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May 10, 2013

U.S. Nuclear Regulatory Commission
ATTN. Document Control Desk
Washington D.C. 20555-0001

Re: Request for License Termination for the Alan J. Blotcky Reactor Facility, License R-57, Docket # 50-131

The Reactor Safeguards Committee for the Alan J. Blotcky Reactor Facility requests License Termination for License R-57, Docket # 50-131. As part of license termination the Reactor Safeguards Committee wishes to submit for Nuclear Regulatory Commission approval the attached revised Decommissioning Plan (May 2013). The Decommissioning Plan has been revised and developed by AECOM for the Alan J. Blotcky Reactor Facility. This revised Decommissioning Plan has been approved by the Reactor Safeguards Committee.

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Department of Veterans Affairs

FSME20

AECOM

Prepared for
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GSA Number GS-10F-0115K
Order Number VA701-BP-004/VA-101-G05020

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Alan J. Blotcky Reactor Facility Decommissioning Plan (Revised)

United States Department of Veterans Affairs
Nebraska-Western Iowa Health Care System
Omaha, Nebraska

7 May 2013

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List of Abbreviations

β/γ	beta-gamma
μCi/mL	microcuries per milliliter
ACOS/R	Associate Chief of Staff for Research
AECOM	AECOM Technical Services, Incorporated
AJBRF	Alan J. Blotcky Reactor Facility
ALARA	As Low As Reasonably Achievable
ALI	annual limit on intake
Ar-41	argon-41
CAA	Controlled Access Area
CAM	continuous air monitor
CAR	Corrective Action Report
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CHP	Certified Health Physicist
Ci	curie
cm	centimeter
cm^2	square centimeters
COS/R	Chief of Staff for Research
COTR	Contraction Officer's Technical Representative
D&D	decontamination and decommissioning
DAC	derived air concentration
DDE	deep dose equivalent
DECON	Decontamination (NRC Decommissioning Method)
DOC	Decommissioning Operations Contractor
DOT	Department of Transportation
DP	Decommissioning Plan
DQO	Data Quality Objective
dpm	disintegrations per minute
EPA	U.S. Environmental Protection Agency
ENTOMB	Entombment (NRC Decommissioning Method)
FSS	final status survey
FSSP	final status survey plan
FSSP	final status survey report
ft	feet
HASP	Health and Safety Plan
HEPA	High Efficiency Particulate Air
HTD	hard-to-detect
ISM	Industrial Safety Manager
km	kilometers
kph	kilometers per hour
kW	kilowatt
lbs	pounds
LDE	lens dose equivalent
LLRW	low level radioactive waste
LSC	liquid scintillation counter
m	meter
m^2	square meters
m^3	cubic meters
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	minimum detectable activity

MDC	minimum detectable concentration
mrem	millirem
MSDS	Material Safety Data Sheets
MSL	mean sea level
NA	not applicable
NCR	Nonconformance Report
NCRP	National Council on Radiation Protection
No.	Number
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Regulatory Guide
Omaha VAMC	Nebraska-Western Iowa Health Care System, Omaha Division
OSHA	Occupational Safety and Health Agency
OSL	Optically Stimulated Luminescence
pCi/g	picocuries per gram
Po-Be	Polonium-Beryllium
PM	Project Manager
QA	quality assurance
QAPP	Quality Assurance Project Plan
RCA	Radiological Controlled Area
RCRA	Resource Conservation and Recovery Act
R/hr	roentgen per hour
RE	Resident Engineer
RHWP	Radiological and Hazardous Work Permits
RP	Radiation Protection
RPP	Radiation Protection Plan
RSC	Reactor Safeguards Committee
RSO	Radiation Safety Officer
RSP	Radiation Safety Program
RWP	Radiation Work Permit
RWT	Radiation Worker Training
SAFSTOR	Safe Storage (NRC Decommissioning Method)
SRE	Senior Resident Engineer
TEDE	Total Effective Dose Equivalent
TODE	total organ dose equivalent
TRIGA	Training, Research, Isotopes, General Atomics
U.S.	United States
USDOE	United States Department of Energy
USGS	United States Geological Survey
VA	U.S. Department of Veterans Affairs
VACO	Veterans Affairs Central Office
VAMC	Veterans Affairs Medical Center

1. Executive Summary

1.1 Licensee/Site Name and Address

The United States Department of Veterans Affairs Nebraska-Western Iowa Health Care System
Alan J. Blotcky Reactor Facility
4101 Woolworth Avenue
Omaha, Nebraska 68105
License Number R-57, Amendment 11 (ML020630192, United States Department of Veterans Affairs 2002a)

1.2 Site Description

The Alan J. Blotcky Reactor Facility (AJBRF) is located in the Omaha Veterans Affairs Medical Center (VAMC) in the City of Omaha, Nebraska. The Omaha VAMC is located in a commercial area within the city limits and the reactor facility is housed in the southwest corner basement of Building One. The reactor facility contained a low-power Mark I Training, Research, Isotopes, General Atomics (TRIGA) nuclear reactor, which 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.

1.3 Summary of Licensed Activities

The AJBRF initial operating license number R-57 was issued on June 26, 1959. The most notable license amendment, Amendment 9, was issued on April 12, 1991, to allow for operation of the reactor at steady state power levels up to a maximum of twenty kilowatts (kW) thermal and for the installation of a microprocessor-based neutron monitoring system. The most recent license renewal was issued as Amendment 11 on August 5, 2002 authorizing operation for an additional twenty years (VA 2002a).

1.4 Nature of Sources

The radioactive materials licensed for use at the AJBRF primarily consisted of the reactor fuels and fission chambers, which have since been transferred to the United States Geologic Survey (USGS) in Denver, Colorado for use in their TRIGA reactor. All licensed radioactive sources have been removed and properly disposed of over the years, with the exception of a single Polonium—Beryllium (Po-Be) source remaining on-site.

1.5 Location and Storage of Licensed Materials

All loose radioactive materials were secured during pre-decommissioning characterization activities performed in May 2011. Radioactive materials were secured in one B-25 container and one 55-gallon drum and stored in the AJBRF room B535A. The single Po-Be source remaining on-site is also stored in the B-25 container.

Impacted reactor components, systems, and a few residually contaminated items also remain. Details concerning these materials are presented in Section 4.

1.6 Decommissioning Objective

The objective of the AJBRF decontamination and decommissioning (D&D) is the termination of License Number R-57 and release of the site for unrestricted use.

1.7 Site Release Criteria

Site release criteria are based on the results of the site characterization performed in late 2002 (Duratek 2003) and 2011 (AECOM Technical Services, Incorporated [AECOM] 2011a), and screening levels provided in the Nuclear Regulatory Commission (NRC) Regulatory Guide (NUREG) 1757, Volume 1, Revision 2, Appendix B, (as detailed in Section 6 of this document). Release criteria have been established for residual radioactive surface contamination and volumetric contamination/activation of concrete and soil.

1.8 Decommissioning Schedule

Decommissioning activities are expected to commence in late Fiscal Year 2012 following NRC approval of this Decommissioning Plan (DP). D&D and the completion of the final status survey (FSS) are expected to take approximately 8 to 12 weeks.

1.9 License Amendment for Decommissioning

The United States (U.S.) Department of Veterans Affairs (VA) Nebraska-Western Iowa Health Care System, Omaha Division, AJBRF requests that its license, Number R-57, be amended to incorporate this DP.

2. Facility Operating History

2.1 License Status

The AJBRF is owned and maintained by the VA Nebraska-Western Iowa Health Care System, Omaha Division (Omaha VAMC). It is licensed pursuant to 10 Code of Federal Regulations (CFR) Part 50 (Domestic Licensing of Production and Utilization Facilities). License Number R-57 was issued on June 26, 1959 and the license was renewed most recently as Amendment 11, issued August 5, 2002, authorizing operation for twenty years from that date (VA 2002a). However, the reactor was permanently shut down on November 5, 2001. The fuel elements were subsequently removed and shipped to the USGS TRIGA reactor in Denver, Colorado in June 2002. No subsequent amendments have been issued.

2.2 License History

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 Comission. Pictures of the construction activities are provided in Figures 2.1 and 2.2. An operating licence 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 20kw 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.

Figure 2.1, Reactor Construction

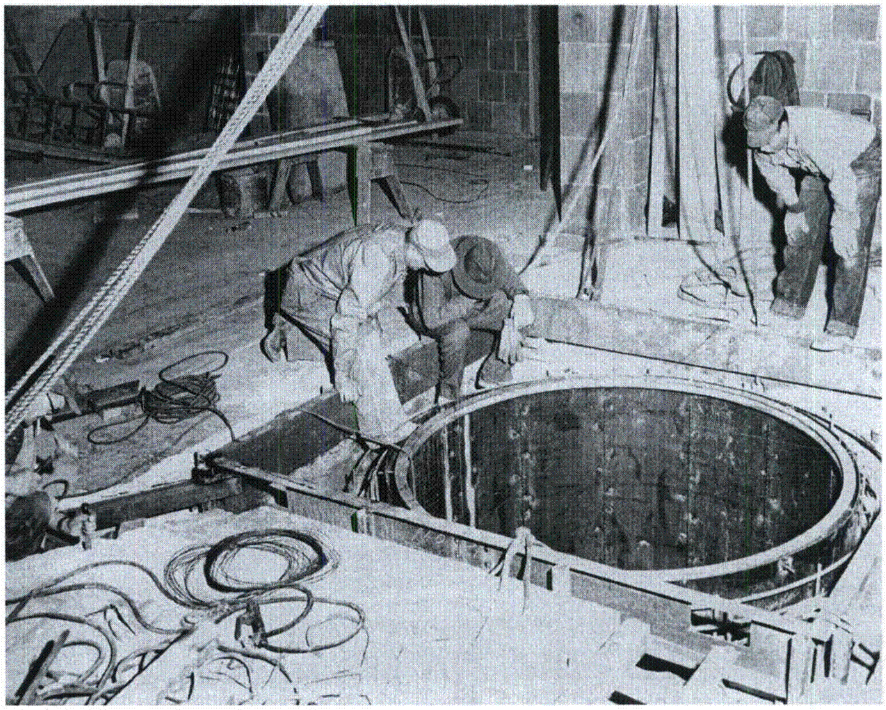
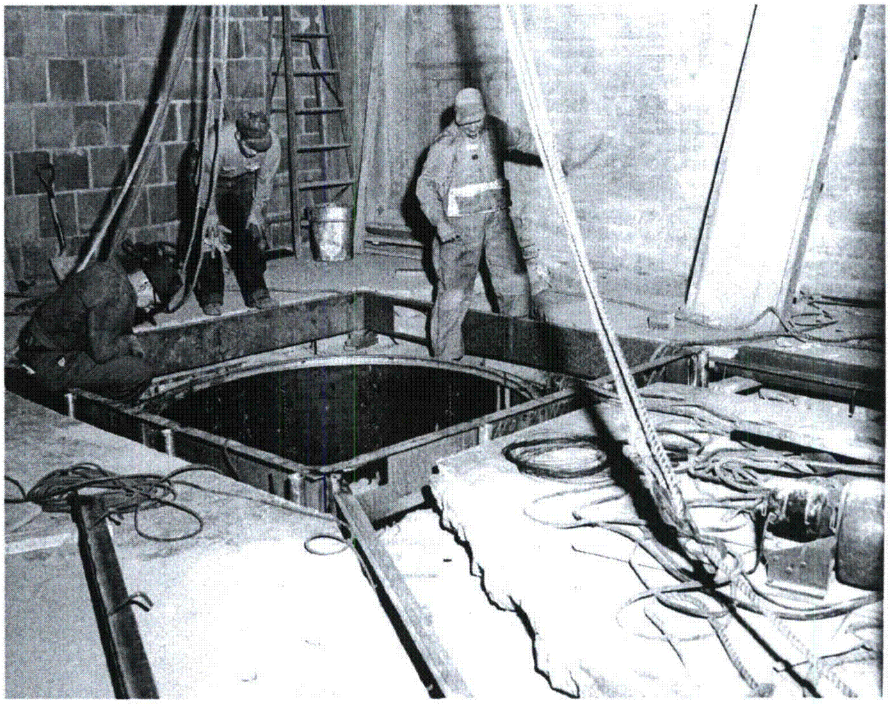


Figure 2.2, Reactor Construction



2.2.1 Authorized Activities

The AJBRF was used to support nuclear medicine and research programs conducted at the Omaha VAMC 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, the reactor was used for training Fort Calhoun Station nuclear power reactor operators.

2.2.2 Location and Storage of Licensed Materials

The AJBRF is located in the basement of the eleven-story wing of the main building housing the Nebraska-Western Iowa Health Care System, Omaha Division. The AJBRF is constructed of brick exterior with a concrete floor and concrete block exterior walls. The normal entrance to the reactor laboratory is through the door marked B526 (see Figure 2.3, Reactor Facility Floor Plan).

The TRIGA reactor assembly is located in a below-ground 20 ft (6.1 m) deep steel reactor pool/tank structure with only vertical access to the core. The tank is lined with gunite and surrounded on the outside by a minimum of 10 inches (25.4 cm) of concrete and on the bottom by 11 inches (27.9 cm) of concrete poured between the steel tank and a corrugated steel casing. The reactor assembly contains a graphite-reflected fixed core constructed of aluminum resting on a platform that raises the lower edge of the assembly about 2 ft (.6 m) above the tank floor. The reactor assembly and portions of the reactor tank and surrounding concrete are radioactive due to neutron activation during reactor operations. The demineralized water within the tank currently provides the required shielding so that radiation levels in and around the reactor room are at background levels. The reactor fuel and fission chambers have been removed from the facility.

Various licensed radioactive sources were used for instrument calibration and reactor start-up and monitoring, and are listed in Table 2.1. All sources were present in solid form. Only one radioactive source, a Po-Be source (serial No. O-178), remains in the reactor facility. It is currently stored within a 5-gallon pail of paraffin, which in turn is located within the radioactive materials storage container (B-25 box) in AJBRF room B535A. The polonium in the source has decayed to negligible activity, but the stainless steel capsule is slightly activated.

During pre-decommissioning activities designed for additional characterization and free-release of items within the facility conducted by VA in May 2011, materials that could not be free-released from the facility were also placed in the B-25 box. These materials are waste items that will be disposed of during decommissioning activities. An inventory of the items in the B-25 box was prepared by VA and is currently affixed to the outer surface of the box. The B-25 is currently stored in Room B535A (former walk-in cooler) as shown in Figure 2.4. A few contaminated lead items were placed in a labeled 55-gallon drum also located in Room B535A. No loose radioactive materials are expected to be located in any other areas of the reactor facility.

Figure 2.3, Reactor Facility Floor Plan

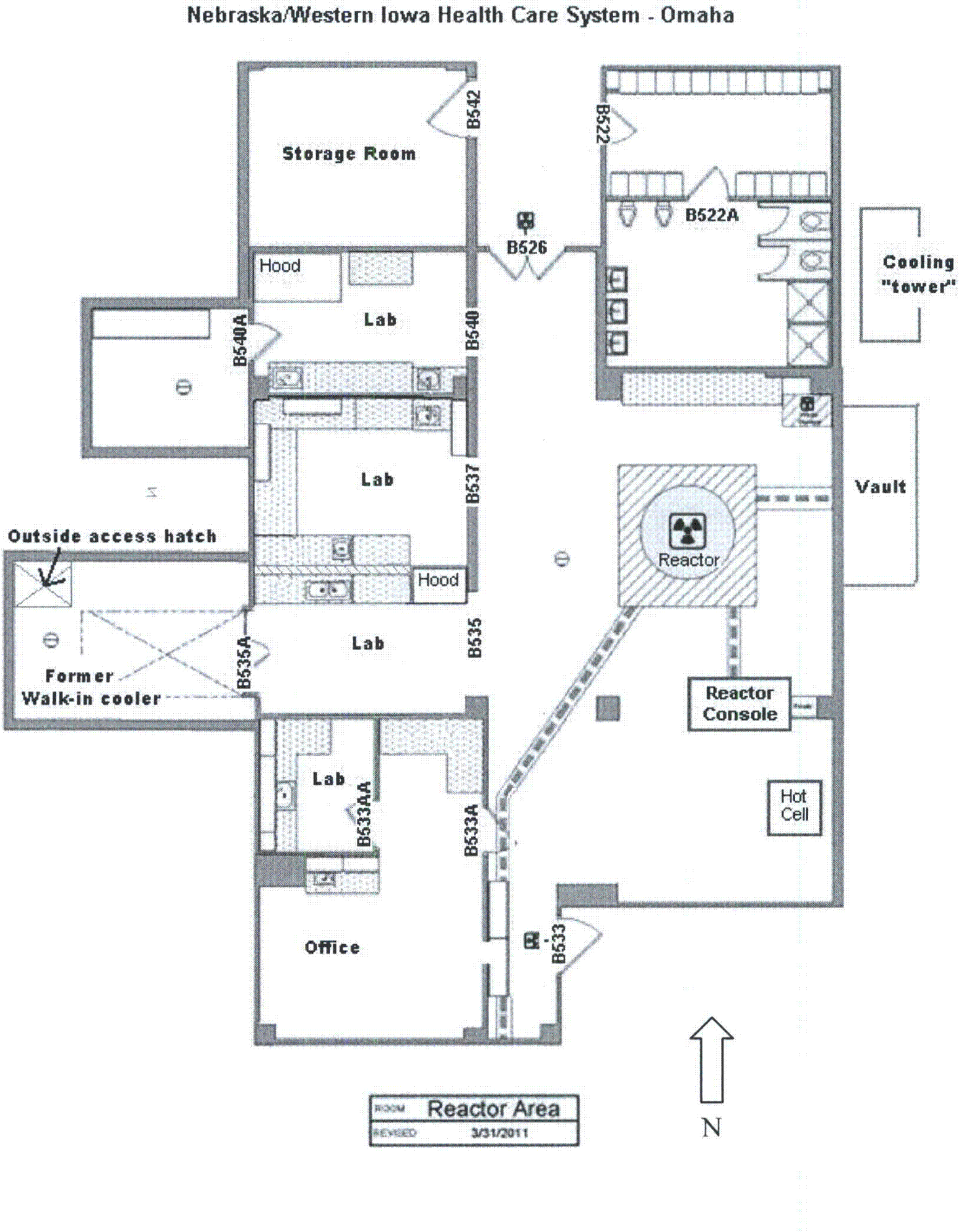
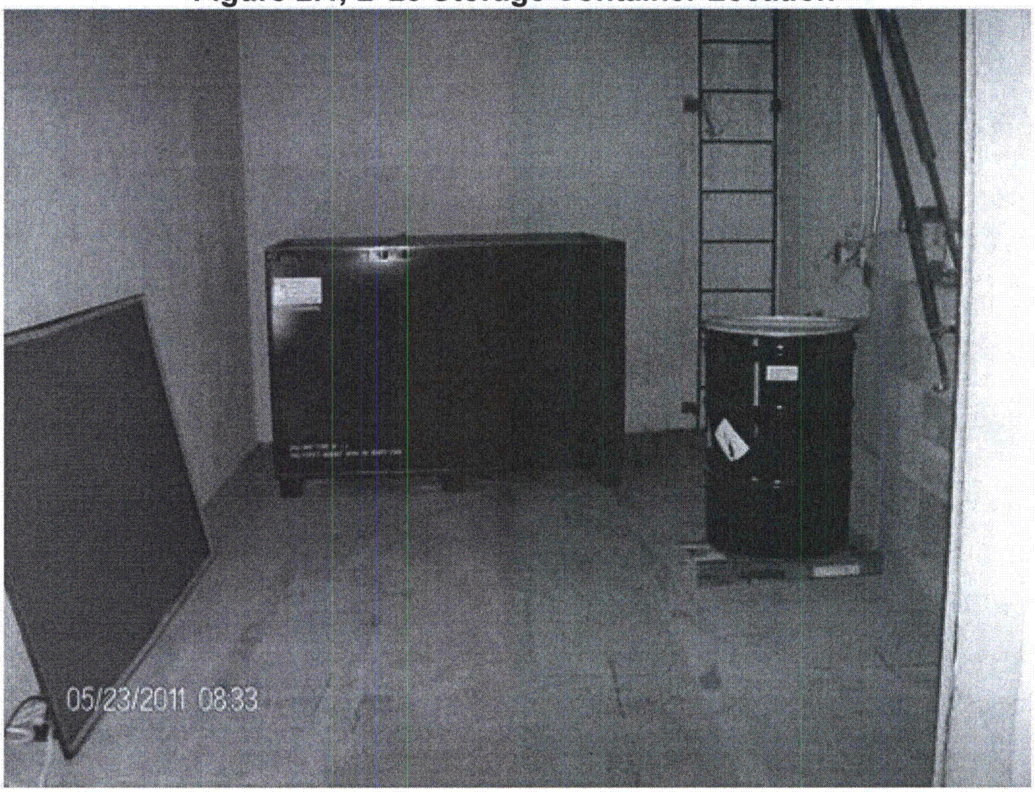


Table 2.1, Specialty Sources

Manufacturer	Received	Source Type	Initial Activity	Serial Number	Transfer/ Disposal Date	Currently In Possession
Isotope Specialties	05/30/1959	Po^{210}/Be	5 curie (Ci)	None	06/01/1963	No
U.S. Nuclear	07/13/1961	Po^{210}/Be	7 Ci	B673	06/01/1963	No
U.S. Nuclear	05/03/1963	Po^{210}/Be	7 Ci	J-102	05/23/1966	No
U.S. Nuclear	07/02/1964	Po^{210}/Be	7 Ci	L-202	05/23/1966	No
U.S. Nuclear	01/29/1968	Am^{241}/Be	2 Ci	618AM372	07/24/2003	No
Siemens	02/27/1987	Am^{241}	12 mCi	2824LA	07/24/2003	No
Siemens	06/24/1987	Am^{241}	12 mCi	5839LV	07/24/2003	No
U.S. Nuclear	01/12/1966	Po^{210}/Be	7 Ci	O-178	-	Yes

Figure 2.4, B-25 Storage Container Location



2.3 Previous Decommissioning Activities

There has been no previous formal decommissioning activity performed at the AJBRF. In June 2002, the fuel rods and fission chambers were transferred to the USGS in Denver, CO for use in their TRIGA Mark I reactor. In May 2011, VA performed pre-decommissioning activities to free-release many non-reactor operational items from the area such as chairs, tables, electronics, lead, etc. Items that could not be free-released were consolidated into storage containers (B-25 box and 55-gallon drum) and secured in the reactor area. The AJBRF was also swept and most horizontal surfaces were wiped down leaving the facility in a relatively clean condition.

2.4 Spills and Burials

Review of available documentation, including AJBRF operating records and annual reports, reveals no evidence of spills or other uncontrolled releases of radioactive materials, or evidence of burials of radioactive materials. The results of the site characterization surveys support this conclusion.

3. Facility Description

3.1 Site Location

The AJBRF is located within the Omaha VAMC in the City of Omaha, Douglas County, Nebraska. The AJBRF site is part of a functioning medical center, including support buildings. The physical features of the hospital site are depicted on Figure 3.1.

The reactor was housed in the basement of the wing of the main building (#15 on Figure 3.1) directly below the optometry clinic. The entire Omaha VAMC grounds are approximately 34.5 acres and the reactor facility occupies approximately 2500 ft^2 (232 m^2) of the hospital basement

The Omaha VAMC site is situated approximately 3 miles (4.83 km) west of the Missouri River. The river level is normally at 880 ft (267.9 m) above mean sea level (MSL) and the rolling hills in and around Omaha rise to as high as 1230 ft (375 m) above MSL. Gently rolling topography surrounds the Omaha VAMC site with the medical center located atop a small knoll. The ground surface at the medical center varies in elevation between 1200 ft (365.8 m) and 1230ft (374.9 m) above MSL. These elevations represent some of the highest ground within the Omaha city limits, sitting approximately 275 ft (83.8 m) above the level of the Missouri River. There are no man-made or natural bodies of water on the medical facility grounds.

To the west of the Omaha VAMC, across 42nd Street, is a residential area of single-family detached homes. To the south, across Center Street, is a commercial district of small businesses and restaurants. To the east is a golf course (the Field Club of Omaha), and to the north, across Woolworth Avenue, is the Douglas County Hospital.

