of the six soil samples, concentrations are very low, less than 0.5 picocuries per gram (pCi/g) in each sample. The NRC's soil screening level for H-3 is 110 pCi/g (NRC 2006). The outside surface and subsurface soils of the reactor facility can be considered as non-impacted.
- Ni-63 and Fe-55 were confirmed as isotopes of concern 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 3:1 and 3.5:1 respectively.
- The naturally occurring isotopes of Thorium 230, Pb-210, and Po-210 have been eliminated as contaminants of concern based on reactor water filter resin and soil samples. Direct alpha contamination surveys also indicate that alpha-emitting radionuclides are not a concern for surface contamination.
- Pu-241 has been eliminated as a contaminant of concern based on reactor water filter resin and soil samples and direct alpha surveys and alpha counting of removable contamination smears.
- Characterization data that was obtained 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 (NRC 2006). Removal of embedded pipes is not expected to be required for decommissioning.
- Characterization of the accessible points of the ventilation system demonstrated that, while there may be detectable levels of removable H-3, C-14, and total beta activity, the activity is generally in the range of the blank samples and well below surface contamination screening criteria (NRC 2006). Removal of only the fume hoods is expected to be required for decommissioning. No removal of additional ventilation system equipment is expected to be required.
- No surveys to date have identified the presence of hot particles.

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