Decommissioning Plan

Buffalo Materials Research Center

Prepared for:

University at Buffalo
The State University of New York
Buffalo Material Research Center
Office of Environment, Health, and Safety Services

Completed by:

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SUMMARY OF CHANGES

Revisions to the Decommissioning Plan will be tracked when revisions are issued. Changed sections will be identified by special demarcation in the margin. A summary description of each revision will be noted in the following table.

Revision Number	Date	Description of Change
0	February 6, 2012	Initial Issue

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ACRONYMS AND ABBREVIATIONS

ACM	Asbestos Containing Material
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
ARA	Airborne Radioactivity Area
BMRC	Buffalo Materials Research Center
BSFR	Bulk Survey for Release
C&D	Construction and Demolition
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
cm^2	square centimeter
D&D	Decontamination and Decommissioning
DAW	Dry Active Waste
DC	Demolition Contractor
DCGL	Derived Concentration Guideline Levels
DDE	Deep Dose Equivalent
DECON-A	complete decontamination and structure demolition option
DECON-B	complete decontamination and release of the structure option
DCF	Dose Conversion Factor
DOC	Design and Oversight Contractor
DOE	United States Department of Energy
DOT	Department of Transportation
DP	Decommissioning Plan
dpm	Disintegration per minute
DQO	Data Quality Objectives
EH&S	Environment, Health and Safety
ENTOMB	Entombment option
ER	Environmental Report
FEMA	Federal Emergency Management Agency
FGR	Federal Guidance Reports
FSS	Final Status Survey
HASP	Health and Safety Program
HAZWOPER	Hazardous Waste Operations and Emergency Response

HEPA	High Efficiency Particulate Air
HIC	High Integrity Container
HMR	Hazardous Materials Regulation
HP	Health Physics
HRA	High Radiation Area
HSA	Historical Site Assessment
HVAC	Heating, Ventilation, and Air Conditioning
ICS	Incident Command System
IF	Irradiation Facility
INEEL	Idaho National Engineering and Environmental Laboratory
ISO	International Standards Organization
LLRW	Low-Level Radioactive Waste
LSA	Low Specific Activity
LSC	Liquid Scintillation Counter
m	meter
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MW_t	Megawatt Thermal
mrem	Millirem
MSHA	Mine Safety and Health Administration
MTR	materials-testing-reactor
N-16	Nitrogen-16
NaI	Sodium Iodide
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NIOSH	National Institute for Occupational Safety and Health
NIST	National Institute of Science and Technology
NMSS	Nuclear Material and Safety and Safeguards
NRC	United States Nuclear Regulatory Commission
NSTC	Nuclear Science and Technology Center
N2S2	Nevada National Security Site
NUREG	Nuclear Regulatory Guide
NYS DOH	New York State Department of Health
pCi/g	picocuries per gram
P.E.	Professional Engineer

PM	Project Manager
PuBe	Plutonium-Beryllium
PULSTAR	Pulse Training Assembled Reactor
OSHA	Occupational Safety and Health Act
QA	Quality Assurance
QAPP	Quality Assurance Program Plan
QC	Quality Control
RA	Restricted Area
RDSC	Reactor Decommissioning Safety Committee
RP	Radiation Protection
RPM	Radiation Protection Manager
RSO	Radiation Safety Officer
RWP	Radiation Work Permits
SAFSTOR	No-Action Alternative
SNM	Special Nuclear Material
SCR	Site Characterization Report
SSC	Structures, Systems, and Components
SUNY	State University of New York
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent dosimeter
UB	University at Buffalo
UBPM	University at Buffalo Project Manager
WRS	Wilcoxon Rank-Sum
yr	year

1.0 SUMMARY OF DECOMMISSIONING PLAN

1.1 Introduction

The Buffalo Materials Research Center (BMRC) is owned by The State University of New York (SUNY) at Buffalo (UB) and is located on the southern edge of the South Campus of UB off of Rotary Drive in Buffalo, Erie County, New York. The BMRC was a Research and Test Reactor Facility with a pool-type reactor that was designed and constructed by AMF Atomics between 1959 and 1961. The initial criticality date for the reactor was March 24, 1961, while the last day of operation occurred on June 23, 1994. The unit has been in Possession Only status since June 6, 1997. In 1998 the unused fuel was shipped to North Carolina State University. In 2005 the spent fuel was shipped to the Idaho National Engineering and Environmental Laboratory. The BMRC is being decommissioned because there is no future need for the facility.

UB is about 20 miles south of the Canadian border at Niagara Falls, New York, 90 miles west of Rochester, New York and 80 miles from the southern border of New York and Pennsylvania. UB is about five miles east of the Niagara River which is a border with Canada. Figure 1-1 shows the location of the BMRC on the UB South Campus.

Figure 1-1 Location of BMRC on the UB South Campus

Decommissioning Plan Synopsis

This Decommissioning Plan (DP) has been prepared in accordance with Chapter 17 of the Nuclear Regulatory Commission (NRC) NUREG-1537 Part 1, *Guidelines for Preparing and Reviewing Applications for Licensing of Non-Power Reactors* and NUREG-1757, *Consolidated Decommissioning Guidance*. This Decommissioning Plan provides guidance on the general process and methods to be used to safely decontaminate, remove and dispose of radioactive materials, equipment, systems, components, and soil associated with the BMRC. Decommissioning activities will result in the complete removal of the BMRC structures from the site location allowing an unrestricted release of the BMRC site by the NRC and be granted termination of the NRC license as allowed by Title 10, Code of Federal Regulations (CFR), Part 20, Section 1401 (10 CFR 20.1401).

Decommissioning will be performed in a safe and orderly manner to prevent undue radiation exposure to the workers, public, and environment. Remediation and demolition tasks include 1) the removal of the reactor core and activated bioshield; 2) removal of reactor and facility systems containing residual radioactive materials; 3) removal of ventilation systems; 4) remediation of concrete surfaces; and 5) disposition of building structural concrete and steel. All structures of the building and facility piping are planned to be removed in their entirety.

A final status survey (FSS) will be implemented to demonstrate compliance with default derived concentration guideline levels (DCGLs) in support of the unrestricted release of the site and termination of the facility.

1.2 Background

General

The reactor at the BMRC was operated by SUNY at Buffalo under an operating license from the NRC. The reactor at UB was placed into operation in 1961 under License Number R-77 and operated until 1963 with materials-testing-reactor (MTR) fuel elements with a maximum steady-state power level of 1 megawatts thermal (MW_t). In 1964 the reactor was shut down and the core and control systems were modified so that the reactor could operate with Pulse Training Assembled Reactor (PULSTAR) fuel at power levels of up to 2 MW_t.

The UB ceased operations of the reactor on June 23, 1994, and shipped the unused PULSTAR fuel to North Carolina State University in 1998. In 2005, UB with the assistance of the U.S. Department of

Energy (DOE) shipped the used PULSTAR fuel to the DOE at the Idaho National Engineering and Environmental Laboratory (INEEL).

The BMRC currently maintains two NRC Licenses, a Possession Only License (R-77) and a Special Nuclear Material License (SNM-273) for possession of a Plutonium-Beryllium (PuBe) neutron source; however, this source has been transferred to the UB's New York State Department of Health (NYS DOH) agreement state license. In addition to the PuBe source, the NYS DOH license (#1051) authorizes the possession of byproduct materials for calibration and check sources as well as byproduct materials from reactor operations.

History

The reactor was constructed in 1959 and 1960 under Atomic Energy Commission (AEC) Construction Permit No. CPRR-39 and went critical on March 24, 1961. The original licensed power was 1 MW_t with an original core loading of MTR fuel. The MTR Fuel was replaced with PULSTAR fuel in 1964 and the facility license was amended to operate at 2 MW_t.

During its operating history, the BMRC was used for training and education, transient fuel performance testing, nuclear component testing and calibration, materials radiation damage research, isotope production, and neutron interrogation through activation analysis, radiography and delayed fission assay.

The following is a chronological history of significant licensing events:

1959	-	Initial construction commenced under AEC License # R-77. Licensee name: The Western New York Nuclear Research Center
1961	-	Construction completed in January
1961	-	Initial criticality achieved on March 24 at 2051 hours Full power first achieved on June 1 at 1220 hours
1964	-	Reactor modifications to utilize PULSTAR fuel; Used MTR fuel transferred to McMaster University in Ontario, Canada in April Initial criticality of PULSTAR core achieved June 22 at 2112 hours;
1973	-	Facility name change: Nuclear Science and Technology Center (NSTC)
1984	-	Vertical 10k tank placed into service; Tank Farm placed into wet layup
1985	-	Facility name change: Buffalo Materials Research Center
1991	-	Reactor and Primary Coolant System modified to present configuration
1994	-	Reactor activities suspended; Last shutdown on June 23 at 0600 hours;
1997	-	NRC license R-77 amended to Possession Only Status
1998	-	Unused PULSTAR fuel shipped to North Carolina State University in July
2005	-	Used fuel shipped to Idaho National Engineering and Environmental Laboratory

Reactor Description

The reactor is a PULSTAR heterogeneous open-pool type water cooled reactor that used solid 6% enriched uranium dioxide fuel clad in Zircaloy similar to commercial nuclear power reactor fuel. The core was cooled by forced convection at higher power levels and by natural convection at lower power levels. The coolant is purified water and the reflector is water and/or graphite.

The core is immersed in a 13,000-gallon aluminum alloy lined pool surrounded by both high density and regular density concrete. Original design specifications used water to cool the core. The water exited the reactor at the pool bottom and was then pumped to a 5,000-gallon Nitrogen-16 (N-16) holdup tank for decay. The water then cycled through a heat exchanger connected to an external cooling tower located on the southeast side of the Administration Building (Laboratory Wing). The cooled water was then circulated back to the top of the pool.

In October 1977 a leak developed in the primary cooling pipe. The leak was repaired by June of 1978 and reactor operations resumed. In 1989 a reoccurrence of a leak at the point where the coolant piping penetrated the bottom (horizontal portion) of the reactor liner directly under the reactor led to the decision that modifications needed to be made. These modifications were completed in 1991 and included adding a second aluminum alloy liner on the Neutron Deck level and rerouting the coolant piping through the original twelve-inch beam tube, just above the floor of Neutron Deck level of the Containment Building. The coolant piping on the Neutron Deck is shielded by high density concrete blocks until it penetrates through the containment wall into the N16 Tank Vault.

All nuclear fuel has been removed from the site. The reactor and its associated components remain in the pool in the normal operational configuration. Reactor components removed during repairs completed in 1991 were stored behind concrete block shielding on the Neutron Deck. These reactor components were removed during the pre-decommissioning cleanout and buried at the EnergySolutions Clive, UT facility..

Associated with the reactor within the Containment Building are the following Irradiation Facilities:

- Flux Trap
- Isotope Tube
- Vertical Standpipe
- Thermal Column
- Pneumatic Conveyor
- Electron Accelerator

BMRC Facility

The BMRC Facility consists of the reactor described above, the Containment Building which encloses the reactor and associated facilities related to use of the reactor, as well as the Administrative Building which contains offices, class rooms, and laboratories. The Administration Building is also referred to as the Laboratory Wing.

Containment Building

The reactor is housed in a tri-level Containment Building as shown in an artist's rendering of the interior of the original Containment Building in Figure 1-2. The three levels (from top to bottom) of the Containment Building include the Control Deck, the Gamma Deck, and the Neutron Deck. Plan views of the Control Deck, Gamma Deck, and Neutron Deck are shown on Figure 1-3, Figure 1-4, and Figure 1-5 respectively. The Containment Building was constructed of reinforced concrete and in a right cylinder shape. The cylindrical Containment Building is approximately 75 feet in diameter and 52 feet high with walls that are two-foot thick reinforced concrete. The roof of the Containment Building varies from four to nine inches in thickness and is supported by concrete beams. The walls and the foundation of the Containment Building are constructed on bedrock.

Within the Containment Building are the following additional facilities:

- Hot Cell and Hot Cell Work Room (Room 105)
- Dry Chamber (Neutron Deck Level)
- Charpy Cell (Neutron Deck level) (Removed)
- Hot and Warm Chemistry Labs (Rooms 103, 104, and 108)
- Neutron Activation Analysis and Counting Lab (Room 109)
- Other Labs (Rooms 202, and 203)
- Electronics Shop (Room 200)

Figure 1-2 Artist's Rendering – Interior Cut Away View of the Reactor and Containment

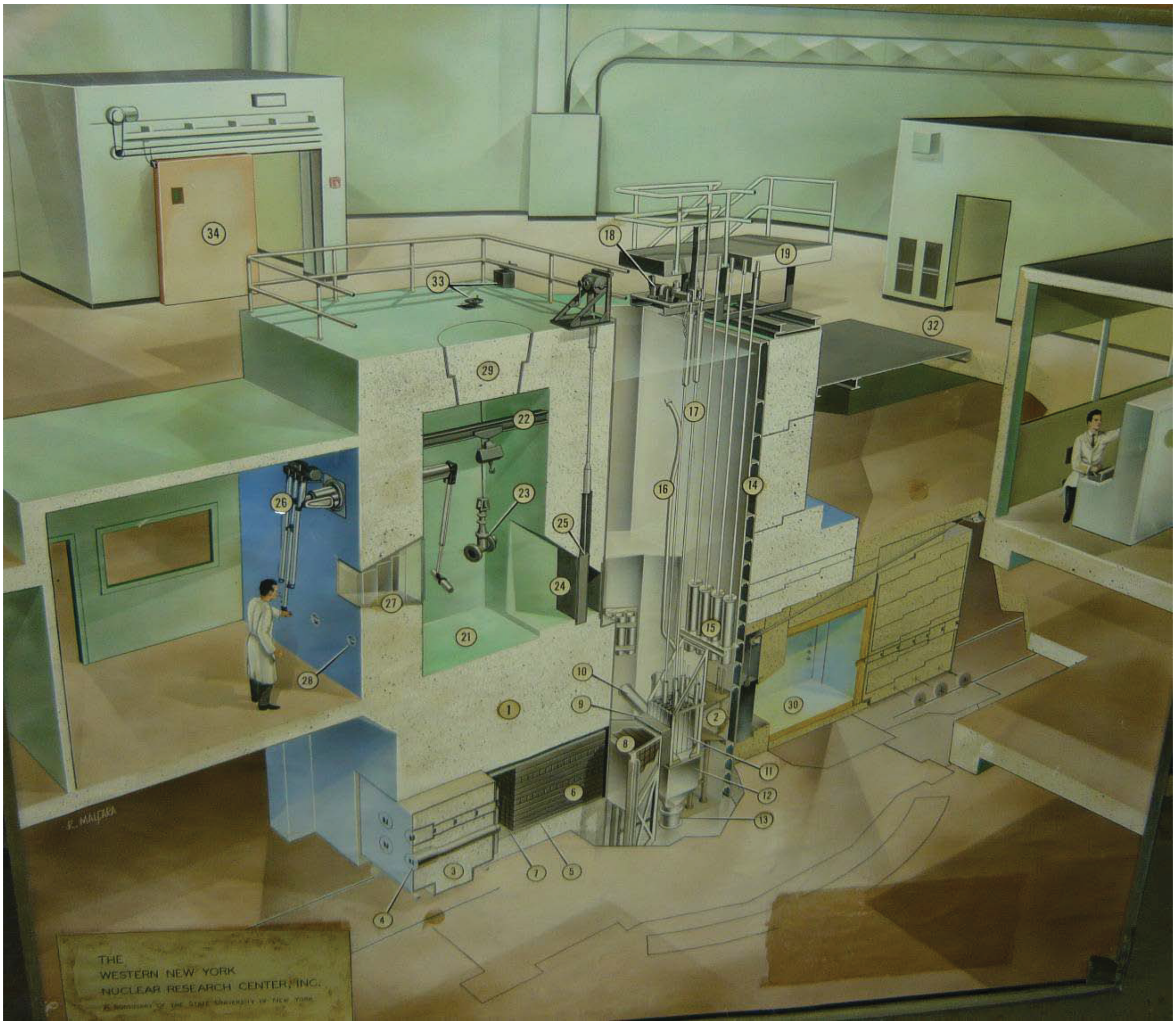


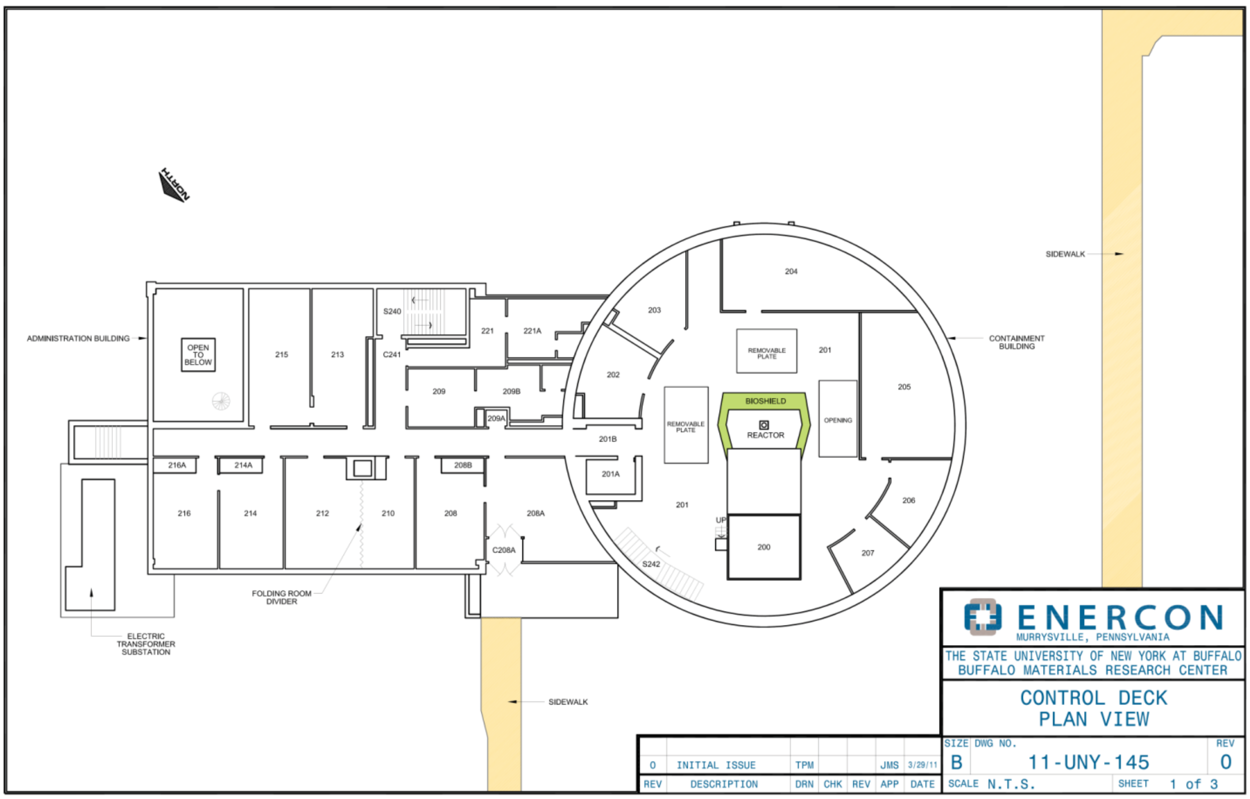
Figure 1-2 Legend:

1. Bioshield
2. Reactor Plenum
3. Thermal Column Plug
4. Beam Tube
5. Thermal Column Support
6. Thermal Column Graphite Blocks
7. Thermal Column Support Plate
8. Thermal Column
9. Flux Trap
10. Beam Tube Chamber
11. Reactor
12. Coolant Outlet Plenum
13. Coolant Outlet Piping
14. Reactor Pool Lining
15. Vertical Standpipe
16. Pneumatic Conveyor Tube
17. Isotope Tube
18. Isotope Tube Drive Mechanism
19. Operating Platform
20. NA
21. Hot Cell
22. Hot Cell Crane and Rail
23. Pass Tube Isolation Valve
24. Hot Cell Lead Shutter
25. Hot Cell Lead Shutter Drive Mechanism
26. Hot Cell Remote Manipulators
27. Radiation Shielding Window
28. Hot Cell Instrumentation Access Port
29. Hot Cell Removable Access Plug
30. Dry Chamber
31. NA
32. Removable Metal Plate Floor Opening Cover
33. Pass Tube Isolation Valve Hand Wheel
34. Control Deck Airlock (Access to/from Administration Building)

Control Deck

The top level of the Containment Building is called the Control Deck. A plan view of the Control Deck is shown on Figure 1-3. The Control Deck houses a dual-hook overhead crane (with a 10-ton and 2-ton hook) at the ceiling level. Crane access to the Gamma Deck is through three floor openings shown on Figure 1-3. Two of the floor openings are covered with removable metal plates. In addition to the reactor pool, the Control Deck houses the Reactor Control Room (Room 205), Superintendent's Office (Room 207), two labs (Rooms 202 and 203), Fan Room (Room 204), another office (Room 206), Electronics Shop (Room 200), and the New Fuel Storage Room in the airlock (Room 201B). The airlock provides for access to the upper level of the Administration/Laboratory Building.

Figure 1-3 Control Deck Plan View



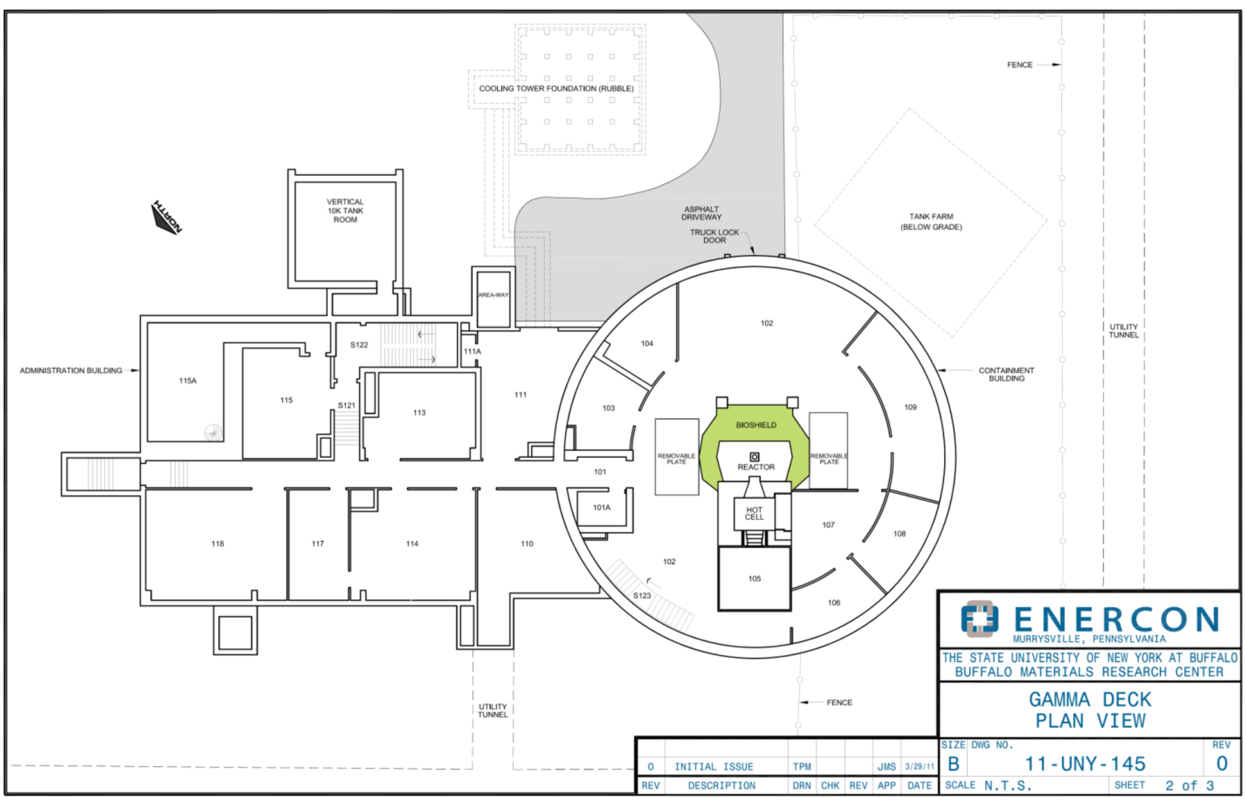
Gamma Deck

The middle level of the Containment Building is called the Gamma Deck. A plan view of the Gamma Deck is shown on Figure 1-4. The Hot Cell, Hot Cell Work Room (Room 105), Locker Room (Room 106), Room 107, Hot Chemical Lab (Room 108), Medium Chemical Lab (Room 104), Activation Analysis Lab (Room 109), Lower Fuel Vault Room (Room 101A), and Chemical Lab (Room 103) are located on the Gamma Deck. This level also has an Airlock (Room 101) for access to the bottom level of the Administration Building and a truck door for access to the asphalt drive located on the southern side of the Containment Building.

The Truck Door provides vehicular and equipment access to the outside. Prior to 1979, the Truck Door area housed a waste compaction system. Wastes generated at the BMRC (NSTC at the time) and under the site-wide radiological program were collected and compacted for volume reduction in this system.

Analysis activities involving the use of mercury in conjunction with radiological constituents occurred in the Activation Analysis room. Mercury may be present in the fume hoods in this room and in the low points or traps in waste drain lines.

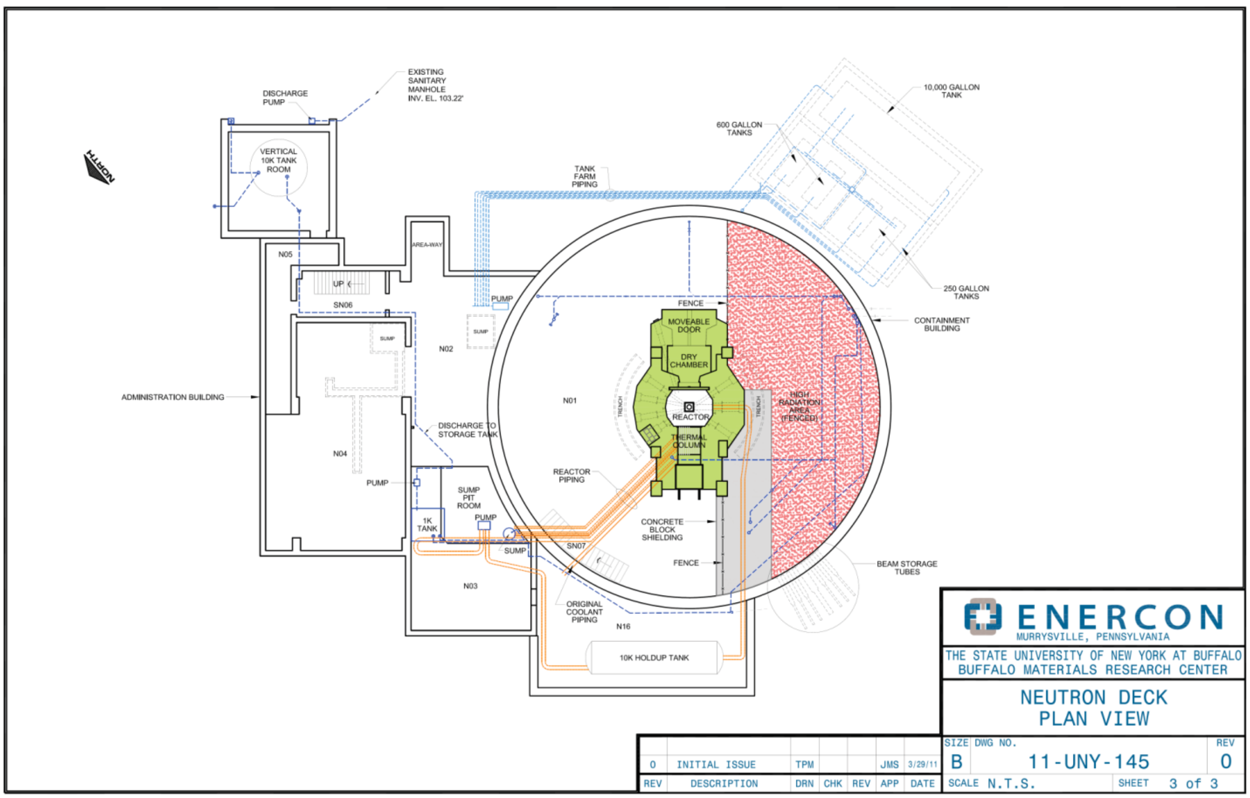
Figure 1-4 Gamma Deck Plan View



Neutron Deck

The lower level of the Containment Building is called the Neutron Deck. A plan view of the Neutron Deck is shown on Figure 1-5. The Neutron Deck is an open area divided by fencing into two sections. The enclosed (fenced) portion of the Neutron Deck, shown in red hatch in Figure 1-5, is currently a high radiation area (HRA). The HRA is being used to store reactor components removed from the reactor during repairs made in 1991 due to a leak in the reactor liner. These reactor components (Thermal Column Nosepiece, Dry Chamber Nosespiece, Grid Plate Bolts, and Neutron Beam Tube ends) were replaced with new items of similar. The rerouted Primary Coolant piping, 18,000 lbs of Depleted Uranium Shield block, and access to the beam storage tubes is also in the HRA. The beam storage tubes extend ten feet through the Containment Building wall into the surrounding soil. They are used as storage-for-decay of high dose reactor components or experiments. At the point where the tubes exit the Containment Building wall, they are approximately ten feet underground.

Figure 1-5 Neutron Deck Plan View



Reactor, Bioshield, Dry Chamber, and Hot Cell

The reactor, bioshield, dry chamber, and hot cell are located inside the Containment Building and portions are located on the various levels of the building. Figure 1-6, Figure 1-7, and Figure 1-8 contain plan views of these items on each level of the Control Deck, Gamma Deck, and Neutron Deck respectively. Figures 1-9 and 1-10 provide elevation views of the reactor and bioshield. These figures reflect the original setup of the reactor and do not include modifications to the reactor tank and coolant piping in 1977 or the tank modification completed in 1991.

The bioshield consists of high density concrete with metal aggregate in lieu of typical stone aggregate. The bioshield section on the Gamma deck facing the Truck Door consists of removable concrete blocks set in place by the facility crane. The remaining sections of the bioshield were cast in place after the entire reactor tank, beam tubes, and piping were constructed. It should be noted that floors and floor supports were poured in place concurrently with the Containment Building walls. Therefore, the Gamma Deck and Control Deck floors do not appear to be structurally supported by the reactor bioshield. The structural independence between the bioshield and the floors should be verified by a New York State Professional Engineer using the original facility design drawings and the chronological historical photos before bioshield demolition.