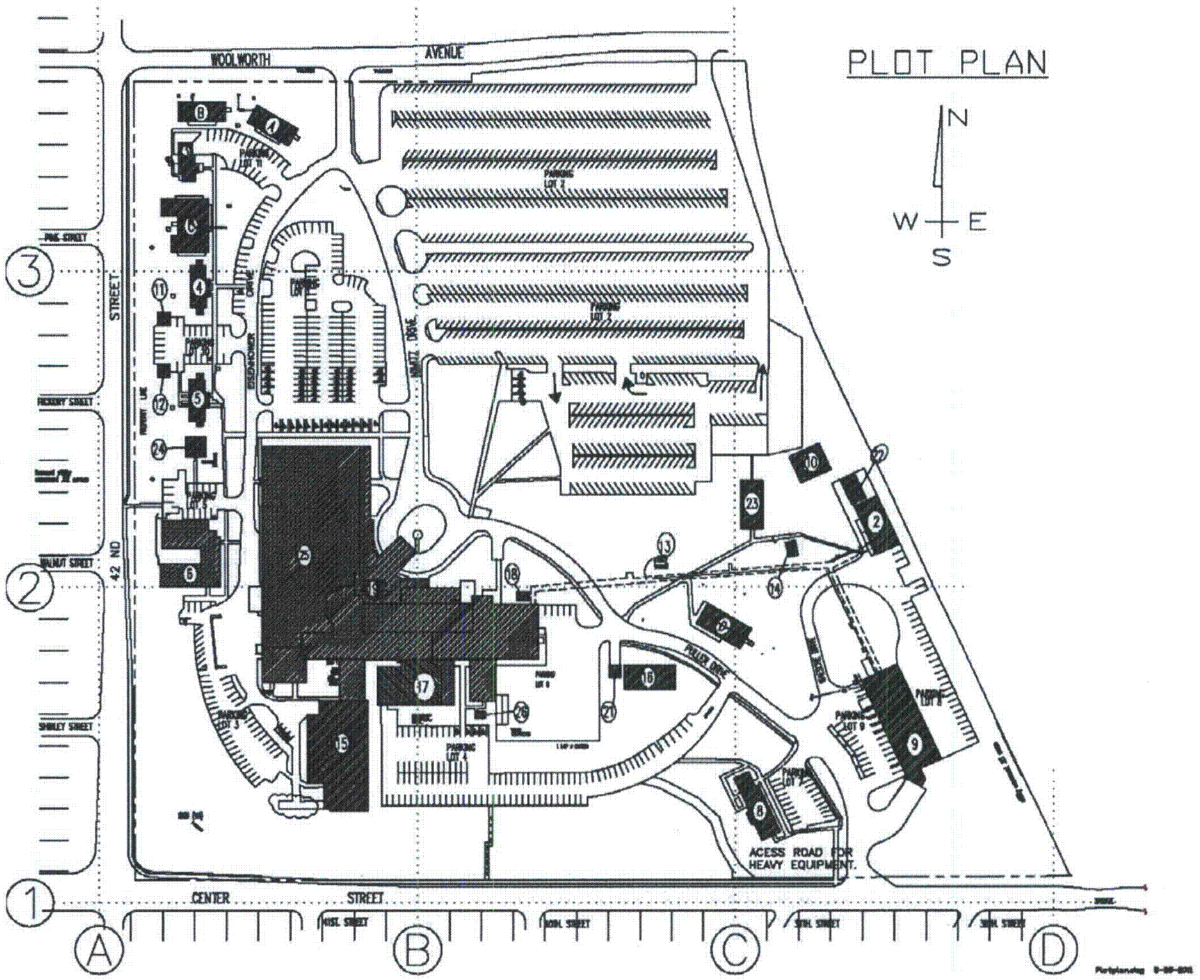
There are no industrial activities in the area that will be impacted by the facility's decommissioning. Approximately 2 miles (3.2 km) east of the site is a large railroad yard and 8 miles (12.9 km) to the southeast is Offutt Air Force Base. The Omaha airport is more than 6 miles (9.7 km) from the facility, and a low altitude airway (3,000 ft [914.4 m] to 17,000 ft [5,181.6 m] MSL) passes near the site. The nearest interstate highways (I-80 to the south and I-480 to the east) are more than 1 mile (1.6 km) from the facility.

3.2 Description of Reactor Area and Reactor Systems

The normal entrance to the AJBRF is through the door to the main reactor room, B526 (Figure 2.3). Samples to be irradiated were typically prepared in either room B540 or B537. Isotopes were stored in the isotope-storage area in room B540A. Gamma counting of irradiated samples was done in a large shielded cabinet in the area marked "hot cell" on Figure 2.3. This cabinet, constructed of steel and possibly lead, has been surveyed and demonstrated free of contamination but still remains on-site (AECOM 2011).

A summary of the current and former functions of the areas/rooms in the AJBRF is provided in Table 3.1.

Figure 3.1, Site Plan of the Omaha VAMC



BUILDING SCHEDULE	
NO.	BUILDING
1	MAIN HOSPITAL BUILDING
2	CENTRAL HEATING PLANT
3	ENGINEERING SERVICE
4	RESIDENT ENGINEER, UNION OFFICE
5	MENTAL HYGIENE
6	DENTAL
7	FLAG POLE
8	DAY HOSPITAL CARE
9	GARAGE AND LAUNDRY
10	OIL STORAGE TANKS
11	STORAGE
12	STORAGE
13	UNDERGROUND TUNNEL
14	EQUIPMENT STORAGE
15	RESEARCH ADDITION
16	COOLING TOWER CENTRAL AIR COND.
17	AIR COND. BUILDING
18	COOLING TOWER I.C.U.
19	WELDING SHOP
21	OXYGEN STORAGE PAD
22	T BUILDING
23	WEATHER SHELTER
24	HIGH VOLTAGE SWITCHING STATION
25	OUTPATIENT CLINIC
26	SMOKE SHELTER
A	WELLNESS FACILITY
B	EMS/SAFETY/
C	CONTRACTING/FINANCE
D	VETRANS RESOURCE & REFERAL CENTER

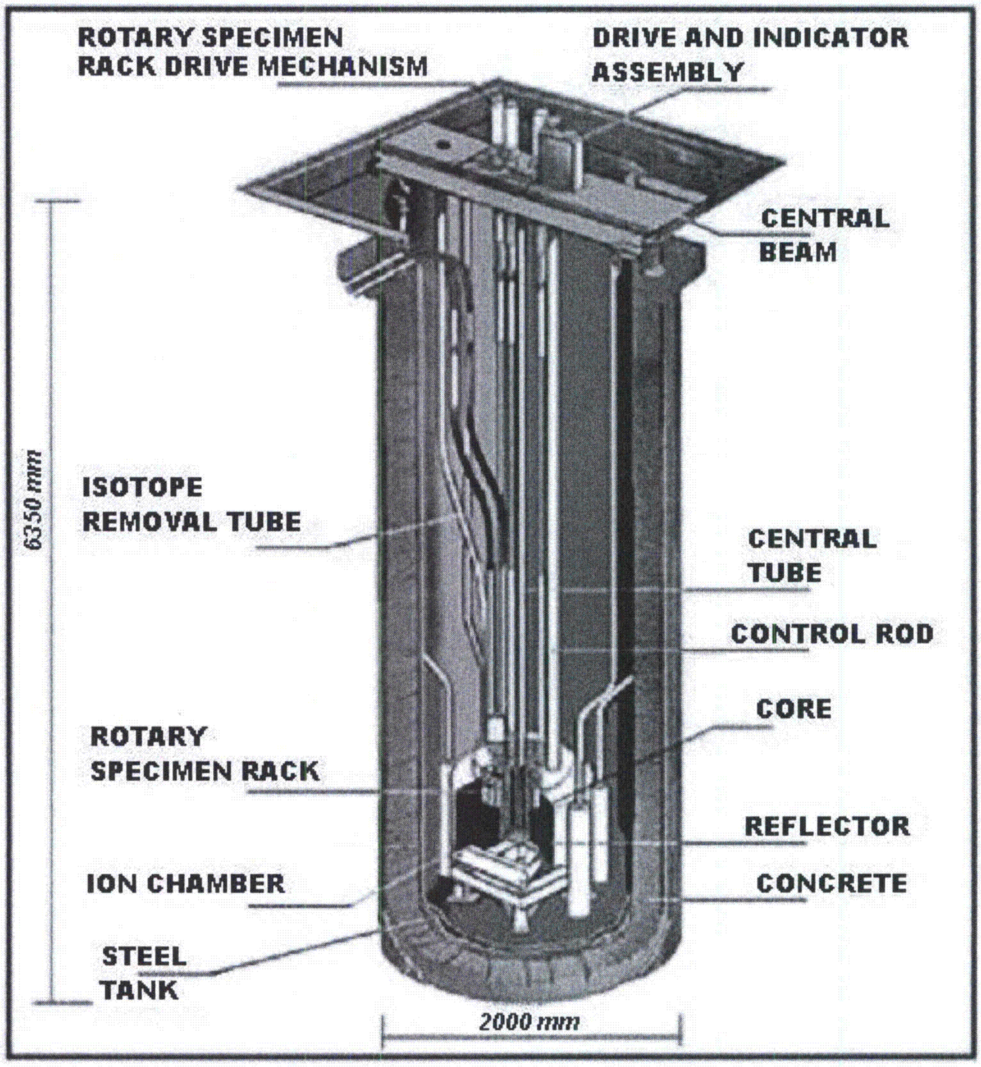
LAND ACREAGE USAGE SCHEDULE		
SYMBOL	TYPE	ACREAGE
	HOSPITAL AND ADMINISTRATION	17.50
	PARKING AREA AND ROADS	10.44
	PATIENT RECREATION	4.48
	HELIPORT AREA	1.11
	UTILITY EASEMENT	0.85
	LICENSED UTILITY EASEMENT	0.02
	BRIDGE REPLACEMENT EASEMENT	0.15
TOTAL MEDICAL FACILITY ACREAGE		34.55

Table 3.1, 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	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
B535	Nuclear research lab	Sample preparation contains one of two fume hoods	None
B535A (SW 2D)	Walk-in cooler	Cold storage	None
B540 (SW 2E)	Nuclear research lab and office	Sample preparation contains 1 of 2 fume hoods	None
B540A (SW 2F)	Isotope and general storage	Storage of irradiated samples	None

The reactor is located near the bottom of a cylindrical pit 20 ft (6 m) below ground level. The pit contains a steel tank with an inner diameter of 6.6 ft (2 m) and a wall thickness of 0.21 ft (6.4 cm). The tank rests on a 1 ft (30.48 cm) concrete slab with at least 10 inches (25.4 cm) of poured concrete surrounding the outside of the tank. The inside of the steel tank is covered on the sides by a layer of gunite (approximately 1 inches [2.54 cm] thick) and on the bottom by poured concrete (approximately 4 inches [10.16 cm] thick).Due to slumping of the gunite, the tank diameter is slightly smaller at the bottom then at the top. The entire inner surface is coated with two applications of a waterproof epoxy resin. The reactor pit was designed to ensure against water leakage. The gunite and its waterproof coating protected the steel tank against corrosion by water. If a small defect in the coating should occur, the steel tank provided a secondary containment vessel. Approximately 16 ft (4.87 m) of water served as shielding above the reactor core. A schematic of the reactor and pit is provided in Figure 3.2. Additional information on the reactor and its facilities can be found in the Safety Evaluation Report for the renewal of the facility operating license (VA 2002b).

Figure 3.2, Reactor and Pit



The core support structure (reflector platform) is bolted to the bottom of the reactor pit and is approximately 1 ft (30.48 cm) long. The core contained 85 fuel-element positions; 57 positions contained active fuel elements, and the other positions were occupied by graphite elements. The assemblies were supported on the top and bottom by grid plates. Both grid plates were 7.5 inches (19.05 cm) of aluminum. The final core consisted of the original 56 aluminum-clad fuel elements and one stainless steel clad element that were added to the core on October 2, 1995. The core was cooled by natural circulation of water, flowing through the core from bottom to top.

A cylindrical reflector that is 1 ft (30.48 cm) thick (inner diameter of 1.4 ft (42.6 cm) and outer diameter of 3.5 ft [1.06 m]) surrounds the core. The graphite reflector is completely encased in a welded aluminum can. Reflectors on the top and bottom of the core were 4 inch (10.16 cm) graphite sections encased in fuel element cans. The reflector assembly rests on the reflector platform, and support is provided by two aluminum channels welded to the bottom of the reflector container. Four tapped holes in the lower flanges of the channels are provided for the leveling screws, which in turn transmit the weight of the reflector to the reflector platform. Four lugs with 2-inch (5.08 cm) diameter holes are provided for lifting the reflector assembly, which weighs approximately 1700 pounds (lbs).

Irradiation facilities were part of the reactor structure and provided for the production of radioisotopes. These include a rotary specimen rack located in the well in the reflector can, a pneumatic transfer tube, and a central thimble. In addition, odd-shaped specimens could be irradiated in the water outside the reflector.

The rotary specimen rack (also referred to as the "Lazy Susan") is an aluminum ring that rotates around the core. Forty aluminum cups, evenly spaced, are hung from the ring and served as irradiation specimen holders. In 2003, Nebraska-Western Iowa Healthcare System employed Scientech, Incorporated of Danbury, Connecticut to transfer the original Lazy Susan to RACE Environmental of Memphis, Tennessee for disposal, but the replacement remains in the reactor core (Figure 3.3).

Approximately 16 graphite ‘dummy’ fuel elements remain in the core assembly and occupy the grid positions that do not contain fuel elements (Figure 3.3), along with an additional 13 in the storage racks on the side of the pool floor. The dummy elements are the same dimensions and construction as the fuel elements, with the exception that they contain graphite instead of fuel. Each element weighs 2.8 lbs and is identified with a blue spacer block.

Three emergency storage pits are located immediately adjacent to the reactor tank. The pits are vertical steel pipes 10 inches (25.4 cm) in diameter and 10 ft long (3.05 m). The pits were intended to store irradiated specimens or failed fuel elements. However, there is no record of them being used for storing fuel or other radioactive materials.

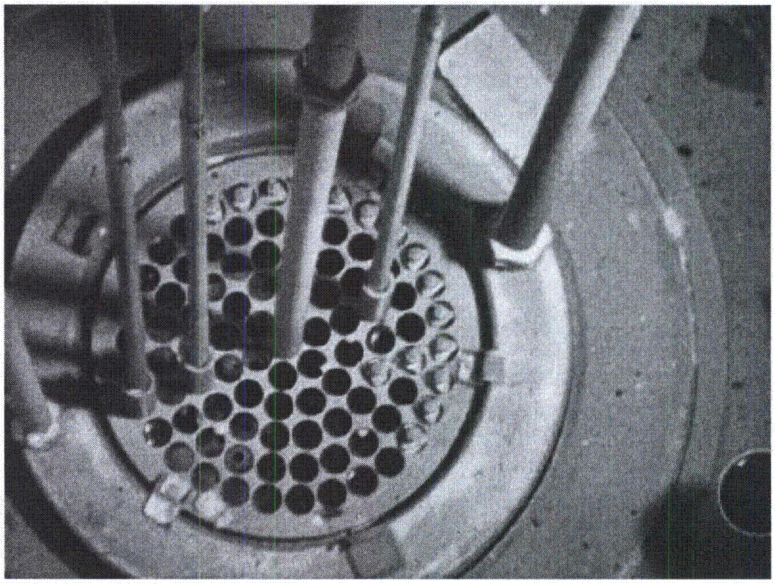
The pneumatic transfer system provided for the production of isotopes with short half-lives. It consisted of two tubes leading down through the water tank to a position at the outer edge of the core, where the tubes were joined. A blower connected to the other tube provided the pressure difference to inject or eject the specimen. Specimens were inserted and removed from the pneumatic transfer system in the reactor laboratory.

The central thimble was installed to provide irradiations or experiments in the region of maximum neutron flux. The thimble consisted of a vertical aluminum tube (with an inner diameter of 1.3 inches [3.3 cm]) leading from the top of the reactor pit through the center of the core and ending at the bottom of the core. The vertical aluminum tube had holes in the region of the core to allow shield water to be removed from the portion of the central thimble above the upper grid plate using air pressure. This provided a highly collimated beam of neutron and gamma radiation for experiments. The central thimble was only used once during reactor operation, for the determination of dose levels.

Three boron carbide control-rods were operated in perforated aluminum guide tubes. Each control rod had an extension tube that connected to the control-rod drive mechanism. The control-rod drive mechanism is located on the bridge at the top of the reactor pool.

The reactor was cooled by natural convection of the pool water. A 5-ton Freon vapor-compression chiller with an air-cooled condenser was used as the heat sink. Water from the reactor went through a water monitor where the temperature, gamma activity, and conductivity of the water were measured. The water was first pumped to the chiller unit, then through a filter and a mixed-bed demineralizer before returning to the tank.

Figure 3.3, Reactor Assembly



A wide-range fission chamber and a boron-lined uncompensated ion chamber provided the reactor core monitoring system. The fission chamber was removed in December 2002 and transferred to the USGS TRIGA Reactor facility in Denver, Colorado.

The reactor room ventilation supply provides 100 percent outside air (heated or cooled) to the reactor laboratory through six ceiling ducts. The exhaust effluent exits the reactor room to the outside air by means of an exhaust fan installed in the outside wall of the building. In addition, two laboratory fume hoods (Figure 3.4) operate continuously and exhaust through two fans installed on the roof of the medical center. Since the blowers for the hood exhaust are on the roof, the entire duct has a negative pressure relative to the adjoining hospital area. Any leakage, if present, would flow into the duct, thus eliminating the potential for exposure within the medical center. The ventilation exhaust point of interest is the area labeled Reactor Lab Hoods on Figure 3.4. As discussed further in Section 7.5.4 of this plan, these hoods will be secured and made non-operational during the D&D activities.

The following facility and system modifications were performed based on ABJRF annual reports and Reactor Safeguards Committee (RSC) minutes and logs back to 1959.

- In 1966, the original rotary specimen rack was replaced. The original rotary specimen rack was stored in the area noted on Figure 2.1 as B535A.
- In the 1960s, the pneumatic transfer system, which was originally routed to the first floor, was cut and re-routed to the reactor laboratory.
- In 1995, a stainless steel TRIGA fuel element was added to the core on October 2, 1995 due to the core fuel burn-up since 1959.
- In 1995, the terminus end of the pneumatic transfer system tube was replaced due to leakage.
- In 1999, the original 1959 General Atomics console was replaced with a solid state General Atomics Mark II console along with safety, shim, and regulating rod drives.

leveling screws, which in turn transmit the weight of the reflector to the reflector platform. Four lugs with 2-inch (5.08 cm) diameter holes are provided for lifting the reflector assembly, which weighs approximately 1700 pounds (lbs).

Irradiation facilities were part of the reactor structure and provided for the production of radioisotopes. These include a rotary specimen rack located in the well in the reflector can, a pneumatic transfer tube, and a central thimble. In addition, odd-shaped specimens could be irradiated in the water outside the reflector.

The rotary specimen rack (also referred to as the "Lazy Susan") is an aluminum ring that rotates around the core. Forty aluminum cups, evenly spaced, are hung from the ring and served as irradiation specimen holders. In 2003, Nebraska-Western Iowa Healthcare System employed Scientech, Incorporated of Danbury, Connecticut to transfer the original Lazy Susan to RACE Environmental of Memphis, Tennessee for disposal, but the replacement remains in the reactor core (Figure 3.3).

Approximately 16 graphite 'dummy' fuel elements remain in the core assembly and occupy the grid positions that do not contain fuel elements (Figure 3.3), along with an additional 13 in the storage racks on the side of the pool floor. The dummy elements are the same dimensions and construction as the fuel elements, with the exception that they contain graphite instead of fuel. Each element weighs 2.8 lbs and is identified with a blue spacer block.

Three emergency storage pits are located immediately adjacent to the reactor tank. The pits are vertical steel pipes 10 inches (25.4 cm) in diameter and 10 ft long (3.05 m). The pits were intended to store irradiated specimens or failed fuel elements. However, there is no record of them being used for storing fuel or other radioactive materials.

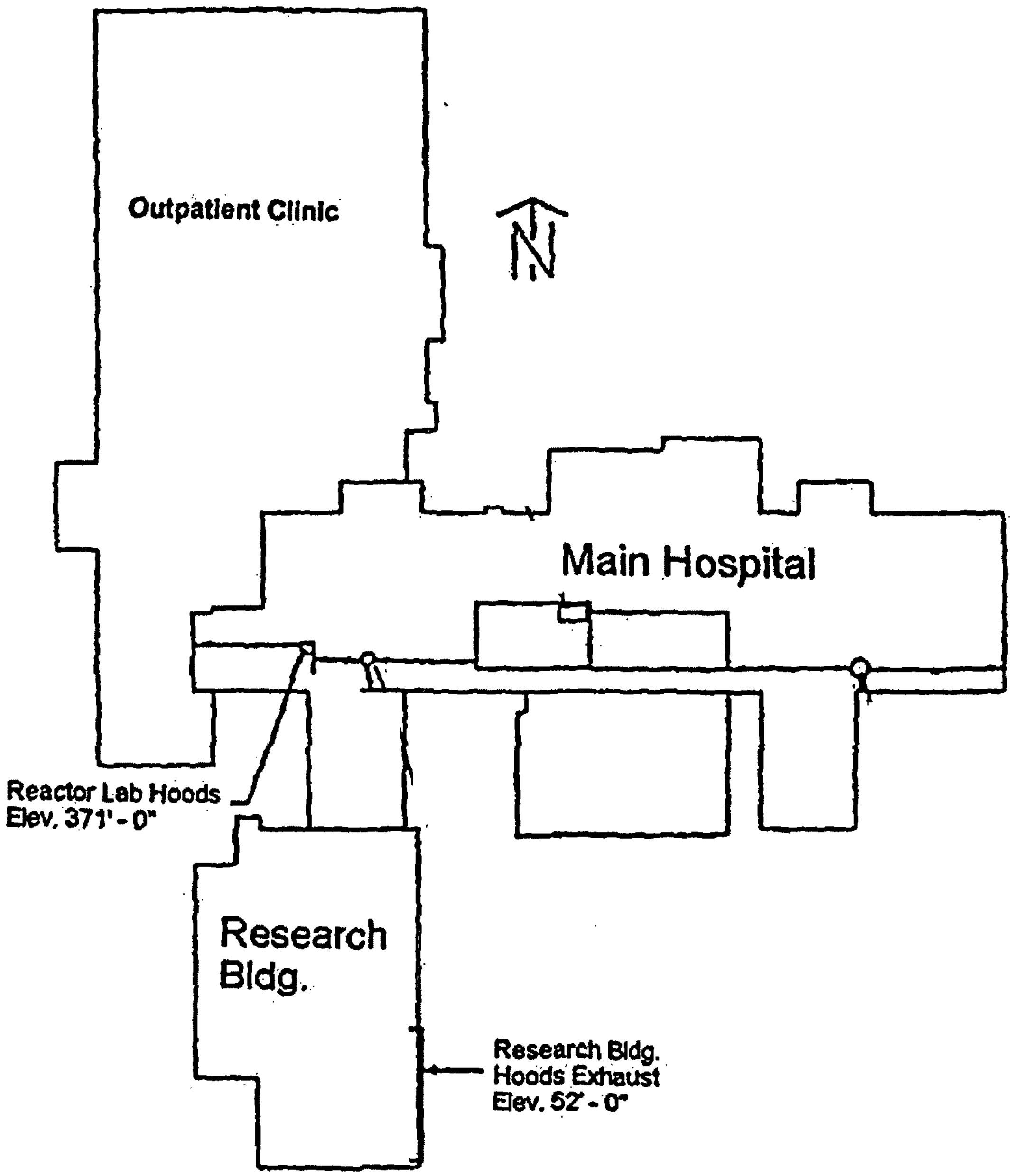
The pneumatic transfer system provided for the production of isotopes with short half-lives. It consisted of two tubes leading down through the water tank to a position at the outer edge of the core, where the tubes were joined. A blower connected to the other tube provided the pressure difference to inject or eject the specimen. Specimens were inserted and removed from the pneumatic transfer system in the reactor laboratory.

The central thimble was installed to provide irradiations or experiments in the region of maximum neutron flux. The thimble consisted of a vertical aluminum tube (with an inner diameter of 1.3 inches [3.3 cm]) leading from the top of the reactor pit through the center of the core and ending at the bottom of the core. The vertical aluminum tube had holes in the region of the core to allow shield water to be removed from the portion of the central thimble above the upper grid plate using air pressure. This provided a highly collimated beam of neutron and gamma radiation for experiments. The central thimble was only used once during reactor operation, for the determination of dose levels.