Figure 1-6 Reactor Plan View – Control Deck Level

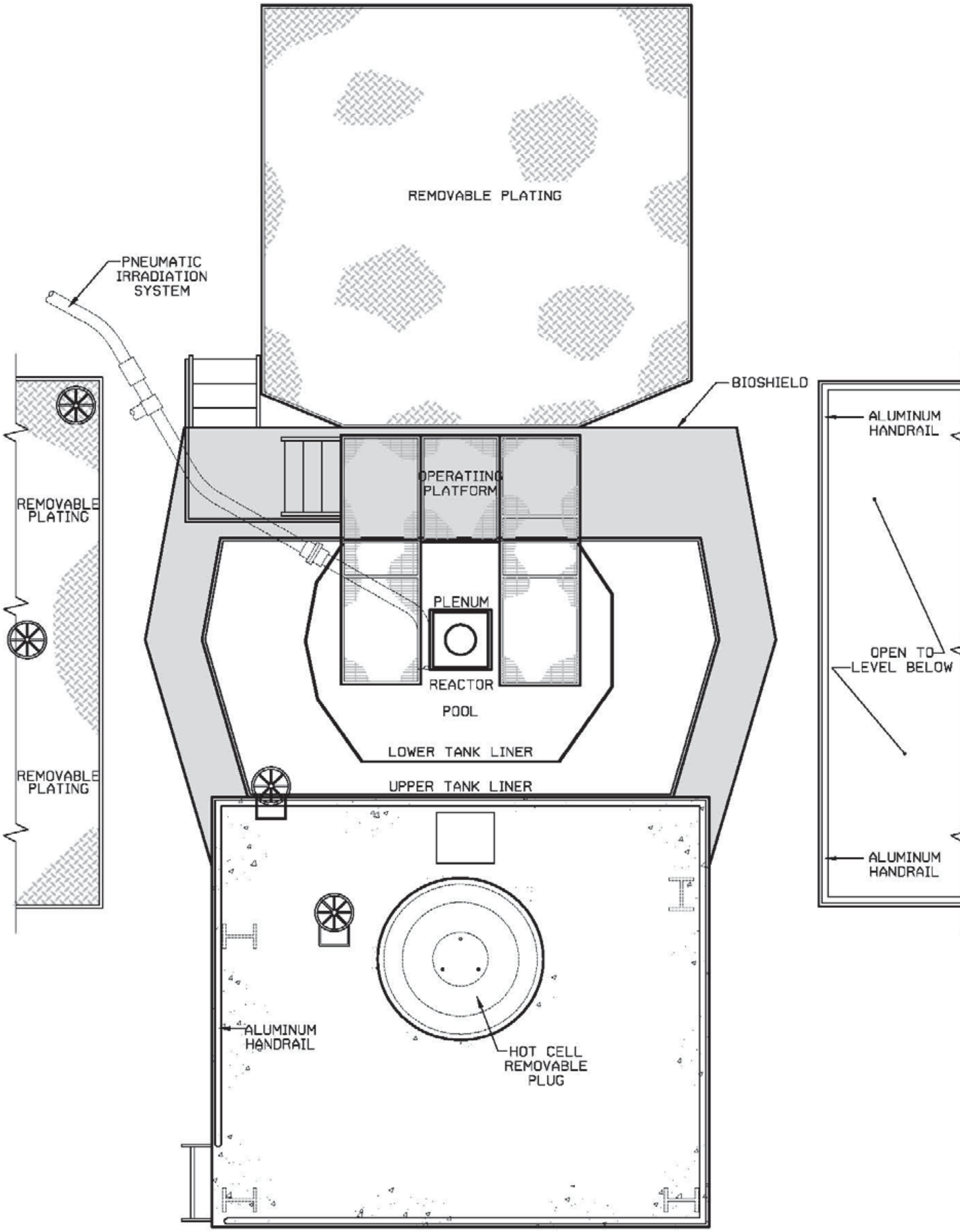


Figure 1-7 Reactor Plan View – Gamma Deck Level

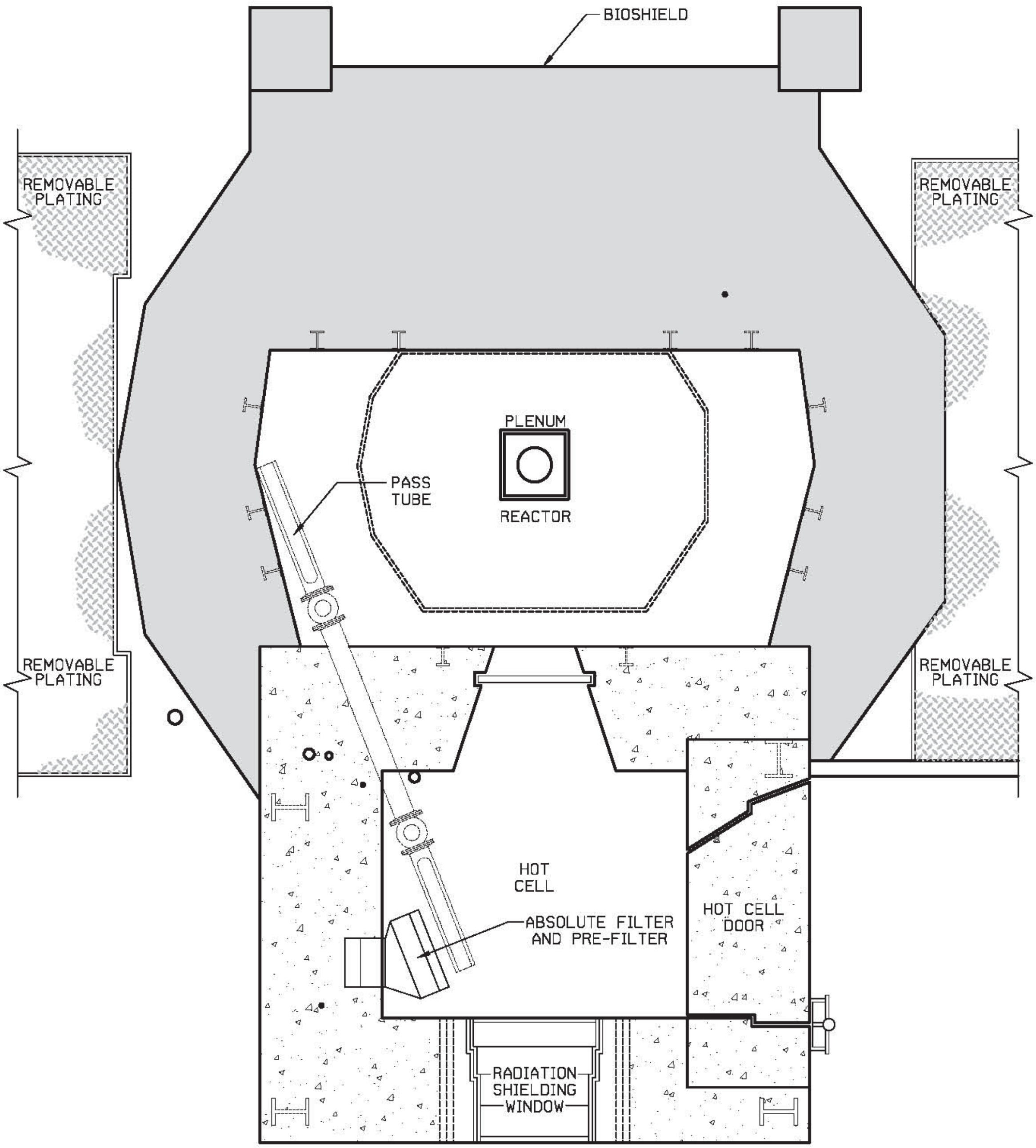


Figure 1-8 Reactor Plan View – Neutron Deck Level

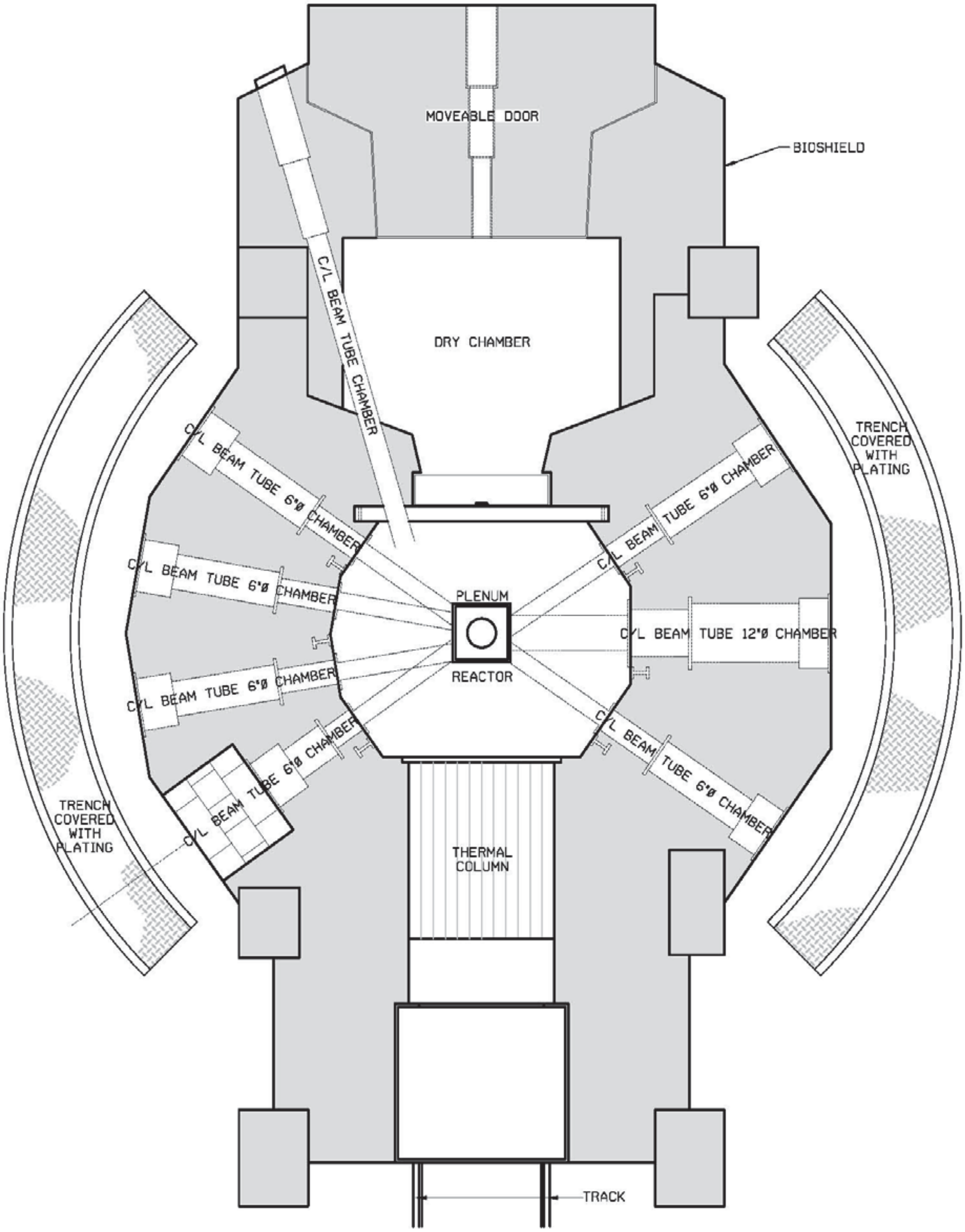


Figure 1-9 Reactor/Bioshield Elevation View (Northeast Direction)

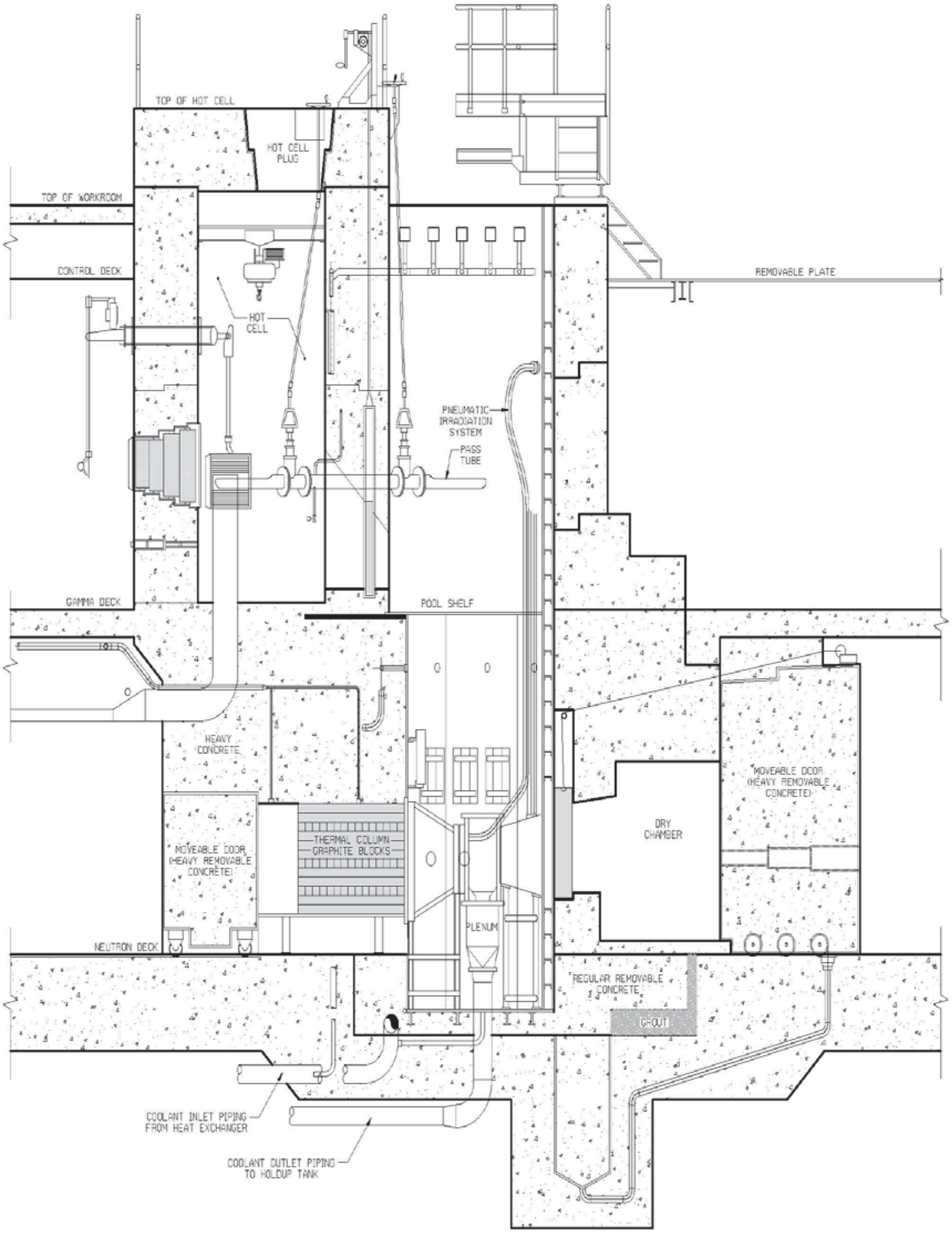
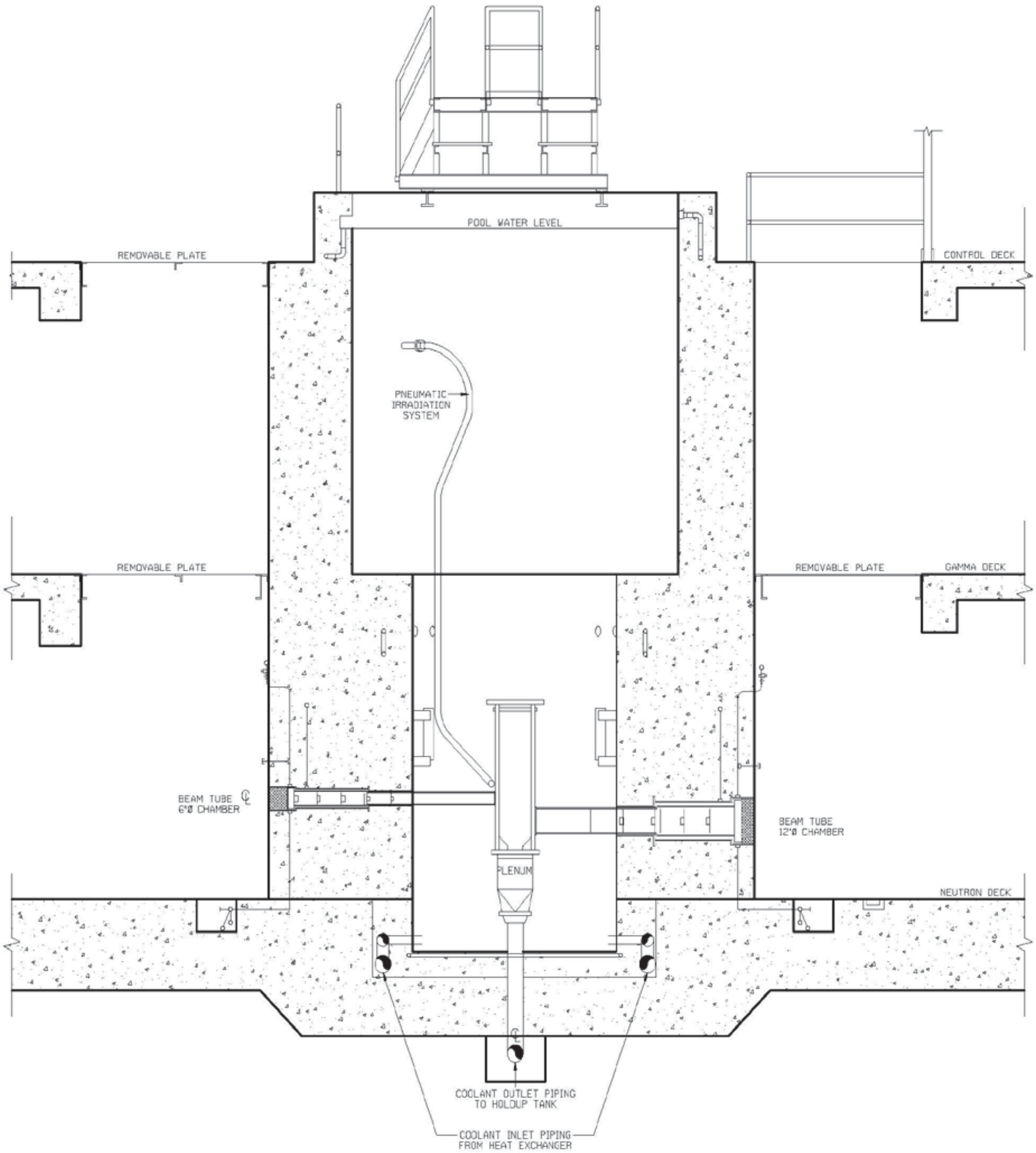

TOP OF HOT CELL
HOT CELL PLUG
TOP OF WORKROOM
CONTROL DECK
HOT CELL
REMOVABLE PLATE
PNEUMATIC IRRADIATION SYSTEM
PASS TUBE
GAMMA DECK
POOL SHELF
HEAVY CONCRETE
MOVEABLE DOOR (HEAVY REMOVABLE CONCRETE)
THERMAL COLUMN GRAPHITE BLOCKS
MOVEABLE DOOR (HEAVY REMOVABLE CONCRETE)
DRY CHAMBER
PLENUM
NEUTRON DECK
REGULAR REMOVABLE CONCRETE
GROUT
COOLANT INLET PIPING FROM HEAT EXCHANGER
COOLANT OUTLET PIPING TO HOLDUP TANK

Figure 1-10 Reactor/Bioshield Elevation View (Southeast Direction)



Ventilation System (Containment Building)

Air from the Heating, Ventilation, and Air Conditioning (HVAC) system from the portions of the Containment Building which housed the medical facilities, hot cell, hot chemistry laboratory, beam storage tubes, rabbit tubes, and the thermal column was discharged via underground piping to the 167-foot high stack on the Mackay Heating Plant located southeast of the BMRC facility (see Figure 1-1). Portions of this piping have been removed during renovation of Rotary Road as detailed in the Historical Site Assessment (HSA).

The remainder of the Containment Building ventilation passes through a bank of roughing and absolute filters prior to being discharged from a point on the southeast side of the Containment Building roof. Per the facility Technical Specifications, ventilation ducts were designed with hydraulic dampers that would isolate the Containment Building from the Administration Building and the outside areas if alarms indicated an airborne event.

Liquid Waste System (Containment Building)

Prior to 1984, liquid waste from the hot cell drain, hot chemistry laboratory, and service area drains was collected in the two 250-gallon tanks located in the tank farm (see Figure 1-5). The tank farm is an underground facility located southeast of the Containment Building. Liquid waste from the Gamma Deck and Control Deck laboratories, change area sinks, and Truck Lock door drain were collected in the two 600-gallon tanks located in the tank farm.

Administration Building (Laboratory Wing)

The Administration Building may also be referred to as the Laboratory Wing in this document. Within the Administration Building are the following facilities:

- Vertical Accelerator (Room 115A)
- Horizontal Accelerator and Counting Cave (Room 118)
- Health Physics Office and Counting Room (Room 114)
- Low Level Labs with Fume Hoods (Rooms 213, and 215)
- Machine Shop (Room 113)

Upper Level – Administration Building

A plan view of the upper level of the Administration Building is shown on Figure 1-3. The upper level of the Administration Building housed the Men's Locker and Change Room (Room 209) and Men's Bathroom (Room 209B), Women's Locker and Change Room (Room 221) and Women's Bathroom

(Room 221A), Conference Room (Room 212), offices (Room 208, Room 210, Room 214, and Room 216), Reception Area (Room 208A), Janitor's Closet (Room 209A) and two low level laboratories (Room 213 and Room 215). There is a fume hood in Room 213 and Room 215. Based on characterization, there has been little to no radiological impact to the upper level of the Administration Building.

Basement Level – Administration Building

A plan view of the basement level of the Administration Building is shown in Figure 1-4. On the basement level of the Administration Building is housed a Horizontal Accelerator (Room 118) with an adjacent cave area. This level also contained a laboratory (Room 115), Classroom (Room 117), Health Physicist Office and Counting Room (Room 114), The Vertical Accelerator Room (Room 115A), Machine Shop (Room 113), and Air Conditioning Equipment Room (Room 111). The Electrical Service Room (Room 110) is also located on this level. A utility tunnel is located north of the Electrical Service Room (Room 110). The utility tunnel connects to the Campus Service Tunnel system and is locked to prevent access.

Subbasement – Administration Building

A plan view of the subbasement of the Administration Building is shown on Figure 1-5. The subbasement level is located under the western portion of the basement. The eastern portion of the basement level is supported by caissons and grade beams. The subbasement contains the 5,000-gallon Nitrogen-16 Holdup Tank Room (Room N16), Cooling Water Equipment Room and Sump Pit Room (Room N03), lower Air Conditioning Equipment Room (Room N02) and adjacent area way, Air Handling Duct Room (Room N04), and access to the vertical 10,000-gallon tank room (Room N05) as well as the vertical 10,000-gallon tank room. The floor of Room N16 is located on bedrock.

Ventilation System (Administration Building)

Air from the HVAC system located in the Administration Building is vented through high efficiency particulate air (HEPA) filters to the roof of the building except for the Machine Shop (Room 113) and Vertical Accelerator Room (Room 115A) fume hoods. Air from these two fume hoods is vented via a duct on the southeast side of the Administration Building.

Liquid Waste System (Administration Building)

Liquids from the bathrooms, floor drains, sinks, and fume hood drains in the Administration Building are discharged to the South Campus sanitary system. Experiments in the Administrative Building have involved use of H-3 and C-14 in the upper level laboratories (Room 213 and Room 215) fume hoods and H-3 in the Health Physicist Office and Counting Room (Room 114) fume hood.

1.2.1 Reactor Decommissioning Overview

UB plans to completely dismantle the reactor and its ancillary support systems, remove radioactive materials from the BMRC, and demolish the facility prior to requesting the termination of license R-77. The facility is being demolished because there is no future need for the facility.

Most of the reactor components and systems are either activated or contaminated and will need to be segregated from non-radiological components and surfaces so that they can be disposed of as low-level radioactive waste (LLRW). The BMRC is primarily constructed of concrete and a vast majority of the waste generated during the decommissioning will be concrete rubble from demolition activities. Waste is disposed of based on characterization for radiological constituents. Based on characterization results, it is anticipated that almost all of the structural concrete will be sent for concrete recycling or will be sent to a local industrial landfill. Some concrete, primarily the floor slabs of the subbasement and the Neutron Deck are anticipated to be disposed of as Bulk Survey for Release (BSFR) waste due to the potential for volumetric contamination. Notable exceptions are the concrete associated with the reactor bioshield on the Neutron Deck level as well as the reactor liner and internal components. Some decontamination is expected to be required for the reactor components, bioshield, and hot cell concrete structures that were out of the main neutron activation zone so that the majority of concrete and metal from these structures will meet the requirements of BSFR waste. The decommissioning cost estimate assumes that the reactor liner and internal components are to be disposed of as Class A LLRW as shown in Appendix A, *BMRC Activations Analysis and Component Characterization*, developed by WMG, Inc. of Peekskill, NY while the old Control Blades are to be disposed of as Class B mixed radwaste.

The following major decommissioning tasks are necessary for site release. The sequence in which these tasks occur may vary (see Section 2.3 for more details):

- Pre-Decommissioning Cleanout - removal of loose legacy equipment and materials (Completed)
- Isolation and removal of inactive systems (Some prior to approval of DP by the NRC)
- Removal of hazardous materials (lead, cadmium) and abatement of asbestos containing material (ACM) (Bulk of the materials to be removed prior to approval of DP by the NRC)
- Perform supplementary characterization (prior to approval of DP by the NRC)

- Installation of temporary systems and preparation of the facility for decommissioning operations (potentially prior to approval of DP by the NRC)
- Removal of the irradiated reactor components
- Removal of and disposition of water in the reactor
- Removal of and disposition of the reactor liner
- Decontamination of concrete associated with reactor, bioshield, and hot and dry cell
- Removal of the reactor, bioshield, and hot and dry cell
- Segregation, packaging and shipment of materials according to radioactivity levels
- Removal of auxiliary systems (rabbit system, water purification, ventilation)
- Decontamination of building surfaces
- Interior release survey using Reg Guide 1.86 limits
- Demolition of the BMRC facility
- Perform the FSS on BMRC footprint
- Submit required reports that demonstrates to the NRC that the facility meets the release criteria
- Request license R-77 termination
- Restore site: Backfill void created by removing below grade structures for future use by UB.

The FSS is to be developed by UB using the criteria provided in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (NRC 2000). An overview of the FSS is provided in Section 4.0 with the Final Status Survey Plan to be submitted and approved prior to commencement.

1.2.2 Estimated Cost

The decommissioning cost estimate is summarized in Table 1-1. It is estimated that approximately $8.9 million is required to decommission the BMRC. The cost estimate includes the Demolition Contractor (DC) costs, subcontractor costs, and other direct costs. The cost includes the costs of shipping and disposing of waste and the final status survey. The cost estimate is subject to cost increases because there are unforeseeable elements of cost within the defined project scope. A contingency of 20% is included in the decommissioning costs in Table 1-1 to ensure that sufficient funds are available to cover costs that may result from unanticipated conditions or unforeseeable elements in the project scope. Typically, these include factors such as waste disposal rates or increased waste volumes from undiscovered or uncharacterized areas. In addition, the time duration between the development of the DP and the inception of decommissioning activities can influence the costs associated with changes in the economy and regulatory requirements.

Table 1-1 Decommissioning Cost Estimate

Item No.	Description	TOTAL
1	Project Planning	$ 238,440
2	Mobilization and Training	$ 67,620
3	Facility Preparation	$ 235,480
4	Neutron Deck remediation and disposal	$ 1,792,092
5	Gamma Deck remediation and disposal	$ 221,882
6	Control Deck remediation and disposal	$ 141,946
7	Administration Building remediation and disposal	$ 51,038
8	Waste and HVAC Systems remediation and disposal	$ 983,429
9	Ancillary systems removal and disposal costs	$ 406,000
10	Impacted Soil excavation and disposal costs	$ 99,270
11	Waste Packaging and Transportation	$ 849,800
12	Demolition of Administration and Containment Buildings	$ 1,801,700
13	Final Status Survey and Report	$ 248,980
14	Backfill and Site Restoration	$ 223,995
15	Demobilization	$ 61,620
	SUBTOTAL	**$7,423,292**
	Contingency @ 20%	$ 1,484,658
	TOTAL DECOMMISSIONING COST	**$ 8,927,950**

1.2.3 Availability of Funds

In accordance with 10 CFR 50.75 (e)(1)(iv), UB is a state institution and as such will provide financial assurance with a statement of intent containing a cost estimate for decommissioning, indicating that funds will be obtained when necessary. The decommissioning of the BMRC is fully funded in the current UB Capital Budget that runs through June 2013.

1.2.4 Program Quality Assurance

1.2.4.1 Independent Contracting and Oversight

The UB will select a qualified DC to assist in the physical remediation and dismantlement of the BMRC reactor and associated facilities. In addition, the UB has contracted with a Design and Oversight Contractor (DOC) to provide the UB with Decommissioning Project Management, Health Physics, Safety, Quality, and other oversight support as needed to augment the UB staff in order to meets its license and regulatory obligations during decommissioning. The selected DOC is Enercon Services, Inc. (ENERCON). The selection of the DOC is discussed in Section 2.6.

The dual-contractor approach has been selected by the UB to assure that decommissioning work activities are subject to continuous and independent compliance oversight. The DOC will report directly to the UB Project Manager (UBPM) for project schedule and cost while reporting to the UB/BMRC Environment, Health and Safety Department in matters affecting quality or safety. The DOC will be responsible for developing a Quality Assurance Project Plan (QAPP) appropriate for the decommissioning of the BMRC and the final status survey. The QAPP will be reviewed and approved by the UB.

The DOC shall monitor UB contracted work to assure compliance with the UB's reactor license, the approved DP, and applicable contract requirements and regulations. The DOC will also provide site-specific training [developed and approved by the BMRC Director and Radiation Safety Officer (RSO)] to the selected contractors to assure their familiarity with the requirements of the UB's reactor license and the BMRC DP.

1.2.4.2 Established Minimum Qualifications for DC

The UB's selection of qualified contractors for the DC role will consider the following criteria as a minimum:

a) Ability of the firm to perform the required tasks as demonstrated by past nuclear reactor demolition experience (5 years) of the firm, its field supervisors, and its teaming partners or subcontractors as appropriate.

b) Qualifications of key individuals based on education, training, experience and professional licensure.

c) The compliance record of the contractor, teaming partners, and key subcontractors; including records of any violations or fines for non-compliance with federal, state or local regulations.

d) Safety record of the contractor and key subcontractors to include a review of past projects and Occupational Safety and Health Act (OSHA) reportable incidents.

e) Relevant experience of contractor, teaming partners, and key subcontractors, particularly with regard to research reactor decommissioning.

f) References and recommendations from previous customers/completed projects.

g) Financial stability of the contractor and key subcontractors to complete the project and their ability to meet the minimum insurance and bonding requirements.

h) Evaluation of the DC's proposed Technical Approach for site demolition.

1.2.4.3 Quality Assurance Program Plan (QAPP)

The DOC will be responsible for preparing and implementing a QAPP to control the decontamination and decommissioning of the BMRC. The QAPP will incorporate industry and regulatory requirements applicable to decommissioning project planning and management, decontamination, dismantling, demolition, radiological sampling, analysis, and surveys. The QAPP shall describe the organization, document approval, and quality assurance (QA) aspects for the BMRC decommissioning project, including the following minimum requirements:

a) A clear description of the project organization, including a definition of the authorities and responsibilities of key personnel involved in the decommissioning.

b) QA actions to be implemented during the BMRC decommissioning project.

c) A list of implementing policies and procedures the DOC and DC will use to complete the project, including detailed project work plans for critical project evolutions.

d) Work approval, communication, and authority guidelines, including guidelines for the interface of the DC, the DOC, and the BMRC/UB.

e) Document control for UB approved work plans and procedures.

f) A plan for performing self-monitoring and independent audits of the DOC and DC's activities.

g) Maintenance of records for project work, calibration, and characterization data.

1.2.4.4 Shipping

The DOC will identify appropriate treatment and disposal facilities for wastes generated from the decommissioning project. The DOC will develop a procedure for compliant shipping of BMRC wastes, identifying methods of compliance with 10 CFR 71 subpart H and Department of Transportation (DOT) Hazardous Materials Regulations (HMR) as appropriate. The DOC is responsible for waste sizing, characterization, segregation, and packaging. The QAPP and associated procedures will address quality requirements for waste preparation and packaging activities.