Three boron carbide control-rods were operated in perforated aluminum guide tubes. Each control rod had an extension tube that connected to the control-rod drive mechanism. The control-rod drive mechanism is located on the bridge at the top of the reactor pool.

The reactor was cooled by natural convection of the pool water. A 5-ton Freon vapor-compression chiller with an air-cooled condenser was used as the heat sink. Water from the reactor went through a water monitor where the temperature, gamma activity, and conductivity of the water were measured. The water was first pumped to the chiller unit, then through a filter and a mixed-bed demineralizer before returning to the tank.

Figure 3.4, Omaha VAMC Roof Exhaust Locations



3.3 Population Distribution

The Omaha metropolitan area includes suburbs in both Nebraska and western Iowa and had a population of 885,350 as of 2010. The city of Omaha had a 2010 estimated population of 432,958 and covers 115 square miles in area. The 2010 census data for Douglas County, Nebraska as distributed by population segments is presented in Table 3.2.

Table 3.2, Douglas County, Nebraska 2010 Census Data

Population, 2010	517,110
Population, percent change, 2000 to 2010	11.5%
Persons under five-years-old, percent, 2009	8.4%
Persons under 18-years-old, percent, 2009	26.3%
Persons 65-years-old and over, percent, 2009	10.7%
White persons, percent, 2009	76.4%
Black or African American persons, percent, 2009	11.6%
American Indian and Alaska Native persons, percent, 2009	0.7%
Asian persons, percent, 2009	2.7%
Native Hawaiian and Other Pacific Islander, percent, 2009	0.1%
Persons reporting some other race, percent, 2000	3.4%
Persons reporting two or more races, percent, 2010	2.8%
Female persons, percent, 2009	50.6%
Persons of Hispanic or Latino origin, percent, 2010	11.2%
White persons, not of Hispanic/Latino origin, percent, 2010	71.9%
High school graduates, persons 25-years-old and over, 2009	89.6%
College graduates, persons 25-years-old and over, 2009	34.7%

As of 11/30/2011, the Omaha VAMC has a staff of approximately 1,335 people and has a typical daily load of 600 outpatients. There are 100 inpatient beds with an average daily inpatient census of 72.

In light of the proposed decommissioning goal and associated 8 to 12 week schedule, no significant population shifts are anticipated during decommissioning activities.

3.4 Current and Future Land Use

The AJBRF is located in the Omaha VAMC, a medical center facility operated by VA. The VA plans to continue such operation of the facility for the foreseeable future. The areas surrounding the reactor in the basement are currently used as laboratories for medical research. It is anticipated that the basement area currently housing the AJBRF will be turned into office space for hospital staff or for use as laboratory space.

The driving force with respect to proposed decommissioning timeline is the likely demolition and re-construction of the entire building in which the reactor area is housed. Omaha VAMC is not contemplating restoration of any research reactor activities at this site in this proposed new construction.

3.5 Meteorology and Climatology

The Omaha climate is typical of the Midwest region of the U.S. with relatively warm summers and cold, dry winters. It is situated midway between two distinctive climatic zones — the humid east and the dry west. Fluctuations between these two zones produce periods of weather conditions that are characteristic of either zone or combinations of both. Omaha is also affected by most storm systems that cross the country. The prevailing winds at the Omaha Airport (Eppley Airfield) are south-southeast during most of the year, shifting to north-northwest during the winter season, with a mean wind speed of approximately 10 miles per hour (16.09 km per hour).

Most of Omaha's precipitation falls during sudden showers or thunderstorms from April to September. Of the total precipitation, about 75 percent falls during this six month period, predominantly as evening/night showers or thunderstorms. Although winters are relatively cold, precipitation is light, with only10 percent of the total annual precipitation falling during the winter.

The mean date of the last freeze in spring is April 14, and the mean date of the first freeze in autumn is October 20. The longest freeze-free period on record is 219 days in 1924, and the shortest period is 152 days in 1885. The average length of the freeze-free period is 188 days.

Omaha and the surrounding region are considered an Attainment Area relative to all airborne pollutants and thus meet the National Ambient Air Quality Standards for those pollutants.

The tornado frequency for Omaha from 1970 through 1995 is summarized in Table 3.3. Tornadoes have been recorded in the general area of the site during this period and have caused minor damage to the hospital; however, no damage has ever occurred to the AJBRF. The AJBRF is in the basement of the Omaha VAMC surrounded by poured concrete walls with no windows and with 4 inches (10 cm) of concrete overhead. Because of this reinforcement, the area in and around the reactor room has been designated as the Omaha VAMC's tornado shelter.

Table 3.3, Tornado Frequency Data, Nebraska

Year	Tornados	Deaths	Injuries
1970	14	0	1
1971	52	0	1
1972	30	0	0
1973	19	0	13
1974	32	0	20
1975	78	4	141
1976	26	0	23
1977	68	0	1
1978	42	0	9
1979	20	0	1
1980	38	5	204
1981	19	0	0
1982	34	0	0
1983	15	0	0
1984	50	0	24
1985	52	0	7
1986	54	0	9
1987	26	0	2
1988	20	2	1
1989	41	0	3
1990	88	0	10
1991	65	0	6
1992	75	0	5
1993	70	0	18
1994	55	0	3
1995	26	0	0

3.6 Geology

The City of Omaha lies within the Dissected Till Plains of the Central Lowland Psysiographic Province of the U.S. The Omaha VAMC is located on a knoll surrounded by gently rolling topography. The medical center sits at an elevation of approximately 1200 ft (366 m) above MSL in a commercial area within the city limits of Omaha. The City of Omaha sits at an elevation of 1000 ft (304 m) to 1300 ft (396 m) above MSL and is bordered on the east by the Missouri River, separating Nebraska from Iowa. Thus, the hospital is on some of the highest ground within the city.

The surface soils in the Omaha area are primarily loess and glacial drift deposits. Two stages of glaciations, the Nebraskan and the Kansan, left thick deposits of till overlying bedrock. It is believed that much of the glacial till has been eroded in the vicinity of the Omaha VAMC and that not more than 98 ft (30 m) remains. The till consists mainly of lean and gravelly clays with a few lenses of sand gravel. The exact depth to bedrock directly below the Omaha VAMC site is not known, but is estimated to vary between 1000 ft (305 m) and 1050 ft (320 m) MSL, on the basis of the nearest top bedrock information.

The loess at the site is of Peorian and Loveland Formations of the late Pleistocene Epoch. The soil classification of the Peorian indicates that the material consists predominantly of clayey silts and lean clay. The soil of the Loveland formation varies from clayey silt to fat clay with minor amounts of sand and clayey sand in the basal part of the formation. At the Omaha VAMC site, the Peorian is from 30 ft (9.1 m) to 45 ft (13.7 m) thick and the Loveland is over 60 ft (18.3 m) thick. The total estimated thickness of the overburden is between 180 ft (55 m) and 213 ft (65 m).

Based on soil boring to a depth of 25 ft (7.6 m) performed during hospital construction, the topsoil layers are Peorian Loess, a wind-blown deposit of clayey silt having low plasticity. The next meter consists of Loveland Loess, a windblown deposit of silty clay having medium plasticity. The bottom 1 ft (0.30 m) of the boring was in glacial clay having a higher plasticity. The test data shows the soils to be strong near the surface, then decreasing to medium strength at a depth just above the very strong clay layer. No water table was encountered.

Bedrock in this area is limestone and shale of the Pennsylvanian period. The surface of the bedrock is very irregular because of erosion that followed the uplift of the area in early Pennsylvanian time and continued on to the Pleistocene period. This uplift brought the granite to within 600 ft (183 m) of the surface in certain areas, forming a ridge known as the Nemaha Ridge or Arch. Also, extensive faulting occurred that developed a major fault, known as the Humboldt fault, which has a throw of over 900 ft (274 m). There is no evidence of activity along this fault in recorded time, and it has not been considered to be a capable fault within the guidance contained in 10 CFR 100.

3.7 Surface Water Hydrology

There is no surface water located on the grounds of the Omaha VAMC. The closest bodies of water to the reactor site are the Big Papillion Creek, approximately 1.86 miles (3 km) to the west, and the Missouri River, approximately 3.1 miles (5 km) to the east.

3.8 Groundwater Hydrology

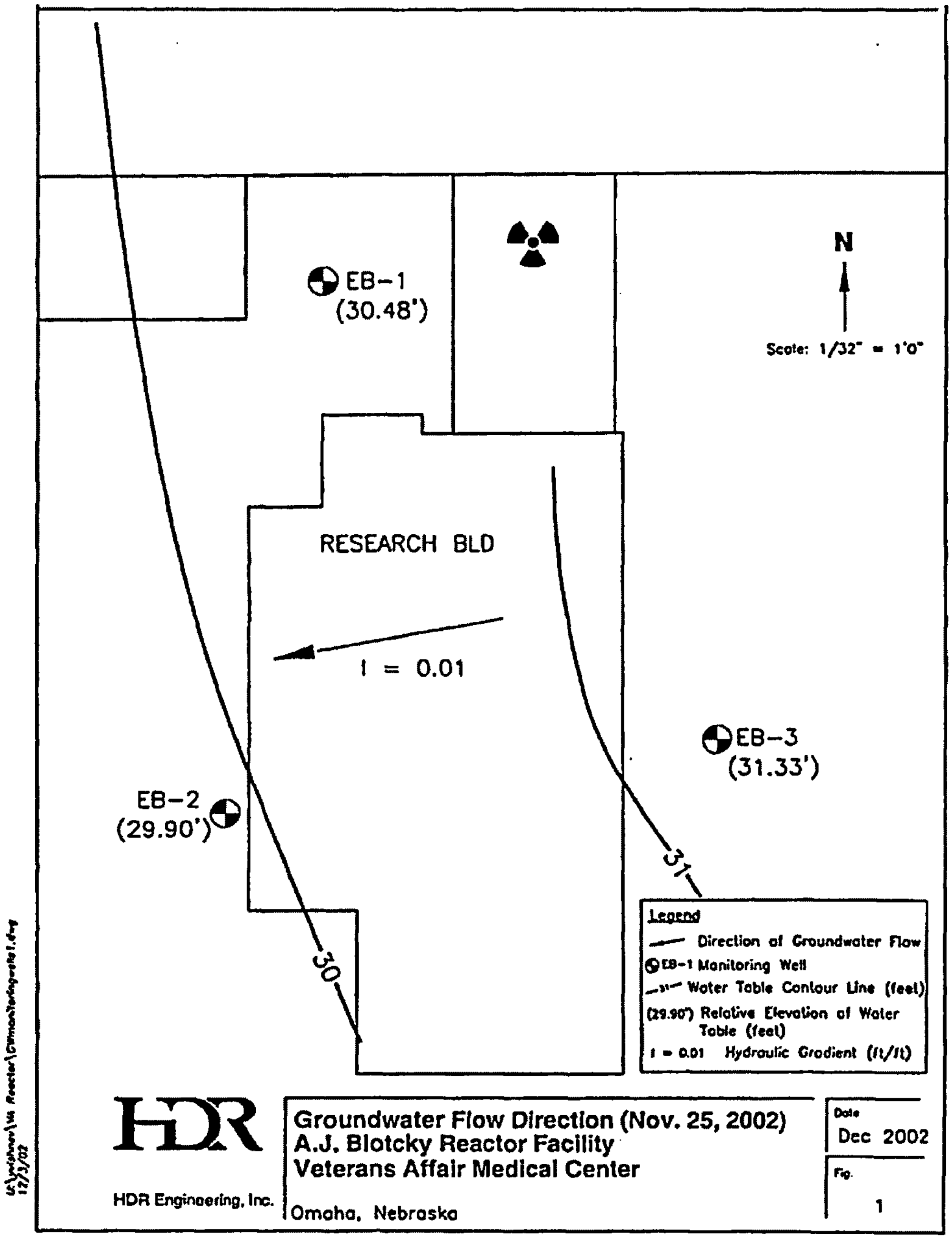
The Big Papillion Creek, which runs in a southeasterly direction, is approximately 3 km west of the site. With the water table troughing along this creek, the underground water would likely migrate along the creek until it returns to the Missouri River South of Offutt Air Force Base (8 miles [13 km] from the site). The Missouri River is approximately 3.1 miles (5 km) due east of the site; however, the topography is such that any transport by groundwater would be towards the Big Papillion Creek.

Based on logs of the 1946 geotechnical borings conducted during design efforts for this site, the zone of saturation was believed to be lower than 65 ft (19.81 m) below grade. Test wells drilled during the site characterization in November 2002 determined water levels to be from 65.6 to 72.1 ft (20 to 22 m) below grade. Drilling locations are presented on Figure 3.5. Based on the three borings performed, the direction of groundwater flow is generally west by southwest with a hydraulic gradient of .03 ft/ft (0.01 m/m). Based upon absence of surface waters, and depth to groundwater, there is little potential that decommissioning activities at the facility will impact neither surface nor groundwater. Groundwater levels are routinely recorded in the three monitoring wells around the facility and they will continue to be monitored throughout the decommissioning and final survey. The well details and discussions of the boring results are provided in the site characterization report (Duratek 2003).

3.9 Natural Resources

There are no commercially or recreationally significant natural resources in the vicinity of the site. Mr. Emmitt Egr of the Papio-Missouri Natural Resources District verified that there are no natural resources, ecology, or endangered species considerations at or near the Omaha VAMC site (personal communication, August 3, 2011).

Figure 3.5, Test Wells, Groundwater Elevations, and Hydraulic Gradient



4. Current Radiological Status of the Facility

VA performed a site characterization of the AJBRF in December 2002 to determine the nature and extent of impacts of the AJBRF operations to the radiological status of the facility and to facilitate the preparation of the draft Decommissioning Plan. No contamination was identified outside the reactor facility. Several areas of contamination were noted in the reactor laboratory area. The highest levels of contamination were found in the room B540 lab hood and in the B535A lab hood drain (Duratek 2003). These results were confirmed during additional characterization efforts conducted in May 2011 and no additional areas of contamination were identified. Additional information on the isotopes of concern was provided from the May 2011 characterization. Also, embedded wastewater drain line piping and laboratory hood exhaust ventilation ducts were sampled (AECOM 2011a). No detectable radioactive contamination above background levels was found in either system.

A list of radionuclides of concern for the AJBRF is provided in Table 4.1. This list consists of common isotopes typically associated with the operation of a TRIGA research reactor and have been confirmed or are suspected to be present within the AJBRF. Because the reactor was shut down in 2001 and fuel removed in 2002, only longer-lived isotopes with a half-life greater than two years are listed. Furthermore, based on site characterization data, operating history, and no known releases, no isotopes associated with reactor fuel or fuel leakages are listed.

Table 4.1, Potential Radionuclides of Concern within the AJBRF

Isotope	Half-Life	Location	Confirmed Present	Screening Level[b]
Tritium (H-3)	12.28 yr	Surface, Soil/Concrete, Dummy Fuel Elements	Yes	1.2×10^8
Carbon-14 (C-14)	5730 yr	Surface, Soil/Concrete, Dummy Fuel Elements	Yes	3.7×10^6
Iron-55 (Fe-55)	2.7 yr	Surface, Soil/Concrete	Yes	4.5×10^6
Cobalt-60 (Co-60)	5.27 yr	Surface, Soil/Concrete, Metal, Dummy Fuel Elements	Yes	7.1×10^3
Nickel-63 (Ni-63)	100.1 yr	Surface, Soil/Concrete	Yes	1.8×10^6
Cesium-137 (Cs-137)	30.17 yr	Surface, Soil/Concrete, Metal	Yes	2.8×10^4
Europium-152 (Eu-152)	13.6 yr	Soil/Concrete, Dummy Fuel Elements	Yes	None
Europium-154 (Eu-154)	8.8 yr	Soil/Concrete, Dummy Fuel Elements	No[a]	None

[a]: Eu-154 is typically found associated with Eu-152, but in lower concentrations.
[b]: Disintegrations per minute per 100 square centimeters (dpm/100cm^2).

4.1 Systems and Structures

There have been two characterization efforts since facility shut down in 2002 (Duratek 2003; AECOM 2011a). No significant areas of surface contamination (total or removable) above the

NRC surface contamination screening levels on structure surfaces were identified during either effort (see Section 6.1).

Removable H-3, C-14, total beta, and alpha activities were generally less than minimum detectable concentration (MDC) for removable contamination measurements (swipes) on floor and wall (structure) and reactor system surfaces. Two of the total 101 swipes taken from the two characterization efforts combined had detectable removable H-3 contamination in excess of 32 disintegrations per minute (dpm), which is 50% of the calculated MDC of 64 dpm. However, all levels were well below 10 percent of the H-3 surface screening criteria (AECOM 2011).

C-14 was identified as the isotope of concern in B535 lab vent hood drain removable contamination smears collected in 2011 (AECOM 2011).

Removable H-3, C-14, and total beta contamination data did not indicate the presence of other hard-to-detect (HTD) isotopes such as Ni-63 or Fe-55 (i.e., the total beta activity was generally equal to the sum of the H-3 and C-14 activity when analyzed in a liquid scintillation counter).

During the 2002 and 2011 characterizations, total surface beta-gamma (β/γ) contamination greater than NRC screening criteria for Co-60 was measured in the lab vent hoods in B535 and B540, the reactor pool cover, and the floor and wall penetrations in B540A (Source Storage Room). Co-60 is the isotope with the lowest screening level of the isotopes of concern presented in Table 4.1(Duratek 2003, AECOM 2011).

The locations and extent of radioactivity detected within the AJBRF during the 2002 characterization efforts relative to structures, systems, and equipment are summarized in Table 4.2. During the site characterization activities in 2011, the AJBRF was swept and most horizontal surfaces were wiped down leaving the area relatively clean. The surface wipe down was performed using Maslin cloths which were surveyed using an alpha/beta phoswich detector prior to disposal. No detectable activity was found during Maslin surveys.

4.2 Reactor Pool and Components

Radiation levels associated with the reactor assembly as measured in December 2002 are depicted on Figure 4.1 (Duratek 2003). These values represent exposure rates at the top of those components measured with an underwater dose rate meter. It is estimated that the vast majority of the dose is from Co-60 present in the Lazy Susan due to activation of the rotary rack bearings, which contain a high concentration of cobalt metal. MicroShield® (Version 8.03) was used to model the dose and estimate the activity of the Lazy Susan and apply it to the reactor assembly as a whole. The model estimates the 2002 activity to be between 0.030 curie (Ci) and 0.045 Ci of Co-60. Co-60 was assumed to be the only significant gamma-emitting isotope. The 0.030 Ci estimate is based on modeling an external dose point on an annular cylinder such that the dose was equivalent to the maximum dose recorded on contact with top of the reactor assembly above the inner edge of the Lazy Susan (1.5 Roentgen per hour [R/hr]). The 0.045 Ci estimate is based on an internal dose point in an annular cylinder filled with water with the dose equivalent to the maximum dose recorded in the center of the top grid plate (0.6 R/hr). Accounting for a decay period of nine years, the estimated Co-60 activity is now between 0.009 Ci and 0.014 Ci. MicroShield® output files for the internal and external dose points are provided in Attachment A.

Table 4.2, Summary of AJBRF Surface Contamination Levels

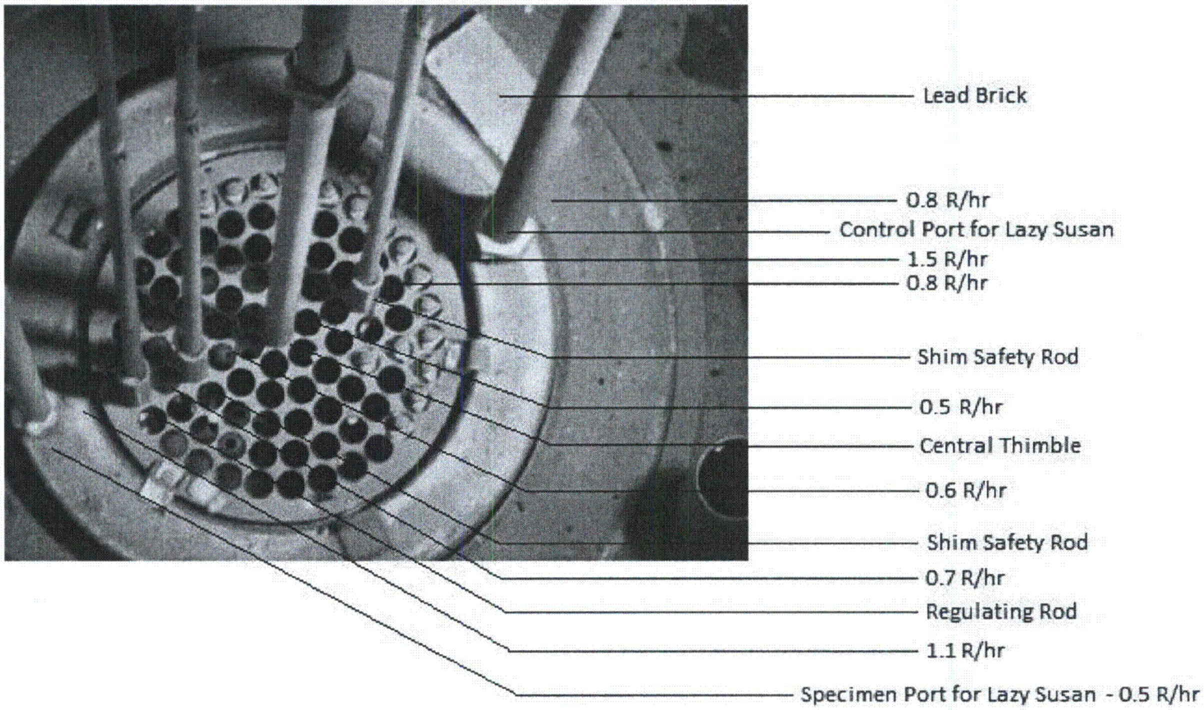
Room Number	Description of Area or Component	Maximum Total β/γ Contamination (dpm/100cm^2)	Maximum Removable β/γ Contamination (dpm/100cm^2)
B526	Reactor Pool Cover	12,116	<MDC
Outside	Cooling System Vault Floor	1,734	<MDC
B537	Nuclear Research Lab and Office, Walls 1 - 4	2,227	<MDC
B540A	Source Storage Room, Concrete Floor	15,682	255
B540A	Source Storage Room, Wall Penetrations 1 - 64	8,748	162
B526	Pneumatic Transfer System, Piping	1,243	<MDC
B540	Lab Hood	72,211	<MDC
B535	Lab Hood Drain	1,554,491	1,080
B535	Sink Drain	9,070	<MDC
B540	Sink Drain	2,017	<MDC

Historically, the activity of the reactor tank water that continues to be circulated has been extremely low and monthly reactor pool water samples since 2006 indicate non-detectable activity. Therefore, loose contamination within the tank or on the reactor assembly and other tank internal components is expected to be minimal. However, H-3, C-14, Fe-55, Co-60, Ni-63, Cs-137, and Eu-152 have all been identified in the reactor water demineralizer filter resins (AECOM 2011). Therefore, the water will require filtering prior to the proposed discharge to the sanitary sewer system. The demineralizer tank and resins, as well as piping and pumps that contacted reactor tank water, will be managed as radioactive waste.

The circulated reactor tank water was cooled by non-contact exchange of heat with a chiller system. This chiller system has been shut down since 2003. The refrigerant was evacuated for re-use elsewhere in accordance with air pollution control regulations. It is expected that components of this chiller system, including the Freon vapor compression chiller (located in the exterior "vault") and the air-cooled condenser (located outside of the vault) are free of internal radioactive contamination.

The steel reactor tank, inner gunite layer, and surrounding concrete are assumed radioactive as a result of neutron activation. However, the vertical extent of the activation (above the reactor core centerline) and the horizontal depth of activation in the surrounding concrete are not yet known. The tank floor and concrete below the reactor are also likely radioactive as a result of activation.

Figure 4.1, Reactor Dose Rates as Measured in December 2002



4.3 Surface Soil

Surface soil samples around the exterior of the AJBRF were collected and analyzed for gamma-emitting radionuclides, H-3, C-14, Fe-55, Ni-63, and other potential isotopes of concern. Sampling and analysis were performed in both the 2003 and 2011 characterization efforts. No isotopes of concern or isotopes of potential concern were detected in the surface soils above background levels (Duratek 2003, AECOM 2011). Sampling locations from the 2002 and 2011 events are depicted on Figure 4.2 and 4.3 respectively.