1.2.4.5 Final Status Survey QAPP

The DOC will develop a QAPP specific to the performance of the FSS and associated documentation (e.g. characterization information used in the design of the final survey). The FSS QAPP will be reviewed and approved by the UB. The FSS QAPP will incorporate the appropriate regulatory requirements applicable to the planning and conduct of radiological surveys necessary for the termination of the BMRC license and the release of the site for unrestricted use. The DOC will perform the FSS in accordance with the QAPP and related procedures, and will submit the completed FSS to the UB for approval and submittal to the NRC.

2.0 DECOMMISSIONING ACTIVITIES

The objective of the BMRC decommissioning activities is to remove licensed radioactive materials from the facility and surrounding grounds necessary to obtain NRC approval for release to unrestricted use of the property and be granted termination of the NRC license. The decommissioning pathway described in this plan is intended to meet the necessary requirements to achieve this objective.

2.1 Decommissioning Alternatives

There are four alternatives available to, and considered by, the UB: 1) the No-Action alternative (SAFSTOR); 2) the entombment option (ENTOMB); 3) complete decontamination and structure demolition (DECON-A); and 4) complete decontamination and release of the structure (DECON-B). The DECON options are recommended by the NRC for non-power reactors. The selected alternative is DECON-A. The four alternatives are summarized in the following sections.

2.1.1 SAFSTOR Alternative

The SAFSTOR alternative would involve the same potential risks and environmental impacts as the proposed action, but for a much greater time period. The alternative would require that the UB maintain current radiological controls, site security, required licensing, a reactor administrator, and the utilities until the DECON option was implemented. The UB would also have to incur all of the expenses associated with maintaining the facility. This alternative would require that the UB apply for and obtain an extension to the current NRC operating license. The SAFSTOR alternative would leave the site as unusable space that would delay any beneficial reuse; and the potential for environmental contamination would still exist. This alternative is not environmentally preferable.

2.1.2 ENTOMB Alternative

The ENTOMB alternative places the same requirements on the UB as the SAFSTOR alternative with the additional requirement that the radioactive material be placed in a long-lived structure, i.e., concrete, or that the current structure proven adequate for long-term storage. This alternative delays the impacts of the proposed action and the potential for environmental contamination would continue to exist. This alternative is not environmentally preferable.

2.1.3 DECON-A Alternative

The facility records and current facility characterization reveal minimal facility contamination from past reactor operations. This alternative requires the site to be released and restored for unrestricted use. The reactor, the containment building, and the administrative building under this option will be disassembled and the radioactive material removed to meet the release criteria. The BMRC facility is located in the center of a growing university and the land area could be reused for future construction; therefore, complete decontamination and demolition (D&D) of the BMRC facility, DECON-A, is the preferred option. This alternative poses minimal risk and impacts to the environment as described the Decommissioning Environmental Report (ER).

2.1.4 DECON-B Alternative

The DECON-B alternative is similar to DECON-A, however, the structure would not be removed after remediation. The risk and impacts to the environment are the same as DECON-A for this alternative, but the UB has determined that the reuse of the building is not in its best interests. Additionally, a final status survey of the soils and bed rock under a standing structure is more complicated and requires a significantly greater level of surveying and sampling than releasing an open excavation. This alternative is not preferred by the UB.

2.2 Facility Radiological Status

2.2.1 Facility Operating History

The BMRC reactor operated from March 24, 1961 until June 23, 1994. The last of the spent fuel assemblies were removed from the facility in 2005. During the operating lifetime of the BMRC it served several functions related to education, health services, neutron activation analysis, radiological chemistry, radiological biology, nuclear science technology information dissemination, services to industry, and contractual research and design. A sampling of some of the tasks performed at the BMRC is listed below.

- Testing of nuclear industry components in a radiological environment.
- Study radiation effects on the solid state properties of carbon.
- Study radiocarbon dating techniques.
- Production of superfluids (He6).
- Investigation of Mercury contamination in fetal brain tissue.
- Study of Mercury in adult brain tissue.
- Cadmium toxicity studies.
- Samples tested for gun powder residue to support a criminal justice case.
- Samples tested for environmental pollution in air, water, and soil.
- Radiation sensitization of microorganisms.
- Isotope production for medical applications.
- The use of radioactive isotopes for visualization of bone lesions for medical applications.
- Testing of neutron detectors and other radiation detection equipment for industry application.
- Neutron activation analysis of blood serum.
- Neutron activation analysis for testing lead contamination and for forensic testing.
- Calibration of instruments for health physics applications.
- Demonstration of safe disposal of liquids, solids, and gaseous radioactive waste.

2.2.2 Current Radiological Status of the BMRC Facility

The BMRC facility has been impacted from the operation of the research reactor. However, the radiological characterization performed to prepare for the development of this DP indicates that practices employed to minimize the spread of radioactive material were effective and therefore the impacts have been limited.

A majority of the waste generated from the demolition of the BMRC facility is expeced to be eligible for release. It should be possible to recycle and reuse the concrete as aggregate or dispose of the concrete as construction and demolition (C&D) waste in a local permitted landfill. These options reduce both the overall cost of decommissioning and the associated safety risk because: (1) the risk of a transportation accident is significantly reduced with local recycling versus interstate transportation for BSFR or Class A LLRW disposal; (2) reduced cost of transportation to a local disposal facility versus distant LLRW landfills or processing facilities; (3) if the concrete is disposed of as C&D waste, the disposal rates are significantly lower than for BSFR waste or Class A LLRW.

Appendix B, *BMRC Site Characterization Report,* provides a detailed status of the radiological conditions at the BMRC. Characterization efforts were guided by a thorough HSA and a characterization plan. Characterization activities included subsurface soil samples down to bedrock, fixed location alpha and beta measurements on building surfaces, waste tank samples, loose activity smears, smears for hard to detect radionuclides, bioshield concrete samples, scan measurements on building surfaces, 10 CFR Part 61 distribution samples, and a gamma walkover survey. Analytical results from subsurface soil samples adjacent the reactor tank were less than the laboratory minimum detectable concentrations (MDCs) for contaminants of concern. In addition, indications of residual activity were not found outside of areas described in the HSA that were expected to contain elevated levels of radioactive materials.

The information obtained from both the site characterization effort and the WMG Activation Analysis and Component Characterization provides sufficient data to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected in the BMRC. Table 2-1 below provides an estimate of the types and quantities of radioactive materials.

Table 2-1 Estimated Volumes and Masses of Radiologically Impacted Components and Systems

Component/System	Primary Material	Waste type	Volume[a] (ft^3)	Mass (lb)
Reactor Internals/Components	Steel, aluminum, graphite	A	200	5000
Reactor and Bioshield Walls[b]	Concrete	A	2,700	401,600
Reactor Foundation	Concrete	A	1,300	185,400
Hot Cell	Concrete	A	1,700	244,500
Reactor Tank liner [b]	6061-T6 alloy (aluminum)	A	100	12,500
Old Control Blades	Nickel Plated Permalloy	B (mixed)	2	200
Containments Walls	Concrete	C&D	19,000	2,714,200
Foundation (Neutron Deck)	Concrete	BSFR	1,600	232,800
Neutron Deck Ceiling	Concrete	C&D	5,100	757,200
Neutron Deck	Concrete	BSFR	2,200	328,700
Gamma Deck Ceiling	Concrete	BSFR	7,200	1,188,000
Bioshield Walls >12 Ft	Concrete	C&D	2,700	324,000
Hot Cell	Concrete	BSFR	22,600	3,389,900
Gamma Deck Rooms	Concrete	C&D	5,500	812,700
Gamma Deck Room 109	Non-concrete	C&D	2,400	403,300
Control Deck Ceiling	Concrete	C&D	1,400	204,800
Control Deck Rooms	Concrete	C&D	300	6,800
Laboratory Wing	Concrete	C&D	6,000	896,400
Laboratory Wing	Non-concrete	C&D	1,600	234,200
Ancillary Structures	Concrete	C&D	31,700	4,603,800
HVAC System	Steel	BSFR	6,800	168,800
Soil	Soil	BSFR	6,600	982,100
	Total	A	6,000	849,000
	Total	B (mixed)	2	200
	Total	BSFR	47,000	6,290,300
	Total	C&D	75,700	10,957,400

(a) Estimated packaged volume.
(b) Assumes a maximum of 12 ft. height in the activation region is removed.

2.2.3 Release Criteria

The decommissioning alternative selected by the UB is DECON-A. This section provides the specific radiological criteria that will be applicable for unrestricted release of the site and termination of NRC license R-77.

DECON-A includes the removal of activated and contaminated materials, equipment and components. Because the BMRC facility will be demolished, the release criteria for the FSS will be for the soil and bedrock that remain on site. The remaining soil and rock surfaces will be released to less than the NRC required 25 millirem (mrem) annual Total Effective Dose Equivalent (TEDE) following guidance contained in MARSSIM. The release criterion will be determined to have been met by demonstrating surface or volumetric activities meet their respective NRC screening values as presented in Tables 2-2 and 2-3.

2.2.3.1 Release Criteria for Structures, Systems and Components

Structures, Systems, and Components (SSCs) destined for reuse, recycling, or disposal as clean waste will be shown to be free of detectable surface contamination in accordance with the guidelines provided by the NRC in IE Circular 81-07 (NRC 1981). Monitoring for residual radioactivity will use instrumentation and techniques (background radiation levels, scan speed, counting times) necessary to detect activity no greater than 5,000 dpm/100cm^2 total and 1,000 dpm/100cm^2 removable beta/gamma contamination. All instruments shall be calibrated with radiation sources having an energy spectrum and instrument response consistent with the radionuclides being investigated. If alpha contamination is suspected, appropriate residual radioactivity measurements capable of detecting alpha activity no greater than 100 dpm/100cm^2 fixed and 20 dpm/100cm^2 removable will be used. Per IE Circular 81-07, the potential dose impacts to members of the public is less than 5 mrem/yr to the maximally exposed individual which will meet the NRC release criteria of 25 mrem/yr. This release criteria is also described in NRC Regulatory Guide (Reg Guide) 1.86, *Termination of Operating Licenses for Nuclear Reactors*, and is known throughout the industry as the Reg Guide 1.86 release criteria.

Properly calibrated survey instrumentation with known efficiencies capable of measuring the radionuclide of concern will be used for release surveys. Removable contamination wipes may be measured in a liquid scintillation counter (LSC) or a wipe/filter counter such as the Ludlum Model No. 3030E or equivalent.

For surface tritium contamination, only removable contamination will be assessed because of the difficulties in measuring total tritium surface contamination directly (ISO 1988). If a removable fraction of 10% is assumed (ISO 1988), analysis for removable tritium must have a minimum detection limit no greater than 500 dpm/100cm^2 so that the total (fixed plus removable) required detection limit of 5,000 dpm/100cm^2 is not exceeded. Tritium wipes shall be measured in an LSC.

2.2.3.2 Release Criteria for Soils

Characterization data indicates that there has been minimal impact to subsurface soils from operation of the reactor. Soils remaining in place will be shown to meet an annual TEDE of 25 mrem/yr by demonstrating the average volumetric activity is less than the NRC screening values as found in Appendix B of NUREG-1757, *Consolidated Decommissioning Guidance* and presented in Table 2-2. The screening values are listed in picocuries per gram (pCi/g) of material.

Table 2-2 NRC License Termination Screening Levels for Surface Soils

Radionuclide	NRC Screening Value (pCi/g)
Ag-108m	8.2*
Am-241	2.1
C-14	12
Co-60	3.8
Cs-137	11
Eu-152	8.7
Eu-154	8
H-3	110
Ni-63	2,100
Pu-238	2.5
Pu-239/240	2.3
Ra-226	0.7
Sr-90	1.7

*See Section 2.2.3.4 for derivation of Ag-108m screening value.

2.2.3.3 Release Criteria for Bedrock Surfaces

During decommissioning, the entire BMRC structure will be demolished and removed from the site. The bottom concrete slab of the Containment building, i.e. the Neutron Deck floor, was poured directly on the bedrock. After the building is demolished, only the bedrock will remain. The bedrock is not volumetrically contaminated; therefore, the FSS will employ survey methods utilized for building surfaces using the NRC screening values for building surfaces as found in Appendix B of NUREG-1757, *Consolidated Decommissioning Guidance* and presented in Table 2-3. The bedrock will be prepped prior to the FSS by removing loose materials, i.e. concrete dust, pieces, and dirt. Release criteria for radionuclides not in Table 2-3 will be consistent with the Reg Guide 1.86 values stated in Section 2.2.3.1. The amount of loose radioactive materials shall not exceed 10% of the release criteria.

Table 2-3 NRC License Termination Screening Levels for Building (Bedrock) Surfaces

Radionuclide	NRC Screening Value (dpm/100cm^2)
Ag-108m	17,000*
C-14	3,700,000
Co-60	7,100
Cs-137	28,000
H-3	120,000,000
Ni-63	1,800,000
Sr-90	8,700

*See Section 2.2.3.4 for derivation of Ag-108m screening value.

2.2.3.4 Derivation of Ag-108m Screening Criteria

The screening criterion provided by the NRC does not list a value for Ag-108m. The NRC's source of calculating a screening value is the computer code DandD developed by Argonne National Laboratory. This computer code does not have Ag-108m listed in its radionuclide library. In order to avoid the lengthy and time burdensome task of calculating site-specific DCGLs using probabilistic RESRAD runs, the UB DOC has calculated a Ag-108m screening value based on the screening value calculation of Ag-110. The Ford Nuclear Reactor at the University of Michigan experienced the same technical issue with Ag-108m and was the first to calculate screening values for Ag-108m.

Isotopes of an element behave the same chemically; therefore, Ag-108m will behave the same as Ag-110 in the environment and human body as well as have the same exposure pathways. The only difference between the two radionuclides in the calculation of their screening value in the DandD computer code is the Dose Conversion Factor (DCF) for each radionuclide.

The DCFs for Ag-108m and Ag-110m were gathered from Federal Guidance Reports (FGR) #11 and #12. The ratio of the two DCFs was used as a dose factor to multiply to the dose calculated for Ag-110m in the

DandD model for each individual exposure pathways. This resulted in a dose per pCi/g for Ag-108m for each exposure pathway. The values are listed in Table 2-4 below.

Table 2-4 Ag-108m Surface Soil Screening Value

Pathway	Ag-110m Dose (mrem/1.0 pCi/g)	Dose factors (Ag-108m/Ag-110m)	Ag-108m Dose (mrem/1.0 pCi/g)
External Exposure	4.58E+00	0.581	2.66E+00
Inhalation	1.22E-05	0.493	6.01E-06
Sec. Ingestion	9.62E-05	0.705	6.78E-05
Agricultural	5.59E-01	0.705	3.94E-01
Drinking Water	1.36E-20	0.705	9.59E-21
Irrigation	1.34E-19	0.705	9.45E-20
Surface Water	7.26E-22	0.705	5.08E-22
		Cumulative mrem/(pCi/g)	3.05
		(pCi/g)/mrem	0.327
		Screening Value @ 25 mrem/y	8.19 pCi/g

Using the Ag-108m/Ag-110m Dose Factor for each exposure pathway, the resultant cumulative dose per unit of volume (mrem per pCi/g) is 3.05 mrem. This means that an average concentration of 1 pCi/g will result in a dose to the public of 3.05 mrem per year. The inverse of this number provides the average concentration that will result in a dose of 1.0 mrem/yr which is 0.327 pCi/g. The average concentration values normalized to 25mrem/yr is 8.19 pCi/g.

In the same process described above, the Ag-108m screening value for building surfaces was calculated as shown in Table 2-5 below.

Table 2-5 Ag-108m Building Surface Screening Value

Pathway	Ag-110m Dose (mrem/1000 dpm/100 cm^2)	Dose factors (Ag-108m/Ag-110m)	Ag-108m Dose (mrem/1000 dpm/100 cm^2)
External Exposure	2.34E+00	0.604	1.41E+00
Inhalation	1.05E-01	0.493	5.18E-02
Ingestion	7.88E-03	0.705	5.56E-03
		Cumulative	1.47E+00
		Screening Value @ 25 mrem/y	17,000 dpm/100 cm^2)

2.3 Decommissioning Tasks

2.3.1 Activities and Tasks

2.3.1.1 Preparation Tasks

Prior to approval of the DP by the NRC, several activities are scheduled to be conducted to prepare the reactor facility for decommissioning. Preparation of the facility and the removal of non-reactor structures, systems, and components can advance the overall decommissioning schedule. The following sections describe the preparation tasks scheduled to be performed by the UB, the DOC, and other contractors.

2.3.1.1.1 Pre-Decommissioning Cleanout

Miscellaneous loose materials and equipment that were located in the BMRC were packaged for disposal to prepare for the decontamination and demolition phase of decommissioning. An inventory of the loose materials and equipment was developed and a Request for Proposal was developed. The items sent for disposal were items that did not require any type of dismantlement, i.e., unbolting from fixed surfaces or torch cutting. The miscellaneous materials included, but not limited to, shield block, lead bricks, lead aprons, old tank samples, inoperable equipment, reactor components replaced during the 1989 rebuild. By removing the miscellaneous equipment and materials from the BMRC, additional work and storage space is available for the D&D phase of decommissioning and additional characterization of the areas that are currently occupied by the waste can be completed.

All materials were sent to a licensed radioactive materials processor for proper disposition of the material, (i.e., BSFR, Class A Radioactive Waste Disposal, lead recycling) as authorized by the processor's radioactive materials license.

The waste is scheduled to be disposed of prior to the submittal of this DP to the NRC and the UB intent to perform this activity was discussed with the NRC and NYS DOH prior to commencement. Project records, shipping manifests, and inventories are available for inspections as necessary.

2.3.1.1.2 Isolation and Removal of Inactive Systems

Inactive systems that are not required by either technical specifications, or for safety, or for support of decommissioning activities may be isolated, de-energized, and drained prior to approval of the decommissioning plan. This stage is often referred to as interference removal. These systems will be removed from the facility to avoid cross contamination during the removal of activated or impacted items. Removal of inactive systems is necessary because typically these systems interfere with the physical dismantling and demolition activities associated with decommissioning. Any inactive radioactively impacted systems will be removed at the direction of the DOC. The remaining (required) systems will be removed after approval of the DP by the NRC.

2.3.1.1.3 Removal of Hazardous Waste and ACM

The UB may contract with a qualified licensed contractor who will be responsible for the disposal of hazardous waste and ACM in compliance with federal, state of New York, and local regulations and requirements. A licensed asbestos abatement contractor will remove ACM utilizing industry practices. The DOC will provide oversight and radiation protection support as necessary during the removal of ACM. To date, characterization activities have not identified radioactively contaminated ACM, however, radiation protection (RP) support will be provided to ensure proper RP controls are in place should they be needed. Similarly, for hazardous waste, the DOC will provide the workers with the appropriate radiation safety training required commensurate with the potential for exposure to radioactive materials and RP support to ensure that radioactively impacted hazardous material will be controlled and segregated from hazardous material that is not radioactively impacted.

2.3.1.1.4 Facility Preparation for Decommissioning

Temporary systems needed to support decommissioning activities such as temporary power, portable lighting, temporary ventilation systems, and air monitoring systems may be installed prior to approval of the DP. If facility modifications are required to facilitate waste handling, they will be performed following approval by the UB.

2.3.1.2 Tasks and Activities for BMRC Decommissioning

The DOC will modify and/or augment existing BMRC and UB procedures to accommodate the decommissioning effort. Additionally, best available industry techniques will be employed during the D&D of the reactor SSCs. These techniques may include, but are not limited to, the use of long handled and standard tools, hydraulic cutters, torches, plasma arc torches, wire saws, needle guns, jackhammers, hand-scabblers, high pressure and ultra high-pressure sprayers and cutters. Any cost-effective tool or technique that achieves the goal of D&D while maintaining the principles of as low as reasonably achievable (ALARA) and considers the end state of the waste will be employed. Procedures and work packages will be developed by the DC designating the specific tool or technique to be employed. These procedures will be reviewed by the DOC and approved by the UB. Methods that will minimize the spread of contamination such as portable high efficiency particulate air (HEPA) ventilation systems or an encapsulation medium may be used. The utilization of temporary shielding will be used to maintain personnel exposures ALARA. Water will be left in the reactor pool/liner during the removal of the reactor internal components.

It is anticipated that decommissioning will initially focus on the items that have the greatest potential to be radioactively impacted. This includes the reactor and its components, the bioshield, the dry cell, and the hot cell. Then, it is anticipated that any radioactively impacted systems will be remediated. After these items have been remediated, a majority of radioactively impacted material will have been removed from the facility. Therefore, decommissioning can proceed at a more efficient pace because the potential for cross-contamination and undue exposure will be limited due to the fact that very little radioactive material will remain. This approach will also allow work to be performed ALARA.

The following activities are presented as an overview (of decommissioning) and the UB may elect to perform the activities in a different sequence. ALARA, safety, cost, and scheduling may dictate that a different sequence be used by the UB. Current BMRC staff has performed the first three activities previously during the rebuild completed in 1991. Their experience and documentation from the rebuild will be used to perform the activities in a manner that is safe and ALARA.

2.3.1.2.1 Removal of the Operating Platform

The operating platform is located above the reactor. It will be removed to allow better access to the reactor and its components. The operating platform will be disposed of based on the release criteria in Section 2.2.3 and Section 2.2.5.

2.3.1.2.2 Removal of Reactor Components

Reactor core components include, but are not limited to, the following: the core frame, plenum, core blades and shrouds, thermal column frame, and sleeve for the fission chamber. Reactor components, irradiation facility (IF) tubes for experiments and other items located within approximately three feet of the core for extended periods of time have been activated to levels that will require disposal as LLRW. The upper portion of reactor components, i.e. Control Blade drives, IF tubes, and instrumentation housings, may be surveyed and released or disposed as BSFR waste.

Removal of the reactor components should be performed while water is still in the reactor pool to provide shielding so that the work can be performed while keeping exposure ALARA. Long-handle tools and remotely operated equipment may be used to disassemble the components. The components will be lifted using the overhead crane or other device, and will be placed in LLRW containers, with appropriate shielding to keep dose rates ALARA and within DOT and disposal facility acceptance limits.

2.3.1.2.3 Water Removal from the Reactor Tank

The water in the reactor tank will be drained, filtered, and discharged to the sanitary system, as approved by the UB using the existing liquid waste system. Radioactive sediments on the bottom of the tank can be removed by agitating the sediments to remove them during pumping, for subsequent filtering. Alternatively, the sediments may be left still and removed manually after the tank is drained. The tank water will be filtered to remove the suspended solids, stored, and sampled according to current procedures to allow discharge of the water to the sanitary system. Any water from any ancillary systems, such as the water purification system or water in the tanks in the Tank Farm, should also be processed in the same manner. Filter media and water not meeting the discharge limits will be managed as LLRW.

2.3.1.2.4 Removal of Activated Portions of the Reactor Tank Liner

In the rebuild completed in 1991, a liner was installed in the bottom half (Neutron Deck Level) of the reactor tank. Both the tank and the liner are made of an aluminum alloy (6061-T6). The aluminum liner may be removed prior to destructive dismantling of the bioshield. It has been assumed that the inner and outer liner will require disposal as LLRW. However, UB may elect to survey the newer liner material and only dispose of material as LLRW only in the activation region. Since the bioshield was poured around the completed reactor tank, it is not feasible to remove the original tank separately from the bioshield.

2.3.1.2.5 Removal of Reactor Bio-Shield, Dry Cell, and Hot Cell

After the activated core components (and possibly the reactor liner) have been removed the concrete associated with the reactor bioshield, dry cell, and hot cell will be removed by destructive methods, i.e. cutting and/or hydraulic shear. As described in Section 1.2, the bioshield, the dry cell, and the walls/ceiling of the hot cell were poured independently from the Containment Building floors and walls. The concrete waste from the demolition of the portions of these structures between the Neutron Deck and the bottom of the Gamma Deck will be disposed of as Class A LLRW. The remaining concrete waste from these structures is expected to be BSFR waste.

The destructive removal technique for the bioshield will be cutting by diamond wire, or equivalent, into pieces that can be lifted by the BMRC overhead crane having a maximum capacity of 10 tons. The dry cell and hot cell walls will primarily be removed by diamond wire cutting, but portions may be cut by a rotary saw or blunt force demolition.

Cut points will be verified by a New York Licensed professional engineer (P.E.) to ensure the structural integrity of the BMRC is not compromised. Concrete dust from destructive removal operations will be controlled and continuously monitored for industrial hygiene and radiological concerns.

2.3.1.2.6 Removal of HVAC system and other impacted systems

HVAC system removal may be necessary prior to Containment Building demolition. Radioactively impacted systems may be disposed of as LLRW if decontamination is not possible or cost effective. Minimally impacted systems will be disposed of as BSFR waste. Any systems that were maintained to support decommissioning activities will be de-energized at this stage to prepare for demolition of the Containment Building. The storm water and sanitary systems will be removed up to the junction point where they tie into other UB South Campus systems. The tanks used in the current and former liquid waste management systems will be disposed of as LLRW. The soil and gravel covering the tanks, along with the concrete enclosure and berms are expected to be disposed of as BSFR waste.

2.3.1.2.7 Waste Systems Removal

The waste systems consist mainly of the Tank Farm which lies in the subsurface adjacent to the containment building and extends towards the southwest of the containment building (see Figure 1-5) and the 10K Vertical Waste Holding Tank. A smaller 1,000-gallon holding tank in the subbasement that is in line with the 10K Vertical tank will also be removed. The removal of the waste systems will include the removal of tanks and the piping associated with both waste systems. The tanks will be size reduced by industry standard methods and disposed of as LLRW. Due to the expected residual radioactive material in the tanks, the tanks will be flushed prior to removal and local ventilation will be used unless the tanks are shipped offsite for processing at a radioactive waste processing facility.

2.3.1.2.8 Demolition of Containment Building

The Containment Building will be safely demolished after the work described in Sections 2.3.1.2.1 through 2.3.1.2.7 are complete, the activated concrete is removed, and building surfaces are remediated where necessary to meet the limits specified in Section 2.2.3.2. The demolition debris, i.e., concrete and rebar, will be sent to a local recycler or to a local C&D landfill. An exception to this is the bottom slab, i.e., the Neutron Deck floor, which is expected to be disposed of by BSFR due to impacts from the reactor leaks. Although cores through the Neutron Deck floor showed no indication of contamination, it is deemed too costly to perform the amount of surveying on the bottom of the concrete surface required for release to a recycler.

Industry standard demolition techniques will be used to demolish the BMRC. The concrete dust will be continuously monitored and controlled to applicable federal, state, and local limits. Air monitoring will also be conducted periodically for radiological parameters.

2.3.1.2.9 Demolition of Administration Building

The Administration Building will be demolished in conjunction with the Containment Building because there is no future need for the BMRC. It is anticipated that some minor decontamination will be required prior to demolition. It is also anticipated that if there are any radioactively impacted systems they will be removed prior to demolition. The building is mostly composed of concrete construction materials so a majority of the waste will be concrete rubble. The demolition debris will be disposed of in the same manner as the Containment building, i.e., the subbasement slab as BSFR and the remaining debris to a local recycling company or local C&D landfill.

2.3.1.2.10 Remediation of Soils and Bedrock

Characterization has not shown that there are radioactively impacted soils at the BMRC. The samples collected in the Room N16 were scrapings of dirt on the bedrock. It is possible that a small amount of soil around the waste tanks have elevated levels of radioactive materials. This material is to be remediated and characterized for proper disposal. In addition, removal of impacted bedrock is required in Room N16 due to historical releases. The amount of bedrock to be remediated is expected to be approximately one cubic yard with residual material to be removed by scabbling or chiseling of the bedrock.

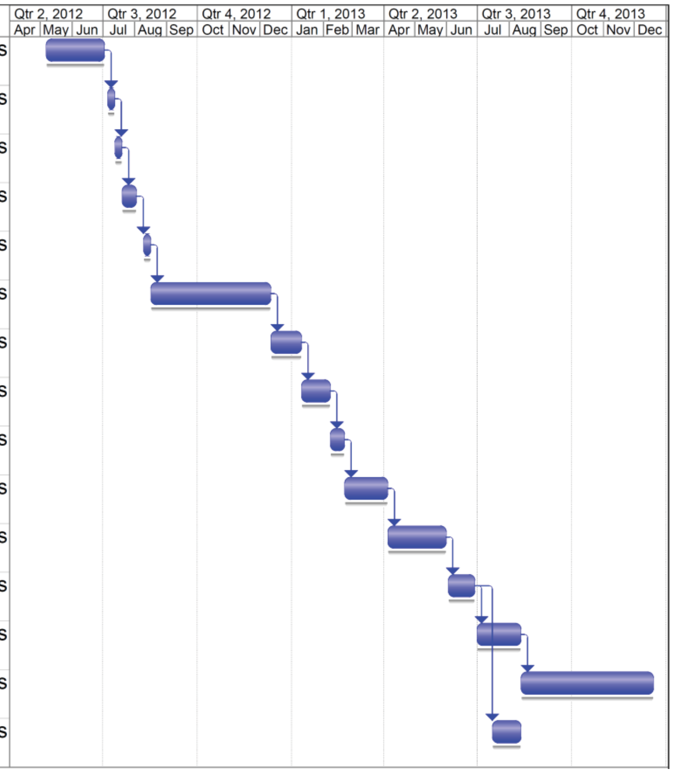
2.3.2 Schedule

The project duration, from DP approval to completion of the FSS, is approximately 15 months. The proposed decommissioning schedule is presented in Figure 2.1. Changes to the schedule may be made at the UB’s discretion to respond to resource management, availability of a radioactive waste burial site, interference with ongoing UB activities, ALARA considerations, and/or temporary on-site radioactive waste storage operations. The current project schedule is consistent with other recently completed university research reactor decommissioning projects.

The schedule includes activities from project planning through NRC review of the Final Status Survey Report and license termination. It is based on a 5-day work week with the number of work days listed for each activity.

Figure 2-1 Proposed Project Schedule

ID	Task Name	Duration
1	Project Planning	40 days
2	Mobilization, Orientation, and Training	5 days
3	Removal of the Operating Platform	5 days
4	Removal of Reactor Components	10 days
5	Drain Reactor Pool Water	5 days
6	Demolish Bioshield, Dry Cell, Hot Cell	80 days
7	Remove HVAC and Drain Lines	20 days
8	Decontaminate Building Surfaces	20 days
9	Remove Vertical 10K Waste Tank	10 days
10	Remove Tank Farm Waste Systems	30 days
11	Demolish Buildings	40 days
12	Final Status Survey/NRC Verification Survey	20 days
13	Develop FSS Report	30 days
14	NRC Review and License Termination	90 days
15	Site Restoration	20 days



2.4 Decommissioning Organization and Responsibilities

The decommissioning of the BMRC facility is under the technical supervision of the UB's Environment, Health and Safety (EH&S) Department who is responsible for assuring that decommissioning activities are conducted in a safe manner and within the requirements of the NRC License, this DP, the UB's Radiation Protection Program, and other state and federal laws and regulations. Personnel in the UB EH&S Department that are providing supervision of the decommissioning for licensing, radiation safety and industrial safety operated the BMRC before it was shutdown in the positions of Director, Operations Manager, Reactor Engineer, and RSO. The UB has appointed a project manager to oversee the decommissioning process.

The Organizational Chart in Figure 2-2 at the end of this section shows the lines of reporting at the UB BMRC. The Decommissioning Radiation Safety Staff reports to the RSO for radiation safety and licensing functions which are represented by the dashed reporting line. Additionally, the Decommissioning Industrial Safety Staff will report to the UB Reactor Engineer for safety and compliance reporting for industrial safety concerns.

2.4.1 University Project Manager

The following duties, at a minimum but not limited to, will be assigned to the UBPM.