4.4 Subsurface Soil

Subsurface soil samples around the exterior of the AJBRF and below the floor of the reactor room (B526) have been collected and analyzed for gamma-emitting radionuclides, H-3, C-14, Fe-55, Ni-63, and other potential isotopes of concern. Sampling and analysis were performed in both the 2003 and 2011 characterization efforts. No isotopes of concern or isotopes of potential concern were detected in the subsurface soils above background levels (Duratek 2003, AECOM 2011).

The locations of soil borings performed in the soil adjacent to the reactor pit in 2002 are depicted on Figure 4.2. Note that soil-boring number IB-4 began outside the facility to allow for access under the reactor pit. While no radioactive contamination was identified in any samples, analysis was not performed for HTD beta-emitting isotopes such as H-3 and Ni-63 (Duratek 2003). Therefore, no data is available to verify the presence (or absence) of some isotopes of concern such as H-3 and Ni-63 close to the reactor tank. Additional soil sampling near the reactor tank will be conducted as part of the FSS.

The locations of subsurface soil samples collected outside the facility in 2011 are shown on Figure 4.3. Soil samples from these locations near the two "hatches" that provide access to the basement area were analyzed for gamma-emitting isotopes as well as HTD isotopes (AECOM 2011). The data indicate that there are no isotopes of concern above background levels in the subsurface soils around the perimeter of the ABJRF.

4.5 Surface Water

There is no surface water either on the Omaha VAMC campus or in the immediate vicinity.

4.6 Groundwater

Shallow groundwater wells approximately 60 ft (18.2 m) below ground surface were installed around the AJBRF in 2001 (Duratek 2003). Groundwater samples from these wells have not been analyzed for radionuclides. However, no soil samples collected during installation of the wells indicated the potential for subsurface contamination (Duratek 2003).

Figure 4.2, 2002 Soil Sampling Locations

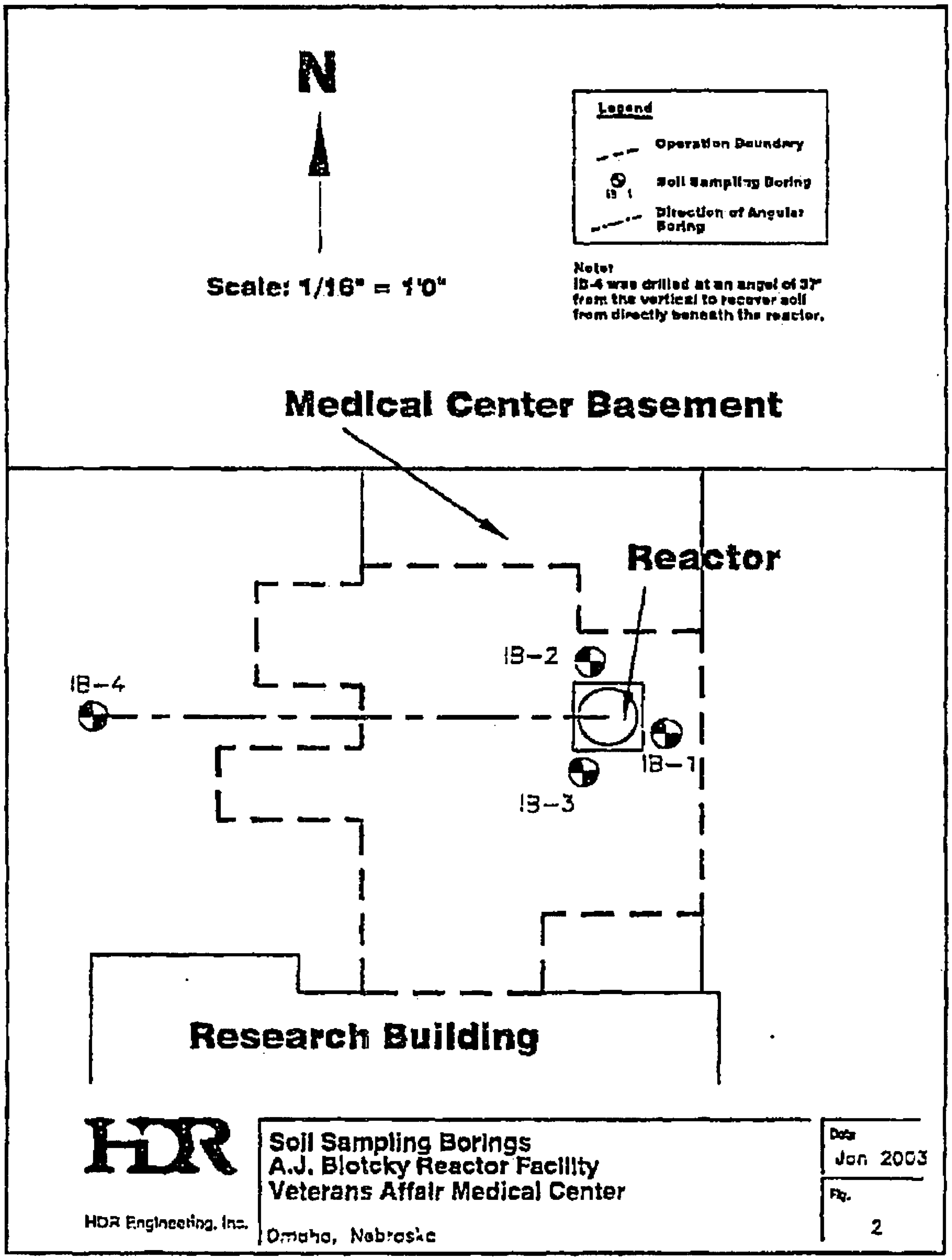
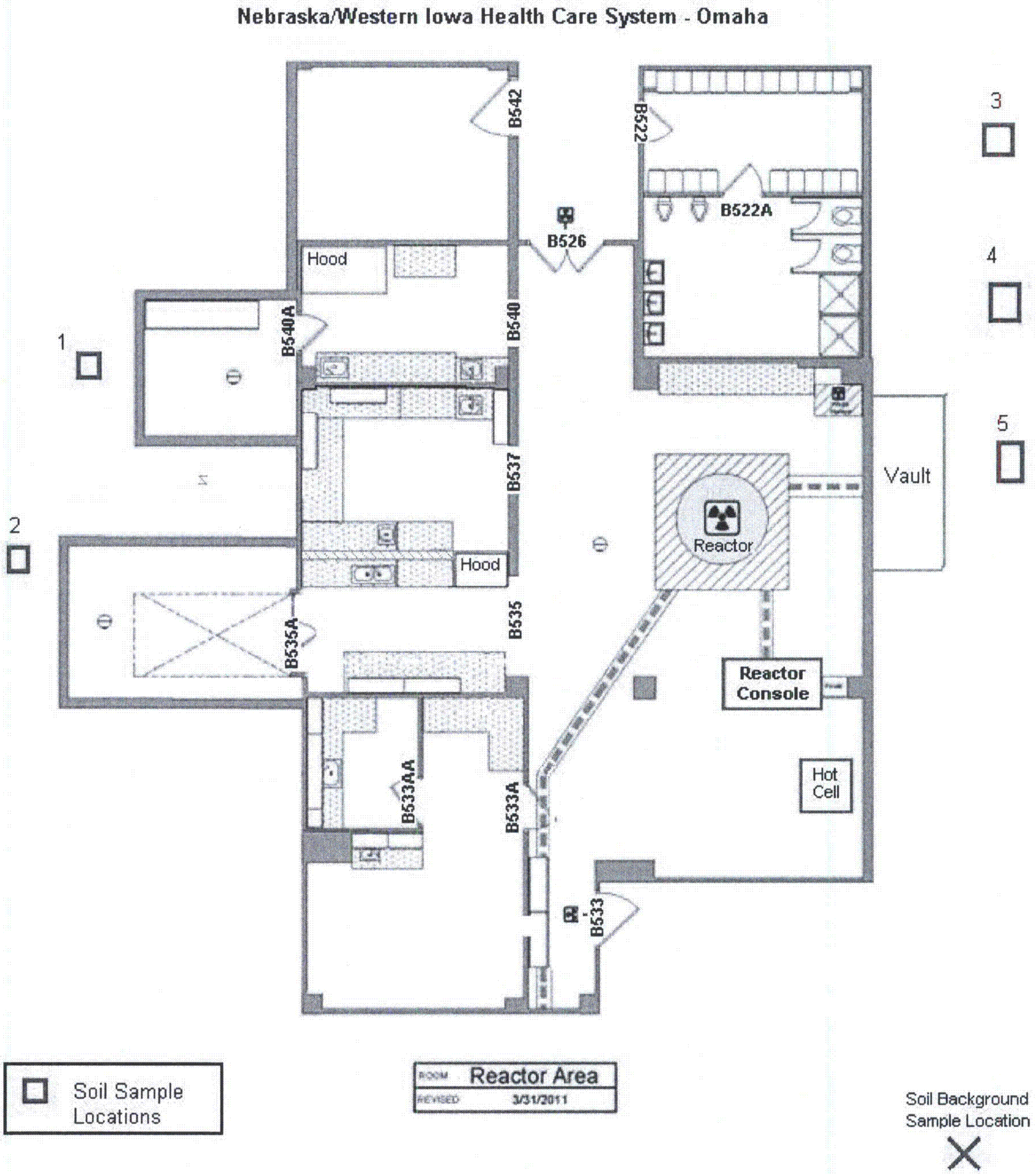


Figure 4.3, 2011 Soil Sample Locations



5. Alternatives Considered and Rationale Supporting the Decommission Strategy Selected

5.1 Decommissioning Objective and Strategy Selected

The objective of the AJBRF decommissioning is termination of operating License Number R-57 and unrestricted release of the reactor site and adjacent areas in accordance with 10 CFR 20.1402. The Omaha VAMC facility where the AJBRF is located will continue to function as a VA hospital post decommissioning. The planned use of the current AJBRF areas will be for storage, laboratory space, or staff offices. Based on the site objective, safe storage (SAFSTOR) and entombment (ENTOMB) decommissioning alternatives were considered and rejected as inconsistent with the planned future use. Decontamination (DECON) is the decommissioning option that has been selected by the Omaha VAMC to best support site requirements.

No impact to the surrounding community or natural resources is anticipated as a result of the DECON alternative execution. The Omaha VAMC's current functions and the existing occupancy will not change as a result of decommissioning activities or outcomes. There is no potential for criticality during any of the DECON activities as all fissile material has been removed from the site.

5.2 Alternatives Considered

SAFSTOR and ENTOMB were considered to satisfy requirements for public protection while minimizing initial commitments of time, money, radiation exposure, and waste disposal capacity. However, both SAFSTOR and ENTOMB strategies require prolonged decommissioning schedules and would result in the current reactor spaces being unavailable for extended periods. In addition, maintenance, security, and surveillance would be continuously required until the final decontamination activities were completed. Therefore, both of these approaches would require continued annual expenditures for operating costs and management oversight. Furthermore, over an increased timeline, the availability of personnel with deep knowledge and expertise in reactor operations would diminish significantly. For these reasons, SAFSTOR and ENTOMB were rejected as decommissioning alternatives.

5.3 Rationale for Chosen Alternative

DECON is the most environmentally preferable alternative and allows for the maximum flexibility in future re-use of the site. Conformity to As Low As Reasonably Achievable (ALARA) principles will minimize radioactive exposures to the Omaha VAMC staff and patients, as well as contractors participating in the DECON process. This process will be followed by a comprehensive final contamination survey demonstrating that the reactor site meets the NRC criteria for release to unrestricted use. The survey results will be documented and submitted to the NRC to support the request for reactor license termination and release to unrestricted use.

6. Release Criteria

6.1 Unrestricted Release of Structures and In-place Systems

VA will reduce contamination levels to levels that are ALARA and use radionuclide screening criteria provided in NUREG 1757, Volume 1, Rev 2 App B (Tables B.1 and B.2), as site-specific release criteria for facility surfaces and in-place systems (floors, walls, ventilation, pipes, etc.) and volumes of concrete and soil. The screening criteria have been calculated to individually correspond to a 25 millirem per year dose limit. Therefore, the sum-of-fractions rule will apply when considering multiple contaminants. By applying the screening criteria, development of site-specific dose modeling is not warranted.

6.1.1 Building and System Surfaces

The criteria that are applicable to the release of building surfaces and surfaces of systems that are to remain in place following decommissioning (e.g., ventilation ducts and embedded pipes) are provided in Table 6.1. For simplicity and conservatism, VA will apply the Co-60 surface activity limit to all detectable β/γ surface activity (7,100 disintegrations per 100 square centimeters [dpm/100cm^2]). The total surface activity screening criterion was developed assuming a 10 percent removable fraction. Therefore, VA will apply a removable β/γ contamination criterion of 710 dpm/100cm^2 for "detectable" removable surface contamination measured in a sample counter such as a Ludlum Model 2929 or equivalent.

Table 6.1, NRC License Termination Screening Levels for Surfaces

Radionuclide	Screening Levels for Unrestricted Release (dpm/100cm^2)	Detectable or HTD
H-3	1.2×10^8	HTD
C-14	3.7×10^6	HTD
Fe-55	4.5×10^6	HTD
Co-60	7.1×10^3	Detectable
Ni-63	1.8×10^6	HTD
Cs-137	2.8×10^4	Detectable

Total surface contamination of HTD isotopes will be evaluated using the results of removable contamination swipes analyzed in a liquid scintillation counter (LSC). Based on characterization data from the 2011 characterization event, HTD isotopes are present in the form of low-level surface contamination or in contaminated/activated materials (AECOM 2011a). VA will assume a removable fraction of 10 percent when applying removable contamination levels to total surface contamination screening levels. Data should be reported for H-3 and total beta activity. The screening value in Table 6.1 for H-3 will be used to evaluate the H-3 results. The total net beta activity will be compared against the Ni-63 screening level in Table 6.1 as it is the most conservative screening level for the HTD isotopes listed. While some of the total net beta activity may be due to "detectable" isotopes, the activity of these isotopes will be accounted for in the non-LSC swipe sample counting.

VA will use the sum-of-fractions rule to evaluate the impact of all measured activity. The following equation demonstrates this approach:

$$\frac{total\ detectable\ \beta/\gamma\ dpm}{7.1E3\ dpm} + \frac{removable\ H3\ dpm}{1.2E7\ dpm} + \frac{removable\ net\ HTD\ \beta\ dpm}{1.8E5\ dpm} < 1$$

As there are no alpha-emitting isotopes of concern listed in Table 4.2 and characterization data do not indicate the presence of alpha contamination, measurements for alpha surface contamination are not required. However, should either total or removable alpha contamination levels be assessed, VA will apply the alpha surface contamination criteria from NRC Regulatory Guide 1.86 of 1,000 dpm/100 cm^2 average, 3,000 dpm/100cm^2 maximum, and 200 dpm/100cm^2 removable.

6.1.2 Soil and Sub-surface Concrete

The screening criteria that will be applied to the release of soil and concrete are provided in Table 6.2. All isotopes can be quantified using laboratory analysis and the sum-of–fractions rule will apply when evaluating data against the screening criteria.

Table 6.2, NRC License Termination Screening Levels for Soils/Concrete

Radionuclide	Default Screening Criteria (pCi/g)	Detectable or HTD
H-3	110	HTD
C-14	12	HTD
Fe-55	10,000	HTD
Co-60	3.8	Detectable
Ni-63	2,100	HTD
Cs-137	11	Detectable
Eu-152	6.9	Detectable
Eu-154	8.0	Detectable

pCi/g - picocuries per gram

After removing activated concrete to a level that is considered ALARA, VA will apply the release criteria in Table 6.3 to the concrete that will remain in place. The determination of ALARA will be based on safety considerations associated with deploying personnel into the reactor tank/pit. Once the structural limits of the concrete have been reached, as determined by a structural or civil engineer, and radioactivity is below the screening criteria, remediation will be considered complete.

While it is currently expected that the surrounding soils are not activated/contaminated based on results from both site characterizations, soil samples will be collected during decommissioning activities to demonstrate compliance with the above criteria. Samples will be collected either horizontally through the concrete or vertically through the basement floor. The sampling approach will be defined in the Final Status Survey Plan (FSSP).

6.2 Criteria for the Free-Release of Items and Articles from the AJBRF

During decommissioning activities, VA will not release items from the AJBRF with detectable residual activity. Detection limits will not be greater than the values provided in Table 6.3. These criteria will be applied to materials that are designated as "clean waste" and materials that are to be released for unrestricted use. These criteria are those given in Table 1 of Reg Guide 1.86 *Termination of Operating Licenses for Nuclear Reactors.*

Table 6.3, Maximum Allowable Limits for Free-Release

Radiation	**Average**	**Maximum**	**Removable**[1]
Beta-gamma	5,000 dpm/100 cm^2	15,000 dpm/100 cm^2	1,000 dpm/100 cm^2

[1]**Reg Guide 1.86 *Termination of Operating Licenses for Nuclear Reactors***

6.3 Alternative Release Criteria

VA may wish to apply alternative release criteria. Any proposed criteria that deviate from those described in this DP will be submitted to the NRC for approval before use.

7. Planned Decommissioning Activities

VA intends to decontaminate and dismantle the AJBRF and associated systems in a safe manner and in accordance with ALARA principles, the AJBRF Radiation Protection Program, and written procedures provided by the Decommissioning Operations Contractor (DOC). The planned decommissioning will include the general activities described in Table 7.1 and detailed in the following sections. The durations noted in the table are as currently projected prior to approval of the DP by the NRC and prior to contracting of a DOC.

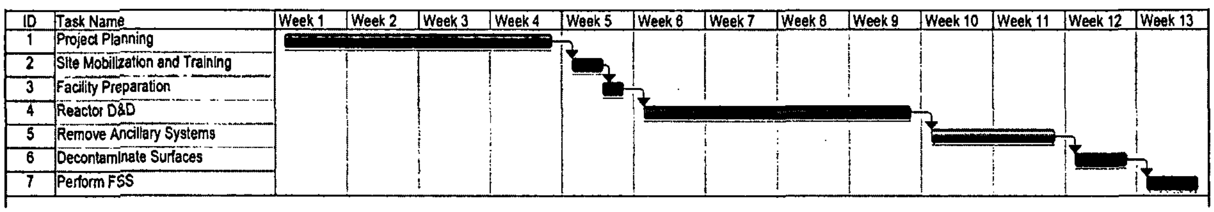
Table 7.1, Planned Decommissioning Tasks

Task	Task Name	Estimated Duration
1	Project Planning (off-site)	4 weeks
2	Site Mobilization and Training	3 days
3	Facility Preparation	2 days
4	Reactor D&D	4 weeks
4A	Remove Control Rod Drives and Bridge	
4B	Remove Fuel Storage Rack and Interference	
4C	Remove Reactor Core Components	
4D	Drain Tank and Process Water	
4E	Remove Gunite, Liner, Tank, Soil	
5	Remove Ancillary Systems	2 weeks
5A	Remove Water Filter System	
5B	Remove Water Cooling System	
5C	Remove pneumatic transfer system	
5D	Remove lab vent hoods and drain lines	
6	Decontaminate Surfaces	1 week
6A	Decontaminate Room 540A (Source Storage)	
6B	Decontaminate Fuel Storage Pits	
6C	Decontaminate Miscellaneous Surfaces	
7	Perform FSS	1 week
8	NRC Confirmation Surveys	Not Applicable (NA)
9	Secure Reactor Pit	NA

The decontamination, dismantling, and demolition of the facility; and the removal of materials (both contaminated and non-contaminated); will be performed by contractors under the supervision of the VA and independent oversight consultants. Organizational information governing the performance of the effort is provided in Section 8. This DP covers only those efforts required for decommissioning of the reactor and associated support equipment. This DP does not address any re-construction efforts (if any). As one of the driving forces with respect to the timeline for decommissioning is the likely demolition and re-construction of the entire building in which the reactor area is housed, it is not clear at present whether this area will be re-constructed or used at all. Since the goal of this DP is unrestricted use, the re-construction (if any) is immaterial to the evaluation of the DP.

The preliminary decommissioning project with the preparatory decommissioning activities, including final survey and license termination, is presented in Figure 7.1. The decommissioning activities will commence with the components and/or structures that are the most contaminated and proceed to those structures and components that are least contaminated. This approach will minimize the potential spread of contamination to areas considered clean and will remove sources of exposure from the job site, therefore, reducing the overall exposure to the decommissioning work crews.

Figure 7.1, Preliminary Decommissioning Project Schedule



7.1 Project Planning

7.1.1 VA

VA will develop a scope of work and bid specifications for prospective DOCs. These documents will include specific requests for decontamination, demolition, radioactive waste packaging and disposal, and final status survey activities. VA will contract a DOC to perform the AJBRF decommissioning in accordance with this DP and the associated bid documents.

During the project planning phase, VA's project management team will coordinate with the DOC to schedule upcoming D&D activities. The planning, preparation and mobilization activities shall ensure that the necessary resources, procedures, work requirements, controls, processes, and Omaha VAMC and contractor roles and responsibilities are clearly identified and documented.

In conjunction with the preliminary engineering tasks, the technical specifications shall be reviewed to determine which, if any, required surveillances will be impacted by the decommissioning activities. Requests for changes to the technical specifications may be prepared and submitted to address such impacts prior to the start of the decommissioning work.

7.1.2 DOC

The DOC will be assigned preliminary engineering tasks to support the decommissioning work. Preliminary engineering tasks will include the design of temporary electrical power, temporary exhaust ventilation, temporary structural supports, as well as the planning for the removal of reactor systems and reactor components. Designs will be reviewed and approved by VA. The

DOC will also prepare a project Work Plan, Draft FSSP, Health and Safety Plan (HASP), Radiation Protection Plan (RPP), and Quality Assurance Project Plan (QAPP) during the project planning phase.

Prior to the start of the D&D activities, the DOC's safety and management team and the Omaha VAMC decommissioning project personnel (i.e., Radiation Safety Officer [RSO] and Safety Manager) may prepare procedures for project planning, control and oversight, and reporting of work.

7.2 Site Mobilization and Training

Initial mobilization of DOC personnel will include the travel of workers to the site, their training, and acquisition and delivery of required equipment for the DP tasks. DOC management staff will establish a site office and storage area in a location determined by Omaha VAMC staff.

Each individual performing decommissioning work, oversight or supervision shall attend site-specific Radiation Worker training in accordance with approved Omaha VAMC procedures and 10 CFR 19.12. Training will be provided by the DOC's radiation safety professional. Additionally, all workers shall attend a safety and securities briefing provided by Omaha VAMC personnel regarding site-specific items.

7.3 Facility Preparation

7.3.1 Establishment of Alternate Access to AJBRF

To minimize disruption of hospital operations during decommissioning and to minimize the potential for delays in accessing the reactor area and delays in removing contaminated materials, an alternate access may be constructed from outside the Omaha VAMC research building (Figure 3.1) directly to the AJBRF. This access could consist of a temporary door in the roof of the former walk-in cooler room (B535A in Figure 2.3) to lift materials directly out of the reactor room area. In this case, site boundary will be extended to include this outside area.

7.3.2 Establishment of Radiological and Hazardous Waste Packaging/Staging Area

An area will be designated inside the AJBRF to package and store industrial, radiological, and hazardous waste. Once packaged and ready for shipment, the waste may be taken via existing or alternate routes for loading onto transport vehicles.

7.3.3 Establishment Exterior Controlled Area Boundary and Radiological Control Points

The areas outside the facility may serve as an equipment staging area (Figure 2.1). These areas may be used for placement of temporary exhaust ventilation fans and high efficiency particulate air (HEPA) filters and for staging trailers for the storage and donning of protection clothing, offices for supervisory personnel, break rooms for workers and the staging of packaged radioactive waste for pickup.

During the D&D effort, Omaha VAMC police will provide security. They regularly patrol the hospital grounds and interior.

7.4 Reactor Decontamination and Demolition

The DOC will develop a detailed project Work Plan for review and approval by VA. The following sections generally describe the anticipated D&D activities. The specific work sequencing will be determined by the DOC.

7.4.1 Control Rod Drives and Bridge

The center channel assembly, control rod drive mechanisms and aluminum grating/Plexiglas tank covers will be manually decontaminated to the extent possible, removed and cut to size for packaging as low-level radioactive waste (LLRW).