- Selecting a Demolition Contractor in accordance with UB procurement guidelines.
- Overseeing the DOC and DC performance relative to the terms of their contract.
- Overseeing the DOC and DC performance relative to plans and procedures.

Minimum required qualifications and experience requirements for the UBPM are as follows:

- Bachelor's degree in Architecture, Civil, Electrical, Mechanical or Structural Engineering or related field AND five (5) years of construction experience which included one year of construction supervisory experience; OR,
- Nine years of progressively responsible construction experience which included one year of construction supervisory experience; OR,
- Any equivalent combination of experience, training and/or education approved by the UB Human Resources department.

2.4.2 BMRC Director

The BMRC Director has management responsibility and technical oversight for facility operations and radiation safety programs. The following duties and responsibilities are assigned to the BMRC Director, or designee:

- Ensuring that decommissioning activities are performed in compliance with applicable regulations and license conditions.
- Approval of plans and procedures required for decommissioning
- Reviewing and submitting to the Reactor Decommissioning Safety Committee (RDSC) needed changes and subsequent plans and procedures that do not change the original intent of the DP or result in an unreviewed safety question.
- Communicating with the NRC, New York State Agencies, and UB Administration.

Minimum required qualifications and experience requirements for the BMRC Director are as follows:

- Advanced degree (MS or PhD) in Nuclear Engineering or related discipline or equivalent experience and five (5) years experience in nuclear reactor operations and/or decommissioning.
- Familiarity with NRC License R-77, the Decommissioning Plan, the BMRC Radiation Protection Program, and with applicable federal and state regulations.
- Trained at the level required by the UB RP Program to be in possession of radioactive materials of the types known to be present at the licensed reactor site.

2.4.3 Radiation Safety Officer

The UB RSO is responsible for monitoring and overseeing radiological safety at the reactor and BMRC. The RSO has the responsibility and authority to stop any plan or activity that has the potential to result in an unacceptable radiological hazard. The following duties and responsibilities will be assigned to the UB RSO, or designee:

- Supervise the implementation of the UB radiation safety program by DOC radiation safety staff.
- Review and approve radiation safety procedures.
- Review and approve radiation work permits.

Minimum required qualifications and experience requirements for the RSO are as follows:

- Advanced degrees (MS or PhD) and/or certified in Health Physics with 10 years experience in radiation safety and health physics.
- Familiarity with NRC License R-77, NYS Radioactive Materials Licenses, the Decommissioning Plan, the BMRC Radiation Protection Program, and with applicable federal and state regulations.

2.4.4 Reactor Decommissioning Safety Committee

The function, responsibilities, and makeup of the RDSC are defined in the Technical Specifications (UB Research Reactor license R-77). Among those responsibilities are, but not limited to:

- Approval of plans required for decommissioning.
- Review and approval of proposed changes to the facility, procedures and Technical Specifications and DP.
- Determination of whether a proposed change to the DP would constitute an unreviewed safety question or a change in the Technical Specifications as required by 10 CFR 50.59, and review and approval of required safety analysis.

2.4.5 Design and Oversight Contractor

The DOC is responsible for the direct field management of the BMRC decommissioning and assisting the UB in the administration of the industrial and radiological safety programs. The duties assigned to the DOC PM include, but are not limited to, the following:

- Manage the safe and regulatory compliant implementation of the UB DP and FSS.
- Assist the UBPM in overseeing the DC performance relative to subsequent plans and procedures.
- Development, implementation, and associated costs of a Radiation Safety program compliant with 10 CFR 20.
- Development, implementation, and associated costs of an OSHA compliant Health and Safety Program.

- Review and approval of work plans and procedures necessary for the safe and compliant decommissioning of the BMRC.
- Assist the UB in acquiring applicable permits for radiological waste disposal and transportation.

The DOC will assign a PM with the following minimum required qualifications and experience:

- B.S. Degree and 10 years combined experience in decommissioning, project management, and radiation safety with at least five (5) of the years specific to nuclear reactor decommissioning.
- Experience in the decommissioning of university research and test reactors.
- Familiarity with applicable federal and state regulations, MARSSIM and NRC decommissioning guidance.

See Section 2.6 for additional criteria used in the selection of the DOC.

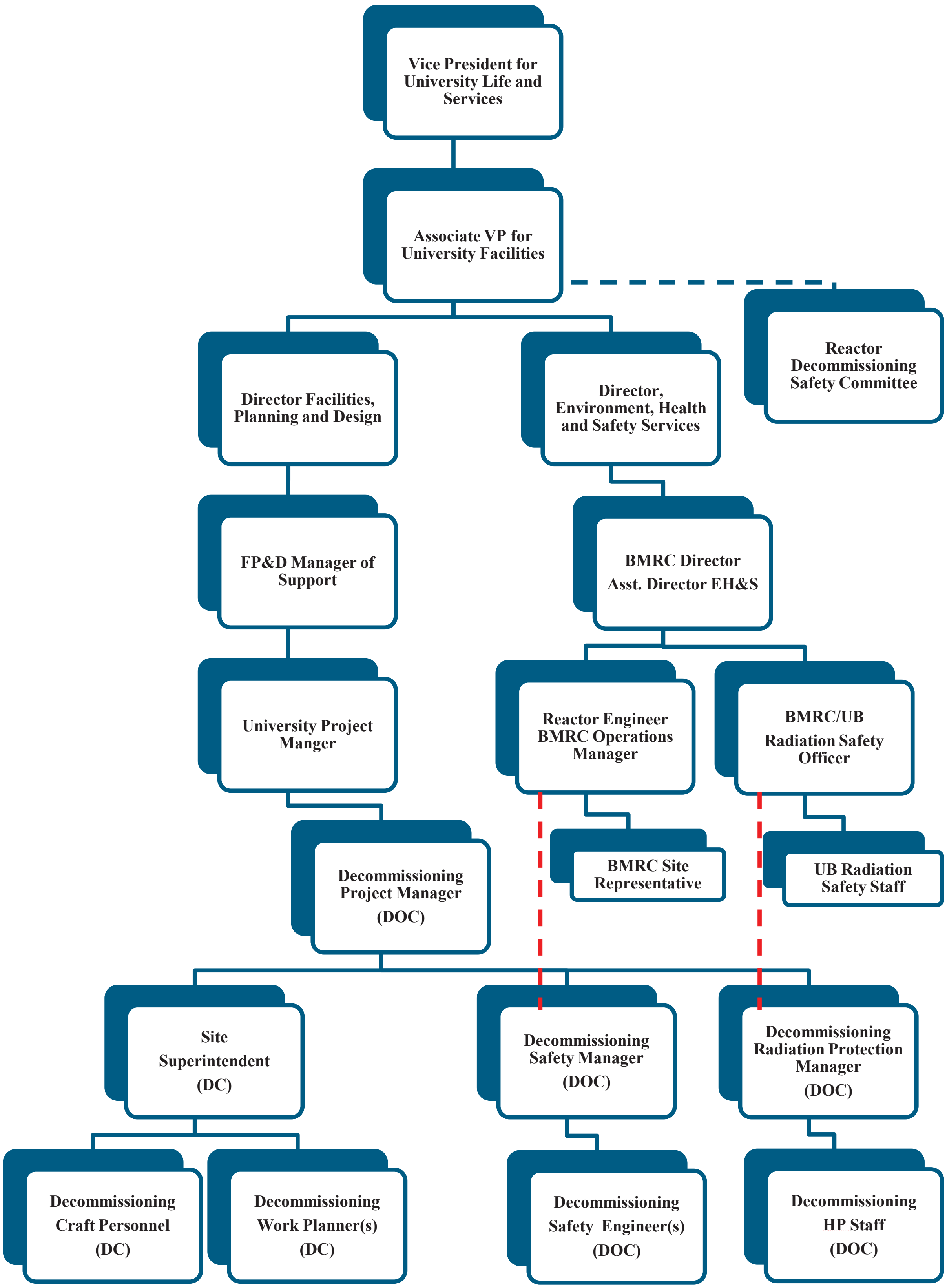
2.4.6 Demolition Contractor

The DC is responsible for the development, implementation, and associated costs of work plans and procedures for the safe and compliant demolition of the UB BMRC. The DC will report directly to the DOC Project Manager for implementation of the DP, but will be contracted directly to the UB. The DC is also assigned the following responsibilities:

- Development and safe implementation of work plans and procedures for the demolition of the BMRC.
- Compliance with the BMRC DP.
- Compliance with the BMRC Decommissioning Radiation Safety program.
- Compliance with the BMRC Decommissioning Health and Safety Program.

The minimum required qualifications and experience for the selection of a DC is discussed in Section 2.6.

Figure 2-2. Organizational Chart



2.5 Training Programs

2.5.1 General Site Training

A general training program will be designed and implemented by the DOC and approved by UB to provide orientation to project personnel and meet the requirements of 10 CFR 19, *Notices, Instructions, and Reports to Workers: Inspection and Investigations*. General site training will be required for personnel assigned on a regular basis to the D&D project. General site training will include but is not limited to:

- Project orientation, security, and access control – such as BMRC Operating Procedure # 56; BMRC Unescorted Access Training for Non-BMRC Personnel
- Introduction to radiation protection
- Quality assurance
- Industrial safety
- Emergency procedures
- Packaging and transport of radioactive materials

The following are examples of additional training that may be required:

- Radiation Worker Training will meet the requirements identified in the BMRC Decommissioning Radiation Protection Plan.
- Hazardous Waste Operations and Emergency Response (HAZWOPER) training will be required for personnel engaged in hazardous substance removal or other activities that potentially expose them to hazardous substances and health hazards.
- Respirator Training and Fit Testing will be performed according to the BMRC Decommissioning Respiratory Protection Program.
- Hazard Communication Training will be provided to personnel exposed to hazardous or potentially hazardous materials.
- Hearing Conservation Training will be provided on the effects of noise on hearing and the purpose, advantages, disadvantages, and attenuation of various types of hearing protective devices.
- Permit-Required Confined Space Entry Training will be required for personnel entering confined spaces.
- Lockout/Tagout Hazardous Energy Control Training for hazardous energy control.
- Trenching and Excavation Training for the purpose of determining the safety and stability of excavations.

For specific tasks that require state licensing or other special qualifications, the qualifications will be reviewed by the DOC PM or Safety Manager. If additional radiation safety training is required, it will be provided by the site RSO.

2.5.2 Radiation Worker Training

The reactor D&D operations will be managed by the DOC and performed by the DC. As such, the DOC will be responsible for the radiation worker training of BMRC decommissioning contractors and subcontractors in accordance with the requirements of 10 CFR 19 with the final approval of qualifications by the RSO. The Decommissioning Radiation Protection Manager (RPM) will be responsible for on-site radiation safety training of workers and verifying pervious training and qualification as approved by the RSO. The DOC's radiation safety training program will be administered by the RPM who will approve, along with the RSO, training materials and qualification of workers. The RSO will provide dose monitoring badges such as thermoluminescent dosimeters (TLD).

The minimum radiation safety training provided to any worker will include, but is not limited to the following subjects:

- Principles of radiation protection
- Radiation monitoring techniques
- Radiation monitoring instrumentation
- Emergency procedures
- Radiation hazards and controls
- Concepts of radiation and contamination
- Provisions of 10 CFR 19 and 20
- NRC license conditions and limitations
- Reporting requirements for workers
- Biological effects of radiation
- Radiation control zone procedures
- Radiation Work Permits (RWP)

A written exam will be required to demonstrate proficiency with the radiation worker training topics. Radiation worker training will also include a practical factors demonstration and evaluation. This evaluation will include a review of the following:

- Proper procedures for donning and removing protective clothing and equipment.
- The ability of the worker to read and interpret self-reading and/or electronic dosimeters (if used).
- Proper procedures for entering and exiting a controlled area, including proper frisking techniques

Persons who have documented equivalent radiation worker training from another site or employer within the previous twelve (12) months may be waived from taking the training but must take the written and practical factors examinations. Radiation Worker training will be refreshed on an annual basis or at the discretion of the RSO.

2.6 Contractor Assistance

The UB has already selected a qualified DOC for the design and oversight of the BMRC decommissioning based on a combination of the following items:

1. Response and demonstration of prior experience (five years minimum) on the tasks listed in the project scope.
   i. Project Management
   ii. Pre-Decommissioning Clean Out
   iii. Site Characterization
   iv. Decommissioning Plan Development
   v. NRC and New York State Regulatory Interface
   vi. Interference Removal
   vii. Decommissioning Drawings and Specifications
   viii. Decommissioning Oversight Activities
   ix. Final Status Survey planning and implementation
2. Prior University Research and Test Reactor Decommissioning Experience
3. Experience of Personnel Dedicated to the Project
4. State Agency/NRC Experience
5. Design / Technical Experience
6. Minority Status / Staffing

In addition to the DOC, the UB will select a qualified contractor to perform the physical demolition of the BMRC. In selecting the contractor, the UB will produce a request for proposal, which will define the qualifications and experience necessary for prospective DCs and subcontractors. Prior history and performance of the prospective contractor on non-power reactor or similar decommissioning projects will be used to help the UB select a qualified contractor to perform the facility D&D.

Qualification and experience requirements that will be used to evaluate the DC are to have at least 5 years prior experience in radiological site decontamination. Specific experience in the decontamination and demolition of test reactors, power reactors, and/or materials licensed sites will be required by the prime contractor. DC submittals of project descriptions, references, and other supporting information will be required prior to contract award. Specific DC project management documentation will be required in the areas of work plan development, training, QA, work management, reactor dismantlement and decontamination, waste packaging, waste shipping, work documentation, and supporting the preparation of the final decommissioning project report. The minimum expectation of the UB for a DC is verification of company experience in these tasks, proof of financial viability, and bonding capacity.

The UB will require the selected DC to support the project with an experienced Superintendent, supervisors, and Work Planner(s) with at least five years prior experience in radiological site decontamination and demolition with preference in test reactors in a college campus setting. Specific individual experience will be required in the areas of work plan development, training, QA, work management, reactor dismantlement and decontamination, waste packaging, waste shipping, work documentation, and supporting the preparation of the final decommissioning project report. The minimum expectation of the UB for a DC is verification of personnel experience in these tasks plus a commitment to provide experienced personnel for the duration of the project.

2.7 D&D Documents and Guides

This decommissioning plan was prepared using the guidance and format specified in Chapter 17 of NUREG-1537 (NRC 1996). The radiological criteria for license termination to allow unrestricted use will be as set forth in 10 CFR 20, Subpart E. The decommissioning project will also be administered according to the applicable section of the following regulations and regulatory guidance documents:

Code of Federal Regulations

10 CFR 19	Notices, Instructions and Reports to Workers; Inspections
10 CFR 20	Standards for Protection Against Radiation
10 CFR 30	Rules of General Applicability to Domestic Licensing of Byproduct Material
10 CFR 50	Domestic Licensing of Production and Utilization Facilities
10 CFR 51	Licensing and Regulatory Policy and Procedures for Environmental Protection
10 CFR 71	Packaging of Radioactive Materials for Transport and Transportation of Radioactive Materials Under Certain Conditions
29 CFR 1910	Occupational Safety and Health Standards
29 CFR 1926	Occupation Safety and Health Standards for Construction
49 CFR 170-199	Department of Transportation Hazardous Materials Regulations

NRC Regulatory Guides

1.86	Termination of Operating Licenses for Nuclear Reactors
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments
8.2	Guide for Administrative Practices in Radiation Monitoring
8.7	Occupational Radiation Exposure Records Systems
8.9	Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable
8.13	Instruction Concerning Prenatal Radiation Exposure
8.15	Acceptable Programs for Respiratory Protection

NUREG-Series Publications

1505	A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys
1507	Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions
1549	Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination, Draft
1575	Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)
1640	Radiological Assessments for Clearance of Materials From Nuclear Facilities
1756	Technology, Safety, and Cost of Decommissioning Reference Nuclear Research and Test Reactors
1757	Consolidated Decommissioning Guidance

Additional project-specific documents will be developed by the DOC and/or the UB prior to starting the D&D project. Such documents may include:

- RP and ALARA Plan
- Site Health and Safety Plan
- QAPP
- Waste Management Plan
- FSS Plan
- Specific Task Plans

3.0 PROTECTION OF WORKERS AND THE PUBLIC

3.1 Radiation Protection

The RPM, under the direct supervision of the RSO, will administer the Health Physics Program (also referred to as the Radiation Protection Program) specifically developed for the BMRC Decommissioning as described in Section 3.1.2. The DOC will supplement the radiation protection program with detailed plans and procedures specific to the radionuclides listed in Table 3-1. The RSO, the RPM and Decommissioning Health Physics (HP) Staff will be responsible for implementing ALARA principles; providing radiation worker training; establishing administrative-level occupational and public dose limits; monitoring personnel for occupational exposures; controlling exposures; providing and maintaining radiation monitoring equipment; performing radiation surveys and monitoring; and maintaining records and generating reports as necessary to comply with regulatory and licensing requirements.

Table 3-1 BMRC Radionuclides of Concern as determined in the Site Characterization Report

Radionuclide	Half Life (yr)	Emission	Area(s) of Interest
Ag-108m	4.18E+02	β, γ	Soil; Tank Water; SSCs
Am-241	4.32E+02	α, γ	Tank sediment
C-14	5.73E+03	β	Laboratory areas
Co-60	5.27E+00	β, γ	Soil; SSCs; Bioshield
Cs-137	3.01E+01	β, γ*	Soil; SSCs; Bioshield
Eu-152	1.36E+01	β, γ	Soil; SSCs; Bioshield
Eu-154	8.59E+00	β, γ	Soil; SSCs; Bioshield
H-3	1.23E+01	β	Soil; Bioshield and Tank Water
Ni-63	1.00E+02	β	Soil; SSCs; Bioshield
Pu-238	8.78E+01	α, γ	Tank sediment
Pu-239	2.41E+04	α, γ	Tank sediment
Pu-240	6.60E+03	α, γ	Tank sediment
Ra-226	1.60E+03	α, γ	Discrete sources
Sr-90	2.88E+01	β	SSCs; Ventilation systems; Soil

* γ emission from Ba-137m progeny

3.1.1 Ensuring ALARA Radiation Exposures

The DOC will prepare an RP and ALARA Plan that will incorporate provisions for minimizing occupational and public radiation exposures. This Plan will describe specific administrative and engineering controls that will be put in place during specific D&D project activities. Examples of administrative and engineering controls include limiting access to certain areas, mock-up training, use of remote-handling devices, temporary shielding, containment structures, portable HEPA filtered ventilation, and specialized protective equipment and respiratory protection.

3.1.2 Health Physics Program

The project Health Physics Program will be implemented under the authority of the RSO with the assistance of the DOC RPM. The Health Physics Program will satisfy the following commitments that should be established by the Radiation Protection Program:

- Implement the procedures defined in the Radiation Protection and ALARA Plan.
- Ensure radiological safety of the public, occupationally-exposed personnel, and the environment.
- Monitor radiation levels and radioactive materials.
- Control the distribution and release of radioactive materials.
- Maintain potential exposures to the public and occupational radiation exposure to individual within administrative limits and the regulatory limits of 10 CFR 20 and ALARA.
- Monitor personnel internal and external exposure in accordance with 10 CFR 20 requirements.

3.1.2.1 Project Health Physics Program - General

UB has procedures in place that will be implemented during the BMRC Decommissioning Project. Additional Health Physics procedures may be required and will be developed and approved in accordance with UB policy and procedure.

UB senior management is readily accessible to ensure timely resolution of difficulties that may need to be addressed prior to regularly scheduled meetings. The RSO, while organizationally independent of the project staff, is a direct report to the BMRC Director. He also has full authority to act in all aspects of protection of workers and the public from the effects of radiation. Conduct of the BMRC Decommissioning Project HP program will be evaluated according to UB policy.

3.1.2.2 Audits, Inspections, and Management Review

During Decommissioning Project work, aspects of the Project may be assessed and reported by the DOC's Quality Assurance Department, through audits, assessments and inspections of various aspects of decommissioning performance, including HP, as described in Section 1.2.4.3 Program Quality Assurance.

Audits of the BMRC Health Physics program are conducted in accordance with the requirements of 10 CFR 20. These audits will include aspects of the BMRC Decommissioning Project. Additional assessments or management reviews may be performed when deemed appropriate by the EH&S Director.

HP equipment and instrumentation suitable to permit ready detection and quantification of radiological hazards to workers and the public will be chosen to ensure the validity of measurements taken during remediation and formal release surveys. The selection of equipment and instrumentation will be based upon detailed knowledge of the radiological contaminants, concentrations, chemical forms and chemical behaviors that are expected to exist as demonstrated during radiological characterization, and as known from process knowledge of the working history of the BMRC. Equipment and instrumentation selection will also take into account the working conditions, contamination levels and source terms that are reasonably expected to be encountered during the performance of decommissioning work, as presented in this Plan.

The following sections present details of the equipment and instrumentation planned for use during the decommissioning. It is anticipated that through retirement of worn or damaged equipment/ instrumentation or increase in quantities of available components or instruments, that new technology will permit upgrades or, at minimum, like-for-like replacements. UB is committed to maintaining conformance to minimum performance capabilities stated in this Plan whenever new components or instruments are selected.

3.1.2.3 Criteria for Selecting Equipment and Instrumentation for Conduct of Radiation and Contamination Surveys and Personnel Monitoring

A sufficient inventory and variety of instrumentation will be maintained on site to facilitate effective measurement of radiological conditions and control of worker exposure consistent with ALARA, and to evaluate the suitability of materials for release to unrestricted use.

Instrumentation and equipment will be capable of measuring the range of dose rates and radioactivity concentrations expected to be encountered during the decontamination and decommissioning activities associated with the BMRC, including implementation of a final status survey.

Decommissioning HP staff will select instrumentation that is sensitive to the minimum detection limits for the particular task being performed, but also with sufficient range to ensure that the full spectrum of anticipated conditions for a task or survey can be met by the instrumentation in use.

Consumable supplies will conform to manufacturer and/or regulatory recommendation to ensure that measurements meet desired sensitivity and are valid for the intended purpose.

3.1.2.4 Storage, Calibration, Testing, and Maintenance of Health Physics Equipment and Instrumentation

Survey instruments will be stored in a common location under the control of BMRC Decommissioning HP Staff. A program to identify and remove from service inoperable or out-of-calibration instruments or equipment as described in HP procedures will be adhered to throughout the BMRC Decommissioning Project. Survey instruments, counting equipment, air samplers, air monitors and personnel contamination monitors will be calibrated at license-required intervals, manufacturer-prescribed intervals (if shorter frequency) or prior to use against standards that are traceable to the National Institute of Science and Technology (NIST) in accordance with approved calibration laboratory procedures, HP procedures, or vendor technical manuals. Survey instruments will be operationally checked daily when in use. Counting equipment operability will be verified daily when in use. The personnel contamination monitors are operationally tested on a daily basis when work is being performed.

3.1.2.5 Specific Health Physics Equipment and Instrumentation Use and Capabilities

Table 3-2 provides details of typical HP equipment and instrumentation planned for use in the BMRC Decommissioning Project. This list is neither inclusive nor exclusive.

Table 3-2 Health Physics Instrumentation

Instrument	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum Model 2221	Ludlum Model 43-68 Gas Proportional (126 cm^2 area)	Beta	Tc-99	Surface Static Measurements; Beta scan measurements
Ludlum Model 2360	Ludlum Model 43-68 Gas Proportional (126 cm^2 area)	Alpha/Beta	Th-230/Tc-99	Alpha/Beta static measurements
Ludlum Model 19	Internal NaI	Gamma	Cs-137	General area exposure rates
Ludlum Model 3030E	Ludlum Model 43-10-1 ZnS internal detector	Alpha/Beta	Th-230/Tc-99	Swipe/smear counting

3.1.2.6 Potential Sources of Radiation or Contamination Exposure to Workers and Public as a Result of Decommissioning Activities

Sources of radiation or contamination exposure were assessed by process knowledge, the HSA, radiological survey data, surveys performed during characterization, previous and current job coverage surveys, or daily, weekly and monthly routine surveys.

Worker exposure to significant external deep-dose radiation fields is possible during this project due to the level of activation of reactor components, the reactor bioshield, and the level of radioactive materials in the waste tank systems. Therefore, work precautions and ALARA techniques such as time, distance, shielding, and pre-job planning will be employed. Worker exposure to airborne radioactivity may occur during decontamination operations/work evolutions involving abrasives or methods that volatilize loose and/or fixed contamination.

Exposure of the public to external or internal radiation from this Decommissioning Project is possible due to the high dose rates of reactor components and the position of the Tank Farm, however, exposures to the public shall be kept below the limits specified in 10 CFR 20 and will be ALARA by using the Administrative and Engineering controls described in the following paragraphs.

The types of exposure controls used will consider the current state of technology and the economics of improvements in relation to the benefits. Control of potential sources of radiation exposure to workers and public as a result of decommissioning activities will be achieved through, but not limited to, the use of administrative and engineering controls.

Administrative controls consist of, but are not limited to:

- Administrative dose limits that are lower than regulatory limits
- Training
- Radiological surveys
- Physical barriers such as radiological warning rope/ribbon, in combination with radiological warning tape, lockable doors/gates as well as information signs and flashing lights or other applicable barriers may also be used

Engineering controls may consist of but are not limited to:

- HEPA ventilation/enclosures
- Protective clothing/equipment
- Containment(s)

3.1.2.7 Health Physics Policies for Contractor Personnel

Contractor personnel will be used during the BMRC Decommissioning Project and will be required, at the discretion of the RSO, to:

- Attend and complete appropriate radiation safety course
- Provide required exposure history information
- Read and sign an applicable RWP and comply with instructions
- Follow special instructions given by the HP

3.1.3 Dose Estimates

The total estimated occupational exposure to complete the BMRC Decommissioning Project is 6.7 person-rem. The dose estimate for decommissioning of the reactor was prepared using the individual work activity durations and work crew sizes, based upon the results of the characterization results to date and based upon recent experience in performing similar activities at the University of Washington and at the University of Arizona, combined with the DOC's experience at numerous other sites. Using these individual work activity durations, and work crew sizes and characterizations results, a dose estimate was generated for each activity. The doses from each activity were categorized and are provided by those categories in Table 3-3.

This estimate is provided for planning purposes only. Detailed exposure estimates and exposure controls will be developed in accordance with the requirements of the ALARA program during detailed planning of the decommissioning activities.

The primary doses expected to be received by D&D project workers will be from external exposure to activated metals and concrete, with little dose expected from internal exposure. External exposure will be monitored using whole-body and extremity TLDs, and electronic dosimeters. External exposures can be kept ALARA due to the availability of long reach tools, remote handling equipment, and the building's overhead crane. Additionally, there is plenty of floor space in the BMRC to maintain safe distances and to use shielding as necessary to keep exposures ALARA. Air sampling will be performed to assess the potential for airborne contaminants and internal doses will be monitored if they are expected to exceed 10% of the annual dose limits specified in 10 CFR 20. However, the committed effective dose equivalent (CEDE), the sum of the external and internal doses, is expected to be equal to the Deep Dose Equivalent (DDE).

The dose estimate to members of the public as a result of decommissioning activities is estimated to be negligible. This is because the area immediately surrounding the facility is under the control of the UB and because the BMRC is not within a high foot-traffic area of the UB South Campus. UB students, faculty and staff will be kept at a safe distance from the BMRC during decommissioning activities using temporary construction fencing.

Table 3-3 Project Dose Estimate

Task	Task Name	Time/Resources		Est. Dose (person – rem)
		Days	People	
1	Project Planning	40	7	0.04
2	Site Mobilization and Training	5	10	0.00
4	Removal of the Operating Platform	5	6	0.01
5	Removal of Reactor Components	15	8	2.88
6	Water Removal from the Reactor Tank	5	8	0.02
8	Removal of Reactor Bio-Shield, Dry Cell, and Hot Cell	80	8	2.56
9	Removal of HVAC and other impacted systems	50	8	0.16
10	Tank Farm Removal	30	8	0.96
11	D&D of Containment Building	20	10	0.02
12	D&D of Administration Building	20	10	0.02
13	Remediation of Soils and Bedrock	20	10	0.02
14	Perform Final Status Survey	20	9	0.01
15	Backfill and Site Restoration	20	8	0.00
			Total Estimated Project Dose	6.70

3.2 Waste Management

Decommissioning will require the handling of a relatively large volume of radioactive materials to reduce the residual levels of radioactivity to levels that allow for license termination and the release of the site for unrestricted use. Materials that are not decontaminated and/or released will be managed as radioactive waste. This section of the decommissioning plan presents the programs used to manage and control the management and disposition of solid, liquid and gaseous radioactive waste.

Radioactive wastes include neutron-activated materials, contaminated materials remaining in the containment building, tools, and disposable equipment and supplies that become contaminated during dismantling activities. Waste disposal costs are directly related to the activity, volume, and weight of the materials requiring disposal. Strategies for minimizing waste include: source reduction, reuse, decontamination, volume reduction, and waste stream segregation.

The DOC will be required to develop and implement a Waste Management Plan for the BMRC Decommissioning project. The Waste Management Plan will include detailed guidance for the characterization, sampling, classification, segregation, handling, packaging, manifesting, transporting and disposal of waste generated by the decommissioning. The plan will be submitted to the RSO and BMRC Director for review and approval prior to the start of any decommissioning work on site that has the potential to generate radioactive waste.

Waste volumes will be minimized through the application of industry-proven methodologies to ensure the segregations of contaminated and non-contaminated materials. These methodologies will include the establishment of radiological controls consistent with the health physics program and the implementation of good contamination control practices. Segregation categories may include: uncontaminated waste acceptable for land disposal or reuse, uncontaminated C&D wastes suitable for land disposal or recycle, Class A LLRW, and Class B activated components. Additionally, mixed wastes and non-radiological hazardous waste, if encountered, will be segregated from LLRW. Based on the site characterization and Activation Analysis of the reactor components, Class C radwaste is not present at the BMRC.

Except for Class B waste, disposal facility options for contaminated or activated materials currently available to BMRC include land disposal at the Energy Solutions facility in Clive, Utah; and mixed waste treatment by Energy Solutions or another qualified service provider, with subsequent disposal at Energy Solutions. BSFR using a bulk survey and disposal procedure that is specially-permitted in Tennessee is also an option for lightly contaminated materials, especially building materials.

Currently, there is no commercial Class B disposal option available for the Old Control Blades. The UB is working with the Nevada National Security Site (N2S2) to determine if a nexus exists between BMRC operations and the federal government in order to establish this as a disposal option. The UB shall properly manage the Class B waste until a disposal option is obtained.

Transportation of radioactive waste will be performed in accordance with applicable NRC and DOT regulations and the Waste Management Plan. Radioactive waste and material will be shipped either by truck, or by a combination of truck and rail. Shipments will be planned in a practical and efficient manner. Personnel with appropriate training and experience will be used to ensure the shipments comply with the BMRC license, applicable regulations, and the receiving site’s license. Packages, packaging, and labeling for radioactive materials and waste shipment will meet applicable regulations and requirements. Personnel involved in the packaging, preparation for shipment, and transportation of licensed material will be required to have training in accordance with 49 CFR 172 Subpart H. The RSO, or designee, shall certify all radioactive waste manifests.

3.2.1 Fuel Removal

There is no longer any reactor fuel, used or unused, at the BMRC. As previously stated, the reactor was permanently shutdown in June of 1994. Unused fuel was transferred to North Carolina State University in 1998, and used fuel was returned to the DOE at INEEL in 2005.

3.2.2 Radioactive Waste Processing

Decommissioning of the BMRC reactor will result in the generation of solid and liquid LLRW, mixed waste, and hazardous waste. Solid radioactive wastes include neutron-activated materials, materials with surface or volumetric residual radioactive materials, and soil. Liquid LLRW includes the water in the reactor pool, waste systems, and the associated piping as well as contaminated water generated during remediation activities. The DP does not contain provisions for gaseous radioactive waste because the reactor has been defueled and has been shut down for over 16 years; therefore, radioactive gases present during operations have long since decayed.

Handling, staging, and shipping of packaged radioactive waste will be performed in accordance with applicable regulations, 10 CFR 20.2006, *Transfer for Disposal and Manifests*; 49 CFR 100-177, *Transportation of Hazardous Materials*; disposal site waste acceptance criteria; and BMRC licenses and permits. Onsite radioactive waste processing will include waste minimization, volume reduction, segregation, characterization, neutralization, stabilization, solidification, and packaging. Wastes may be shipped to a licensed processing facility for survey and release or decontamination and release, or may be disposed of directly at a licensed facility. Each shipment of radioactive waste will be accompanied by a shipment manifest as specified in Section I of Appendix G to 10 CFR 20, *Requirements for Transfers of Low-Level Waste Intended for Disposal at Licensed Land Facilities and Manifests*. Radioactive waste generated from BMRC decommissioning activities will be manifest in a manner consistent with its waste classification.

3.2.3 Radioactive Waste Disposal

3.2.3.1 Solid

Solid waste generation will primarily be the direct result of the decontamination and dismantlement of the BMRC reactor components, activated and contaminated systems, and structures. The bulk of the radwaste is Class A with approximately 2 cubic feet of Class B waste and no Class C wastes based on site characterization data and activation analysis data for the reactor. Disposition paths for solid radioactive wastes include decontamination and free release, BSFR, and direct landfill disposal. Wherever possible, volume reduction strategies will be explored to reduce waste disposal handling, exposure, and cost; including offsite processing for volume reduction if appropriate.

The Class B Mixed Waste consists of the Old Control Blades made of nickel plated permalloy which is an alloy consisting of silver, indium and cadmium. Disposal sites licensed to accepted Class B waste are not available to the BMRC at the time this DP was developed. The UB and DOC are initiating the process with the N2S2 to determine if the Class B waste can be accepted for disposal. After determination is made for disposal at N2S2, or another site, the Waste Management Plan is to be revised to incorporate the waste acceptance criteria for the disposal site.

Irradiated reactor hardware may require size reduction to facilitate loading. Depending on the dose rate exhibited by irradiated hardware, it will be shipped either in a shielded Type A container, or it will be loaded into a High Integrity Container (HIC) or liner then placed in an approved, shielded shipping container (Type A or B as dictated by the waste characterization) for transport and subsequent disposal. Contaminated reactor system piping and hardware may be land disposed as LLRW. Pre-disposal volume reduction processing may be performed off site at a subcontracted processing facility if it is determined to be cost effective.

Activated or contaminated concrete removed in large sections may be packaged as Low Specific Activity (LSA) material in approved shipping containers for direct shipment to the licensed land disposal facility operated by Energy Solutions at Clive, Utah. Concrete may also be eligible for processing using an option such as the BSFR program offered in Tennessee, or potentially N2S2.

Mixed waste lead bricks will either be decontaminated and released, or processed to allow for direct land disposal.

3.2.3.2 Liquid

Decommissioning the BMRC reactor involves radioactively contaminated water primarily in the form of low-level radioactively contaminated water in the reactor pool and associated piping. Additional contaminated water may be generated during decommissioning operations (e.g. concrete cutting). These waters will be disposed by discharge to the public sewer system operated by the City of Buffalo; assuming the discharged liquid can be shown to meet the requirements for sewage disposal established by the BMRC license.

Pre-discharge treatment may include the use of existing or temporary filtration units or demineralizers, coupled with tanks to store processed water prior to discharge. After it has been verified that the stored processed water meets the allowable discharge limits, the water may be released. In addition to testing the stored water after treatment, effluent monitoring instrumentation may be used to monitor discharges of liquid effluent as required, and to demonstrate compliance with applicable regulations. Contaminated filter media will be disposed as LLRW, after onsite treatment to meet LLRW disposal facility waste acceptance criteria.

In the event that discharge to the sanitary sewer or onsite treatment is not feasible, the waters may be treated offsite. There are several licensed radioactive waste processors provide specialized services for volume-reducing or treating radioactive liquid waste, including demineralization, direct incineration, ground application, evaporation, and survey and release. BMRC may elect, or find it necessary, to transfer all or some of the liquid radioactive waste from decommissioning to a licensed waste processor based upon onsite treatment effectiveness, discharge permit limits, stakeholder input, or cost.

3.2.4 General Industrial Safety Program

Industrial safety and hygiene personnel, such as Certified Safety Professionals or Certified Industrial Hygienists, along with project management personnel, will be responsible for ensuring that the D&D project complies with applicable federal safety requirements and general safe work practices. These personnel will be provided by the DOC, but will report to the UB EH&S department. The DOC will prepare a site specific Health and Safety Plan (HASP) to document safety requirements and accident response procedures.

All personnel working on the D&D project will receive health and safety training in order to recognize and understand potential hazards and risks. Training requirements for subcontractors will be determined by the Decommissioning Safety Manager based on the specific task the subcontractor is performing.

The HASP will be reviewed and approved by the UB EH&S Department. The HASP will direct site activities necessary for ensuring that the reactor D&D project meets occupational safety and health requirements for protection of project personnel. The functional responsibility of the HASP will be to ensure compliance with the OSH Act of 1973. New York adopts federal OSHA standards by reference and enforces OSHA standards contained within 29 CFR 1910 and 29 CFR 1926 for General Industry and Construction Industry, respectively. The DOC assists the UB in oversight of the site HASP.

As a minimum, the HASP will include the following:

- Hazards assessment
- General site safety procedures
- A requirement for a daily site safety meeting
- Site inspection procedures
- Emergency response procedures
- Emergency contact telephone numbers
- Material Safety Data Sheets for hazardous materials present on-site
- Training requirements for specific activities such as permit-required confined space entry or hot work
- Local emergency medical information

3.3 Radiological Accident Analyses

Potential radiological accidents during reactor decommissioning were evaluated by determining reactor components and areas that contain the highest radioactive material inventory. The proposed decommissioning activities and methods in which radioactive material could be released to the work area or environment were considered. Since all special nuclear material has been removed decommissioning, the majority of the accidents discussed in the current license are not applicable. The accident identification process was supplemented by reviewing experiences at other non-power reactor decommissioning projects. The following radiological accidents were considered to present the highest potential consequences:

- Fire
- Spill liquid radwaste into the environment
- Release airborne contamination to the environment
- Transportation accident

The accidental dropping of an activated reactor component was also considered as a potential accident. However, because the more highly activated components are located under water, 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 most likely result in additional unplanned external exposures. There are no, and will not be, fissile materials located on site that could result in a criticality incident.

3.3.1 Fire

The consequences of a fire during decommissioning of the reactor were considered and are not significantly different than the consequences of a fire during reactor operations. Most materials are metals, concrete, or similar non-combustible materials. Although some torch cutting operations may be performed during decommissioning, the likelihood is low that a fire would start or that a fire could become intense enough to release radioactive material.

Dry Active Waste (DAW) will be collected and packaged, to limit the volume of DAW available for consumption by fire and lower the potential for a fire to consume additional waste collections. The UB will implement a routine fire loading inspection program during decommissioning activities. DAW would have very low quantities of radioactive materials; therefore, the radioactivity is not high enough to result in a significant release in the case of a fire.

3.3.2 Spill Contaminated Water

The consequences of a spill during decommissioning of the reactor were considered and are not significantly different than the consequences of a spill during reactor operations. The spilling of contaminated water could occur during pool water pumping or liquid removal operations from the waste systems. Hoses could leak or break, resulting in an uncontrolled release. To mitigate the extent of such releases, processes involving contaminated liquids will only be operated with personnel present. Personnel will watch for leaks and spills and respond by shutting down the activity. This will not allow for additional water to leak from the system. In addition to the use of secondary containments or berms, a spill kit will be readily available to respond to any incidents.

As evidenced in prior reactor leaks, contaminated water does not infiltrate the subsurface soils or bedrock. Should radioactive liquids be spilled, the soils and/or building materials can be remediated with little to no impact to the public or the environment.

3.3.3 Release of Airborne Contamination

The consequences of an airborne contamination event during decommissioning of the reactor were considered and are not significantly different than the consequences of an airborne contamination event during reactor operations. An uncontrolled release of airborne radioactivity could occur during cutting and demolition activities involving contaminated or activated materials, such as removal and segmentation of reactor components, or removal of tank steel and concrete. The primary method of cutting the activated bioshield is a wet process; therefore, eliminating most if not all of the airborne hazard. Such activities may take place inside temporary containment structures equipped with local HEPA filter ventilation systems. Additionally, non abrasive cutting methods, i.e., hydraulic shears, will be used where possible to limit abrasive dusts and/or activated metal fragments where feasible.

Temporary containment systems with local HEPA filter systems will likely vent to the BMRC rooms or tie into existing building ventilation. A failure in the HEPA filter system could result in the uncontrolled release of airborne radioactive materials. A Continuous Air Monitor will be used to monitor effluent air and will be set to alarm at 10% of the allowable effluent criteria. Operations inside the containment structure will immediately stop and an evaluation conducted to determine the nature of the alarm.

While the actual concentrations of airborne radioactive materials are unknown at this time, the scenario is similar to accident analyses contained in the current BMRC Technical Specifications. Safety management operations (standard engineering and administrative controls) are sufficient for protecting against such accidents.

3.3.4 Transportation Accidents

Various forms and quantities of radioactive waste will be shipped from the reactor during the D&D project. The dose consequence from transportation accidents could be higher than the contamination accident scenarios described above because high-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. The transportation accident risk is further reduced by using local disposition methods for the concrete, i.e., a local C&D landfill or recycling.

4.0 PROPOSED FINAL STATUS SURVEY PLAN

An FSS Plan will be submitted to the NRC for approval following submittal of this DP and prior to implementation of the FSS. The following sections describe the methodology and QA requirements to be implemented during the FSS. The DOC is responsible for the planning and implementation of the FSS.

4.1 Survey and Sampling Approach

The reactor and support facilities will be removed prior to site release. Consequently, the Final Status Survey (FSS) will include only the exposed soils and bedrock surfaces in areas covered by license R-77. The expected condition of the area post remediation will be exposed subsurface soils and bedrock. The non-porous bedrock, being similar to a concrete structure, will be surveyed using survey methodologies for surface release of concrete and the results compared to the NRC surface screening values. Prior to the FSS on bedrock material, the loose dirt and debris is to be removed to ensure there is no interference with the survey measurements. The BMRC Site Characterization report (ENERCON 2011) demonstrates that the subsurface soils are not volumetrically contaminated; therefore, the exposed subsurface soils will be surveyed to surface soil screening criteria to prove that the surface of these exposed soils were not contaminated during the building demolition process. There will be no BMRC buildings or structures that will require a FSS.

The FSS will be developed following the guidance provided in NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NRC 2000) to demonstrate compliance with the release criteria provided in Section 2.2.3. The MARSSIM process emphasizes the use of data quality objectives (DQO), proper classification of survey areas (survey units), a statistically-based survey and sampling plan, and an adequate quality assurance/quality control (QA/QC) program.

The FSS will be performed in accordance with an FSS Plan by trained DOC technicians experienced in performing a MARSSIM FSS. The technicians will follow written procedures regarding surveys and sampling, sample collection and handling, chain-of-custody, and recordkeeping. The FSS Plan will define sampling locations, required analysis, and survey types. Any additional release criteria set forth by the UB or NYS will be contained within the FSS Plan which will direct surveys or sampling efforts required to demonstrate compliance with such criteria.

The FSS may include surface gamma surveys using sodium-iodide (NaI) gamma scintillation detectors and gas flow proportional detectors. Surface soil samples will be collected using either a random-start grid pattern or randomly generated locations as appropriate commensurate to the classification of the survey area. Soil samples will be analyzed for contaminants of concern using standard analytical methods including liquid scintillation counting for hard-to-detect beta-emitting radionuclides (i.e., Carbon-14 and tritium) and gamma spectroscopy for gamma-emitting radionuclides.

4.2 Data Quality Objectives

The object of the FSS is to demonstrate that the radiological conditions of the reactor site satisfy the decommissioning criteria provided in Section 2.2.3. The DQO's in the MARSSIM survey approach will provide a 95% confidence level for the false negative (Type I error) in demonstrating that the site meets the criteria. Typically, the false positive (Type II error) will also be defined as a 95% confidence level, but may be modified to apply to a specific situation. Therefore, the Type I decision error will be 5-percent. The decision error rates are used in determining the required number of samples necessary in each survey unit as well as the required minimum number of data points used for the final nonparametric statistical test performed to evaluate contaminant concentrations in the survey units against release criteria. DQOs, will be fully described in the FSS Plan and will include limits on the sensitivities of survey and analytical methods.

The QAPP will incorporate standard regulatory and industry measures applicable to the FSS. The QAPP will be reviewed and approved by the Reactor Committee.

4.3 Identification and Classification of Survey Units

4.3.1 Method for Classification

Survey units are classified based on contamination potential according to the methods described in the MARSSIM. In general, there are two overall classifications, non-impacted and impacted. Non-Impacted areas have no reasonable potential for residual contamination because there was no known impact from facility operations.

Impacted areas may contain residual radioactivity from facility operations. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. Class 1 areas have the greatest potential for residual activity while Class 3 areas have the least potential for impacted areas. Each classification will typically be bounded by areas classified one step lower to provide a buffer zone around the higher class. Exceptions occur when an area is surrounded by a significant physical barrier, such as a wall, that would make transport of residual activity unlikely from one area to the adjacent area. In such cases, each area will be classified solely on its own merit using the most reliable information available. The class definitions provided below are from Section 4.4 of the MARSSIM.

Class 1
Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys). Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material high specific activity. Note that areas containing contamination in excess of the $DCGL_w$ prior to remediation should be classified as Class 1 areas.

Class 2
These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_w$. To justify changing an area's classification from Class 1 to Class 2, the existing data (from the HSA, scoping surveys, or characterization surveys) should provide a high degree of confidence that no individual measurement would exceed the $DCGL_w$. Other justifications for this change in an area's classification may be appropriate based on the outcome of the DQO process. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (e.g., process facilities), 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of some buildings or rooms subjected to airborne radioactivity, 5) areas where low concentrations of radioactive materials were handled, and 6) areas on the perimeter of former contamination control areas.

Class 3
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 $DCGL_w$, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

The size of a survey unit is directly affected by its classification. Section 4.6 of MARSSIM provides suggested sizes for survey units. However, as stated in MARSSIM, the suggested survey unit sizes were based on a finding of reasonable sample density and consistency with commonly used dose modeling codes. MARSSIM limits the size of a survey unit. Table 4-1 summarizes MARSSIM recommendations for survey unit sizes based on their type and classification.

Table 4-1 Recommended Survey Unit Sizes

Classification	Minimum/Maximum	
	Buildings	Open Land
Class 1	10-m^2 / 100-m^2	100-m^2 / 2,000-m^2
Class 2	100-m^2 / 1,000-m^2	2,000-m^2 / 10,000-m^2
Class 3	1,000-m^2 / No limit	10,000-m^2 / No limit

4.3.2 BMRC MARSSIM Classifications

Following remediation and demolition of the buildings, there will be no structures remaining that will require FSS. Using the size restrictions in Table 4-1, the exposed bedrock may include only one Class 1 survey unit. The exposed subsurface soils surrounding the Class 1 survey unit will likely be designated as a single Class 2 survey unit to a maximum survey unit size of 10,000 m^2. The remaining surface soils will be designated as a Class 3 survey unit.

4.4 Data Collection

Survey methods are applied differently depending on the data requirements of a survey area. For example, removable activity measurements provide little, if any, benefit when attempting to assess the radiological conditions in an excavation. Conversely, assessing a building surface via volumetric sampling would provide the necessary data, but at great costs of time and money. This section will discuss the steps necessary to strike a reasonable balance between data needs and ease of survey performance based on the data needs of the survey area.

4.4.1 Buildings, Equipment, and Components

Buildings, equipment, and components are not expected to remain after license termination.

4.4.2 Soils

To demonstrate soil areas meet the appropriate release criteria, gamma scans and volumetric samples are required as demonstrated in the following two sections.

4.4.2.1 Surface Soil Scans

Soil areas require gamma scan measurements as part of the FSS process at appropriate coverage rates and speeds that ensure hot spots and/or hot particles are detected. The coverage rates and speeds will be described in the FSS plan and subsequent report to ensure adequate MDCs for hot spots and/or particles.

4.4.2.2 Volumetric Samples

Volumetric samples are required to demonstrate a soil area meets the appropriate release criteria. In lieu of volumetric samples, soil areas may receive direct measurements using in-situ gamma spectroscopy, as equipment and trained personnel are available. Volumetric sampling differs slightly depending on the situation for which the sample is desired. The required quantity of volumetric samples for an open land survey unit is a calculated value that is discussed in MARSSIM.

4.4.3 Bedrock

The Neutron Deck of the BMRC was poured directly on the bedrock with no leveling material (e.g., gravel or sand) utilized between the bedrock and concrete. The bedrock, described in Section 8.0, is non-porous and similar to a concrete slab. Radiological impacts to the bedrock are expected to be limited to the surface and after the concrete is removed from the bedrock, loose material such as soil and concrete dust will be removed prior to the FSS.

4.4.4 Bedrock Surface Scans

The bedrock surfaces, after cleaning, will be scanned using beta instrumentation such as a gas flow proportional detector. The coverage rates and speeds will be described in the FSS plan and subsequent report to ensure adequate MDCs for hot spots and/or particles. Volumetric samples may be collected to verify residual radioactive materials are only on the surface of the bedrock.

4.5 Data Evaluation

Data evaluation is performed on FSS results for individual survey units to determine whether the survey unit meets the release criteria. Appropriate tests will be used for the statistical evaluation of survey data. Tests such as the Sign test and Wilcoxon Rank-Sum (WRS) test will be implemented using unity rules, surrogate methodologies, or combinations of unity rules and surrogate methodologies, as described in the MARSSIM and NUREG-1505 Chapters 11 and 12.

If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign test will be used. If background is a significant fraction of the DCGL the WRS test will be used. It is anticipated that the sign test will be the only statistical test applied to the collected data because of the small fraction of the DCGL that background radionuclides will contribute.

4.6 Final Status Survey Report

The FSS report will provide a summary of the survey results and the overall conclusions to demonstrate that the BMRC site meets the radiological criteria for release. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results will be included in the report. The level of detail is to be sufficient enough to clearly describe the FSS program and to certify the results. The basic outline of the final report will be similar to the following:

- Overview of the Results
- Discussion of Changes to FSS
- FSS Methodology
    - Survey unit sample size
    - Justification for sample size
    - Survey Instrument MDCs
- FSS Results
    - Number of measurements taken
    - Survey maps
    - Sample concentrations
    - Statistical evaluations, including power curves
    - Judgmental and miscellaneous data sets
    - Elevated Measurement Comparisons (if used)
- Conclusion for each survey unit

5.0 TECHNICAL SPECIFICATIONS

This DP is an addendum to Revision 27 of the BMRC Technical Specifications. Revision 27 has been submitted to the NRC via a license amendment request to perform decommissioning activities in accordance with this plan. The Technical Specifications include a license condition requiring an FSS Plan to be submitted and approved by the NRC prior to conduct of license termination surveys.

6.0 PHYSICAL SECURITY PLAN

The regulations in Section 73.67(c)(1) of Part 73 require facilities to maintain a physical security plan when they possess special nuclear materials of moderate strategic significance or 10 kg or more of special nuclear material of low strategic significance. There is no nuclear fuel remaining at the BMRC; therefore, a part 73 physical security plan is not required.

A site security plan is maintained to comply with the regulations in Subpart I, *Storage and Control of Licensed Material*, of 10 CFR 20 are applicable to the remaining radioactive materials possessed by the UB. Licensed materials in storage are secured from unauthorized access or removal. Additionally, licensed materials not in storage will be under the control and constant surveillance of authorized personnel.

7.0 EMERGENCY PLAN

Overview

BMRC ceased operation in 1994, with the used reactor fuel returned to the DOE in 2005 and unused reactor fuel shipped in 1998. Current activities in this building are minimal, with concentration on maintaining surveillance, monitoring of environmental conditions and security of the facility. During decommissioning of the site, management of significant emergency incidents by the UB will be accomplished through implementation of a response framework as specified by Federal Emergency Management Agency (FEMA) using the Incident Command System (ICS). The UB is committed to an All Hazard Emergency Management approach, which designates areas of responsibility and defines the administrative framework to respond to all emergency incidents.

Actions

The start of the decommissioning work raises the level of concern regarding response to an incident at the BMRC. Even though research has ended, the fuel returned to the DOE, and the level of risk reduced, the importance of protecting the surrounding community remains a priority to the UB.

Without research activity or a source of fuel, the existing equipment is the highest risk at the BMRC. Access to the facility is limited and daily monitoring of activities at the BMRC is conducted under the supervision of the UB and DOC project management team, minimizing the risk potential. This team carefully monitors the status of activities and implements controls to minimize exposure. The Site Security Plan, along with the UB police monitoring capabilities, maintains a safe environment and protects the surrounding community.

The UB Communications Department plays an important role in response to any emergency at BMRC, providing accurate information to all media resources. Utilizing the Crisis Communication Plan from the Division of External Affairs, this information flow will maintain community confidence and supporting the UB's response to all incidences at the BMRC.

Utilization of the ICS, under FEMA guidelines, will provide a quick response mitigating the incidents of concern at the BMRC. Unified command under ICS format will afford the UB the opportunity to bring experts from multiple agencies together to mitigate impacts and allow for a smooth transition back to normal activities at the BMRC until decommissioning.

8.0 ENVIRONMENTAL REPORT

The Decommissioning ER was prepared in accordance with the guidance provided in Chapter 6.0 of the NRC Office of Nuclear Material and Safety and Safeguards' (NMSS) NUREG-1748, Environmental Review Guidance for Licensing Actions Associated with NMSS Programs (NRC 2003b). The ER is provided in Appendix C and is provided for use by the NRC to conduct its environmental assessment of the potential impacts of this proposed DP in accordance with the National Environmental Policy Act (NEPA) of 1969. NEPA requires Federal agencies, as part of their decision-making process, to consider the environmental impacts of actions under their jurisdiction. The NRC's NEPA requirements are provided in 10 CFR 51.

9.0 CHANGES TO THE DECOMMISSIONING PLAN

Following NRC review and approval of the DP, the DP will be incorporated as an amendment to license R-77. Minor changes to the DP that do not change the original intent of the DP and which do not involve an unreviewed safety question may be approved by the Operating Committee, a sub-group of the RDSC, as defined in the BMRC Technical Specifications and this section. The licensee may make changes to the DP without prior NRC approval provided the proposed changes do not:

i. Require Commission approval pursuant to 10 CFR 50.59;
ii. Use a statistical test other than the Sign test or WRS test for evaluation of the FSS;
iii. Increase the radioactivity level, relative to the applicable derived concentration guideline level, at which an investigation occurs;
iv. Reduce the coverage requirements for scan measurements;
v. Decrease an area classification (i.e., impacted to unimpacted; Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3);
vi. Increase the Type I decision error;
vii. Increase the DCGLs and related MDCs (for both scan and fixed measurement methods); or
viii. Result in significant environmental impacts not previously reviewed.

If a significant change to the DP is required, the BMRC RDSC will apply the criteria identified in 10 CFR 50.59 (March 2001) as it applies to non-power reactors in decommissioning. 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)

If the RDSC determines 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.

Changes to the DP are to be listed in the Summary of Changes table at the beginning of this DP with special demarcations in the margins next to the revised text. A report of changes made to the DP without NRC approval is to be maintained for review by the NRC during routine decommissioning inspections. Records of all changes to the DP are maintained until license termination.

10.0 REFERENCES

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6. NRC. 1996. *Guidance for Preparing and Reviewing Applications for Licensing of Non-Power Reactors.*NUREG-1537, Part 1. February.

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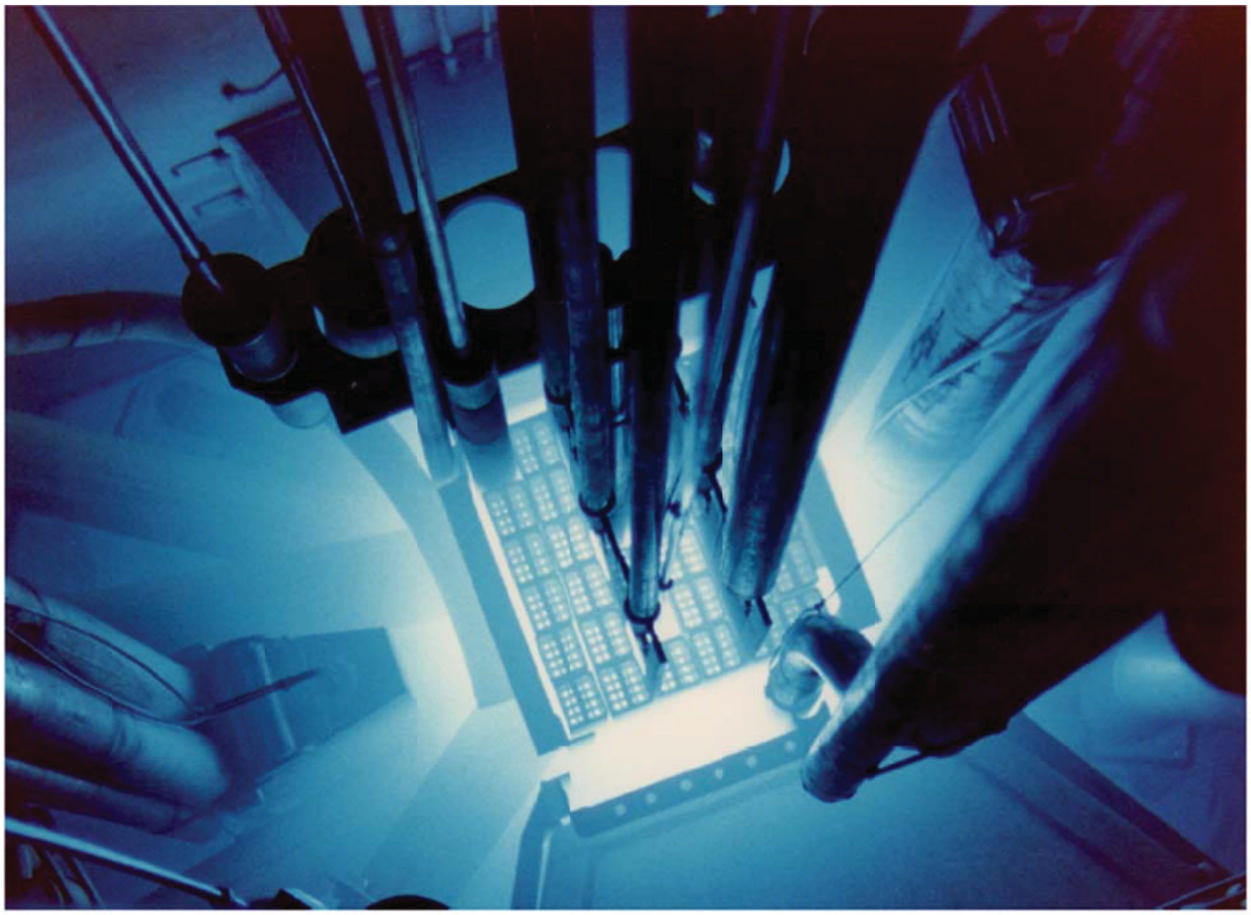
10. NRC. 2003. *Environmental Review Guidance for Licensing Actions Associated with NMSS Programs.* Office of Nuclear Material and Safety and Safeguards. NUREG-1748

11. NRC. 2011. *Standards for the protection against radiation* Title 10, Part 20, Code of Federal Regulations.

APPENDIX A – ACTIVATION ANALYSIS AND COMPONENT CHARACTERIZATION

The STATE UNIVERSITY of NEW YORK at BUFFALO
BUFFALO MATERIALS RESEARCH CENTER REACTOR
ACTIVATION ANALYSIS
AND COMPONENT CHARACTERIZATION

Report 10-102D-RE-138
Revision A



May 2011

WMG Project
10-102D
Prepared by:

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FOREWORD

This report summarizes the activation analysis work performed by WMG, Inc. to support The State University of New York (SUNY) in the preparation of the decommissioning plan for the University at Buffalo's Reactor. This work was performed by WMG, Inc. under SUNY Contract T-000401.

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1.0 INTRODUCTION

In October 2010, The State University of New York at Buffalo (SUNY Buffalo) engaged WMG to perform an activation analysis and segmentation planning for the University's Buffalo Materials Research Center (BMRC) reactor. This report summarizes the preliminary results of WMG's demonstrated and approved analytical methodologies used to complete this phase of work. The preliminary results present the activation product content within the activated reactor components.

The results presented in this report will require further normalization after final surveys have been obtained with adequately scaled measurement equipment. The reported results are considered preliminary until all (including the control rod shrouds and lower core plate) the calculated activation activities are normalized to measured dose rates, and surface contamination, which covers the surfaces of the reactor components, is evaluated.

Based on WMG's extensive experience, the preliminary activation results are conservative and provide a reasonable basis for decommissioning planning. In addition, surface contamination typically does not dictate the NRC waste classification of highly activated reactor components. However, the surface contaminant activity (which can contain activation products, fission products and transuranics) can be significant for reactor components with relatively low activation activities and must be considered prior to classification as Low Level Radioactive Waste (LLRW) for disposal in accordance with 10 CFR Part 61.

The relevant activation product radionuclide concentrations determined from these results are used to classify the activated reactor components in accordance with 10 CFR Part 61. The activation product scaling factors are also presented in this report and are used to quantify the hard-to-detect radionuclide concentrations important to final classification under 10 CFR Part 61. The reported scaling factors are Co-60 based and considered final since the ratios of impurities in the initial material compositions are assumed to be fixed and consistent throughout the components, as shown in Appendix A. Therefore, the individual radionuclide concentrations from neutron activation are directly proportional to the Co-60 concentration.

The analysis methods used for this project are discussed in Section 2. The component specific results in terms of estimated activation activity and 10 CFR Part 61 classification as of June 1, 2011 are presented in Section 3. The packaging requirements for the activated components are discussed in Section 4. References are presented in Section 5.

Historical summary

The BMRC is a light water pool type reactor constructed by the University that was originally fueled with Materials Test Reactor (MTR) fuel from 1961 to 1963. The BMRC was reconfigured and used AMF/Hittman/IRM Pulstar fuel from 1964 through shutdown. Active core height is approximately 24-inches and core mid-plane is located approximately 16-inches above the top face of the Grid Plate. Many reactor core items were modified and re-used during the reconfiguration. The core is configured to accept a 6 x 6 array of fuel assemblies but did not operate with each grid location fueled. The BMRC operated with relatively high research reactor flux levels approaching 10^{13} n/cm^2-s. Water cooling, experimental facility, and reactor tank configurations were also changed during the operational history of the facility without substantial changes to the reactor core components.

The facility operating history forms the basis for the activation analysis presented in this document. Facility operating records through June 1994 are included in this report.

2.0 NEUTRON TRANSPORT AND ACTIVATION METHODOLOGY

WMG used University supplied neutron flux levels at locations of interest. A summary of the provided neutron flux data is included in Section 2.2. The fluxes and initial material compositions shown in Appendix A are used as inputs to the ORIGEN2.2 (Ref. 5) computer program to perform activation analysis on individual components.

A similar activation analysis methodology has been used by WMG to support decommissioning activities at 18 domestic power and research reactors. The methodology has been refined and benchmarked over the years and has been found to provide reasonable characterization results for the components of interest.