7.4.2 Remove Fuel Storage Rack and Interference

The fuel storage rack will be removed from the reactor tank after the control rod drives and bridge. In order to remove the rack, guide tubes for the control rods, any remaining lights, and any other interference must be removed from the reactor tank. These items are expected to be activated and, therefore, size-reduced as necessary for packaging as LLRW.

7.4.3 Remove Reactor Core Components

The core internal components, including the reflector and the upper and lower support plates will be removed from the reactor tank. The reactor core assembly forms a cylinder approximately 43 inches in diameter by 23 inches high, and rests on a platform that raises the lower edge of the assembly approximately 2 ft off the tank floor. A photo of the reactor core components is provided as Figure 3.3.

The reactor internal components consist of:

- Reflector — an aluminum cylinder 1 ft (30.5 cm) thick, 3.5 ft (107 cm) outer diameter, and 1.83 ft (56 cm) high containing graphite.
- Top and Bottom Grid Plates — aluminum grids approximately .75 inches (2 cm) thick that support and align the fuel elements.
- Rotary Specimen Rack - an aluminum ring that contains forty evenly spaced aluminum cups that are hung from the ring and served as irradiation specimen holders. The rotary specimen rack is completely enclosed in a welded aluminum box.
- Pneumatic Transfer Tube and Central Thimble — aluminum tubs that either enter from the side of the tank or vertically from the top. The pneumatic transfer tube and central thimble were used to send samples into, and to retrieve samples from, the core.
- Dummy Fuel Elements — aluminum-clad elements in which the uranium-zirconium-hydride fuel has been replaced by graphite. Approximately 16 of these are still contained in the core assembly with another 13 stored in the storage racks on the side of the pool floor.

A containment enclosure may be constructed over the reactor tank, with a dedicated ventilation and HEPA filter system to prevent the spread of contamination into the rest of the facility. To prevent the generation of airborne contamination and provide shielding for the workers, these components should be cut and/or disassembled as necessary underwater utilizing remote tooling. After cutting/disassembly, the pieces will be raised to the surface and placed inside shipping containers with adequate shielding to meet package dose rate limits defined in

Department of Transportation (DOT) regulations. Components may require air drying prior to loading and additional radiation safety controls may be required during drying activities.

During this evolution the reactor water circulation and demineralizer system will remain in operation to maintain water clarity and remove contaminated particles generated during the operation. Continuous air monitors (CAM) will be used and all workers will utilize appropriate protection clothing and respiratory protection equipment.

7.4.4 Drain Reactor Tank and Process Water

After the reactor internal components are removed, the reactor coolant water will be pumped from the reactor, through an ion exchange filter and into a tank staged outside the temporary access. The water will be sampled to verify that the water meets liquid effluent release limits and then it will be discharged directly to the sanitary sewer system. Water that does not meet the release criteria will be stabilized and managed as LLRW. The circulation pump for the cooling system, by design, does not produce enough suction head to drain the reactor tank, so an auxiliary pump with sufficient suction will need to be supplied and connected to the system.

7.4.5 Potential Remove Gunite, Liner, Tank and Soil

The reactor tank structure will be removed only as required due to activation or radioactive contamination. If there is no radiological rationale for its removal, the radiological status will be documented and it will be physically secured and left in place for removal as directed by VA. Omaha VAMC is currently slated for a several hundred million dollar major renovation which is likely to include demolition of this entire building wing which houses the research reactor facility.

The reactor tank structure consists of a carbon steel tank 0.25 inches (0.64 cm) thick with an inner diameter of 6.8 ft (208 cm) resting on an 11 inch (28 cm) thick concrete slab. Approximately 9.85 inches (25 cm) of poured concrete surrounds the outside of the tank. The inside of the tank is covered on the sides by a 2 inch (5 cm) thick layer of Gunite and on the bottom by 4 inches (10 cm) of poured concrete. The entire inner surface is coated with two applications of a waterproof epoxy resin coating. Due to slumping of the Gunite, the tank diameter is smaller at the bottom than at the top by between 1 to 2 inches.

Upon removal of the reactor internal components and water (as described in Sections 7.4.2 through 7.4.4), core bores will be performed into the surrounding material to determine the depth of irradiated materials (i.e., epoxy/Gunite, steel, concrete and soil). A portable high-purity germanium gamma spectroscopy unit (such as Canberra's ISOCS/LabSOCS) is capable of minimum detectable activities (MDAs) low enough of observing the entire suite of gamma producing isotopes in question and could be used to analyze material samples in the field. Additional analysis for alpha and HTD isotopes will require analysis at an off-site laboratory.

If contamination/activation levels are such that removal of some or all of the pit structure is required (levels exceed release criteria in Table 6.2), then contaminated materials will be removed. The Omaha VAMC research wing, including the AJBRF, sits on 5 ft (1.5 m) thick concrete footers that effectively surround the reactor pit, approximately 1.5 ft (46 cm) from the outer diameter of the reactor tank. To ensure that the pit does not collapse upon itself during the decommissioning, engineered supports may need to be installed to support the sides of the pit.

The reactor pit will be classified as a confined space. Each individual who works in the reactor pit shall be trained for confined space entry. All work in the reactor pit will be in accordance with written procedures that incorporate industrial safety as well as radiological controls that have been approved by the RSC.

Compliance with decommissioning criteria can be demonstrated by sampling materials around the reactor pit at various depths by penetrating the pit wall horizontally or by sampling vertically from the reactor room floor. The selection of the method is up to the decommissioning contractor which will provide the details of the sampling method in their FSSP.

Upon completion of the reactor pit decommissioning, the pit will be placed in a safe condition with respect to fall protection while awaiting final radiological clearance. Engineering supports and bracing inside the pit (if used during decontamination efforts) will remain in place until confirmation is received from the NRC that the pit meets the release criteria. Following NRC concurrence, the pit will be backfilled and physically closed (possibly by others following departure from site of DOC).

7.5 Remove Ancillary Systems

7.5.1 Remove Water Filter System

The reactor water filter system consists of a circulation pump, water filter, mixed bed demineralizer and piping. The resin contained in the demineralizer will be discharged and managed as LLRW, and the shell will be surveyed for contamination and also likely managed as LLRW. Because the internals of the pump are difficult to survey, the pump will likely be managed as LLRW.

Piping between the reactor and the demineralizer penetrate the east wall of AJBRF Room B526 approximately 12 ft (3.65 m) above the floor. The system piping will be cut out and piping managed as LLRW. Pipes should be cut and ends crimped and taped to limit the spread of potentially contaminated materials. The penetrations in the east wall of room B526 will be surveyed, decontaminated if necessary, and sealed.

7.5.2 Remove Water Cooling System

The reactor cooling system consists of a 5-ton Freon vapor-compression chiller with an air-cooled condenser, and piping. The chiller is located in the East Cooling Pit, outside the building. Heat was exchanged between the reactor cooling water and the refrigeration system in a non-contact closed system.

The refrigerant has already been recovered from the coolant system as reactor is non-operational. Cooling system components will be disconnected and surveyed to the extent practical to demonstrate non-contamination. Internal and external surfaces of the cooling system components are not expected to be contaminated.

VA will apply the same release criteria presented in Table 6.1 for surface materials to inaccessible areas such as the coolant pipes. Contaminated systems (accessible or inaccessible) that do not meet the release criteria will be removed and managed as LLRW.

7.5.3 Remove Pneumatic Transfer System

The pneumatic transfer system tubing that exists outside the reactor pit will be removed and/or cut, surveyed, and decontaminated. VA surveyed the embedded piping, sewer lines, and the pneumatic transfer system as part of the May 2011 characterization. No contamination above the system detection limits or background levels was identified in these systems (AECOM 2011). The surface contamination release criteria provided in Table 6.1 will be applied to the pneumatic transfer system pipes and trench.

All pneumatic tubing on the basement level will be removed. This will require the partial demolition of a wall to access vertical tubing. Removal techniques should consider methods that will crimp the end of the tubing to avoid the potential release of contamination. All cutting will be done in conjunction with a dedicated exhaust ventilation system and HEPA filters to prevent the spread of airborne contamination. All removed tubing will be managed as LLRW. The trench will be decontaminated as necessary to meet release criteria.

7.5.4 Remove Lab Vent Hoods and Drain Lines

The lab hoods located in rooms B540 and B535 will be dismantled (Figure 2.3). Additional characterization efforts in 2011 indicated that the laboratory hood structures were likely not contaminated above release criteria; however the sink drains inside hoods had elevated levels of carbon-14 and require handling and disposal as LLRW. Any sections not meeting the Table 6.1 release criteria will be managed as LLRW. Contaminated or suspect drain lines will also be managed as LLRW. The drain line plumbing within several feet of the sink discharge did not show evidence of contamination. Note that the interior panels of the fume hoods were found to contain asbestos during the 2001 characterization survey (Duratek 2003).

7.6 Decontaminate Surfaces

7.6.1 Decontaminate Room 540A (Source Storage)

Room B540A was a vault room used primarily for storage of radioactive sources and radioactive waste. The south wall contains 64 horizontal cylindrical vaults, each approximately 25 cm deep. Additionally, the floor of this room contains ten vertical storage pits previously used for storage of sources. These tubes extend 20 feet below the floor. The thickness of the floor is unknown. The floor, south wall and vaults showed contamination in 2001 (Duratek 2003). The wall vaults and floor pits have steel liners that will be manually decontaminated in place or completely removed. Any contamination remaining after liner removal will be scabbled or ground out of the inner concrete. Initially, the floor of room B540A will be decontaminated using non-hazardous cleaners and abrasive pads and, if necessary, the contaminated concrete will be removed by scabbling.

7.6.2 Decontaminate Fuel Storage Pits

Three emergency fuel element storage pits are located immediately adjacent to the reactor tank. The pits are vertical steel pipes 9.8 inches (25 cm) in diameter and 10 ft (305 cm) long, and are lined with an organic coating. No evidence of fixed or removable contamination was discovered in these storage pits and they are expected to be free of contamination above the Table 6.1 release criteria. After the FSS, these pits will be secured in a similar manner as the reactor tank or they may be completely removed.

7.6.3 Decontaminate Miscellaneous Surfaces

7.6.3.1 General Contamination, Room B526

The only location of removable contamination on the floor of room B526 is in the northeast corner of the room in the area identified with a radiation symbol in Figure 2.3. This area may be decontaminated with aggressive cleaning techniques using strong nonhazardous cleaners and abrasive pads or, if necessary, by removal of the floor tiles and adhesive, followed by decontamination of the underlying concrete by aggressive cleaning or scabbling.

7.6.3.2 Pneumatic Transfer System Trench

A trench for the pneumatic transfer system piping runs along the exterior of rooms B533A and B533AA and diagonally to the reactor pit. The trench surface is concrete. Measureable levels of radioactivity above 1,000 dpm/100cm^2 have been identified (Duratek, 2003, AECOM 2011). The concrete will be decontaminated as necessary to meet release criteria and may be scabbled to remove residual contamination to levels below the Table 6.3 release criteria.

7.6.3.3 Room B537

Characterization data indicates that residual contamination on the walls in Room B537 is below the surface contamination release criteria (Table 6.3). However, should additional contamination be identified above the criteria, the walls of B537 will be decontaminated with aggressive cleaning and subsequent removal of contaminated material by cutting out of the contaminated areas.

7.6.3.4 Heat and Ion Exchanger Pit (East Cooling Pit)

Characterization surveys indicated elevated residual contamination levels on the floor of the Heat and Ion Exchanger Pit (or Vault). However, the elevated surface contamination levels may be a result of radon and radon decay daughters. Regardless of the source of the elevated activity, this area will be decontaminated with aggressive cleaning followed by removal of any of the surface concrete by scabbling if necessary. The decontamination of the structures (walls and floor) will occur after the removal of the piping and components, and sealing of the piping penetrations through the east wall of the facility.

7.7 Final Status Survey

The FSS will be conducted by the DOC in accordance with an approved FSS Plan. Additional information on the FSS is provided in Section 13.

7.8 NRC Confirmatory Surveys

VA will work with the NRC to schedule confirmatory surveys and sampling following or concurrent with the FSS.

7.9 Backfill Reactor Pit

Following NRC confirmatory surveys and sampling and review of the confirmatory data, VA will backfill the reactor pit, likely by backfilling with gravel, clean fill soil, or concrete and secure the top surface likely with either a concrete cap or steel plates. As this DP covers only decontamination closure of the reactor and associated appurtenances for unrestricted use, VA considers final closure of the reactor pit outside the scope of this DP. This final reactor pit closure efforts may be accomplished outside the scope of the contract with the DOC. VA reserves the right to backfill the pit at risk, should there be delays in receiving confirmatory survey and/or sampling data from the NRC.

7.10 Schedule

Decommissioning will occur sequentially as outlined in Figure 7.1. It should be noted that the final schedule will be agreed upon by VA and the DOC. A revised schedule that contains more accurate and firm dates will be provided to the NRC when available. VA will continue to provide an updated schedule to the NRC as necessary.

8. Project Management and Organization

VA is committed to a safe decommissioning effort. VA retains ultimate responsibility for the performance of all work in accordance with the AJBRF license requirements. The decommissioning project management and organization is designed to ensure that the appropriate management control and oversight will be exercised during decommissioning activities to ensure compliance with federal, state, and local regulatory requirements as well as AJBRF license requirements.

In order to accomplish these efforts, the decommissioning team, will be composed of three main elements as provided below. The reporting relationships of these main elements are shown on Figure 8.1.

VA:

- Omaha VAMC Director – overall responsibility for VAMC Omaha operations including the reactor operations
- Omaha VAMC Industrial Safety Manager (ISM) – insures safety of non-project personnel, patients, and hospital visitors
- Omaha VAMC RSC – overall responsibility for radiation safety and adherence to the reactor license requirements
- Omaha VAMC Reactor Manager - provides day-to-day interactions with VA's technical representatives contractor and the DOC
- VA Central Office (VACO) Project Manager – provides overall guidance with respect to project planning, scheduling and execution, and interactions with VA funding source, VA contracting, and NRC Project Manager

VA Technical Representative (contracted through AECOM)

- Provide D&D Project RSO
- Technical oversight of DOC ensuring compliance with DP and safety plans
- Provide VA with technical input for communications with the NRC

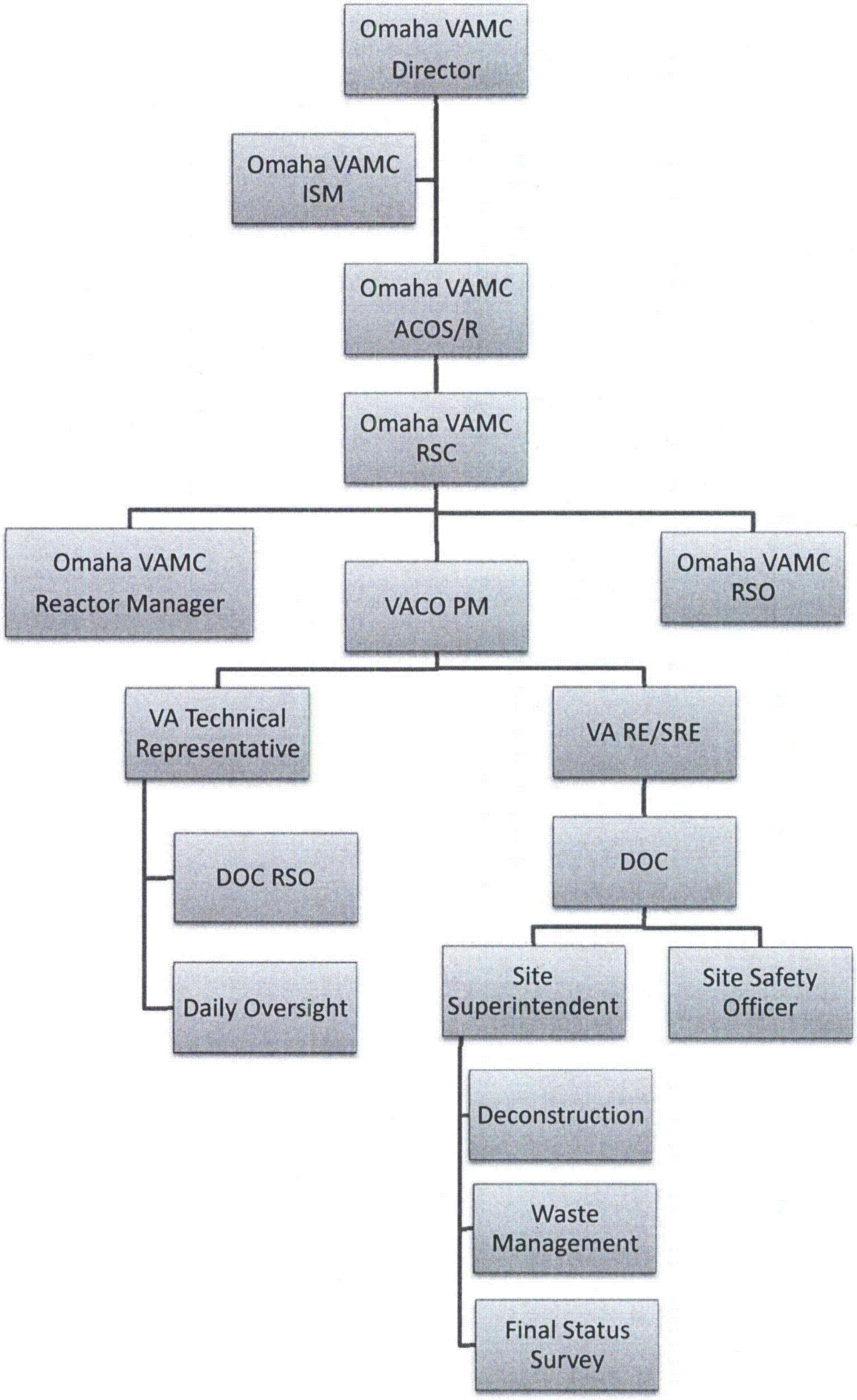
Decommissioning Operations Contractor (contractor not yet selected)

- Execute the DP in a safe and responsible manner
- Provide a DOC Site Safety Officer
- Provide VA and VA's Technical Representative with pertinent information and communication regarding schedule, progress, concerns, incidents, etc.
- Oversight of subcontractors engaged on the project
- Manage waste in accordance with applicable procedures and regulations

8.1 Omaha VA Management Organization

During decommissioning planning, the AJBRF Decommissioning Organization will be formally organized and appropriately staffed. Project governance will be defined prior to project initiation including definition of roles and responsibilities, division of authority, reporting structures, and lines of communication. In addition, project administrative procedures will be prepared and documented in manner to support staff training prior to project initiation. The following decommissioning project management and oversight organizations will be established in order to ensure that the decommissioning activities of the facility are completed safely and to the highest standards of quality and performance. The proposed staffing and lines of reporting are provided on Figure 8.1, Decommissioning Organization, below.

Figure 8.1, Decommissioning Project Organization with VAMC Oversight



8.1.1 Director, Omaha VAMC

The Director of the Omaha VAMC has ultimate responsibility for all functions of the Omaha VAMC including all licensed activity at the AJBRF. The Director shall ensure that adequate funds are available to complete the decommissioning and final radiation survey activities in an efficient and timely fashion. The Director has delegated authority for the overall management and oversight of the decommissioning activities at the AJBRF to the Omaha VAMC Associate Chief of Staff for Research (ACOS/R). This delegated responsibility is to ensure the proper level of management oversight is given to the performance of all decommissioning activities. Additionally, to further ensure proper management oversight of the decommissioning activities, the RSC shall report the results of its audits and meetings to the Director.

8.1.2 Associate Chief of Staff for Research

The Associate Chief of Staff for Research (ACOS/R) is responsible for ensuring that all decommissioning activities are conducted safely and that the control of radioactive materials and radiation exposure is in accordance with regulatory requirements and ALARA. The ACOS/R shall be responsible for the conduct of the Quality Assurance program and for the effectiveness of the Radiological and Industrial Safety programs. The ACOS/R shall have final approval authority of minor changes to the Decommissioning Plan and to procedures (that do not involve an un-reviewed safety question), and for the daily conduct of the AJBRF decommissioning. The ACOS/R shall also act as the Chairman of the RSC. The ACOS/R shall ensure that:

- Decommissioning activities comply with applicable federal, state and local regulations;
- Decommissioning activities are performed in a manner to protect the public, hospital staff, patients and the environment;
- Proper resources and qualified personnel (or organizations) are assigned adequately to safely achieve the decommissioning of the AJBRF;
- Decommissioning work is performed safely; and
- Radiological control and industrial safety program and personnel are effective and have adequate support from management.

8.1.3 Omaha VAMC Industrial Safety Manager

The Omaha VAMC ISM shall be responsible for the development, communication and implementation of a safety culture at the Omaha VAMC and among the decommissioning contractor. The ISM has the authority and responsibility to stop or suspend any activity deemed unsafe or lacking in adequate safety controls and measures. The ISM shall report to the Director on all decommissioning matters.

Specific responsibilities of the ISM include, at a minimum:

- Review of the D&D project HASP to ensure proper controls are in place during implementation of the decommissioning activities, in accordance with the requirements of the Omaha VAMC Safety Program and Federal, state, and local regulations;
- Ensure the safety of non-project VA personnel as well as facility patients and visitors;
- Investigate, document, and evaluate any injuries that may occur during the decommissioning, and develop and monitor implementation of corrective measures;

- Assist in the identification of hazardous materials within the work boundary of the AJBRF and maintain an inventory of such materials; and
- Inspect and monitor work sites and worker performance to identify any hazards and ensure procedural and safety program compliance.

8.1.4 Reactor Safeguards Committee

The RSC has broad responsibility to provide independent reviews and audits of decommissioning activities for safety and proper controls. The committee will review decommissioning procedures, decommissioning activities dealing with radioactive material and radiological controls as well as review and approve changes to the decommissioning plan. The RSC shall evaluate all changes to determine if un-reviewed safety questions exist.

The RSC shall report to the Omaha VAMC Director through the ACOS/R. The RSC can require work to be interrupted and/or suspended if audit or review findings indicate unsafe or uncontrolled conditions or operations.

The RSC shall perform the following functions:

- Review proposed decommissioning plan or procedure changes, including changes in monitoring or control equipment, systems, or testing to determine if there are safety questions as defined in 10 CFR 50.59;
- Review all new procedures and major revisions having safety significance;
- Oversee and review any operations that could potentially release radioactivity to the environment;
- Review all reports of violations of technical specifications, license, and procedures having personnel and/or public safety significance;
- Review NRC inspection reports and responses/corrective actions as necessary;
- Perform audits of plans and programs (such as ALARA, Quality Assurance [QA], and industrial safety) to ensure compliance with procedures;
- Review of the dosimetry program and radiation exposures to decommissioning workers;
- Review and approve planned releases of radioactive material to the environment;
- Review any unmonitored release or suspected unmonitored release of radioactive material to the environment; and
- Perform audits and review decommissioning tasks and operations, as the Committee deems necessary.

The AJBRF technical specifications and the RSC charter govern these functions.

The autonomous oversight of the RSC is essential for safe decontamination and decommissioning of the facility and the protection of the health and safety of the public. In accordance with the NRC License and Technical Specifications requirements, the RSC has the following standing members, at a minimum:

- ACOS/R (Chairman) for Research,
- Omaha VAMC RSO,
- and Reactor Facility Manager.

Other, ex-officio members shall be included as appropriate to provide specific expertise in the areas of decommissioning, radiological engineering, environmental controls, and/or industrial safety.

8.1.5 VA Central Office Project Manager

The VACO Project Manager (PM) is responsible for ensuring funding and that contracts are approved and awarded in a timely fashion for satisfactory completion of the AJBRF decommissioning. The VACO Project Manager serves as the Contraction Officer's overall Technical Representative (COTR).

The VACO Project Manager shall be responsible for monitoring contractor schedule adherence, schedule changes due to unforeseen issues, work scope expansion, or contractor performance. The VACO PM is an ex-officio member of the RSC.

During the D&D operations the VACO Project Manager will provide the following:

- Technical leadership for the overall D&D project;
- Act as primary overall project point of contact for the Omaha VAMC, Technical Representative, Resident Engineer(RE)/Senior Resident Engineer (RE/SRE), and DOC;
- Provide organizational leadership to accomplish decommissioning of the reactor;
- Incorporate the contractor work schedules into a master schedule for the decommissioning project;
- Provide overall communications coordination with NRC Project Manager.