Several assumptions were made for the activation analysis, based upon information provided by the University or available from similar projects performed by WMG in the past. The activation analysis and dose profiling at the Brookhaven National Laboratory High Flux Beam Reactor indicated Co-60 production in 6061 Aluminum components. Similar effects to a lesser degree are present at the BMRC. These assumptions are stated in the subsequent sections describing each portion of the activation analysis methodology.

2.1 Input Parameters

Use of the ORIGEN2 computer program requires accurate modeling of the reactor components in terms of their physical characteristics and neutron exposure histories.

2.1.1 Material Compositions

Elemental material compositions are required inputs for the ORIGEN2 computer program. A summary of the initial material compositions is included in the appendices. Detailed drawings with material heat data were lacking, and WMG used drawings and technical descriptions to discern materials of construction. The Grid Plate and Control Rod Shrouds are fabricated primarily of 1100 grade Aluminum with integral stainless steel fasteners. Aluminum components outside the core were confirmed or assumed to be fabricated from less expensive and more common 6061 grade Aluminum. All stainless steel was assumed to be type 304. Nickel clad, silver-indium-cadmium Control Rod material was well documented. Reactor grade graphite was used in the thermal column. The reactor tank is constructed of a 6061 Aluminum tank and liner interior to a poured 225 pcf concrete pit.

The initial material composition data for SS304 and concrete are taken from NUREG CR 3474 (Ref. 6). The graphite composition is compiled from work performed for the Brookhaven National Laboratories' research reactors. The Aluminum Alloys 6061 and 1100 are compiled from work performed at the University of Michigan and the NASA Plumbrook facility. Material composition data for the Control Rod Blades was provided by the University,

2.1.2 Operating History

A detailed operating and power history was provided by the University for the BMRC. The information was provided in terms of MW-hrs produced on a monthly basis from start-up (July 1961) until shutdown (June 1994). The cumulative power produced over the lifetime of the reactor through June 1994 was 68,543.4 MW-hrs.

An irradiation power history was compiled from this information for each calendar year of operation, which is considered a "cycle of operation" in this analysis. The capacity factor for each "cycle" is determined using the maximum MWs/day (cycle 17, 1987). Significant down times are accounted for in the operating history. The operating history is summarized in Table 2-1 below.

Table 2-1
BMRC Operating History

					Total		
Origen Power History:			"Cycle"	Decay	Year	MWs per	Capacity
YearCycle	Start	Stop	Length	Days	MWs	day**	Factor
1961 (7-12)	07/01/61	01/01/62	184		413.3	2.2	0.074
1962	01/01/62	01/01/63	365		1,295.6	3.55	0.118
1963	01/01/63	06/30/63	180	641	753.0	4.18	0.139
1965	04/01/65	01/01/66	275		3,255.8	11.84	0.392
1966	01/01/66	03/31/66	89	6485	1,085.3	12.19	0.404
1984	01/01/84	01/01/85	366		10,598.0	28.96	0.960
1985	01/01/85	01/01/86	365		10,922.7	29.93	0.992
1986	01/01/86	01/01/87	365		10,117.0	27.72	0.919
1987	01/01/87	01/01/88	365		11,010.4	30.17	1.000
1988	01/01/88	01/01/89	366		9,894.4	27.03	0.896
1989	01/01/89	11/30/89	333	914	9,025.8	27.10	0.899
1992	06/01/92	12/31/92	213	366	3,684.6	17.30	0.573
1994 (1-3)	01/01/94	03/31/94	89	62	1,580.8	17.76	0.589
1994 (6)	06/01/94	06/30/94	29	6083	0.8	0.03	0.001
Survey:	2/24/2011						

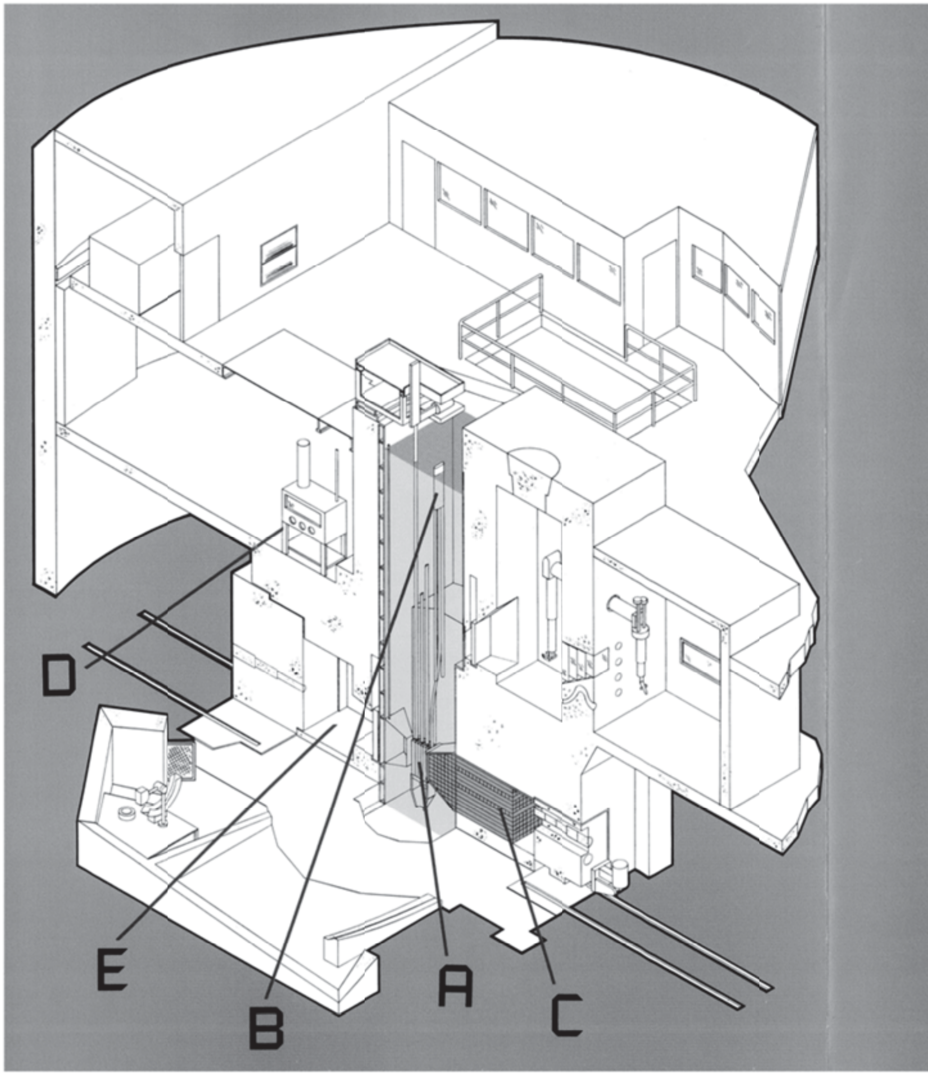
Notes:

- *Time Steps (i.e., "Cycles") are established based on the applicability of actual operating history at BMRC vs Origen computer code*
- *** MWs/day is necessary due to the inconsistent Cycle lengths*
- *Capacity Factor = [MWs/day ---- Cycle(i)] / (Max. MWs/day)*

2.2 Neutron Flux Data

A summary of the neutron flux data used in the activation analysis is shown in Figure 2-1 below. This data represents typical thermal and fast flux data for the irradiation facilities.

Figure 2-1
University of Buffalo Irradiation Facility Flux Data



IRRADIATION FACILITIES

Diagram Code	Facility	Max. sample size (inches)	Typical Flux Thermal	Typical Flux E>1MeV	Wet/Dry	Comments
A	Flux Trap	2.5 x 2.9 x 24	3×10^{13}	8×10^{12}	Wet	temperature-controlled sealed capsules available
A	Isotope tube	.81 diam x 3 long	3×10^{13}	$<5 \times 10^{12}$	Wet	short or long irradiatons
B	Vertical standpipe	4 diam	1×10^{11}	$\leq 8 \times 10^{11}$	Dry	2.5" tube with optional cadmium shielding available
C	Thermal Column	48 x 48 x 60	3×10^{10}	nil	Dry	Pure low energy neutrons available
C	Thermal Column	1.25 dian x 7 long	10^{9}	nil	Dry	Stringer facility; bismuth shielding available
D	Pneumatic conveyor	1.25 dian x 7 long	$3\text{-}5 \times 10^{12}$	$<1 \times 10^{12}$	Dry	Accurate irradiations; rabbits can be inserted from 20 seconds to 8 hours
E	Dry chamber	72 x 78 x 66	10^{10} - 10^{11}	$<8 \times 10^{9}$	Dry	Large sample volume
–	Electron accelerator	15 x 84	10^{4} - 10^{8} R/h		–	Electron bombardment
		18 x 84	10^{3} - 10^{6} R/h		–	X-ray bombardment

ADDITIONAL FACILITIES INCLUDE

Hot Cell
Large cell with 25,000 Curie Co-60 equivalent capacity; 110,000 pound materials test frame

Charpy Cell
Remote operated, fully instrumented Charpy impact machine calbrated to Watertown standards

Hot and Warm Chemistry Labs
Four laboratories in containment area for radioactive materials work

Counting Laboratory
Supports gamma spectroscopy, low background beta/alpha analysis

Electronics Shop
Complete facility for repair and calibration of radiation detection instruments

Machine Shop
Fully equipped; operated by ful time equipment designer

2.3 ORIGEN 2 Neutron Activation Calculations

The ORIGEN2 computer code, version 2.2, was used to calculate the activation and depletion of radionuclides in components exposed to neutron flux. Each component was irradiated based on the reactor operating history summarized in Table 2-1 using the appropriate University-supplied flux for the component location and the initial material compositions.

Some reactor component assemblies, notably the Grid Plate and Control Rod Shrouds, are constructed from multiple materials. Neutron activation differs significantly for each material and due to the different material composition, resulting in different dose rates and different waste classes for the respective component materials. However, the components are treated as assemblies for classification and decommissioning purposes in many cases due to the practical tradeoffs and difficulties in material separation. For example, the Grid Plate, primarily constructed of 1100 grade aluminum, includes stainless steel dowel pins that would be difficult and dose intensive to remove. Therefore, the classification results in Section 3 for such component assemblies reflect the different material constituents of the assembly.

3.0 ESTIMATED COMPONENT RADIOACTIVITY

This section presents the estimated activity for each reactor component of interest as of February 2011. These preliminary estimates are based on activation analysis results and, as stated previously in this report, will be normalized to measured dose rates when such data is available. The activities presented in this section represent the activation products only.

3.1 Overview

The activation results of all reactor components presented in this section are based on ORIGEN output using University provided flux levels and energy spectra. The ORIGEN output is in terms of specific activity (Ci/g) and the individual nuclide activities are calculated using the activated component weight and nuclide specific activity. Normalization to more accurately measured dose rates will change the activation activity for each component but will not change the component-specific scaling factors. Based on WMG's past experience, normalization to measured dose rates at the time of actual decommissioning is expected to reduce component total activity.

Table 3-1 presents a summary of the preliminary results in terms of waste weight, waste volume, total activation activity and materials of construction. The estimated activity for all reactor components should be less than 1,200 curies as of June 1, 2011. Component waste weights, waste volumes and materials of construction were determined from the drawings, documents and data provided by the University, supplemented as necessary by WMG's engineering experience. Please note that the waste volumes included in this section of the analysis are displacement volumes and do not reflect the "as-packaged" for disposal, or "envelope" volumes. This will be addressed further in Section 4.0.

TABLE 3-1
University at Buffalo BMRC
Component Characterization Summary

Component	Waste Weight (lbs)	Waste Volume (ft^3)	Total Activity (Ci)	Materials of Construction
Reactor Structural/Peripheral Components				
Plenum	54.8	0.324	7.29E-01	6061 Aluminum with 304 SS Bolting
Flapper	19.0	0.113	2.27E-01	1100 Aluminum
Plenum Chamber Stand	31.9	0.190	4.41E+00	6061 Aluminum
Ion Chamber Guide	7.2	0.043	1.00E+00	6061 Aluminum
Fission Chamber (worst case)	1.5	0.009	1.18E+00	6061 Aluminum with Uranium Loading
Reactor Tank				6061 Aluminum
Subtotals			**<10**	
Reactor Core Components				
Grid Plate	88.4	0.522	7.00E+01	1100 Aluminum with 36 SS Dowell Pins
Grid Plug (ea)	2.3	0.014	1.67E+00	6061 Aluminum
Fission Chamber Guide	10.8	0.063	8.30E+00	6061 Aluminum
Control Element (ea, typical)	11.2	0.025	5.53E+01	Ni clad Ag-In-Cd with Al & 304 SS fixtures
Control Element Shroud (ea, typical)	87.1	0.503	8.85E+01	6061 Aluminum with SS Fasteners
Control Element Drive Shaft (ea, typical)	87.1	0.503	6.29E+01	6061 Aluminum
Subtotals			**<1,000**	
Experimental Facilities				
Pneumatic Conveyor	10.3	0.061	1.43E+00	6061 Aluminum
Thermal Column	3,162.0	22.0	1.16E+00	Rx Grade Graphite & 6061 Aluminum
Dry Chamber				6061 Aluminum & Lead Shutter
Isotope Tubes (ea, typical)	1.8	0.011	1.33E+00	6061 Aluminum
Vertical Standpipes (ea, typical)	23.5	0.140	3.27E+00	6061 Aluminum
Becquerel Standpipe	33.1	0.196	4.59E+00	6061 Aluminum
Subtotals			**<50**	
Totals			**<1100**	

3.2 Component Radioactivity

Individual component estimated activation activities are decay corrected to June 1, 2011. The reactor components are divided into three groups: the Reactor Structural/Peripheral Components, the Experimental Facilities and the Reactor Core Components. The Reactor Core Components hold the fuel elements in place (Grid Plate) or reside within the reactor core (Control Rod Shrouds, Core Plugs, etc.). Based on the empirical data from previous decommissioning activities, the additional effect of surface contaminants could increase component dose rates marginally but is unlikely to change waste classification.

The radionuclide activities presented in this report are those considered "significant" per the guidance presented in NUREG/BR-0204 "Instructions for Completing NRC's Uniform Low-Level Radioactive Waste Manifest" (Ref. 7). In summary, a radionuclide is considered significant when it is contained in the waste in concentrations:

- Greater than 1% of the total activity of the component
- Greater than 1% of the Class A limit for nuclides listed in 10 CFCR 61.55
- Greater than 7.0 μCi/cc for nuclides not listed in 10 CFR 61.55

As discussed previously, final surveys of the reactor components are necessary to update the preliminary results presented herein to provide final characterization results.

3.2.1 Reactor Structural/Peripheral Components

The reactor structural and peripheral components include the Concrete Reactor Tank (Bioshield Concrete), Aluminum Reactor Tank and Liner, Plenum and Flapper, and Core Support.

The Reactor Tank is a ¼ inch thick quasi-hexagonal 6061 plate aluminum construction nominally 8 ft wide at core elevation increasing to 14 ft wide approximately 14 ft up from the tank bottom and approximately 29 ft deep. A ¾" thick Liner was installed interior to the Reactor Tank during the operational history atop support strips/spacers. The Bioshield concrete supporting the tank varies from 250 to 150 pcf with heavy 250 pcf concrete around core elevation. The BMRC also has moveable lead shields just outside the reactor tank at and above core elevation. Preliminary activation analysis results for selected components are presented below.

Plenum Components

The Plenum, and Plenum Chamber Stand are located beneath the active core and supported the core while facilitating coolant flow. The Plenum is fabricated from 3/8" thick 6061 aluminum and interfaces with other components with stainless steel fasteners. Component isotope abundances and scaling factors are summarized in Tables 3-2 and 3-3 below.

Table 3-2
Plenum Chamber
Characterization Results

Component ID	03a			
Description	Plenum Chamber			
Waste Weight	54.82	lbs	2.49E+04	g
Waste Volume	0.324	ft^3	9.16E+03	cc
	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	T2
H-3	1.02E-04	1.08E-04		0.00
C-14	1.51E-03	1.61E-03	0.02	
Fe-55	1.11E-02	1.18E-02		0.00
Co-60	9.40E-01	1.00E+00		0.11
Ni-59	3.52E-04	3.75E-04	0.00	
Ni-63	4.74E-02	5.05E-02		0.11
Sr-90	1.99E-14	2.12E-14		0.00
Nb-94	3.86E-05	4.11E-05	0.15	
Tc-99	2.35E-07	2.50E-07	0.00	
Total	1.00E+00		0.17	0.22

Table 3-3
Plenum Chamber Stand
Characterization Results

Component ID	03c			
Description	Plenum Chamber Stand			
Waste Weight	31.93	lbs	1.45E+04	g
Waste Volume	0.190	ft^3	5.37E+03	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	1.22E-11	1.28E-11		0.00
C-14	3.00E-03	3.15E-03	0.31	
Fe-55	1.38E-02	1.45E-02		0.02
Co-60	9.52E-01	1.00E+00		1.12
Ni-59	2.49E-04	2.62E-04	0.01	
Ni-63	3.08E-02	3.23E-02		0.72
Sr-90	1.09E-15	1.14E-15		0.00
Nb-94	2.25E-05	2.37E-05	0.93	
Tc-99	5.77E-10	6.06E-10	0.00	
Total	1.00E+00		1.24	1.86

Flapper

The Flapper is a hinged and counterbalanced plate that was shut dung during forced convection and open during natural circulation. The inch thick component was fabricated from 1100 grade aluminum with some 304 stainless steel hinge parts. Component isotope abundances and scaling factors are summarized in Table 3-4 below.

Table 3-4
Flapper
Characterization Results

Component ID	03b					
Description	Flapper					
Waste Weight	19.00	lbs	8.62E+03	g		
Waste Volume	0.113	ft^3	3.19E+03	cc		

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	4.89E-12	5.00E-12		0.00
C-14	1.49E-03	1.53E-03	0.01	
Fe-55	5.54E-03	5.67E-03		0.00
Co-60	9.77E-01	1.00E+00		0.10
Ni-59	1.10E-04	1.13E-04	0.00	
Ni-63	1.54E-02	1.58E-02		0.03
Sr-90	1.25E-14	1.28E-14		0.00
Nb-94	3.84E-05	3.93E-05	0.14	
Tc-99	5.52E-09	5.65E-09	0.00	
Total	1.00E+00		0.15	0.13

Ion Chamber Guide

The ion chamber guide is the framework of 2 inch, 6061 aluminum angles above the Grid Plate bolted to the Plenum to flange to locate the ion chambers. Component isotope abundances and scaling factors are summarized in Table 3-5 below.

Table 3-5
Ion Chamber Guide
Characterization Results

Component ID	04			
Description	Ion Chamber Guide			
Waste Weight	7.215	lbs	3.27E+03	g
Waste Volume	0.043	ft^3	1.21E+03	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	1.61E-11	1.70E-11		0.00
C-14	3.29E-03	3.46E-03	0.34	
Fe-55	1.37E-02	1.45E-02		0.02
Co-60	9.49E-01	1.00E+00		1.12
Ni-59	2.72E-04	2.87E-04	0.01	
Ni-63	3.32E-02	3.50E-02		0.78
Sr-90	1.30E-15	1.37E-15		0.00
Nb-94	2.47E-05	2.60E-05	1.02	
Tc-99	6.33E-10	6.67E-10	0.00	
Total	1.00E+00		1.37	1.92

Fission Chamber

The Fission Chamber is a moveable component located at a core corner and withdrawn from the core during power operation. The fission chamber is sized to fit inside it's guide and is fabricated from 6061 aluminum. Despite it's relatively light weight, transmutation of Uranium to Plutonium is not expected to drive waste class because the chamber was withdrawn during power operation. Component isotope abundances and scaling factors are summarized in Table 3-6 below.

Table 3-6
Fission Chamber
Characterization Results

Component ID	05a				
Description	Fission Chamber				
Waste Weight	1.5	lbs	6.80E+02	g	
Waste Volume	0.009	ft^3	2.55E+02	cc	
	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	T2	
H-3	7.25E-10	7.65E-10		0.00	
C-14	3.59E-03	3.79E-03	2.06		
Fe-55	1.49E-02	1.58E-02		0.10	
Co-60	9.48E-01	1.00E+00		6.20	
Ni-59	2.31E-04	2.44E-04	0.05		
Ni-63	3.34E-02	3.52E-02		4.37	
Sr-90	1.50E-14	1.58E-14		0.00	
Nb-94	2.71E-05	2.86E-05	6.20		
Tc-99	8.57E-10	9.04E-10	0.00		
Total	1.00E+00		8.31	10.67	

3.2.2 Reactor Core Components

These components are subject to the greatest neutron flux and are consequently the most activated components. The reactor core components consist of the Grid Plate, Grid Plate Plugs, Fission Chamber Guide & Fission Chamber, Control Elements, Control Element Shrouds, and lower portions of the Control Element Drive Shafts.

Grid Plate

The Grid Plate is a 5 inch thick slab of 1100 grade aluminum with a 6 by 6 array of 2.4” diameter holes milled to accommodate the lower fuel end fittings. 25 ¾” holes are also milled to accommodate the Control Element Shroud locating studs. ¼” diameter stainless steel dowel pins are fixed on the upper face near each fuel hole to orient the fuel assemblies. Component isotope abundances and scaling factors are summarized in Table 3-7 below.

Table 3-7
Grid Plate
Characterization Results

Component ID	01					
Description	Grid Plate					
Waste Weight	88.36	lbs	4.01E+04	g		
Waste Volume	0.522	ft^3	1.48E+04	cc		

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	4.71E-05	5.25E-05		0.00
C-14	3.44E-03	3.83E-03	2.04	
Fe-55	2.06E-02	2.29E-02		0.14
Co-60	8.98E-01	1.00E+00		6.07
Ni-59	5.37E-04	5.98E-04	0.12	
Ni-63	7.72E-02	8.60E-02		10.43
Sr-90	9.31E-14	1.04E-13		0.00
Nb-94	2.40E-05	2.67E-05	5.68	
Tc-99	1.37E-08	1.53E-08	0.00	
Total	1.00E+00		7.83	16.64

Grid Plate Plugs and Weep Hole Plugs

Grid Plate holes not occupied by fuel, Control Element Shroud studs or experiments were filled with Plugs. WMG conservatively assumed the plugs are fabricated from 6061 Aluminum. Component isotope abundances and scaling factors are summarized in Table 3-8 below.

Table 3-8
Grid Plate Plug
Characterization Results

Component ID	02			
Description	Single Grid Plug			
Waste Weight	2.315	lbs	1.05E+03	g
Waste Volume	0.014	ft^3	3.89E+02	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	7.36E-10	7.77E-10		0.00
C-14	3.72E-03	3.93E-03	2.00	
Fe-55	1.55E-02	1.64E-02		0.10
Co-60	9.46E-01	1.00E+00		5.81
Ni-59	2.38E-04	2.52E-04	0.05	
Ni-63	3.44E-02	3.63E-02		4.22
Sr-90	4.00E-15	4.23E-15		0.00
Nb-94	2.61E-05	2.76E-05	5.61	
Tc-99	5.20E-10	5.49E-10	0.00	
Total	1.00E+00		7.65	10.13

Fission Chamber Guide

The Fission Chamber Guide is a 7½ foot long 3 inch diameter 6061 aluminum tube with a fuel assembly lower end fitting. It is located at a core corner and is used to guide the elevation of the Fission Chamber. Component isotope abundances and scaling factors are summarized in Table 3-9 below.

Table 3-9
Fission Chamber Guide
Characterization Results

Component ID	05a			
Description	Fission Chamber Guide			
Waste Weight	11.03 lbs		5.00E+03 g	
Waste Volume	0.065 ft^3		1.85E+03 cc	
	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	T2
H-3	7.25E-10	7.65E-10		0.00
C-14	3.59E-03	3.79E-03	2.06	
Fe-55	1.49E-02	1.58E-02		0.10
Co-60	9.48E-01	1.00E+00		6.20
Ni-59	2.31E-04	2.44E-04	0.05	
Ni-63	3.34E-02	3.52E-02		4.37
Sr-90	1.50E-14	1.58E-14		0.00
Nb-94	2.71E-05	2.86E-05	6.20	
Tc-99	8.57E-10	9.04E-10	0.00	
Total	1.00E+00		8.31	10.67

Control Elements

The neutron reaction is controlled by the elevation of the Control Elements. The 0.18Control Elements are fabricated from a nickel-plated 80% silver – 15% indium – 5% cadmium alloy. A flat aluminum transition is used to couple the plate-like Control Element absorber with a 1 inch diameter Drive Shaft. The absorber is only 0.18 inches thick (excluding plating) by 4.85 inches wide by 29 inches tall. WMG assumed the five rivets attaching the transition to the absorber are made from stainless steel. Two sets of Control Elements exist with the set currently stored in the Reactor Tank having a shorter exposure history than the first set. Control Element waste class is most heavily influenced by the activation of the thin nickel cladding. Also, cadmium is considered a toxic waste, so the activated Control Elements should be considered mixed waste, particularly the first set of Control Elements, whose flat transitions were severed from the absorber. Component isotope abundances and scaling factors are summarized in Table 3-10 below.

Table 3-10
Control Element
Characterization Results

Component ID	14a			
Description	Single Control Element			
Waste Weight	11.150	lbs	5.06E+03	g
Waste Volume	0.025	ft^3	7.10E+02	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	5.64E-05	5.95E-04		0.00
C-14	1.17E-04	1.24E-03	0.07	
Fe-55	1.13E-02	1.20E-01		0.07
Co-60	9.47E-02	1.00E+00		0.62
Ni-59	7.05E-03	7.44E-02	1.47	
Ni-63	8.87E-01	9.36E+00		116.48
Sr-90	1.11E-14	1.17E-13		0.00
Nb-94	8.89E-07	9.38E-06	0.20	
Tc-99	5.93E-09	6.26E-08	0.00	
Ag-108m	1.46E+01	1.54E+02		
Ag-108	1.30E+00	1.37E+01		
Total	1.00E+00		1.75	117.18

Control Element Shrouds

Control Element location is facilitated by frame-like Control Element Shrouds which act as a guide for the absorbers regardless of elevation. The Control Element Shrouds have a rectangular cavity 5 inches wide by 0.305 inches thick and are approximately 6 feet tall. The locating studs and fasteners are stainless steel while the remainder of the component is assumed to be made from 1100 grade aluminum. Component isotope abundances and scaling factors are summarized in Table 3-11 below.

Table 3-11
Control Element Shroud
Characterization Results

Component ID	14b			
Description	Control Element Shroud			
Waste Weight	87.12 lbs		3.95E+04 g	
Waste Volume	0.503 ft^3		1.42E+04 cc	
	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	1.48E-04	1.87E-04		0.00
C-14	2.80E-03	3.54E-03	2.17	
Fe-55	3.60E-02	4.56E-02		0.32
Co-60	7.91E-01	1.00E+00		7.02
Ni-59	1.18E-03	1.49E-03	0.33	
Ni-63	1.69E-01	2.14E-01		30.02
Sr-90	2.84E-13	3.59E-13		0.00
Nb-94	1.93E-05	2.44E-05	5.98	
Tc-99	4.18E-08	5.28E-08	0.00	
Total	1.00E+00		8.49	37.36

Control Element Drive Shafts

Control Element elevation is facilitated by the Drive Shafts. The Control Element Drive Shafts are 1 inch diameter 6061 aluminum tube, the lowermost part is activated. The upper portion of the shaft is not activated. Component isotope abundances and scaling factors for the lower shaft are summarized in Table 3-12 below.

Table 3-12
Control Element Drive Shaft
Characterization Results

Component ID	14c			
Description	Control Element Drive Shaft			
Waste Weight	1.17	lbs	5.30E+02	g
Waste Volume	0.007	ft^3	1.94E+02	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	7.36E-10	7.77E-10		0.00
C-14	3.72E-03	3.93E-03	2.00	
Fe-55	1.55E-02	1.64E-02		0.10
Co-60	9.46E-01	1.00E+00		5.81
Ni-59	2.38E-04	2.52E-04	0.05	
Ni-63	3.44E-02	3.63E-02		4.22
Sr-90	4.00E-15	4.23E-15		0.00
Nb-94	2.61E-05	2.76E-05	5.61	
Tc-99	5.20E-10	5.49E-10	0.00	
Total	1.00E+00		7.66	10.13

3.2.3 Experimental Facilities

Special neutron irradiation facilities are provided for the production of radioisotopes and materials testing. Experimental facilities are located in, around and adjacent to the active core. These experimental facilities are assumed to be purged of any experimental materials or isotope production material. Experimental facilities include the Pneumatic Conveyor, Thermal Column, Dry Chamber, Isotope Tubes, movable Vertical Standpipes, and the Becquerel Standpipe. The results for each experimental facility are discussed separately below.

Pneumatic Conveyors

The Pneumatic Conveyors are double 6061 aluminum tube rabbit assemblies that could move materials from the hot cell to immediately adjacent to the core in the horizontal plane using differential air pressure. Only the sections closest to the core are activated. Component isotope abundances and scaling factors for the lower shaft are summarized in Table 3-13 below.

Table 3-13
Pneumatic Conveyor
Characterization Results

Component ID	07			
Description	Single Pneumatic Conveyor			
Waste Weight	10.34 lbs		4.69E+03 g	
Waste Volume	0.061 ft^3		1.74E+03 cc	
	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	T2
H-3	1.22E-11	1.28E-11		0.00
C-14	3.00E-03	3.15E-03	0.31	
Fe-55	1.38E-02	1.45E-02		0.02
Co-60	9.52E-01	1.00E+00		1.12
Ni-59	2.49E-04	2.62E-04	0.01	
Ni-63	3.08E-02	3.23E-02		0.72
Sr-90	1.09E-15	1.14E-15		0.00
Nb-94	2.25E-05	2.37E-05	0.93	
Tc-99	5.77E-10	6.06E-10	0.00	
Total	1.00E+00		1.24	1.86

Thermal Column

The Thermal Column used a 6061 aluminum exterior, graphite filled nosepiece immediately adjacent to the active core to provide thermal neutrons to a large experimental space beneath the hot cell. 6061 aluminum bails and rods facilitated placement of the reactor grade graphite blocks in the experimental space. Only the section closest to the core is substantially activated. Component isotope abundances and scaling factors for the activated section are summarized in Table 3-14 below.

Table 3-14
Thermal Column
Characterization Results

Component ID	08			
Description	Thermal Column			
Waste Weight	3162.0	lbs	1.43E+06	g
Waste Volume	21.98	ft^3	6.22E+05	cc
	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	T2
H-3	1.51E-07	1.60E-07		0.00
C-14	1.57E-03	1.66E-03	0.00	
Fe-55	4.03E-02	4.26E-02		0.00
Co-60	9.46E-01	1.00E+00		0.00
Ni-59	1.07E-04	1.13E-04	0.00	
Ni-63	1.25E-02	1.32E-02		0.00
Sr-90	0.00E+00	0.00E+00		0.00
Nb-94	8.22E-06	8.70E-06	0.00	
Tc-99	4.34E-11	4.59E-11	0.00	
Total	1.00E+00		0.00	0.00

Dry Chamber

Opposite the Thermal Column was a similar but slightly larger experimental space that was not filled with graphite. The Dry Chamber had a similar nosepiece, but had a large lead shield close to the tank wall. Additional operational history information is needed to characterize Dry Chamber components other than the nosepiece.

Isotope Tubes, Vertical Standpipes, and the Becquerel Standpipe

Straight and curved 6061 aluminum tubes of different diameters were used for experimental purposes. Some of the tubes' locations could be changed relative to the core. Only the portions of the tubes closest to the core are activated. Component isotope abundances and scaling factors for the activated sections are summarized in Tables 3-15 to 3-17 below.

Table 3-15
Isotope Tube
Characterization Results

Component ID	10					
Description	Isotope Tube					
Waste Weight	1.81	lbs	8.20E+02	g		
Waste Volume	0.011	ft^3	3.04E+02	cc		

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	7.29E-10	7.70E-10		0.00
C-14	3.69E-03	3.89E-03	2.01	
Fe-55	1.53E-02	1.62E-02		0.10
Co-60	9.47E-01	1.00E+00		5.91
Ni-59	2.36E-04	2.50E-04	0.05	
Ni-63	3.41E-02	3.60E-02		4.26
Sr-90	6.15E-15	6.50E-15		0.00
Nb-94	2.64E-05	2.78E-05	5.76	
Tc-99	6.09E-10	6.43E-10	0.00	
Total	1.00E+00		7.82	10.27

Table 3-16
Vertical Standpipe
Characterization Results

Component ID	11			
Description	Single Vertical Standpipe			
Waste Weight	23.5	lbs	1.07E+04	g
Waste Volume	0.140	ft^3	3.96E+03	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	1.61E-11	1.70E-11		0.00
C-14	3.29E-03	3.46E-03	0.34	
Fe-55	1.37E-02	1.45E-02		0.02
Co-60	9.49E-01	1.00E+00		1.12
Ni-59	2.72E-04	2.87E-04	0.01	
Ni-63	3.32E-02	3.50E-02		0.78
Sr-90	1.30E-15	1.37E-15		0.00
Nb-94	2.47E-05	2.60E-05	1.02	
Tc-99	6.33E-10	6.67E-10	0.00	
Total	1.00E+00		1.37	1.92

Table 3-17
Isotope Tube
Characterization Results

Component ID	12			
Description	Bequerel Standpipe			
Waste Weight	33.08	lbs	1.50E+04	g
Waste Volume	0.196	ft^3	5.56E+03	cc

	Fractional Abundance	Co-60 SF	NRC Class A Fractions T1	NRC Class A Fractions T2
H-3	1.61E-11	1.70E-11		0.00
C-14	3.29E-03	3.46E-03	0.34	
Fe-55	1.37E-02	1.45E-02		0.02
Co-60	9.49E-01	1.00E+00		1.12
Ni-59	2.72E-04	2.87E-04	0.01	
Ni-63	3.32E-02	3.50E-02		0.78
Sr-90	1.30E-15	1.37E-15		0.00
Nb-94	2.47E-05	2.60E-05	1.02	
Tc-99	6.33E-10	6.67E-10	0.00	
Total	1.00E+00		1.37	1.92

4.0 COMPONENT PACKAGING REQUIREMENTS

Detailed radiation surveys must be performed to finalize the preliminary results presented in this analysis. Of particular importance is to obtain surveys of the highly activated components closest to the active core. February 2011 component survey use was hampered by detector saturation when approaching the the highly activated core components. In addition, the radiation surveys identify any “hot spots” on the reactor components which need to be taken into consideration during packaging of the components for transport and disposal.

The Control Element results and discussion are likely to drive packaging options for the BMRC. NRC Class A mixed waste can be disposed at Clive, UT. However, based on results not yet normalized to measured dose rates, the Control Element waste class exceeds that acceptable at the Clive disposal site. Even after normalization to dose rates to be obtained in May 2011 of a representative Control Element, nuclide concentration could still exceed NRC Class A limits. Only the Nevada Test Site (NTS) is currently available for disposal of such mixed waste. The Plum Brook Reactor Facility had a very similar predicament that, after regulatory effort, did result in acceptance and disposal of non–Department of Energy mixed waste. Key to regulatory acceptance at NTS is showing a nexus of waste responsibility between the BMRC and federal government entities.

Preliminary curie estimations indicate the Grid Plate, Control Elements, and Control Element Shrouds, may exceed the definition of Low Specific Activity [II] (LSA II) material necessitating transport in an NRC licensed Type B cask. A single 8-120 cask liner has sufficient volume to accommodate all the highly activated components with minimal segmentation. The shielding afforded by the 8-120 is also sufficient for all the highly activated components. The large payload capacity of and 8-120 minimizes handling of components, reducing the time and dose necessary to package the components for disposal. Tubing sections and other items exceeding 5 feet long need to be segmented to accommodate the internal height of the disposal liner and furthermore, to increase packaging efficiency of the activated components. A packaging efficiency of up to 15% (waste volume / liner internal volume) can be achieved with segmentation of long components and careful packaging. .

The minimally activated components of the reactor structure, periphery and experimental facilities can be packaged for disposal as Low Specific Activity (LSA) material. Disposal liner/container and transportation package selection will be based upon the detailed radiation surveys obtained during decommissioning. Activation of the reactor tank and other minimally activated components may have localized hotspots. However, the activation levels in these components are low enough such that localized hotspots will not change the overall approach to removal and disposal of these components, nor approach any regulatory limits.

5.0 REFERENCES

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2. DLC-185, BUGLE-96, Coupled, 47-Neutron, 20 Gamma-Ray Cross Section Library derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications, Oak Ridge National Laboratory Radiation Shielding Information Center, July 1999.
3. ORNL/CCC-371, ORIGEN 2.2, Isotope Generation and Depletion Code, Matrix Exponential Method, Oak Ridge National laboratory, May 2002.
4. Evans, J.C., et al., NUREG 3474, Long-Lived Activation Products in Reactor Materials, Pacific Northwest Laboratories, August 1984.
5. NUREG/BR-0204, Instructions for Completing NRC's Uniform Low-Level Radioactive Waste Manifest, July 1998.

APPENDIX A
ORIGEN Material Compositions

304 Stainless Steel (NUREG 3474)

- Density = 8.03 $\frac{g}{cm^3}$, 501 $\frac{lbs}{ft^3}$

Average NUREG 3474 304 Stainless Steel Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
Li	30000	1.30E-07	Sr	380000	2.00E-07
N	70000	4.52E-04	Y	390000	5.00E-06
Na	110000	9.70E-06	Zr	400000	1.00E-05
Al	130000	1.00E-04	Nb	410000	8.90E-05
Cl	170000	7.00E-05	Mo	420000	2.60E-03
K	190000	3.00E-06	Ag	470000	2.00E-06
Ca	200000	1.90E-05	Sb	510000	1.23E-05
Sc	210000	3.00E-08	Cs	550000	3.00E-07
Ti	220000	6.00E-04	Ba	560000	5.00E-04
V	230000	4.56E-04	La	570000	2.00E-07
Cr	240000	1.84E-01	Ce	580000	3.71E-04
Mn	250000	1.53E-02	Sm	620000	1.00E-07
Fe	260000	7.06E-01	Eu	630000	2.00E-08
Co	270000	1.41E-03	Tb	650000	4.70E-07
Ni	280000	1.00E-01	Dy	660000	1.00E-06
Cu	290000	3.08E-03	Ho	670000	1.00E-06
Zn	300000	4.57E-04	Yb	700000	2.00E-06
Ga	310000	1.29E-04	Lu	710000	8.00E-07
As	330000	1.94E-04	Hf	720000	2.00E-06
Se	340000	3.50E-05	W	740000	1.86E-04
Br	350000	2.00E-06	Pb	820000	6.70E-05
Rb	370000	1.00E-05	Th	900000	1.00E-06
			U	920000	2.00E-06

Carbon Steel (NUREG 3474)

- Density = 7.87 $\frac{g}{cm^3}$, 491 $\frac{lbs}{ft^3}$

Average NUREG 3474 Carbon Steel Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
Li	30000	3.00E-07	Sr	380000	1.50E-07
N	70000	8.40E-05	Y	390000	2.00E-05
C*	80000	2.90E-03	Zr	400000	1.00E-05
Na	110000	2.30E-05	Nb	410000	1.88E-05
Al	130000	3.30E-04	Mo	420000	5.60E-07
Cl	170000	4.00E-05	Ag	470000	2.00E-06
K	190000	1.20E-05	Sb	510000	1.10E-05
Ca	200000	1.40E-05	Cs	550000	2.00E-07
Sc	210000	2.60E-07	Ba	560000	2.73E-04
Ti	220000	2.00E-06	La	570000	1.00E-07
V	230000	8.00E-05	Ce	580000	1.00E-06
Cr	240000	1.70E-03	Sm	620000	1.70E-08
Mn	250000	1.02E-02	Eu	630000	3.10E-08
Fe	260000	9.80E-01	Tb	650000	4.50E-07
Co	270000	1.22E-04	Ho	670000	8.00E-07
Ni	280000	6.60E-03	Yb	700000	1.00E-06
Cu	290000	1.27E-03	Lu	710000	2.00E-07
Zn	300000	1.00E-04	Hf	720000	2.10E-07
Ga	310000	8.00E-05	Ta	730000	1.30E-07
As	330000	5.32E-04	W	740000	5.50E-06
Se	340000	7.00E-07	Pb	820000	8.20E-04
Br	350000	8.50E-07	Th	900000	1.80E-07
Rb	370000	4.80E-05	U	920000	2.00E-07

* The carbon weight percent is taken from Page 2, Grade 65 [Grade 450], for typical pressure vessel steel because there is no carbon listed in NUREG 3474.

Bioshield Concrete **(NUREG 3474)**

- Density = 2.30 $\frac{g}{cm^3}$, 144 $\frac{lbs}{ft^3}$

Average NUREG 3474 Bioshield Concrete Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
H	10000	6.10E-3	Rb	370000	3.50E-5
Li	30000	2.00E-5	Sr	380000	4.38E-4
B	50000	2.00E-5	Y	390000	1.82E-5
N	70000	1.20E-4	Zr	400000	7.10E-5
O*	80000	5.02E-1	Nb	410000	4.30E-6
Na	110000	7.39E-3	Mo	420000	1.03E-5
Mg*	120000	2.46E-3	Pd	460000	3.00E-6
Al	130000	3.10E-2	Ag	470000	2.00E-7
Si	140000	1.68E-1	Cd	480000	3.00E-7
P	150000	5.00E-3	Sn	500000	7.00E-6
S	16000	3.10E-3	Sb	510000	1.80E-6
Cl	170000	4.50E-5	Cs	550000	1.30E-6
K	190000	7.50E-3	Ba	560000	9.50E-4
Ca	200000	1.83E-1	La	570000	1.30E-5
Sc	210000	6.50E-6	Ce	580000	2.43E-5
Ti	220000	2.12E-3	Sm	620000	2.00E-6
V	230000	1.03E-4	Eu	630000	5.55E-7
Cr	240000	1.09E-4	Tb	650000	4.10E-7
Mn	250000	3.77E-4	Dy	660000	2.30E-6
Fe	260000	3.90E-2	Ho	670000	9.00E-7
Co	270000	9.80E-6	Yb	700000	1.40E-6
Ni	280000	3.80E-5	Lu	710000	2.70E-7
Cu	290000	2.50E-5	Hf	720000	2.20E-6
Zn	300000	7.50E-5	Ta	730000	4.40E-7
Ga	310000	8.80E-6	W	740000	1.40E-6
As	330000	7.90E-6	Pb	820000	6.10E-5
Se	340000	9.20E-7	Th	900000	3.50E-6
Br	350000	2.40E-6	U	920000	2.70E-5

* The oxygen and magnesium weight percents are calculated since these nuclides are excluded from NUREG 3474.

Aluminum 6061

- Density = 2.7 $\frac{g}{cm^3}$, 169 $\frac{lbs}{ft^3}$

Average Aluminum 6061 Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
B	50000	8.43E-7	Fe	260000	7.00E-3
N	70000	1.37E-4	Co	270000	1.72E-4
Mg	120000	1.00E-2	Ni	280000	4.04E-4
Al	130000	9.66E-1	Cu	290000	2.75E-3
Si	140000	6.00E-3	Zn	300000	2.50E-3
Ti	220000	1.50E-3	Nb	410000	3.40E-5
Cr	240000	1.95E-3	Mo	420000	5.60E-7
Mn	250000	1.50E-3			

Graphite

- Density = 1.7 $\frac{g}{cm^3}$, 106 $\frac{lbs}{ft^3}$

Average Graphite Composition

Element	RSICC Number	Weight Fraction (g/g)	Element	RSICC Number	Weight Fraction (g/g)
H	10000	3.89E-5	Mg	120000	1.00E-4
B	50000	5.00E-7	Cl	170000	2.50E-5
C	60000	9.89E-1	Ti	220000	1.00E-3
O	80000	1.29E-3	Fe	260000	8.00E-3
F	90000	2.50E-5	Co	270000	5.00E-5
Na	110000	1.00E-4	Cu	290000	1.00E-4

APPENDIX B – SITE CHARACTERIZATION REPORT

Site Characterization Report

Buffalo Materials Research Center

Prepared for:

University at Buffalo
The State University of New York

Buffalo Material Research Center
Office of Environment, Health, and Safety Services

Completed by:

ENERCON
Excellence—Every project. Every day.

4490 Old William Penn Hwy
Murrysville, PA 15668

July 30, 2011

Summary of Changes

Revisions to the BMRC Characterization Report will be tracked when revisions are issued. Changed sections will be identified by special demarcation in the margin. A summary description of each revision will be noted in the following table.

Revision Number	Date	Description of Change
0	July 30, 2011	Initial Issue

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List of Acronyms and Abbreviations

AEC	Atomic Energy Commission
ACM	Asbestos Containing Material
BF	Bioshield Face
BMRC	Buffalo Materials Research Center
D&D	Decontamination and Dismantlement
DCGL	Derived Concentration Guideline Level
DP	Decommissioning Plan
DPM	Disintegrations per Minute
DQO	Data Quality Objectives
FSS	Final Status Survey
HSA	Historical Site Assessment
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MW_t	Megawatt Thermal
NaI	Sodium Iodide
NIST	National Institute for Standards and Testing
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guide
PULSTAR	Pulse Training and Reactor
R-77	Reactor License Number 77
RFP	Request For Proposal
RSO	Radiation Safety Officer
SSCs	Systems Structures Components
SUNY	State University of New York
UB	University of Buffalo

1. EXECUTIVE SUMMARY

This report describes the current radiological status of the facility to support the development of the Buffalo Materials Research Center (BMRC) Decommissioning Plan (DP). The sections that follow describe the scope, background, and technical approach used during the characterization of the BMRC. The BMRC contains a 2-Megawatt Thermal (MW_t) pool-type research reactor licensed by the United States Nuclear Regulatory Commission (NRC) under reactor license number 77 (R-77) and is located on the South Campus of the State University of New York (SUNY) at Buffalo, NY.

The data collected during this characterization will be utilized to support the development of a DP in accordance with the following NRC regulatory guidance documents: 1) NUREG-1537, Part 1, Chapter 17, *Guidelines for Preparing and Reviewing Applications for Licensing of Non-Power Reactors*; 2) NUREG-1757, *Consolidated NMSS Decommissioning Guidance*; and 3) NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*.

The data will specifically assist in the development of radiation safety protocols and project dose estimates; estimation of expected radiological waste volumes; methodologies for removal and packaging of radiological waste; and disposal options for radioactively impacted materials generated during decommissioning activities conducted under NRC License R-77. This data will also be incorporated into the decision making process for the development of Final Status Survey (FSS) Data Quality Objectives (DQO's) described in MARSSIM. The FSS provides the closeout documentation required by the NRC in order to terminate the BMRC Reactor License.

For the characterization effort, ENERCON developed a Characterization Plan to establish the measurement and sampling technical approach along with the project quality assurance requirements. This plan was reviewed and approved by the BMRC Operating Committee. Radiological work permits were issued by the University's Radiation Safety Officer (RSO) for all survey areas, as necessary. The BMRC facility was separated into survey units with survey instructions for each unit. The survey instructions were implemented by a qualified Health Physics technician and the data was documented on a radiological survey. This report only contains a summary of the survey results. All completed survey packages are stored at the BMRC for future reference.

Radiological measurements indicate that levels of residual materials on the building surfaces are less than release criteria established by the NRC in Regulatory Guide 1.86. Soil samples indicate that remediation of soils will not be required outside of the facility; however, radionuclides above the release criteria were detected in the N16 Vault which does not have a concrete floor covering the bedrock. Residual radioactive materials were only found in areas indicated by the Historical Site Assessment. The structures and components that contain the majority of the residual radioactive materials are the activated reactor components and bioshield, the former and current liquid waste tanks, and the reactor water cooling system. Residual radioactive materials were also identified in the Hot Cell and laboratory fume hoods.

2. SCOPE AND BACKGROUND

2.1 Scope

The BMRC characterization scope was to determine the nature and extent of residual radioactive materials and was accomplished through the following methods:

- Radiological measurements on building surfaces and structures
- Concrete cores on the reactor bioshield to determine concrete activation
- Exterior sub-surface soil borings
- Interior concrete floor cores
- Liquid samples from legacy and current waste holding tanks

The activation analysis of reactor components and pool liner is not included in this characterization report. The Component Activation Analysis scope of work was contracted by the University to WMG, Inc. of Fishkill, NY. The Component Activation Analysis Report, along with this characterization report, will be appended to the BMRC DP for approval by the NRC.

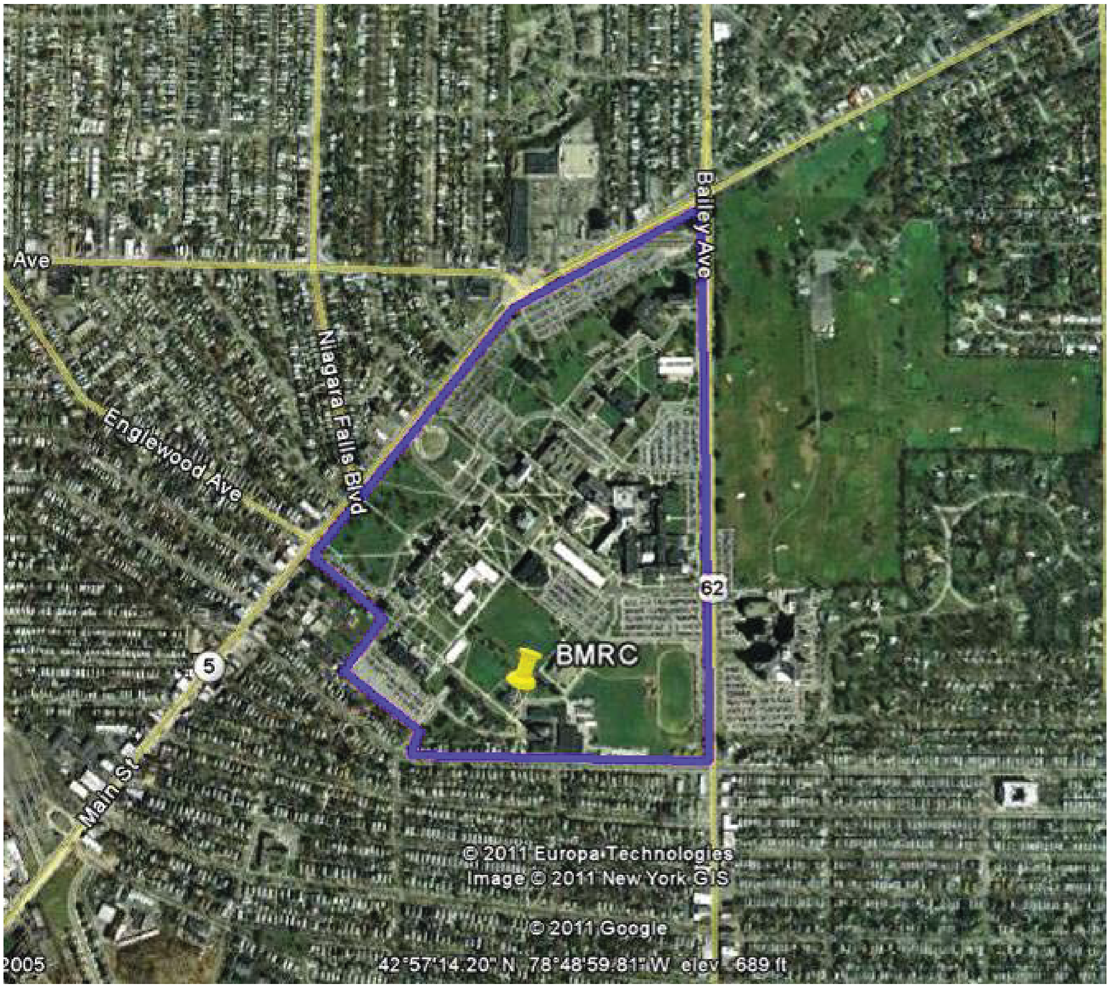
2.2 Facility Background

The UB is licensed by NRC to possess radioactive materials within the BMRC. The BMRC reactor is located on the South Campus of the University at Buffalo (UB) in the city of Buffalo, New York, as shown in Figure 1.

The reactor was defueled in 2005 and is beginning the Decontamination and Dismantlement (D&D) process to decommission the facility for eventual termination of the NRC license. This Characterization Report is in support of a DP developed in accordance with Chapter 17 of the Nuclear Regulatory Commission (NUREG -1537) Part 1 *Guidance for Preparing and Reviewing Applications for Licensing of Non-Power Reactors* (NRC 1996).

This report describes the purpose, scope and technical approach used for the characterization of the BMRC. The purpose of the characterization is to assess the current radiological status of the facility for the development of the Decommissioning Plan. This data is also incorporated into the decision-making process for development of FSS DQOs described in MARSSIM.

Figure 1: University of Buffalo South Campus



The BMRC building is comprised of an administrative/laboratory wing (administrative wing) and a vapor containment building (containment building) with three levels in each. The three levels are: neutron deck (lowest level, i.e. sub-grade), gamma deck (2nd level, i.e. middle level), and control deck (top floor). The neutron deck of the administrative wing (a.k.a. sub-basement) houses numerous systems, structures, and components (SSCs) that support operation of the reactor. The remainder of the administrative wing contains 20 rooms which are a mixture of offices, laboratories, utility rooms, and classrooms. See Figures 2, 3, and 4 at the end of this section.

The facility was constructed in 1959, went into operation on March 24, 1960, and was operated by The Western New York Nuclear Research Center, Inc, a subsidiary of the State University of New York, under Atomic Energy Commission (AEC) license number R-77. The facility licensee was changed to the Nuclear Science and Technology Center followed by a change to the Buffalo Materials Research Center in 1973 and 1985, respectively. The licensed power was one MW_t without forced convection cooling and up to two MW_t possible for short times. In 1991, the reactor was retrofitted with a 0.75 inch thick aluminum liner. This new liner supplements the original 0.25 inch thick aluminum liner and there is a 0.75 inch air gap between the two liners.

The reactor core is located in a pool-type tank that is 29 feet deep with horizontal dimensions of 13 feet by 7 feet, located in the containment building of the BMRC. The tank contains approximately 13,700 gallons of demineralized water. The reactor fuel consisted of standard zircalloy clad Pulse Training and Reactor (PULSTAR) type fuel elements.

Several events have occurred at the BMRC during its operational history that has led to the determination that the buildings, structures, and immediate surrounding soils are impacted from historical operations at BMRC. The following events are documented in the BMRC Historical Site Assessment (HSA).

- In 1975, and again in 1989, a leak developed in the primary cooling piping at the point where the pipe penetrates the reactor pool tank. The leakage is thought to have traversed the primary piping exterior to a point where it emerged from the foundation concrete in Room N16 and then to the sumps associated with the Cooling Water Equipment Room (Room N03).

- In 1993, the Primary Coolant Heat Exchanger located in Room N02 of the Laboratory Wing lower level, developed a primary to secondary leak that permitted small amounts of coolant to transfer to the water cascading in the cooling tower. The cooling tower and its above grade concrete structure have since been removed. The concrete basin was left in place and backfilled to grade.

- In the Laboratory Wing, the drains in the Vertical Accelerator Room (Room 115) are posted as contaminated as a result of drains backing up during operation.

- Events were discussed during interviews that relate to airborne contamination events within the Containment Building. These events resulted in areas or levels of the Containment Building being designated in whole or part as contaminated. One event was an airborne release of Co^{60} to the entire Neutron Deck. Decontamination occurred in all areas of the Containment Building except for areas near the ceiling and the ceiling. The second event was a release of Sr^{90} that contaminated areas of Room 202. The walls adjacent to the California hood in this room have plexiglass mounted to provide shielding from the strontium. There was also one event where H^3 was released in the Control Deck Upper Airlock and Upper Fuel Vault.

Figure 2: Control Deck Layout

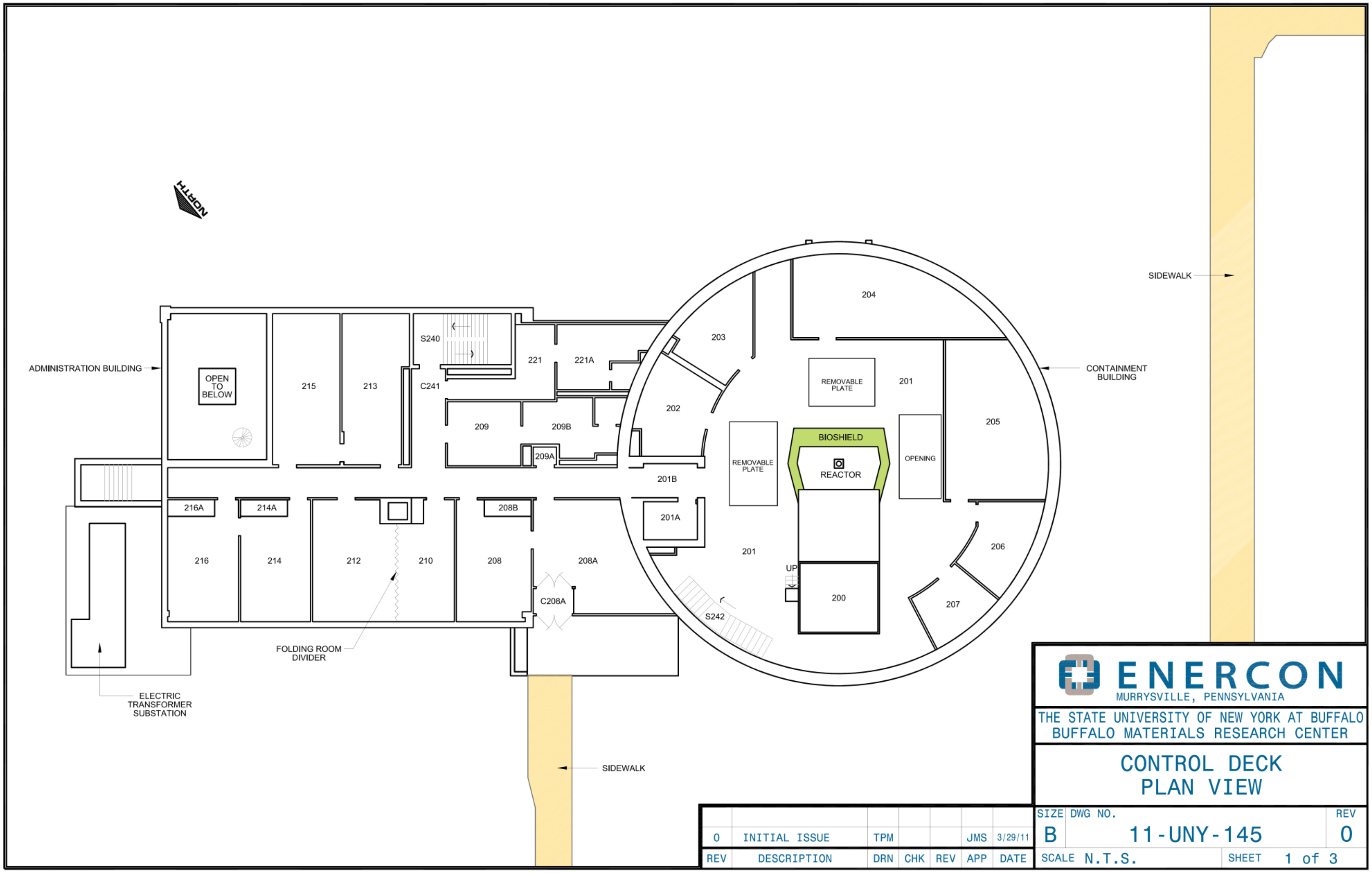


Figure 3: Gamma Deck Layout

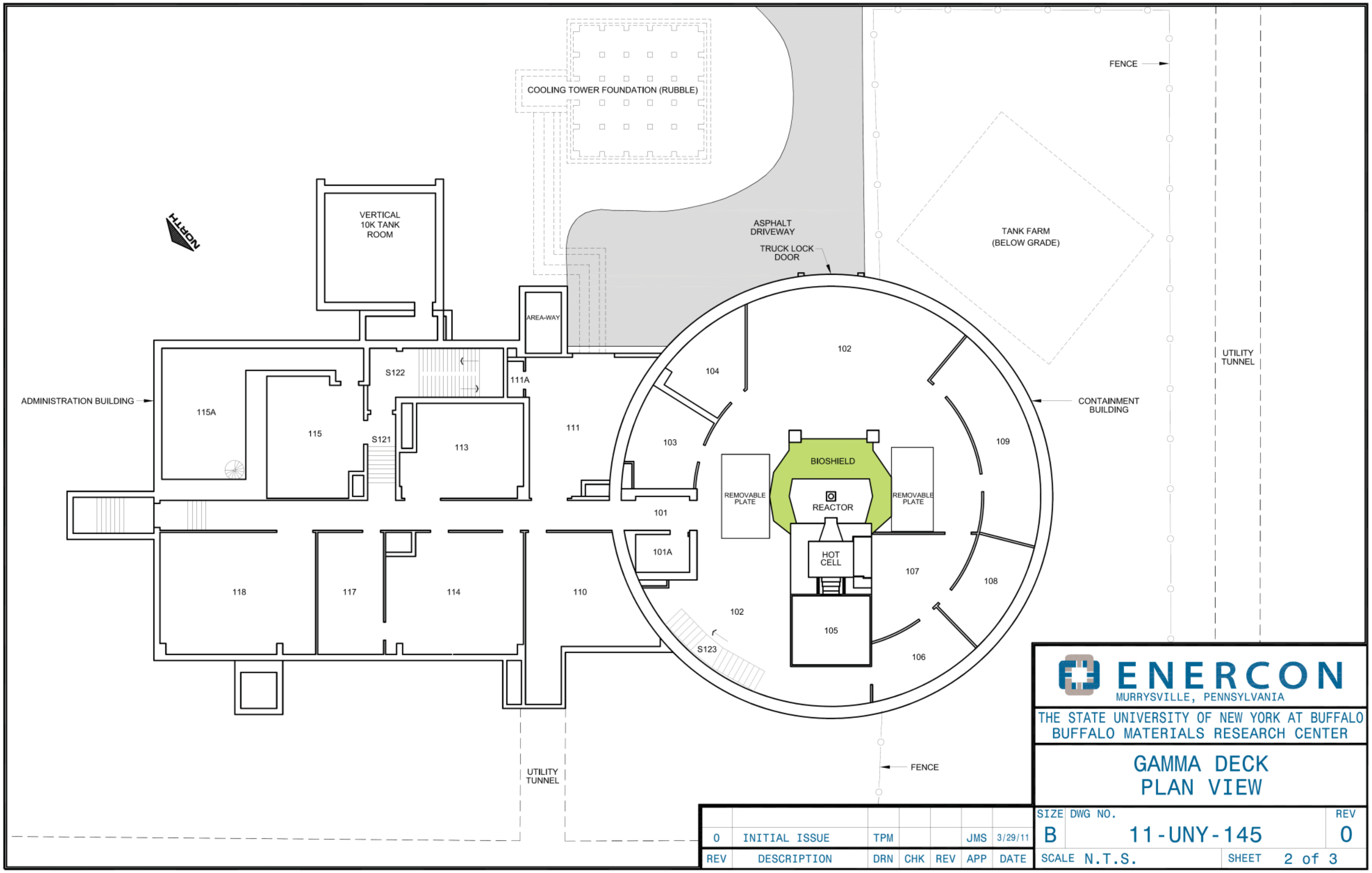
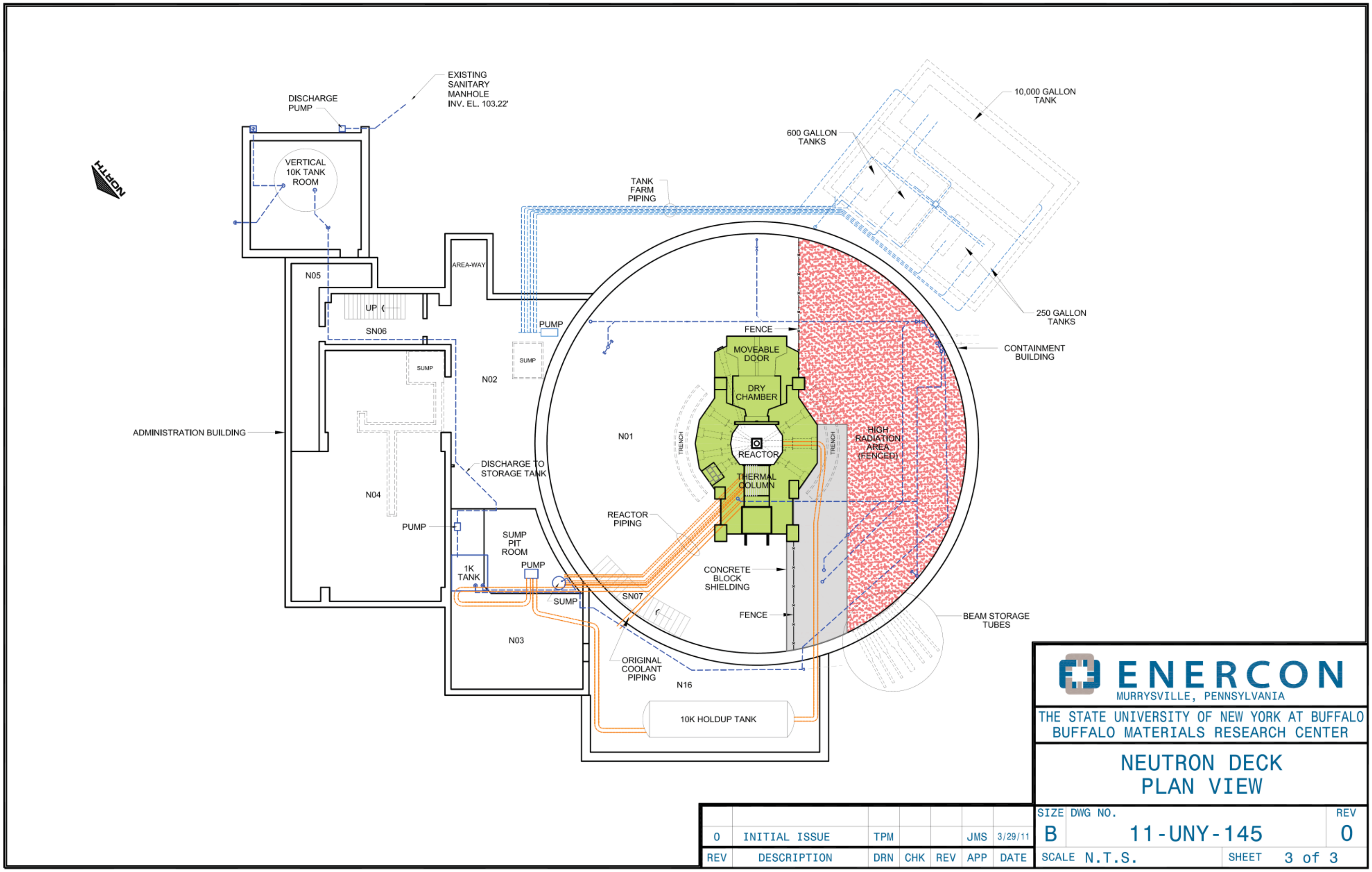


Figure 4: Neutron Deck Layout



3. SUMMARY OF TECHNICAL APPROACH

Radiological characterization surveys were performed to establish the current radiological status of the BMRC. The site characterization survey included radiological surface measurements on building surfaces and equipment in addition to volumetric measurements on surrounding soils, floors, and the reactor bioshield. The characterization surveys were designed, performed, evaluated, and documented in accordance with the following sections.