8.1.6 Resident Engineer / Senior Resident Engineer

The RE/SRE shall be responsible for monitoring contractor schedule adherence, monitoring overall safety concerns to the hospital at large, and be a source of information to the RSC and the DOC. Specific responsibilities include the following:

- Act as construction site RE/SRE and Contracting Officer's Technical Representative (COTR) for the DOC, including stop/start work authority;
- Coordinate vendor work plans and logistics to minimize interference with VAMC Omaha operations and delays to work efforts;
- Ensure, with support from VA Technical Representative Contractor, that neither VA personnel nor patients or visitors to the Medical Facility are endangered by these decommissioning activities; and
- Track budget and schedule progress and prepare reports on progress, variance, and trends.

8.1.7 Reactor Facility Manager

The Reactor Facility Manager is responsible for ensuring compliance with the AJBRF license and technical specifications, including performance of necessary surveillance, testing, and inspections. The Reactor Facility Manager will review all decontamination and dismantlement plans to ensure that the reactor facility is maintained in a safe and licensed condition. The Reactor Facility Manager shall provide technical oversight during decontamination and removal of reactor components and systems.

During D&D operations, the Reactor Facility Manager will provide the following:

- Technical representation for the Omaha VAMC RSC; and

- Day-to-day interaction with VA's Technical Representative Contractor, the DOC, and various elements within VA Omaha operations.

8.1.8 Omaha VAMC Radiation Safety Officer

The Omaha VAMC RSO is responsible for ensuring the radiological safety of the patients and staff of Omaha VAMC, primarily with regards to radiation treatment and diagnosis procedures. The Omaha VAMC RSO will play no direct role in the decommissioning of the reactor. However, the Omaha VAMC RSO is a standing member of the RSC.

8.2 VA's Technical Representative Contractor

8.2.1 D&D Project Radiation Safety Officer

The D&D Project RSO shall have the responsibility and authority for the daily implementation of policies and practices regarding the safe decontamination, demolition, and disposition of radioactive material at the AJBRF. This service will be contracted through VA Technical Representative Contractor (AECOM). The D&D Project RSO, non-voting consultant to the RSC, has the authority and responsibility to suspend any activity involving radioactive material and/or work performed in radiation areas if the methods and/or procedures used are determined to be unsafe. This would include activities that are not in accordance with ALARA principles, may result in an unmonitored or unplanned release of radiation to the environment, and/or are contrary to applicable regulations. Responsibilities of the D&D Project RSO include the following:

- Verify contractor-provided radiation detection instruments used for measurement of radiation levels and contamination levels are properly maintained and calibrated;
- Maintain and implement the ALARA and personnel dosimetry program for VA and DOC project staff;
- Evaluate and distribute the results of dosimeter measurements;
- Maintain an inventory of the radioactive material possessed within the jurisdiction of the AJBRF license;
- Review radioactive material/waste shipping documents;
- Oversee the environmental monitoring programs;
- Provide or approve radiation safety training for all personnel, contractors and staff working at the facility in accordance with the Omaha VA Radiation Protection (RP) Program;
- Ensure consistency of DOC radiological protection procedures with those of the Omaha VA RP Program;
- Ensure that the activities involving potential radiological exposure are conducted in compliance with the AJBRF license, approved procedures, the decommissioning plan, and federal, state and local regulations;
- Review and approve all DOC Radiation Work Permits (RWP);
- Evaluate procedural and RWP compliance by workers and supervision;
- Prepare and maintain the documentation required by 10 CFR 20 and the AJBRF RP Program; and
- Advise the RSC about all matters regarding radiation monitoring and radiation safety during decommissioning activities.

8.2.2 Daily Oversight and Consultation

VA's Technical Representative Contractor will provide daily radiological safety oversight of the DOC to ensure that the DOC is operating within the requirements of the DP, Radiation Safety Program (RSP), and other applicable operation and safety plans and procedures. VA's on-site technical representative will be a health physicist and/or radiological engineer with more than ten years of D&D experience, including experience with decommissioning research reactors. On-site personnel will be supported by a Certified Health Physicist (CHP) with similar D&D experience. The technical representative contractor will also provide VA with interface with the NRC Project Manager and on-site NRC inspectors.

During D&D operations, VA's on-site Technical Representative will be responsible for the following:

- Ensuring the DOC and subcontractor compliance with the VA-approved QAPP;
- Initiating Nonconformance Reports (NCRs)/Corrective Action Reports (CARs) as required and ensuring corrective actions are identified and performed;
- Reviewing decontamination and dismantling plans and procedures to ensure proper levels of controls during the execution of the decommissioning activities and compliance with the requirements set forth in the facility license and applicable regulations;
- Reviewing dismantling plans to ensure proper configuration control is maintained for safety;
- Reporting to the RSC on the status of quality assurance, compliance with plans and procedures, and safety; and
- Maintaining records of quality and safety audits or inspections.

8.3 Decommissioning Operations Contractor

The DOC shall be responsible for the performance of the decommissioning in accordance with Federal and State regulations, the AJBRF Technical Specifications, and the contractual requirements of the decommissioning contract. The DOC shall provide the personnel and engage the necessary subcontractors to provide all the services and skills required for the decontamination, dismantlement, disposal, and reconstruction activities. Additionally, the DOC shall provide the radiological and radioactive waste services to support the decommissioning activities, including handling and shipment of radioactive and hazardous waste as well as performance of the decommissioning final survey.

The DOC is responsible for the development and execution of the detailed work plans, policies and procedures for the decommissioning project with ongoing management of all activities and operations. The DOC is responsible for managing the engineering progress, mobilization, decontamination, demolition, removal of materials including radioactive and hazardous waste, and the final release survey. The DOC is also responsible for preparation of the FSSP and Final Status Survey Report (FSSR) in support of the license termination applications to the NRC. Specific responsibilities include, but are not limited to the following:

- Directing and supervising D&D activities and resolving work problems;
- Coordinate, verify, and validate the proper implementation of work control and oversight programs;
- Manage D&D services and waste management services;

- Prepare radioactive waste manifests;
- Prepare and implement the FSSP; and
- Support the analysis and reporting requirements of the RSC.

8.3.1.1 Site Superintendent

The DOC site superintendent shall directly oversee the decommissioning work and will conduct daily communications with VA's oversight representative. The DOC site superintendent shall report directly to the VA CO and has the authority and responsibility to stop or suspend work of any personnel performing D&D work. The site superintendent is responsible for implementation of the DP and the FSSP and will describe the day's proposed tasks during the daily tailgate meetings.

8.3.1.2 Decontamination and Dismantlement Services

The DOC will be responsible for decontamination and dismantlement services including the execution of the Decommissioning Plan in accordance with the engineering and radiological control procedures and RWPs. Specific services shall include the decontamination, cleanup, dismantlement, removal and preparation for disposal of all materials that comprised the AJBRF. The DOC will provide the skilled labor such as decontamination and health physics technicians, and equipment either directly or via subcontractors.

8.3.1.3 Radiological and Radioactive Waste Services Contractor

Radiological and radioactive waste services shall include radiation protection personnel, instrumentation, training, and controls in support of the AJBRF decommissioning. Specific services shall also include provision of personnel, equipment, and transport services for the removal of radioactive waste from the AJBRF, and the arrangements for disposal at licensed facilities. These support services shall be conducted in accordance with Federal and State regulations, and the AJBRF license and radiation protection program.

The radiological and radioactive waste services shall also include the management of technical personnel, equipment, and procedures for the completion of the AJBRF termination survey. The Final Survey Supervisor shall report to the DOC and shall advise on the adequacy of the decontamination effort prior to the performance of the final radiation survey.

8.4 Decommissioning Task Management

Decommissioning task management will consist of two components: a project level work plan and the processes for controlling daily work activities. The development of detailed specific plans for the decontamination, demolition, radioactive and chemical/hazardous waste removal, and final survey will be the basis for bids supplied by qualified contractors to VA. These plans provide a detailed work breakdown that will allow the Omaha VAMC management and RSC personnel to understand the sequence and timing of tasks that require oversight and support.

Daily work control will be via Radiological and Hazardous Work Permits (RHWP) or RWP prepared by the DOC. In addition to controlling personnel exposure to radiation and radioactive materials, specific task details and instructions will be identified and attached to the RHWP/RWP request for that work. Work techniques will be specified to ensure that exposure rates for all personnel are maintained ALARA, and to ensure that the work techniques are suitable and effective in accomplishing the tasks specified. RHWPs/RWPs will not replace work procedures, but will be a supplement. A daily accounting for work hours, equipment usage and exposure will be provided in the RHWPs/RWPs.

The DOC site superintendent shall prepare the RHWPs/RWPs along with a detailed breakdown of planned tasks. The DOC will submit RHWPs/RWPs to the Project RSO for approval. The Project RSO will review RHWPs/RWPs for compliance with Omaha VAMC and DOC safety programs and procedures. The RSC shall review and approve all RHWPs/RWPs that have an estimated collective total exposure of 2.5 person-rem or greater.

Additionally, appropriate work procedures will be utilized for welding, cutting, rigging, decontamination, and other required tasks and will be specified on the RHWP/RWP. The DOC will acquire Hot Work Permits through the Reactor Facility Manager, as required.

8.5 Training

Unescorted individuals including employees, contractors and visitors who require access to the AJBRF Controlled Access Areas (CAA) will be trained in radiation protection and industrial hazards in accordance with the AJBRF Radiation Protection and the Omaha VAMC Safety Manual.

Health physics, industrial safety and health criteria, and other standards that guide the activities of the decommissioning plan are discussed in detail in Section 9.

Training will consist of formal training sessions conducted upon initial employment or on-site arrival for contractors. Informal training sessions will be conducted during weekly tail-gate sessions for all AJBRF contractors and staff.

8.5.1 Radiation Worker Training

Radiation protection training provides the necessary information for workers to implement sound radiation protection practices and is required for personnel working in restricted areas. Radiation protection training will be provided to all personnel granted unescorted access to the radiological controlled area (RCA). The training will ensure that decommissioning project personnel have sufficient knowledge to perform work activities in accordance with the requirements of 10 CFR.19.12 and the AJBRF Radiation Protection program. The principle objective of the training program is to ensure that personnel understand the responsibilities for minimizing exposure to radiation, required techniques for safe handling of radioactive materials, and the required techniques for minimizing exposure to radiation.

General Radiation Worker Training (RWT) will be required for decommissioning project personnel working in restricted areas and will be commensurate with the duties and responsibilities being performed. Initial or annual refresher training must be documented and approved by the D&D Project RSO. RWT will include the following:

- Fundamentals of radiation;
- Biological effects of radiation;
- External radiation exposure limits and controls;
- Internal radiation limits and controls;
- Contamination limits and controls;

- ALARA philosophy and program;
- Respiratory Protection requirements, philosophy, and program;
- Management and control of radioactive waste, including waste minimization practices;
- Response to emergencies; and
- Worker rights and responsibilities.

Additionally, participants in RWT are required to participate in the following demonstrations:

- The proper procedures for donning and removing a complete set of protective clothing (excluding respiratory protection equipment);
- The ability to read and interpret self-reading and/or electronic dosimeters;
- The proper procedures for entering and exiting a contaminated area, including use of proper frisking techniques; and
- An understanding of the use of RHWP by working within the requirements of a given RHWP.

The following are examples of the training programs applicable to decommissioning activities:

- Safety controls, including: personnel monitoring, radiation surveillance and monitoring of controlled areas, ventilation, posting of hazardous and/or radiological areas, access controls and health physics administrative controls;
- Training on the principles and techniques of D&D activities;
- Proper use of personal monitoring devices; and
- Job-specific training, including mockup simulation or pre-performance briefings to ensure proper equipment usage and achievement of ALARA.

Personnel who have documented equivalent RWT from another site may be exempted from participating in training, except for site-specific training on administrative limits and emergency response. However, such personnel will be required to pass a written examination and attend demonstration exercises.

Records of training will be maintained by the Reactor Facility Manger. Training records will include each trainee's name, dates of training, type of training, test results, authorization for protective equipment use and instructor's name(s).

8.5.2 Respiratory Protection Training

Personnel whose work assignments require the use of respiratory protection devices will receive respiratory protection training in the devices and techniques that will be required for use. The training program will follow the requirements of 10 CFR 20 and will consist of a lecture session and a simulated work session. Personnel who have documented equivalent respiratory protection training may be exempted from this training.

8.5.3 General Site and Safety Training

The primary goal of the DOC's safety program is to ensure compliance with the Occupational Safety and Health Administration (OSHA) requirements and the Omaha VAMC safety requirements during implementation of the DP. The Omaha VAMC ISM will review the DOC's safety plans and may conduct periodic safety inspections. The DOC site supervisor is responsible for ensuring that the decommissioning activities are conducted in accordance with occupational health and safety requirements for both project personnel and the general public as defined in the approved safety plans. VA's Technical Representative Contractor has the role of verifying training and documentation.

All personnel working on the decommissioning project will have appropriate health and safety training. The training will provide personnel with the means to recognize and understand the potential risks involved with personnel health and safety issues associated with the work at the facility. Additionally, the training will help to ensure compliance with the requirements of the NRC (10 CFR), the Environmental Protection Agency (EPA) (40 CFR 24.311), and OSHA (29 CFR). Workers and regular visitors will be familiarized with plans, procedures, and operation of equipment required for safe performance of their work. In addition, each worker must be familiar with procedures that provide for good quality control. Specific responsibilities of each worker include conducting an industrial training program to instruct employees in the following:

- Training on this DP;
- General safe work practices;
- Communication of radiological, chemical, and demolition-induced hazards;
- Worker rights and responsibilities;
- Use of personnel protection equipment;
- Hearing conservation training; and
- Fall protection.

Daily tailgate sessions will include training on procedural changes, deficiencies discovered during audits and surveillances, and applicable industry events or occurrences. This training will be documented by retention of the sign in/attendance sheets and the agenda for the particular tailgate meeting. Also, tailgate meetings shall be held in response to any violation of safety or radiation protection procedure/policy violations, lost-time, injuries, or work procedure non-compliances.

9. Radiation Safety and Health Program

9.1 Occupational Dose During Decommissioning

As previously discussed, the reactor facility and associated outside areas of the facility are small in nature and do not contain wide-spread nor significant levels of residual radioactivity. Accordingly, the site is considered to have low residual concentrations of radioactivity present in small areas of the facility and not wide-spread throughout the facility.

An estimate of the occupational dose to the DOC staff during decommissioning operations less than 1 person-rem for each task is provided in Table 9.1. The dose estimate was prepared using the individual work activity durations and work crew sizes, based upon the results of the characterization results. Dose estimates were also determined using professional experience involving similar work, task durations, projected man-loading, and schedule and are considered conservative for many tasks where dose rates are assumed to be 0.1 and 0.05 mrem/hr.

Table 9.1, Occupational Dose Estimate

Task	Task Name	Time (hours)	Number of People	Average Dose Rate (mrem/hr)	Collective Dose (person-rem)
1	Project Planning (off-site)	NA	NA	0	0
2	Site Mobilization and Training	24	6	0.01	0.001
3	Facility Preparation	16	6	0.05	0.005
Reactor D&D (est. 4 weeks)					
4A	Remove Control Rod Drives and Bridge	24	6	0.5	0.072
4B	Remove Fuel Storage Rack and Interference	16	6	1	0.096
4C	Remove Reactor Core Components and Lazy Susan	1	4	11	0.044
4D	Drain Pool and Process Water	20	2	0.8	0.032
4E	Remove Gunite, Liner, Tank, Soil	120	6	0.8	0.576
Remove Ancillary Systems (est. 2 weeks)					
5A	Remove Water Filter System	40	3	0.1	0.012
5B	Remove Water Cooling System	40	3	0.1	0.012
5C	Remove pneumatic transfer system	40	3	0.1	0.012
5D	Remove lab vent hoods and drain lines	40	3	0.1	0.012
Decontaminate Surfaces (est. 1 week)					
6A	Decontaminate Room 540A (Source Storage)	20	3	0.05	0.003
6B	Decontaminate Fuel Storage Pits	20	3	0.05	0.003
6C	Decontaminate Miscellaneous Surfaces	20	3	0.05	0.003
7	Perform FSS	40	3	0.05	0.006
8&9	Confirm and backfill	NA	NA	NA	NA
				Total	**0.889**

9.2 Radiation Safety Program

9.2.1 Management Policies

Decommissioning activities at the AJBRF involving the use and handling of radioactive materials will be conducted in a controlled manner in order to minimize all exposures to radiation. As described in the AJBRF Radiation Protection Program, the Omaha VAMC is committed to a RSP that controls internal and external radiation dose to workers and members of the public in a manner that avoids unnecessary and accidental doses, and that maintains effluents and doses to workers below regulatory limits. The DOC will prepare a Radiation Protection Plan (RPP) and a HASP for VA review and approval. The implementing safety procedures for the D&D project will be provided in the approved DOC documents.

9.2.2 ALARA Policy

The AJBRF D&D activities will be conducted in such a manner as to minimize exposures to workers and the public ALARA. The basic philosophy of radiation protection is to achieve radiation exposures and effluent releases below applicable regulatory limits. All decommissioning personnel are expected to be knowledgeable of work activities and to abide by all ALARA requirements documented in work instructions. In addition, each individual is responsible for minimizing their own dose as well as and dose to others.

9.2.3 Respiratory Protection Program

Respirators are to be used to control personnel exposure to airborne radioactive materials when administrative and engineering controls are not effective and the use of respirators result in Total Effective Dose Equivalent (TEDE) being ALARA. The DOC shall implement a respiratory protection program in compliance with the requirements of 10 CFR 20, Subpart H.

9.2.3.1 Workplace Air Sampling

The DOC shall describe respiratory protection requirements in a written Respiratory Protection Program in support of the RPP. Respiratory protection equipment shall be selected using allowed respiratory protection factors to ensure that individual limits on intake or exposure are not exceeded. The DOC's respiratory protection program shall be based on requirements in 10 CFR 20.1103 for radiation protection and 29 CFR 1910.134 for non-radiological hazards and comply with the guidelines in the NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection" and NUREG 0041, "Manual of Respiratory Protection Against Airborne Radioactive Material."

9.2.3.2 Breathing Zone Air Sampling

Localized air samples including breathing zone air samples shall be collected where it is expected that the concentrations of airborne radioactivity are likely to exceed the criteria of 10 CFR 20.1502(b). Alarm set points for CAMs will be set at 10 percent of the derived air concentration (DAC). If monitors or grab samples show airborne contamination greater than 10 percent of the DAC, the work will be suspended and radiation protection personnel will be notified. Eu-152 has the most-limiting inhalation DAC at 1E-8 microcuries per milliliter (μCi/mL) of all AJBRF radionuclides of concern.

9.2.3.3 Respiratory Protection Selection Considerations

Respiratory protection equipment will normally be selected that has a protection factor greater than the anticipated peak airborne concentration expressed as a multiple of total DAC. Respiratory protection equipment may be selected that has a protection factor less than the anticipated peak airborne concentration expressed as a multiple of total DAC provided that use of that equipment is expected to result in a lower TEDE. This evaluation shall be documented in the RPP or the RWP. Protection factors for respiratory protection equipment shall be assigned in accordance with Appendix A of 10 CFR 20.1103

9.2.3.4 Medical and Training Requirements for Use of Respiratory Protection Equipment

Personnel who require the use of respiratory protection equipment shall receive a physical examination, and be certified by a physician as qualified to wear respiratory protective equipment prior to wearing any respiratory protection device. Personnel shall be medically evaluated to ensure that they possess the physical and psychological capabilities necessary to perform tasks while wearing a respirator. This medical evaluation shall use 29 CFR 1910.134 and NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection" as guidance in determining if an individual is medically qualified to wear respiratory protection equipment.

9.2.3.5 Maintenance and Storage of Respiratory Equipment

Personnel responsible for the cleaning and maintenance of respiratory protection equipment shall receive documented training necessary for fulfilling their responsibilities.

9.2.4 Internal Exposure Determination

Internal exposure monitoring will be conducted in accordance with approved procedures. 10 CFR 20.1204 (a) states, "For purposes of assessing dose used to determine compliance with occupational dose equivalent limits, the licensee shall, when *required* under 10 CFR 20.1502, take suitable and timely measurements of (1) concentrations of radioactive materials in air in work areas; or (2) quantities of radionuclides in the body; or (3) quantities of radionuclides excreted from the body; or (4) combinations of these measurements." The DOC RPP shall include air sampling and *in vitro* bioassay and/or *in vivo* bioassay monitoring in order to comply with this requirement.

Based on recommendations of the National Council on Radiation Protection and Measurements (NCRP) and on regulatory requirements, controls shall be established for the protection of the embryo/fetus during a female worker's pregnancy. These controls shall ensure compliance with regulatory requirements and protect the rights of the female worker.

Declaration of pregnancy is entirely at the discretion of the female project staff member and does not require medical proof. To declare pregnancy, the woman shall inform the RSO, in writing, of the pregnancy and an estimated date of conception so that the estimated dose to the embryo/fetus prior to declaration can be determined. A declared pregnant worker shall not be permitted to enter airborne radioactivity areas nor be assigned to tasks that could lead to internal radionuclide intakes.

9.2.5 Medical Radionuclide Intake

Occupational exposure does not include exposure due to medical administration of radionuclides. Therefore, individuals shall be required to inform the RSO before receiving medical treatments involving radionuclides. After being informed of a medical intake, documentation shall be obtained, signed by the individual stating the date of treatment, radionuclide used, amount of intake, and medical procedure. The RSO shall perform an assessment to determine what work restrictions may be necessary until the medical radionuclides have cleared to avoid problems with frisking/portal monitors, exposure to co-workers, or exposure to external dosimeters.

9.2.6 External Exposure Determination

10 CFR 20 establishes a TEDE limit and a total organ dose equivalent (TODE) limit for occupationally exposed individuals. Monitoring of an individual's external radiation dose is required by the regulations if the external occupational dose is likely to exceed 10 percent of any dose limit appropriate for the individual. External radiation monitoring is also required by 10 CFR 20 for any individual entering a high or very high radiation area.

The DOC shall provide appropriate monitoring for the accurate determination of occupational radiation exposure to their staff that enter and/or work in the restricted area. VA will be responsible for monitoring its oversight personnel. Official determination of external exposures shall normally be made using primary dosimeter results except as otherwise noted.

External radiation dose monitoring may be accomplished through the use of a combination of Optically Stimulated Luminescence Dosimeters (OSLs). The official record of external dose to beta, gamma, and neutron radiation shall be obtained from primary dosimetry (e.g., OSLs). Secondary dosimetry (e.g., self-reading pocket dosimeters and alarming dosimeters) can be used to track dose between OSL-processing periods and may also be used as a backup to the OSL. The OSLs or other primary dosimetry devices shall be capable of measuring the deep dose equivalent (DDE) at a tissue depth of one centimeter, measuring the lens dose equivalent (LDE) at a tissue depth of 0.3 cm, and measuring the skin dose equivalent at a tissue depth of 0.007 cm. All dose records for project personnel will be provided to VA for record retention.

9.2.6.1 Exposure Limits

To provide assurance that individuals do not exceed the federal limits specified in 10 CFR 20, administrative dose control levels will be established for the AJBRF that are less than the federal limits. When an individual approaches or has reached his/her administrative dose control level, that individual's access to radiation areas shall be restricted to minimize further occupational dose to ensure that the administrative dose control level and the federal limit are not exceeded.

9.2.6.2 Dose Assessment

Dose assessments shall be reviewed and approved by the RSO prior to assigning a dose other than that measured by a primary dosimetry device.

9.2.7 Summation of Internal and External Doses

Internal and external doses shall be summed whenever positive doses are measured. The dose to the lens of the eye, skin, and extremities are not included in the summation.

9.2.8 Contamination Control

Radioactive material and contamination control measures shall be established to prevent the spread of contamination to clean areas, minimize the need for respiratory protection devices, and maintain personnel internal and external exposures ALARA. The primary means of preventing the spread of contamination is to contain it at its source and to minimize the number of contaminated areas and the amount of loose surface contamination in those areas. Minimization of the potential for spread of contamination will be accomplished by instituting good work practices.

9.2.8.1 Surveys

Routine contamination surveys shall be conducted at frequencies and locations dependent upon the work activity performed. At a minimum, contamination surveys of AJBRF shall be performed on a weekly basis to monitor the potential spread of contamination and to identify areas that warrant decontamination. Decontamination will be performed either as "good housekeeping" or due to an unusually high level of beta/gamma contamination or any detectable alpha contamination.

Non-routine contamination surveys are conducted as deemed necessary by the RSO to detect the presence of or prevent the spread of contamination and, as necessary, to prepare RHWP/RWPs and monitor associated work. These surveys shall be used to ensure proper contamination control and respiratory protective measures are being applied. Enhanced personnel contamination surveys will be required (e.g., nasal and face smears, sampling of hair) when the work performed results in unexpectedly high levels of loose or airborne contamination.

9.2.8.2 Contamination Action Levels

It is the policy of the Omaha VAMC management that detectable contamination on personnel is maintained ALARA. The individual site conditions and isotopes of concern will dictate the monitoring requirements. The contamination limits shall be specified in the DOC RPP.

Areas shall be identified and controlled as contaminated when removable contamination levels exceed 1,000 dpm/100 cm^2 of beta/gamma -emitting radionuclides or 20 dpm/100 cm^2 of alpha-emitting radionuclides. Equipment, materials, and tools shall be controlled when total contamination exceeds 100 counts per minute above background using a detector at least as sensitive as a pancake GM detector or removable contamination exceeds 1000 dpm/100 cm^2 of beta/gamma-emitting radionuclides or 20 dpm/100 cm^2 of alpha-emitting radionuclides. Internal surfaces that have been exposed to radioactive contamination shall also meet these limits. Final release of materials will be in accordance the criteria discussed in Section 6.2.

9.2.9 Radiation Monitoring Instrumentation Program

Various types of radiological and hazardous measurement instrumentation will be used at the AJBRF for radiation protection and survey and monitoring purposes. The DOC's RPP or associated standard operating procedures will include procedures for inventory, issuance and control, calibration, operation, response testing, maintenance, repair, and quality control of radiation protection and hazardous instrumentation and equipment. The procedures will ensure compliance with applicable federal regulations and Omaha VAMC policies.

9.3 Nuclear Criticality Safety

All fissile material (i.e., fuel and fission chambers) have been removed from the AJBRF and transferred to the USGS TRIGA facility. There is no fissile material at the AJBRF and thus no nuclear criticality issues of concern.

9.4 Health Physics Audits, Inspections, and Record Keeping

Records related to health and safety of radiation workers and individual members of the public shall be maintained by the AJBRF RSO as required by procedures and reactor license technical specifications. Record-keeping is important in demonstrating regulatory compliance issues as well as providing means to resolve potential liability issues that may arise. All records shall be prepared and maintained in accordance with the AJBRF Radiation Protection Program and implementation procedures.

9.5 General Industrial Safety Program

The DOC is responsible for ensuring that the decommissioning process meets the Omaha VAMC occupational health and safety requirements applicable to project personnel and the general public. The primary function of the DOC's HASP is to ensure that all applicable OSHA requirements are implemented and that full compliance is achieved. The HASP will be reviewed and approved by the ISM and the RSC.

10. Environmental Monitoring and Control Program

Decommissioning activities at the AJBRF will be conducted in a controlled manner to minimize both public and occupational radiation exposures. Based on the site characterization surveys, most structures and surface areas within the AJBRF are not radioactively contaminated. Also, based on the characterization surveys, areas outside the AJBRF are not contaminated above natural background levels. The survey results indicated that the reactor components and some reactor and ancillary systems have levels of radioactivity that will need to be decontaminated or removed as radioactive waste. The removal of these components and materials will be carried out in such a manner that the spread of contamination will be contained and minimized and thereby reduce the chances of elevated releases to the environment.

Environmental monitoring will consist of routinely measuring the quantity of direct radiation and radioactive material releases outside of the AJBRF. The monitoring results will be used to establish the radiological conditions as a function of time, provide for a permanent record of these conditions, and permit evaluation of radiological trends over time. Environmental monitoring will be instituted prior to commencement of decommissioning activities. The environmental sampling and measurement results obtained during the site characterization survey will serve as the baseline data benchmark for comparison with in-progress and final survey results following completion of the decommissioning efforts.

10.1 Environmental ALARA Radiation Exposures Evaluation Program

10.1.1 Management ALARA Commitment

As described in the AJBRF Radiation Protection Program, the Omaha VAMC management is committed to maintaining occupational and public exposures, as well as effluent discharges ALARA. This will be accomplished through implementation of a sound radiation protection and environmental monitoring program. This program will include the use of engineering controls and ALARA principles commensurate with the radiation and contamination levels associated with activities performed during the D&D.

10.1.2 ALARA Policy

The Omaha VAMC shall maintain a zero release policy for liquids and particulate airborne effluents associated with the AJBRF. Specifically, Omaha VAMC management's position is that no hazardous or radioactive liquids will be further discharged by way of sinks and drains or other means of discharge to the environment during the decommissioning. This philosophy is undertaken to ensure exposures during decommissioning efforts and exposures to facility personnel and members of the public will be minimized.

10.2 Monitoring Program

10.2.1 Direct Radiation Monitoring

Direct radiation levels in the AJBRF will be monitored by VA with Thermoluminescent Dosimeters or OSLs continuously positioned at a minimum of five pre-determined sample locations in the environment outside the AJBRF. Four of the locations will be the following:

- the hall outside Room B526,
- the area outside the reactor cooling system vault,
- the location of radioactive waste loading,
- the optometry clinic located directly above Room B526.

A fifth dosimeter will record background and will be placed in a location at least 100 m from the closest point to the AJBRF, but not along a transportation route. Dose rate in millirem per hour will be computed from the time of installation to the time of removal. The dosimeters will be processed according to the current area monitoring program schedule implemented by the Reactor Facility Manger.

10.2.2 Airborne Monitoring

Laboratory hoods located in the AJBRF that were previously used in analyzing samples removed from the reactor will be disabled, as the site characterization survey showed that the ventilation system supporting the operation of these hoods were free of radioactive contamination (AECOM 2011). Continued operation of these hoods during D&D activities would potentially contaminate the exhaust system. Therefore, the ventilation of these hoods will be discontinued at the start of D&D operations.

The primary method of removing exhaust air from the areas of the AJBRF is through an exhaust fan that vents to the cooling vault on the east wall of the reactor facility. Additionally, the two laboratory hoods exhaust continuously through dedicated ductwork that runs through the Omaha VAMC facility and exhausts at the main building's 12th floor roof. During reactor operations the primary concern for airborne contamination control purposes was related to short-lived gaseous emitters. During D&D operations a temporary HEPA filtration system, with the exhaust capacity required by AJBRF Technical Specifications, will be installed and operated in place of the building ventilation system and discharge to the ground level outside the AJBRF.

Monitoring the air handling discharge point will be accomplished by installing a portable air monitor at this location for purposes of collecting routine and special air samples from the ventilation exhaust point. The instrumentation to be used for this monitoring will have the capabilities of a CAM for real-time monitoring as well as capabilities to collect samples for laboratory analyses.

Additionally, local engineering controls, such as containments and HEPA filters, will be temporarily installed to minimize the release of airborne radioactivity to general areas within the AJBRF. These controls will likely be used during removal of the gunite and concrete in the reactor tank/pit. Local ventilation may also be used in the event that additional concrete outside of the reactor pit requires decontamination by scabbling.

Airborne radioactivity action levels will be established at an administrative control level of 50 percent of the applicable criteria for Eu-152 in Table 2 of 10 CFR 20, Appendix B or 1.5E-11 μCi/ml. The Eu-152 value is used because it and Co-60 are the most likely contaminates in the activated concrete but the release limit for Eu-152 is lower than the limit for Co-60.

10.2.3 Liquid Effluent Monitoring

As described in Section 7.4.4, the water from the reactor tank is anticipated to be discharged from the AJBRF via the sanitary sewer following treatment through the existing demineralizer system. Prior to discharge, a water sample will be collected and analyzed to verify that the tank water is free of radioactive contaminants of concern greater than 50 percent of the applicable isotope-specific criteria contained in Table 3 of 10 CFR 20, Appendix B.

10.3 Release Control Program

10.3.1 Corrective Actions

Administrative controls will require immediate notification to the AJBRF License RSO and the Omaha VAMC ACOS/R for any airborne or liquid effluent levels that exceed the action levels set forth above. The AJBRF License RSO will initiate or perform an investigation and response consisting of one or more of the following actions:

- Verify applicable laboratory data and supporting calculations;
- Analyze and review probable causes;
- Determine the need for re-analysis or additional analyses on original sample;
- Evaluate the need for re-sampling;
- Evaluate the need for sampling of other pathways;
- Determine the need for notifications and reporting as required by internal procedures, technical specifications and federal requirements;
- Document all actions, analyses and evaluations in applicable logs or files;
- Perform applicable exposure assessments; and
- Develop corrective/preventive action recommendations to prevent future releases.

10.3.2 Quality Assurance

Administrative controls will be established to ensure that samples collected from the discharge areas are representative. Duplicate and blank samples will be prepared for tank water analysis. Air samples analyzed on-site will be periodically recounted at a frequency of 20 percent. Procedures will be developed to ensure sample integrity will be maintained from the time of collection to the time of analysis.

11. Waste Management Program

11.1 General Waste Management Program

11.1.1 Waste Management Statement

The Omaha VAMC is committed to the safe D&D of the AJBRF. The primary objective of the Omaha VAMC's waste management program is to protect workers, visitors and the environment from potential effects of various radiological and hazardous waste streams. The Omaha VAMC is committed to strict compliance with all federal and state radioactive and hazardous waste handling and disposal requirements.

11.1.2 Waste Management Philosophy

A Waste Management Plan addressing radioactive, hazardous, and mixed-waste streams will be formally established and implemented by the DOC during the entire decontamination and decommissioning process. The Waste Management Plan will be approved by the RSC.

Programs for the management of waste and pollution prevention will be defined prior to beginning D&D. The program will be implemented by written administrative and technical implementation procedures to meet the requirements of the AJBRF license Technical Specifications and applicable Federal and State regulations.

11.2 Radioactive Waste Management

The AJBRF D&D processes will require the handling and processing of volumes of radioactive waste typical of a research reactor of this type to reduce residual radioactivity to a level permitting release of the facility/site for unrestricted use. Equipment and materials that cannot be decontaminated to the level below the radioactive release criteria will be processed as radioactive waste.

The DOC will ensure that appropriate processing, packaging, and monitoring of solid and liquid wastes generated during the decommissioning process are performed in accordance with formally approved administrative and technical implementation procedures. These programs and procedures will be maintained and controlled in compliance with the AJBRF technical specification requirements and this plan.

Based on recent experience at the University of Arizona, all radioactive waste generated during D&D is expected to meet the definition of Class A waste as defined in 10 CFR 61. Disposal of Class A waste may occur at Energy Solutions in Clive, Utah. However Waste Control Specialist in Andrews, TX can accept Class A waste and under certain circumstances can import these wastes from out-of-compact states. It is the DOCs responsibility to determine cost effectiveness of the waste disposal options while maintaining compliance with all applicable regulatory requirements.

11.2.1 Solid Radioactive Waste Management

Solid radioactive waste handling at the AJBRF will be divided into three phases: packaging, on-site material staging, and shipment. Each of these phases will be conducted in strict adherence with the AJBRF Technical Specifications requirements, Waste Management Plan, and applicable federal, state, and disposal site requirements.

Solid radioactive waste generated during the decommissioning of the AJBRF will be primarily comprised of low-level radioactive waste. The solid waste will primarily consist of activated aluminum/graphite from the reactor internals, concrete and structural materials from walls, exhaust hoods, steel and concrete/epoxy/gunite from the reactor tank, steel/rust from laboratory drains, and a small volume of mixed waste consisting of contaminated lead paint and contaminated asbestos-containing floor tiles. The various components, activity and estimated volumes that may be generated as a result of the D&D operations (exclusive of void space in waste containers) are listed in Table 11.1.

Table 11.1, AJBRF Waste Disposition

Area Description		Material/Surface Type	Decontamination Technique	LLRW Disposal Estimate (m^3)
Structures				
B526 Reactor Console Area	Floor	Tiled Concrete/Glass Covers	Mechanical/Rad. Trash	0.37
B526 Reactor Console Area	Reactor Floor Trenches	Concrete	Scabble (0.5 cm)	0.05
B526 Reactor Console Area	East Cooling Pit	Concrete	Scabble (0.5 cm)	0.01
B537 Lab Area with Hood	Hood	Steel and Plastic	Dispose of Entire Hood	3.00
B540A Source Room	Floor	Concrete	Scabble (0.5 cm)	0.30
B540A Source Room	Horizontal Storage Vaults	Metal	Mechanical/ Rad. Trash	0.82
B540A Source Room	Vertical Storage Vaults	Metal	Mechanical/Rad. Trash	0.35
B540 Lab Area with Hood	Hood	Steel & Plastic	Dispose of Entire Hood	3.00
Systems				
	Pneumatic Transfer System	Steel	Mechanical	0.39
	Reactor Coolant System	Steel	Mechanical	3.00
	Ion Exchanger	Steel	Remove resin, dispose	1.7
Reactor Components				
	Water From Tank	Water	Filter and discharge to sewer	0
	Five Micron Filters	Metal Filter Media	Mechanical	0.1
	Gunite and Concrete I/S and O/S Tanks and Rebar	Concrete, Gunite, Epoxy	Mechanical	12.5
	Reactor Tank	Steel	Mechanical	0.27
	Core Components	Metal	Mechanical	0.75
	Bridge	Metal	Mechanical	0.5
Miscellaneous				
	PPE & Tools	Various	Dispose	4.0
	Contaminated Lead	Lead	Dispose	0.2
			Total Waste Volume	31.31

m^3 - cubic meter

11.2.2 Solid Radioactive Waste Processing

Radioactive waste processing at the AJBRF will be performed in controlled areas that both minimize the radiation exposure to personnel and the movement of contaminated equipment and materials. These areas will be controlled and monitored to minimize worker exposure and the spread of contamination to the extent practicable.

All radioactive waste is expected to meet the definition of Class A waste as defined in 10 CFR 61. Waste packages and packaging will meet the applicable requirements of the U.S. DOT (DOT; 49 CFR 173) and the NRC (10 CFR 20 and 10 CFR 71), as well as the selected disposal facility's criteria for transportation and disposal for each decommissioning waste stream.

Instructions for determining the 10 CFR 61 waste classification of radioactive waste and for determining the radionuclide content of a container through a combination of direct measurements, radiation shielding calculations, and the use of appropriate scaling factors will be provided in the Waste Management Plan.

11.2.3 Solid Radioactive Waste Storage Awaiting Shipment

Solid radioactive waste awaiting shipment will be stored in posted controlled areas and away from personnel traffic or work areas. Storage will be in accordance with the requirements of 10 CFR 20. Periodic inspections of these designated areas and those containers stored within the controlled area boundaries will be performed to ensure that package integrity is maintained.

Large packages awaiting shipment may be stored in designated areas outside the actual reactor facility prior to shipment. Appropriate security controls and radiological controls will be implemented, as necessary, to prevent unauthorized access to the area and the radioactive materials stored inside. Additionally, precautionary measures will be implemented to ensure that the components are adequately protected from on-site hazards (e.g., extreme weather conditions and heavy load movement).

11.2.4 Solid Radioactive Waste Shipment

Solid radioactive waste will be shipped by a certified radioactive waste broker in accordance with formally established implementation procedures. These procedures will be implemented to comply with applicable federal, state, and licensed waste processor/disposal site requirements. Prior to each shipment of a radioactive material package, the quality control requirements of 49 CFR 173.475 will be applied.

Vendors may be utilized, as necessary, for radioactive waste processing during the various stages of the decommissioning process. For example, vendors may be needed to manage the Po-Be source or provide survey/characterization services for bulk wastes.

11.2.5 Estimated Volumes of Solid Radioactive Waste

Based upon the results of the site characterization surveys (Duratek, 2003, AECOM 2011a), preliminary radioactive waste estimates have been determined. The total radioactive waste estimate is approximately 41.8 cubic yards (32 cubic meters). A breakdown of the waste volume estimate is provided in Table 11.1.

11.2.6 Liquid Radioactive Waste Processing

The vast majority of the liquid on-site is within the reactor tank. This water will be filtered and discharged as previously described in Section 7.5. However, there will likely be residual water in the water filter/cooling systems than cannot be discharged because it does not meet the discharge limits. This residual liquid waste will be stabilized and solidified and shipped as solid waste. Management of this residual liquid waste will be described in the Waste Management Plan.

11.2.7 Mixed Low-Level Radioactive/Hazardous (Mixed Waste)

The handling and disposal of hazardous materials and hazardous wastes will be controlled through the Omaha VAMC hazardous waste management program and described in the Waste Management Plan.

Samples were taken during the 2001 site characterization (Duratek 2003) for lead-based paint and asbestos. Asbestos was found in the lab counters, fume hood panels, floor tiles, pipe insulation, and in the floor mastic adhesive. Paint samples taken from three locations had a measurable lead content: beige paint from dumbwaiter frame, beige paint from the south stairwell doorframe, and beige paint from the southeast corner column.

Known mixed waste includes contaminated lead. Potential mixed waste includes contaminated lead-based paint applied to the reactor walls during the construction of the AJBRF and contaminated asbestos-containing floor tiles in Rooms B540A and B526. The lead and asbestos containing materials will be classified and disposed of at an EPA/NRC authorized facility by a certified waste broker, if appropriate.

No hazardous chemicals or other hazardous substances are anticipated to be used during decommissioning operations that could generate a mixed waste.

11.2.8 Radioactive Waste Minimization

Radioactive waste minimization will be practiced at the AJBRF throughout the decommissioning process. This effort will be achieved by management focus, worker training, and written procedures to ensure volume reduction is considered in tasks. Tools used in the AJBRF will be controlled and re-used to minimize the number of contaminated tools and all equipment and materials provided by outside vendors shall be required to be designed to facilitate decontamination. An aggressive program of minimizing packaging and other unnecessary materials inside the controlled areas of the AJBRF will be implemented.

Packaging of radioactive waste during the dismantlement process will be performed in accordance with the Waste Management Plan. A primary objective of packing methods is to ensure that voids in packages are minimized. Void minimization helps to ensure that the radioactive waste volume generated has been reduced to the maximum extent practicable. Such volume reduction minimizes the project costs, disposal, and transportation risk.

11.3 Non-Radioactive Hazardous Waste Management

When operational, the reactor was classified as a Conditionally Exempt hazardous waste generator. Routinely generated hazardous waste included lighting waste and chemicals. Non-hazardous waste generated during routine operations included items such as lubricating oils and trash. A general cleanup at the AJBRF has resulted in the removal and disposal of all non-radioactive chemicals and other potentially hazardous waste (Duratek 2003). The 2002 characterization generated a substantial volume of equipment and materials from the AJBRF for either disposal as clean waste or reuse. Additionally, almost 3 tons of lead was surveyed and verified free of residual contamination and transferred to a lead recycler (AECOM 2011a).

11.3.1 Hazard Communication

The OSHA Hazard Communication (29 CFR 1910.1200) requires the Omaha VAMC to provide information to its employees and contractors concerning the hazardous substances to which they may be exposed. The Omaha VAMC's Right to Know Program was implemented to meet these requirements. Site-specific safety training will apprise contractors of the types and identification of hazardous materials used at the AJBRF.

During the dismantlement phase, a waste handling storage area will be established to facilitate the inventory, collection, categorization, and disposition of waste materials in containers. Adequate area for containment and segregation of potentially incompatible or reactive wastes will be provided, and will minimize the potential for environmental release.

Chemicals used during decommissioning will be evaluated for hazardous constituents or properties using Resource Conservation and Recovery Act (RCRA) criteria and the chemicals' Material Safety Data Sheets (MSDS). Decontamination agents expected to be used during decommissioning activities include detergents and solvents. Detergents and water-based solvents that would generate a non-hazardous waste will generally be used for cleaning.

All chemicals and materials used during the decommissioning will be subjected to a chemical control review by the Omaha VAMC Industrial Safety Manager to determine if a non-hazardous or a less toxic chemical may be substituted to prevent the generation of hazardous or mixed wastes. In the event that hazardous chemicals or materials must be used, waste minimization techniques will be applied during usage. Steps will be taken to ensure that if a potentially hazardous material must be used, controls are in place to ensure these materials are not inadvertently contaminated with radioactivity. If any hazardous material becomes radioactively contaminated, it will be considered a mixed waste, and subject to applicable NRC, EPA,, and state regulations. Any such mixed waste generated will be managed according to Subtitle C of RCRA, and applicable state and local permitting and operating requirements.

11.3.2 Specific Waste Management

The requirements for the disposition of equipment and materials will depend on whether the material or equipment is categorized as a product or waste. The categorization of the waste will be based on: (1) Federal RCRA, (2) State of Nebraska Title 128 Requirements, (3) its physical and chemical properties, and (4) the disposition of the material.

Based on waste stream inventory data, wastes that have the potential to be hazardous will be classified through MSDS information or analysis. A sampling, analysis and composite method will be utilized to properly classify and group similar wastes. Following classification, appropriate regulatory waste disposition options will be evaluated and selected. Selection of disposition methods, in order of priority, will focus on: (1) reuse/recycle, (2) on-site elementary neutralization of acids or bases, and (3) off-site treatment.

11.3.3 Mercury

Instruments and switches which contain mercury will be consolidated and either drained or the entire assembly shipped off-site to a licensed mercury reclamation or disposal facility. Mercury drained from equipment will be reclaimed or processed by an authorized contractor.

11.3.4 Lead-Based Paints

Historically, lead-based paints may have been used to coat steel components, concrete structures, and the reactor liner. During the operating life of the reactor facility, some lead-based paints may have been covered with several coats or types of paint. Paint samples taken from three locations in 2001 showed measurable lead content: beige paint from dumbwaiter frame, beige paint from the south stairwell doorframe, and beige paint from the southeast corner column (Duratek 2003).

11.3.5 Lead

Lead from various locations of the reactor facility, mostly used for shielding in the form of casks, pigs, and bricks that met site release criteria have been sent to a lead recycler (AECOM 2011). Remaining lead items that contain measureable levels of contamination have been placed in one 55-gallon drum and stored in Room B540A. Lead containing material that cannot meet the radioactive release criteria will be characterized and disposed of at a licensed facility.

11.3.6 Insulation/AsbestosBearing Type Materials

Asbestos-type material may exist in floor tiles both in radioactively and non-radioactively contaminated areas of the reactor facility. In 2011, asbestos was identified in the lab counters, fume hood panels, floor tiles, pipe insulation, and in the floor mastic adhesive (Duratek 2003). The presence of asbestos will be confirmed in suspect materials removed for disposal.

Radioactively contaminated asbestos materials will be segregated and managed in accordance with the selected disposal facility's waste acceptance criteria. Asbestos will be packaged for shipment and disposed of at an authorized disposal site. Asbestos handling and disposal regulations will be enforced. Generally, asbestos-containing materials will be handled in a manner as to minimize air emissions and collected in plastic bags labeled "Asbestos-Containing Waste." Disposal of radioactively clean asbestos will be prearranged at a licensed Type II solid waste disposal landfill.

11.3.7 Scrap Metal

During dismantlement, non-radioactive structural metal components, plates, piping, and wire cables will be removed and sent off-site for scrap metal recycling.

Radioactive contaminated metals that can economically be decontaminated to meet the radioactive release criteria will be decontaminated and sent to scrap metal recycling facilities. Metals that cannot be decontaminated practically will be characterized and disposed as radioactive waste.

11.3.8 Miscellaneous NonHazardous Solid Waste

Decommissioning and dismantlement will require the disposal of system and building wastes. These wastes will include materials that were never radiologically contaminated or otherwise meet the radiological-release criteria, and that are not classified as hazardous waste. Non-radioactive, non-hazardous wastes are expected to include:

- Duct work and associated equipment including, ducts, fans, and supports;
- Electrical systems and equipment, such as cables, conduit and motors;
- Concrete flooring and bituminous pavement (asphalt-type); and
- Office furniture and fixtures.

The priority options for these materials are: (1) re-use within the Omaha VAMC, (2) recycle, and (3) disposal at a licensed Type II landfill.

11.3.9 Non-Hazardous Liquid Wastes

Following classification as non-hazardous waste, contained liquids will be categorized to facilitate recycling, re-use, and/or disposal. Liquids transported off-site will be manifested and handled by approved disposal/recycling facilities.

11.4 Potential Environmental Response

The existing Omaha VAMC procedures for spill control and mitigation will be revised to address the additional handling of radioactive or potentially radioactive materials during the AJBRF decommissioning. An AJBRF-specific amendment to the facility Spill Prevention Control and Countermeasures Plan will be prepared and remain in effect until such time as all of the materials in the plan or additional polluting materials resulting from D&D activities have been removed from site. If a spill occurs during the decommissioning process, it will be handled according to the plan and written implementation procedures.

11.5 Spent Fuel Management

The AJBRF does not contain any nuclear fuel. The reactor was de-fueled in June of 2002. Fuel, including fission chambers, was transferred to the USGS research reactor facility in Denver, Colorado.

12. Quality Assurance Program

VA is committed to comprehensive and effective QA program as an integral part of the D&D effort. QA will provide a systematic approach to the execution and review of key activities and will assist in providing assurance that program activities are conducted in a manner that complies with established policies, procedures, and recognized good practices. Objectives of the Omaha VAMC's D&D effort will include:

- The prevention of uncontrolled release of radioactive materials outside of the reactor facility site;
- Limiting radiation exposure to workers and to the public at large to levels below those stipulated in 10 CFR 20;
- The packaging and shipping of radioactive and/or hazardous wastes or materials and the characterization and measurement of waste for appropriate disposal within the guidelines provided in 10 CFR 61 and 71, and 49 CFR 172 and 173;
- Controlling work practices and procedures to comply with federal, state, and local requirements;
- Preventing the unnecessary spread of radioactive and/or hazardous contamination to uncontaminated areas; and
- Conducting work in a safe manner.

To implement QA, VA will exclusively utilize contractors with demonstrated experience in D&D projects of similar size and scope and require the DOC to have a QAPP. The QAPP will be reviewed and approved by the RSC. At a minimum, the following will be defined in the QAPP:

- Contractor Authority and Responsibility: Including, but not limited to, written definitions of authority, duties, and responsibilities of contractor managerial, operation, and safety personnel; a defined organizational structure; and assigned responsibility for review and approval of plans, designs, procedures, data, and reports.
- Personnel Training: Including, but not limited to, training program to provide staff trained and qualified in principles and techniques of jobs assigned, aware of the nature and goals of the QA and demonstrated proficiency maintained by retraining and/or periodic performance reviews.
- Procedures: Written procedures for decommissioning activities (including, but not limited to, operational activities, surveys, sampling activities, sample chain of custody, calibration of instruments, and equipment maintenance and calibration) that are prepared, reviewed, and approved by knowledgeable/responsible persons.
- Documentation and Data Management: Documentation of activities performed and to track and control tasks in progress from commencement to conclusion.
- Corrective Action Process: Process to investigate and correct recognized deficiencies and document corrective actions.

12.1 Corrective Action

The key responsibility of the ACOS/R is to ensure that contractors perform their duties according to the requirements of the contract, in a manner that is safe, and in a manner that will minimize possible exposure to radioactive and/or hazardous materials. Omaha VAMC staff or

VA's Technical Representative Contractor will meet regularly with the DOC and will institute the following processes and procedures.

- As a standard part of the regular meetings with the DOC, VA will require the DOC to identify any possible QA issues. If issues are identified, the ACOS/R will be notified and VA will require that the contractors determine appropriate corrective measures.
- Upon determination of corrective measures presented by the contractors, the ACOS/R, in conjunction with the RSC, will review and approve proposed corrective actions upon concurrence of the adequacy of the corrective action.
- The DOC will be required to maintain a record of issues and corrective actions taken. In addition, the DOC will be required to report issues and corrective actions as part of regular management reporting to VA support staff.

12.2 Records Management

The Omaha VAMC D&D team is committed to developing and maintaining a comprehensive record of all QA actions, documents, and policies and procedures. To that end, the ACOS/R, in conjunction with the RSC, will institute the following policies and procedures.

- Fundamental QA record management will be the responsibility of the DOC. The DOC will be required to have in place a documented records management program. The program shall be defined within the QAPP.
- At the completion of the D&D project, the DOC will be required to present the ACOS/R with a complete inventory and duplicate copies of all QA-related documents related to the work completed.
- Upon receipt and review of the documents referenced above, the ACOS/R will transfer the documents to a location physically separated from the project environment. The location will be a secure location.

13. Final Status Survey

The purpose of the FSS will be to document that the AJBRF has been decommissioned to the extent necessary to meet the unrestricted use release requirements as specified in 10 CFR 20.1402 "Radiological Criteria For Unrestricted Use." The FSS will be developed based on applicable guidance contained in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM).

The FSSP will be developed and implemented by the DOC in such a manner that the requirements of this DP and associated implementation procedures are met, thus allowing for license termination. The FSSP design will be based on the assumption that the remaining activity after decommissioning is essentially normally distributed relative to both the interior and exterior areas of the AJBRF. If the results of the survey demonstrate that this assumption is not true, then the final survey implementation procedures will be revised to accommodate the different conditions.

The FSSP will be submitted for NRC review and approval separately from the DP. The FSSP will be finalized after DP submittal and after D&D activities have started because the physical condition of the site is not known at the time of DP submittal. Also, VA may wish to propose alternative release criteria that would impact the implementation of the FSS.

13.1 Data Quality Objectives

The Data Quality Objectives (DQO) process will consist of a series of planning steps, using a graded approach, to ensure that the types, quantity, and quality of radiological data used in decision-making are appropriate for the intended application. DQOs are qualitative and quantitative statements that clarify the process objective, define the most appropriate data to collect, determine the most appropriate conditions for collecting the data, and specify acceptable levels of decision errors that will be used to establish the quantity and quality of data needed to support the decision. The DQOs intended to be used for the AJBRF are as follows:

- Proper selection and independent verification of survey unit classification;
- Collection of sufficient data of high quality to ensure that a comparison can be made with the release criteria for each survey unit to determine if residual radioactivity in each unit has been reduced to a level below the release criteria;
- Ensuring that the potential radiological risk from the AJBRF as a whole is below that represented by the dose limit release criteria.

13.2 Survey Area Classification

The final survey classification for the AJBRF will be subdivided into three distinct classes of impacted areas:

- Class 1 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination in excess of the screening criteria.
- Class 2 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the screening criteria.

- Class 3 Areas: Any impacted areas that are not expected to contain any residual radioactivity or are expected to contain levels of residual radioactivity at a small fraction of the screening criteria.

Class 1 areas will receive the highest degree of survey efforts during the final survey receiving a 100 percent surface scan along with direct surface measurements or material samples at locations determined using the methods in MARSSIM. Class 2 areas receive scan surveys on 10 percent to 100 percent of the surface and direct measurements. Class 3 areas generally receive only judgmental scans and direct measurements. The anticipated areas of the AJBRF in each class are provided in Table 13.1.

Table 13.1, AJBRF MARSSIM Classifications for FSS

Survey Area		MARSSIM Classification
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, B540, and B540A	Floors	1
	Walls < 2 meters	2
	Walls > 2 meters and ceilings	3
Rooms B533A, B533AA, B537, and B535A	Walls < 2 meters	2
	Walls > 2 meters and ceilings	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
Outside areas	All	Non-impacted

13.3 Conduct of the FSS

13.3.1 Instrumentation and Sampling

Radiation monitoring instrumentation to be utilized in the final status survey will have the necessary sensitivities and capabilities to detect the radiation of interest and at levels below the screening criteria. The instrumentation will also be calibrated, operated, and maintained in accordance with written procedures.

Swipe samples and other material samples will also be collected and analyzed in accordance with a written procedure. Rigorous chain-of-custody practices will be implemented to ensure control and the reproducibility of samples.

13.3.2 Background Determination

Prior to conducting surveys, instrument backgrounds will be established for the specific types of surfaces being surveyed. Background measurements will be taken in non-impacted areas of the Omaha VAMC.

13.3.3 Measurement Frequency

The estimated sample sizes for each survey area will be determined using the guidance in Table 5.3 of MARSSIM (NRC, 2000). However, if warranted, a larger number of samples will be added to survey units on a case-by-case basis based on the judgment of the DOC, the VA RSO and the ACOS/R.

13.3.4 Quality Assurance/Quality Control

The DOC will include the scope of the FSS in the required QAPP or include a robust quality assurance/quality control (QA/Quality Control) section in the FSSP. Activities affecting quality will have controls established to ensure that the appropriate equipment, required environmental conditions, and prerequisites for the given activity have been met. Specific quality assurance criteria applicable to the final status survey will be identified in a stand-alone quality assurance plan.

Quality control will be implemented through the QAPP or FSSP. Quality control methods will ensure the quality and accuracy of the survey data. These methods will include quality control over field and laboratory instrumentation, control of samples, sample analysis, use of radioactive reference sources, and data processing, including management and control.

13.4 FSS Report

Following completion of the FSS, a FSS Report will be developed and submitted to the NRC. The report will contain information needed by the NRC to make a decision on termination of the AJBRF license and authorization of the site for unrestricted use. The report will be prepared using the applicable guidance contained in Section 8.6 of MARSSIM (NRC, 2000).

14. Radiological Accident Analysis

There is a potential for radiological accidents during the ABJRF D&D project resulting from the uncontrolled release of radioactive materials to the work area or the environment. These releases are most likely associated with the management of contaminated liquids in the reactor tank and the ancillary water cooling and filter systems.

Uncontrolled releases of airborne contamination could also occur during the demolition and segmentation of activated and/or contaminated reactor components such as the reactor tank or contaminated items such as the fume hoods. An uncontrolled release of radioactive material could also occur during a transportation accident.

The accidental dropping of an activated reactor assembly was also considered as a potential accident. However, because the more highly activated components are located under water and can be washed off within the reactor tank, the surface contamination on these parts would not be sufficiently high to release significant quantities of radioactive materials during such an incident. Such an incident would mostly likely result in additional unplanned external exposures involved with perhaps up righting the reactor assembly and re-rigging it for loading into a waste container.

Fire is another possible source of an uncontrolled release of radioactive materials. However, the majority of the combustibles that will be present on-site will be non-radioactive contaminated materials. Potentially contaminated combustibles will include dry active waste such as personal protective clothing and rags and towels used for site cleanup and decontamination. The radioactivity contained in these materials would not be high enough to result in a significant release during such an incident.

There are no fissile materials located on-site that could result in a criticality incident.

The consequence levels discussed in the following paragraphs are described in the U.S. Department of Energy (USDOE) Standard DOE-STD-I 120-2005, "Integration of Environment, Safety, and Health Into Facility Disposition Activities" (USDOE 2005), as well as NUREG 1537, Part I, Appendix 17.1, Section 3.3.

14.1 Release of Contaminated Liquid

An uncontrolled release of radioactively contaminated liquids could result in the contamination of workers, the ABJRF, or the environment. Liquids containing radioactive suspended solids containing Co-60, Cs-137, and HTD isotopes such as H-3, Fe-55, and Ni-63 are present in the reactor tank. These liquids will be drained or pumped out during the D&D project and filtered to remove the suspended radioactive contaminants.

Accidents could occur during the draining or pumping activities. Hoses or pipes could burst or come free from pumps resulting in an uncontrolled release. To mitigate the extent of such releases, processes involving contaminated liquids will only be operated with personnel present and personnel will actively monitor for leaks and spills. In the event of an accident, personnel will respond by shutting down the activity. Also, pumps and filters will primarily be located in the existing vault. The vault is small and a release would have little impact outside of the vault. A spill kit will be readily available to respond to any incidents.

An uncontrolled release of the contaminated water may result in only incidental ingestion, short term dermal contact, and external exposures. The oral ingestion annual limit on intake (ALI) in

Appendix B of 10 CFR 20 for Co-60 is 200 µCi (the lowest ALI of the contaminants of concern). The ALI corresponds to committed effective dose equivalent (CEDE) of 5 rem. To approach the oral ingestion ALI, more than 1.7 gallons (6.4 liters) of contaminated water at 0.06 µCi/ml Co-60 (Duratek 2003) would need to be ingested. To exceed the public dose limit of 100 millirem (mrem), 0.13 liters of contaminated water would need to be ingested.

External exposures would also be far less than the current dose measured dose-rate in an accident scenario because the activity would be diluted over a large area. A plausible accident scenario may result in the ingestion of several milliliters of contaminated water and exposure to the material for an 8-hour period, The resulting occupational CEDE would then be about 40.2 mrem. Therefore, the resulting dose in an accident involving the release of contaminated liquids would be far less than one rem to off-site receptors and 25 mrem to on-site workers. As such, safety management operations such as standard engineering and administrative controls are sufficient for protection against such accidents.

14.2 Release of Airborne Contamination

An uncontrolled release of airborne radioactivity could occur during cutting and demolition activities involving contaminated or activated materials. Such activities may take place inside temporary containment structures equipped with HEPA filter ventilation systems or using localized ventilation ducts. The failure of the containment structure or ventilation system could result in the release of airborne radioactive materials into the ABJRF. If the negative pressure is maintained in the AJBRF as described in this DP at the time of such an incident, the HEPA air filter system would prevent release to the environment. If the air flow system in the ABJRF is not operating at the time of such an incident, airborne radioactive material could be released directly to other areas of the Omaha VAMC or to the environment. Alarming CAMs will be used in the work areas to warn against the release of airborne radioactivity.

Eu-152 has the most-limiting inhalation ALI of the contaminants of concern and the DAC is 1E-8 µCi/mL. The DAC concentration is the air concentration that results in 1 ALI, or 5 rem, to the exposed individual in a 2,000-hour work year. The Eu-152 concentration in the reactor tank Gunite or concrete is not known at this time, but assuming the concentration is 10 pCi/g or 1E-5 microcuries per gram. If concrete dust at its worst case concentration were to become airborne as a result of an uncontrolled release, the breathing air would be limited to a respirable particulate loading of 1.0 milligram per milliliter (1.0 gram per liter) of air before the DAC was exceeded. Given that the interior free volume of reactor room is approximately 10,000 cubic ft (28,000 liters), approximately 61.7 lb (28 kilograms) of contaminated material would have to become airborne to reach the DAC level. The effluent limit for control of public dose is 3E-11 µCi/mL. Because the DAC level and effluent concentration limits are based on an exposure duration of one year, an uncontrolled release as described above to the air outside the facility would have minimal dose consequence due to the short duration of such an accidental release.

14.3 Hot Particles

VA has never had any failed fuel within its facility and does not expect to encounter hot particles. Furthermore, hot particles have not identified during any site characterization efforts (Duratek 2003, AECOM 2011). However, there will be daily, weekly, and monthly surveillance surveys and constant Heath Physics Technician coverage on all ongoing job activities. Also, engineering controls such as the constant recirculation of the reactor water, HEPA vacuum cleaners, and general radiation monitors will be in use for all project actions.

14.4 Transportation Accidents

Various forms and quantities of radioactive waste will be shipped from the ABJRF during the D&D project. The dose consequence from transportation accidents could be higher than the contamination accident scenarios described above because higher-activity reactor components could be involved. As such, there is a potential for a moderate dose consequence of between 1 and 25 mrem for the public following a transportation accident. However, adherence to NRC and DOT radioactive material packaging and transportation requirements is considered a sufficient control measure for mitigating transportation-related incidents.

15. Financial Assurance

The Omaha VAMC in coordination with VA's Director of Finance, Office of Research and Development, is responsible for developing and submitting all budgeting and legislative requests necessary to operate, maintain, and ensure the ultimate proper disposition of the AJBRF. In accordance with 10 CFR 50.75(e)(1)(iv), VA has requested and received funds consistent with a more than $2,000,000 project. An estimated project cost totaling $1,364,328 is summarized in Table 15.1.

Table 15.1, Decommissioning Project Cost Estimate

Task	Price
Site Decommissioning and Final Status Survey	$358,390
Waste Transportation and Disposal	$569,384
Planning and Management	$163,688
Subtotal	**$1,091,462**
25% Contingency	*$272,866*
Total Project Estimate	**$1,364,328**

Should additional funds above and beyond those designated for the decommissioning be required, VA will engage in the required budget processes necessary to ensure that required funds for decommissioning are provided consistent with the work described in Section 7.

16. Changes to the Decommissioning Plan

The DP will be approved by the NRC and incorporated as a license amendment. Minor changes to the DP, which do not change the original intent of the plan and which do not involve an issue that may impact radiation safety or worker/public dose, may be approved by the ACOS/R.

If a significant change to the DP is required, the RSC will apply the test identified in 10 CFR 50.59 (effective date March 2001) as it applies to non-power reactors in decommissioning. Should the RSC determine that the change is significant and could pose a significant increase in potential worker, public, or environmental impacts, NRC approval will be obtained prior to implementing the change. Guidance on implementing the requirements 10 CFR 50.59 is provided in the following documents:

- NRC Regulatory Guide 1.187 "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments"
- Nuclear Energy Institute (NEI) Guidance NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, September 2000
- NRC Inspection Guidance (Part 9900)

All changes to the DP will be documented and records of changes will be maintained until license termination. All changes to the DP will be described in the FSSR.

17. Technical Specifications

The AJBRF currently operates under technical specifications that are included in NRC License Number R-57 Amendment 11, Appendix A. These technical specifications are in place to ensure the safe operation of the reactor facility. However, most of the technical specifications do not apply to the reactor when it is not in operation. VA does not anticipate the need for any additional technical specification change requests following the NRC's approval of the DP and license amendment.

18. References

AECOM. 2011a. Allen J Blotcky Reactor Facility Additional Characterization Report. July.

AECOM. 2011b. Allen J Blotcky Reactor Facility Free Release Materials Report. June.

Duratek, Inc. 2003. Allen J Blotcky Reactor Facility Omaha Veterans Administration Medical Center, Omaha, Nebraska, Characterization Survey Report. February. (ML061140054)

U.S. Department of Energy (USDOE). 2005. "Integration of Environment, Safety, and Health Into Facility Disposition Activities." DOE-STD-I 120-2005.

US Department of Veterans Affairs (VA). 2004. Decommissioning and Decontamination Plan for the Allen J Blotcky Reactor Facility. March. (ML042740512)

US Department of Veterans Affairs (VA). 2002a. Safety Evaluation Report Related to the Renewal of the Operating License for the Alan J. Blotcky Reactor Facility. August 2. (ML020630192)

US Department of Veterans Affairs (VA). 2002b. Amendment No. 11, Renewal of Facility Operating License No. R-57. Alan J. Blotcky Reactor Facility. August 2. (ML020630192)

US Nuclear Regulatory Commission (NRC). 2000. Multi-Agency Radiation and Site Investigation Manual (MARSSIM). Revision 1 August.

US Nuclear Regulatory Commission (NRC). 2006. Consolidated Decommissioning Guidance: Decomissioning Process for Material Licensees. September.

US Nuclear Regulatory Commission (NRC). 2008. Request for Additional Information Regarding Allen J Blotcky Reactor Facility Decommissioning Plan. May 13. (ML103210235)

Attachment A
Microshield Dose Model Output

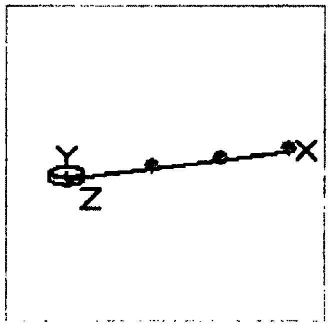
MicroShield 8.03
AECOM (8.03-0000)

Date	By	Checked

Filename	Run Date	Run Time	Duration
Lazy Susan 3 Ext Points.msd	August 19, 2011	3:06:13 PM	00:00:00

Project Info	
Case Title	Lazy Susan Estimate
Description	Lazy Susan Dose Estimate (2011) - Exteral Dose Points
Geometry	12 - Annular Cylinder - External Dose Point

Source Dimensions	
Height	10.0 cm (3.9 in)
Inner Cyl Radius	22.5 cm (8.9 in)
Inner Cyl Thickness	0.0 cm (0 in)
Outer Cyl Thickness	0.0 cm (0 in)
Source	1.0 cm (0.4 in)



Dose Points			
A	X	Y	Z
#1	125.0 cm (4 ft 1.2 in)	5.0 cm (2.0 in)	0.0 cm (0 in)
#2	225.0 cm (7 ft 4.6 in)	5.0 cm (2.0 in)	0.0 cm (0 in)
#3	325.0 cm (10 ft 8.0 in)	5.0 cm (2.0 in)	0.0 cm (0 in)

Shields			
Shield N	Dimension	Material	Density
Cyl. Radius	22.5 cm	Air	0.00122
Source	1445.133 cm³	Air	2.7
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input: Grouping Method - Actual Photon Energies				
Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
Co-60	1.4000e-002	5.1800e+008	9.6877e+000	3.5844e+005

Buildup: The material reference is Source Integration Parameters	
Radial	10
Circumferential	20
Y Direction (axial)	20

Results - Dose Point # 1 - (125,5,0) cm					
Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	8.450e+04	2.126e-01	2.817e-01	4.105e-04	5.438e-04
1.1732	5.180e+08	2.361e+03	2.891e+03	4.218e+00	5.166e+00
1.3325	5.180e+08	2.724e+03	3.283e+03	4.726e+00	5.696e+00
Totals	**1.036e+09**	**5.085e+03**	**6.174e+03**	**8.945e+00**	**1.086e+01**

Results - Dose Point # 2 - (225,5,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	8.450e+04	6.384e-02	8.613e-02	1.233e-04	1.663e-04
1.1732	5.180e+08	7.127e+02	8.811e+02	1.274e+00	1.575e+00
1.3325	5.180e+08	8.233e+02	1.000e+03	1.428e+00	1.735e+00
Totals	**1.036e+09**	**1.536e+03**	**1.881e+03**	**2.702e+00**	**3.310e+00**

Results - Dose Point # 3 - (325,5,0) cm					
Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**
0.6938	8.450e+04	3.026e-02	4.118e-02	5.842e-05	7.950e-05
1.1732	5.180e+08	3.387e+02	4.208e+02	6.053e-01	7.520e-01
1.3325	5.180e+08	3.915e+02	4.776e+02	6.792e-01	8.285e-01
Totals	**1.036e+09**	**7.302e+02**	**8.984e+02**	**1.285e+00**	**1.581e+00**

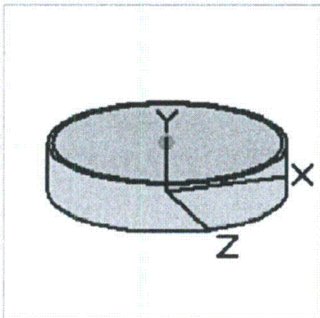
MicroShield 8.03
AECOM (8.03-0000)

Date	By	Checked

Filename	Run Date	Run Time	Duration
Lazy Susan Int Point.msd	August 19, 2011	3:10:34 PM	00:00:00

Project Info	
Case Title	Lazy Susan Estimate
Description	Lazy Susan Estimate (2002) - Top of Grid Plate
Geometry	11 - Annular Cylinder - Internal Dose Point

Source Dimensions	
Height	10.0 cm (3.9 in)
Inner Cyl Radius	22.5 cm (8.9 in)
Inner Cyl Thickness	0.0 cm (0 in)
Source	1.0 cm (0.4 in)



Dose Points			
A	X	Y	Z
#1	0.0 cm (0 in)	10.0 cm (3.9 in)	0.0 cm (0 in)

Shields			
Shield N	Dimension	Material	Density
Cyl. Radius	22.5 cm	Water	1
Source	1445.133 cm^3	Aluminum	2.7

Source Input: Grouping Method - Actual Photon Energies				
Nuclide	Ci	Bq	$\mu Ci/cm^3$	Bq/cm^3
Co-60	4.5000e-002	1.6650e+009	3.1139e+001	1.1521e+006

Buildup: The material reference is Cyl. Radius Integration Parameters	
Radial	10
Circumferential	10
Y Direction (axial)	20

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate $MeV/cm^2/sec$ No Buildup	Fluence Rate $MeV/cm^2/sec$ With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	2.716e+05	3.479e+00	1.494e+01	6.717e-03	2.885e-02
1.1732	1.665e+09	5.671e+04	1.586e+05	1.013e+02	2.834e+02
1.3325	1.665e+09	7.117e+04	1.822e+05	1.235e+02	3.161e+02
Totals	**3.330e+09**	**1.279e+05**	**3.408e+05**	**2.248e+02**	**5.995e+02**