The technical approach for the characterization of the building surfaces of the BMRC was developed to meet MARSSIM criteria for FSS. A minimum quantity of survey locations for each survey unit was defined to meet a 95% certainty of detection using nonparametric statistical analyses contained in MARSSIM. Survey locations within the building were selected in Visual Sample Plan Version 6.0 using either a random or systematic triangular grid placement method. Survey technicians were also instructed to use professional judgment in identifying additional biased locations (i.e., fume hoods, elevated measurements, lab benches) to assure that the nature and extent of any residual radioactive materials present were adequately defined.

The technical approach for exterior soils was developed using MARSSIM guidance to determine the impacts, if any, to surface water and groundwater.

3.1 Radionuclides of Concern

Based upon facility operating history and historical radiological analyses, both alpha and beta/gamma emitting radionuclides may be present in the BMRC. Evaluation of radionuclides of concern identified in NUREG/CR-3474, NUREG/CR-4289, and those detected via laboratory analyses from samples collected at the facility yields a substantial list of 60 radionuclides. This list is presented in Table 3-1.

Table 3-1 Radionuclides of Concern

Nuclide	Half Life (yr)	Citation Source	Detected at BMRC
Ac-228	7.00E-04	N/A	Yes
Ag-108m	4.18E+02	NUREG/CR-3474	Yes
Ag-110m	6.84E-01	NUREG/CR-3474	No
Am-241	4.32E+02	NUREG/CR-4289	No
Ar-39	2.69E+02	NUREG/CR-3474	No

Nuclide	Half Life (yr)	Citation Source	Detected at BMRC
Ba-133	1.05E+01	NUREG/CR-3474	No
C-14	5.73E+03	NUREG/CR-3474	Yes
Ca-41	1.03E+05	NUREG/CR-3474	No
Ce-141	8.90E-02	NUREG/CR-3474	No
Cl-36	3.01E+05	NUREG/CR-3474	No
Cm-244	1.81E+01	NUREG/CR-4289	No
Co-57	7.44E-01	N/A	Yes
Co-58	1.94E-01	NUREG/CR-3474	Yes
Co-60	5.27E+00	NUREG/CR-3474 NUREG/CR-4289	Yes
Cr-51	7.58E-02	NUREG/CR-3474	No
Cs-134	2.06E+00	NUREG/CR-3474	No
Cs135	2.30E+06	NUREG/CR-3474	No
Cs-137	3.02E+01	NUREG/CR-3474 NUREG/CR-4289	Yes
Eu-152	1.36E+01	NUREG/CR-3474 NUREG/CR-4289	Yes
Eu-154	8.59E+00	NUREG/CR-3474 NUREG/CR-4289	Yes
Eu-155	4.96E+00	NUREG/CR-3474	No
Fe-55	2.70E+00	NUREG/CR-3474	No
Fe-59	1.22E-01	NUREG/CR-3474	No
H-3	1.23E+01	NUREG/CR-3474 NUREG/CR-4289	Yes
Hf-178m	3.00E+01	NUREG/CR-3474	No
Ho-166m	1.20E+03	NUREG/CR-3474	No
I-129	1.57E+07	NUREG/CR-3474 NUREG/CR-4289	No
Kr-81	2.10E+05	NUREG/CR-3474	No
Kr-85	1.07E+01	NUREG/CR-3474	No
Mn-53	3.70E+06	NUREG/CR-3474	No
Mn-54	8.56E-01	NUREG/CR-3474	No
Mo-93	3.50E+00	NUREG/CR-3474	No
Nb-92m	2.78E-02	NUREG/CR-3474	No
Nb-94	2.03E+04	NUREG/CR-3474 NUREG/CR-4289	No
Ni-59	7.50E+04	NUREG/CR-3474 NUREG/CR-4289	No
Ni-63	1.00E+02	NUREG/CR-3474 NUREG/CR-4289	Yes
Np-237	2.14E+06	NUREG/CR-4289	No
Pb-205	1.51E+07	NUREG/CR-3474	No
Pm-145	1.77E+01	NUREG/CR-3474	No
Pu-238	8.78E+01	NUREG/CR-4289	Yes
Pu-239	2.41E+04	NUREG/CR-3474 NUREG/CR-4289	Yes
Pu-240	6.60E+03	NUREG/CR-4289	Yes
Ra-226	1.60E+03	N/A	Yes
Sb-124	1.65E-01	NUREG/CR-3474	No

Nuclide	Half Life (yr)	Citation Source	Detected at BMRC
Sc-46	2.29E-01	NUREG/CR-3474	No
Se-79	6.50E+04	NUREG/CR-3474	No
Sm-146	1.00E+08	NUREG/CR-3474	No
Sm-151	9.30E+01	NUREG/CR-3474	No
Sn-121m	5.00E+00	NUREG/CR-3474	No
Sr-89	1.38E-01	N/A	Yes
Sr-90	2.88E+01	NUREG/CR-3474 NUREG/CR-4289	Yes
Tb-158	1.50E+02	NUREG/CR-3474	No
Tc-99	2.13E+05	NUREG/CR-3474 NUREG/CR-4289	No
Th-228	1.91E+00	N/A	Yes
Th-232	1.41E+10	N/A	Yes
U-233	1.59E+05	NUREG/CR-3474	No
U-234	2.46E+05	N/A	Yes
U-238	4.47E+9	N/A	Yes
Zn-65	6.69E-01	NUREG/CR-3474	No
Zr-93	1.53E+06	NUREG/CR-3474	No

Thirty-six (36) of the 60 listed radionuclides have not been detected in samples collected from the facility. Removing those radionuclides which have not been detected; radionuclides with a half-life less than one (1) year; and, naturally occurring radionuclides which have been detected only at background concentrations pares this list down to a site-specific list of fourteen (14) radionuclides as presented in Table 3-2.

Table 3-2 Site-Specific Radionuclides of Concern at BMRC

Radionuclide	Half Life (yr)	Emission	Area(s) of Interest
Ag-108m	4.18E+02	β, γ	Soil; SSCs
Am-241	4.32E+02	α, γ	Tank sediment
C-14	5.73E+03	β	Laboratory areas
Co-60	5.27E+00	β, γ	Soil; SSCs; Bioshield
Cs-137	3.02E+01	β, γ*	Soil; SSCs; Bioshield
Eu-152	1.36E+01	β, γ	Soil; SSCs; Bioshield
Eu-154	8.59E+00	β, γ	Soil; SSCs; Bioshield
H-3	1.23E+01	β	Soil; SSCs, Bioshield
Ni-63	1.00E+02	β	Soil; SSCs; Bioshield
Pu-238	8.78E+01	α, γ	Tank sediment
Pu-239	2.41E+04	α, γ	Tank sediment
Pu-240	6.60E+03	α, γ	Tank sediment
Ra-226	1.60E+03	α, γ	Discrete sources
Sr-90	2.88E+01	β	SSCs; Ventilation systems; Soil

* γ emission from Ba-137m progeny

3.2 Release Criteria

The BMRC characterization effort included evaluation of SSCs and of soils surrounding the facility. While release criteria are not required for the characterization process, consideration of likely criteria in selecting field instrumentation and determining action levels is important. Two sets of criteria were considered; one set for surface soils, and one for SSCs.

3.2.1 Soils

NRC License Termination Screening Levels contained in Appendix H of NUREG-1757 Volume 2 for surface soils were considered during the characterization process at BMRC. These criteria, listed in Table 3-3, provide a point of comparison for the soil data collected around the BMRC. In addition to the NRC Screening Value, a second screening value column has been added to compare the NRC values at 25 mrem/year to the State of New York's 10 mrem/year dose release criteria established in Radioactive Materials Guidance Document, DSHM-RAD-05-01, *Cleanup Guidelines for Soils Contaminated with Radioactive Materials*, issued by the Department of Environmental Conservation. Of the site-specific radionuclides of interest, surface soil screening levels are available for all but Ag-108m. Additional evaluation of criteria for this radionuclide will be required in the DP for the FSS.

Table 3-3 NRC Surface Soil Screening Levels

Radionuclide	NRC Screening Value for Surface Soils (pCi/g)	Screening Value Scaled for NY State* (pCi/g)
Ag-108m	None	None
Am-241	None	None
C-14	12	4.8
Co-60	3.8	1.52
Cs-137	11	4.4
Eu-152	8.7	3.48
Eu-154	8	3.2
H-3	110	44
Ni-63	2,100	840
Pu-238	2.5	1
Pu-239	2.3	0.92
Pu-240	None	None
Ra-226	0.7	0.28
Sr-90	1.7	0.68

*New York State dose requirement is 10 mrem/yr (40% of the NRC Values)

3.2.2 Surfaces

For the BMRC characterization effort, unrestricted release of material and equipment prior to a final status survey considers criteria from NRC Regulatory Guide 1.86. These criteria were considered in calculating the instrumentation counting sensitivities in Section 3.1.6. The Regulatory Guide 1.86 criteria are more conservative than the NRC screening levels for building surfaces presented in Appendix H of NUREG-1757 Volume 2 and are listed in Table 3-4. Final Status Survey criteria, if needed, will use those criteria in NUREG-1757 presented in Table 3-5.

Table 3-4 NRC Regulatory Guide 1.86 Levels for Surfaces

Radionuclide[a]	Average[b c] dpm/100 cm^2	Maximum[b d] dpm/100 cm^2	Removable[b e] dpm/100 cm^2
U-nat, U-235, U-238 and associated decay products	5,000	15,000	1,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000	15,000	1,000

[a] Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface.

Table 3-5 NRC Screening Levels for Building Surfaces

Radionuclide	NRC Screening Value for Building Surfaces (DPM/100cm^2)
Ag-108m	None
Am-241	None
C-14	3,700,000
Co-60	7,100
Cs-137	28,000
Eu-152	None
Eu-154	None
H-3	120,000,000
Ni-63	1,800,000
Pu-238	None
Pu-239	None
Pu-240	None
Ra-226	None
Sr-90	8,700

Consideration of the release criteria for building surfaces is important when selecting remediation strategies and for developing the final status survey. The release criteria for material and equipment are also important when estimating the quantity of material and equipment that will be disposed as radioactive waste.

3.3 Survey Design

Characterization surveys were designed to determine the presence of residual radioactive materials at the BMRC. In each area, professional judgment was used in conjunction with a systematic or random survey design for selection of locations that may have increased potential for residual radioactive materials. These locations were then surveyed for total alpha, total beta, removable alpha, and removable beta and general area gamma radiation dose rates.

In the survey design, interior surfaces of the administrative wing were initially considered a minimum of MARSSIM Class 3. The entire containment wing was considered MARSSIM Class1 with the exception of the containment ceiling, which was considered Class 2. At a minimum, 29 survey locations were identified per survey package. Survey packages could consist of more than one room and a survey unit could contain more than one survey package. Additional survey locations were sometimes identified by the survey technician during the survey for collection of biased survey data. As such, a minimum of 29 survey locations were identified per survey package.

The identified survey locations within a survey package received a static total alpha/beta measurement, an alpha/beta removable wipe sample, and a beta/gamma scan of the 1 square meter area surrounding the location. Gamma dose rate measurements were performed 1-meter above identified floor locations.

Background measurements were collected in air within the selected room, or alternately, during the source check in the established technician staging area (Room 210/212).

The status of these rooms at the time of the survey is documented in the photographs in Appendix A.

3.4 Survey Package Identification

During the investigation, a consistent identification system was used to ensure both the uniqueness and clarity in evaluation location and an identification of data collected. This section describes the protocol that was employed in naming the evaluation locations. Each location was assigned a unique identifier that included:

- Class (1, 2, or 3)
- Building Level
    - C for the control deck level
    - G for the gamma deck level
    - N for the neutron deck level
- Survey unit number – sequential number relative to the class and building level
- Room identifier as historically identified

This resulted in a unique identifier for each survey package. For example, the survey package 2C1-215 represents Room 215 as being survey Class 2 on the control deck in survey unit 1. In this case, adjacent Room 213 is also part of survey unit 1 and is included in the survey package 2C1-213.

3.5 Survey Instrumentation

Radiological survey instrumentation was selected to ensure that sensitivities were sufficient to detect the expected radionuclides at the minimum detection requirements. The instrumentation was calibrated in accordance with approved procedures to National Institue of Science and Technology (NIST) traceable standards. The calibration was checked daily using Th-230 and Tc-99 for alpha and beta/gamma measurements, respectively. A list of the survey instrumentation, along with the type of radiation detected, calibration sources, and the instrument use, is provided in Table 3-6.

Table 3-6 Survey Instrumentation

Instrument	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum Model 2221	Ludlum Model 43-68 Gas Proportional (126 cm^2 area)	Beta/Gama	Tc-99	Surface Static Measurements; Beta/Gamma scan measurements
Ludlum Model 2360	Ludlum Model 43-89 ZnS coated Plastic Scintillator (126 cm^2 area)	Alpha/Beta	Th-230/Tc-99	Alpha static measurements
Ludlum Model 19	Internal NaI	Gamma	Cs-137	General area exposure rates
Ludlum Model 3030E	Ludlum Model 43-10-1 ZnS internal detector	Alpha/Beta	Th-230/Tc-99	Swipe/smear counting

3.6 Minimum Detectable Activity

The minimum detectable activity (MDA), also referred to minimum detectable concentration (MDC) is the concentration of radioactivity that an instrument can be expected to detect at a 95 percent confidence level. For instruments performing direct measurements and for laboratory analyses, the MDA goal was 10-50% of applicable release criteria. For static (direct) surface measurements, with conventional detectors, the MDA was calculated using the formula:

$$\text{MDA (dpm/100cm}^2\text{)} = \left[\frac{3+3.29\sqrt{(R_b)(T_s)(1+T_s/T_b)}}{(T_s)(\varepsilon_t)}\right]$$

<u>Variables:</u>

R_b = Background count rate (cpm)
T_b = Background count time (min)
T_s = Sample Count Time (min)
ε_t = Total Instrument Efficiency (MARSSIM section 6.6.1)

The data used to calculate the MDA for the instrumentation used during the BMRC characterization are shown on the survey records in stored at the BMRC. The typical *a priori* MDA is listed in Table 3-7.

Table 3-7 Instrumentation MDA

Instrument	Detector Type	Radiation Detected	Average MDA (dpm/100 cm^2)
Ludlum Model 2360	Ludlum Model 43-89	Alpha	47
		Beta	612
Ludlum Model 2360	Ludlum Model 43-68	Alpha	59
		Beta	393
Ludlum Model 3030E	Ludlum Model 43-10-1	Alpha	12
		Beta	159
Ludlum Model 2221	Ludlum Model 43-68	Beta	271

3.7 Preliminary Survey Units and Classifications

The BMRC was segmented into logical areas of similar history and/or construction as survey units. Utilizing guidance available in the MARSSIM, an initial classification of either impacted or non-impacted was assigned to each survey unit. Non-impacted areas are areas where historical information and/or process knowledge does not indicate the existence of residual radioactive material, and does not indicate any reason to suspect the presence of residual radioactive materials. Impacted areas are areas where historical information and/or process knowledge indicates a potential presence of residual radioactive materials. During the characterization phase of the BMRC, all areas within the building were considered as impacted. Based on the levels of potential residual radioactive materials, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. Class 1 areas have the greatest potential for residual activity while Class 3 areas have the least potential for impacted areas. Each classification will typically be bounded by areas classified one step lower to provide a buffer zone around the higher class. Exceptions occur when an area is surrounded by a significant physical barrier that would make transport of residual activity unlikely from one area to the adjacent area. In such cases, each area is classified solely on its own merit using the most reliable information available. The class definitions provided below are from Section 4.4 of the MARSSIM.

Class 1

> "Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys). Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material high specific activity. Note that areas containing contamination in excess of the $DCGL_w$ prior to remediation should be classified as Class 1 areas."

Class 2

> "These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_w$. To justify changing an area's classification from Class 1 to Class 2, the existing data (from the HSA, scoping surveys, or characterization surveys) should provide a high degree of confidence that no individual measurement would exceed the $DCGL_w$. Other justifications for this change in an area's classification may be appropriate based on the outcome of the DQO process. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (e.g., process facilities), 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of some buildings or rooms subjected to airborne radioactivity, 5) areas where low concentrations of radioactive materials were handled, and 6) areas on the perimeter of former contamination control areas."

Class 3

> "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 $DCGL_w$, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification."

The preliminary list of survey units is presented in Table 3-8.

Table 3-8 Survey Unit Summary

Class	Deck	Survey Unit	Room ID	Room Description
1	C	1	200	Electronics shop
1	C	2	201NW	Control Deck Northwest quadrant
1	C	3	202	Laboratory
1	C	4	203	Laboratory
1	C	5	204	Fan Room
1	C	6	205	Reactor Control
1	C	7	206	Office
1	C	8	207	Superintendent Office
1	C	9	201A	Fuel Storage Vault
1	C	9	201B	Airlock
1	C	10	201NE	Control Deck Northeast quadrant
1	C	11	201SE	Control Deck Southeast quadrant
1	C	12	201SW	Control Deck Southwest quadrant
1	C	12	S242	Stairs from Gamma Level to Control Level
1	C	13	201HC	Hot Cell Roof
1	C	14	201U	Area above airlock
1	C	15	203U	Area above rooms 202/203
1	C	16	204U	Area above room 204
1	C	17	205U	Area above room 205
1	C	18	206U	Area above room 206
1	C	19	207U	Area above room 207
2	C	1	213	Laboratory
2	C	1	215	Laboratory
2	C	2	Ceiling	Containment ceiling
3	C	1	208	General Manager's Office
3	C	1	209	Men's Locker and Change Area
3	C	1	210	Conference Room
3	C	1	212	Conference Room(cont)
3	C	1	214	Secretary Office
3	C	1	216	Directors Office
3	C	1	221	Women's Rest Area
3	C	1	208A	Reception Area
3	C	1	208B	General Manager Closet
3	C	1	209A	Janitor's Closet
3	C	1	209B	Men's Toilet Room
3	C	1	214A	Secretary Office Closet
3	C	1	216A	Directors Office Closet
3	C	1	221A	Women's Toilet Area
3	C	1	C241	Corridor
3	C	1	OTB	Vertical Accelerator upper
3	C	1	S240	Stairs Hall
1	G	1	101	Airlock
1	G	1	101A	Janitor's Closet
1	G	2	102NW	Gamma Deck
1	G	2	S123	Stairs

Class	Deck	Survey Unit	Room ID	Room Description
1	G	3	103	Chemistry Lab
1	G	4	104	Medical Lab
1	G	5	105	Work Room
1	G	6	106	Change Room
1	G	7	107	Hot Cell Service Room
1	G	8	108	Hot Chemistry Lab
1	G	9	109	Storage Area
1	G	10	113	Shop
1	G	11	115	Laboratory
1	G	12	114	Health Physics/Counting Room
1	G	13	115A	Vertical Accelerator lower
1	G	14	102NE	Gamma Deck NE area
1	G	15	102E	Gamma Deck East area
1	G	16	102S	Gamma Deck South area
1	G	17	Hotcell	Hot Cell
1	G	18	102W	Gamma Deck West area
2	G	1	117	Classroom
2	G	1	118	Horizontal Accelerator
3	G	1	110	Electrical Room
3	G	1	111	Air Conditioning Equipment Room
3	G	1	111A	Janitor’s Closet
3	G	1	S121, S122, & C119	Stair Hall
1	N	1	N01 –A	Neutron Deck
1	N	2	N16	N16 tank room
1	N	3	SN07	Stairs
1	N	4	N02AIR	Air shaft to exterior
1	N	5	N03	Cooling Water Equipment Room
1	N	6	N03PIT	1000 gal tank pit
1	N	7	N02-North	Air Conditioning Equipment Room – North section
1	N	8	N02-South	Air Conditioning Equipment Room – South section
1	N	9	N01-B	Neutron Deck
1	N	10	N01-C	Neutron Deck
1	N	11	N01-D	Neutron Deck
1	N	12	N01-E	Neutron Deck
1	N	13	N01-F	Neutron Deck
1	N	14	N01-G	Neutron Deck
2	N	1	10K	10,000 gal Tank Room
3	N	1	N04	Air Conditioning Equipment Room
3	N	1	N05	Air Plenum
3	N	1	SN06	Stair Hall

4. CURRENT RADIOLOGICAL CONDITIONS

The following sections summarize the results of the radiological characterization effort. Per MARSSIM guidance all measurement results, whether positive, negative, or less than MDA, were included in the calculations of the mean, maximum, and standard deviation. The results are summarized by survey unit. The completed survey packages have been retained for future reference, but not included in this report.

4.1 Administrative Wing

Several rooms in the administrative wing have radioactive materials stored in them. Due to elevated background from the radioactive material, it was difficult to determine the presence of residual radioactive materials. It is ENERCON's belief that these areas do not have residual radioactive materials present and after the stored radioactive material is packaged and shipped off site, these measurements will be verified.

The rooms were surveyed for total alpha, total beta, removable alpha, and removable beta. Twenty-nine (29) building surface locations were selected in each room. In addition, biased locations were selected (e.g. fume hoods, drains, and sinks) at the discretion of the survey technician.

Each subsection below contains a table which summarizes characterization survey results for that area.

4.1.1 Control Deck

The following rooms on the control deck of the administrative wing show no elevated measurements of residual radioactive materials:

- Locker Rooms (Rooms 209, 221 and 221A)
- Conference Room (Room 212)
- Offices (Rooms 208, 210, 214 and 216)
- Reception Area (Room 208A)
- Men's Toilet Room (Room 209B)
- Janitor Closet (Room 209A)
- Two labs (Rooms 213 and 215)

Table 4-1 Administrative Wing Control Deck Survey Results Summary

		Removable Alpha Measurement Results (DPM/100cm²)			Removable Beta Measurement Results(DPM/100cm²)			Static Alpha Measurement Results(DPM/100cm²)			Static Beta Measurement Results(DPM/100cm²)		
Survey Unit	#	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev
2C1	172	1	15	3	11	101	23	19	67	16	218	1537	341
3C1	441	0	6	1	8	90	17	17	139	28	258	4310	673

4.1.2 Gamma Deck

The following rooms on the gamma deck of the administrative/laboratory wing show no elevated measurements of residual radioactive materials:

- Electrical Service Room (Room 110)
- Mechanical Room (Room 111)
- Machine Shop (Room 113)

Several rooms on the gamma deck of the administrative/laboratory wing contain radioactive materials. These rooms are:

- Horizontal Accelerator (Room 118) with an adjacent cave area
- Class Room (Room 114)
- Health Physicist Office (Room 117)
- Vertical Accelerator Room (Rooms 115 and 115A)

Table 4-2 Administrative Wing Gamma Deck Survey Results Summary

		Removable Alpha Measurement Results (DPM/100cm²)			Removable Beta Measurement Results (DPM/100cm²)			Static Alpha Measurement Results (DPM/100cm²)			Static Beta Measurement Results (DPM/100cm²)		
Survey Unit	#	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev
1G10	76	1	6	1	4	64	11	27	89	17	89	659	175
1G11	76	1	12	2	2	21	5	37	67	19	200	979	210
1G12	78	0	3	1	2	37	6	23	84	17	619	4436	1072
1G13	76	0	3	1	8	59	15	31	75	19	1019	17159	3007
2G1	144	0	6	1	12	96	24	18	50	13	242	960	272
3G1	174	0	6	1	3	48	10	27	58	14	100	1330	217

4.2 Sub-Basement

The sub-basement, located below the gamma deck of the administrative/laboratory wing includes rooms that house the Holdup Tank Cubicle (Room N16), the Cooling Water Equipment Room and the Facility Sumps (Room N03), the lower Air Conditioning Equipment Room (Room N02), the Air Handling Duct Room (Room N04) and access to the 10K tank enclosure (Room N05). Room NO3 houses a 1000 gallon tank. This tank contains approximately 6 inches of sludge with elevated levels of radioactive materials. Room N16 houses a 5,000 gallon N16 decay tank which is internally contaminated with radioactive materials.

Table 4-3 Sub-Basement Survey Results Summary

		Removable Alpha Measurement Results (DPM/100cm^2)			Removable Beta Measurement Results (DPM/100cm^2)			Static Alpha Measurement Results (DPM/100cm^2)			Static Beta Measurement Results (DPM/100cm^2)		
Survey Unit	#	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev
1N2	29	1	6	2	47	112	27	6	25	8	4664	6941	1076
1N4	33	1	6	2	2	32	7	27	67	18	2477	49163	8823
1N5	38	1	3	1	5	48	12	27	58	17	3569	16147	3511
1N6	27	1	3	2	13	213	41	410	823	99	24499	364513	74158
1N7	33	1	6	2	16	324	58	24	75	20	5472	55469	12542
1N8	34	1	3	1	1	21	5	8	50	12	241	1056	272
2N1	30	0	0	0	43	149	35	21	50	15	No Data	No Data	No Data
3N1	93	1	6	2	3	64	11	7	69	14	299	3463	577

4.3 Containment Wing

The containment wing of the BMRC is accessible via airlocks from the administrative wing control and gamma deck levels. Exterior access is via a truck door on the gamma deck.

4.3.1 Control Deck

The Control Deck (top level) houses a dual-hook over head crane with one 10-ton and one 2-ton hook on tracks at the ceiling level. Crane access to the lower levels is via a floor plate on the Control Deck and a webbing cover on a floor penetration from the Gamma Deck to the Neutron Deck. In addition to the reactor pool, the Control Deck houses:

- Reactor Control Room (Room 205)
- Superintendent's Office (Room 207)

- Two labs (Rooms 202 and 203)
- Fan Room (Room 204)
- Office (Room 206)
- Airlock (Room 201B) for access to the upper or First Floor of the Administrative Wing
- New Fuel Storage Room (Room 201A) located within the doors of the airlock

Most of these rooms were being used to store legacy radioactive wastes when surveys were performed. The elevated background from the radioactive material storage areas made it difficult to determine the presence of residual radioactive materials on building surfaces in these areas. The limited radiological measurements collected in these areas did not identify removable or fixed radiological activity; therefore, the building surfaces are not believed to be contaminated. Following the removal of these legacy materials for offsite disposal, additional measurements will be collected in these areas. Room 202 was the location of a release of Sr-90 that contaminated the walls adjacent to the California hood. These walls are affixed with Plexiglas coverings as a mitigating measure to provide beta shielding.

Table 4-4 Containment Wing Control Deck Survey Results Summary

		Removable Alpha Measurement Results (DPM/100cm^2)			Removable Beta Measurement Results (DPM/100cm^2)			Static Alpha Measurement Results (DPM/100cm^2)			Static Beta Measurement Results (DPM/100cm^2)		
Survey Unit	#	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev
1C1	30	1	3	1	7	64	15	126	238	60	231	2084	508
1C2	40	1	3	1	8	74	16	80	179	37	200	704	241
1C3	41	2	9	3	41	138	36	64	149	39	12226	241518	42125
1C4	38	1	12	3	21	96	23	41	139	34	83	2001	355
1C5	38	1	6	2	18	74	21	122	218	54	225	649	196
1C6	38	1	6	2	23	96	24	109	248	46	148	1470	265
1C7	38	2	12	3	28	170	36	101	188	37	476	13016	2094
1C8	45	1	6	1	6	48	12	84	218	55	220	704	212
1C9	29	4	18	5	14	80	20	531	1280	320	10904	71732	17243
1C10	34	1	12	3	10	96	21	111	208	45	470	2692	477
1C11	38	2	9	2	11	59	17	95	407	95	127	697	177
1C12	68	4	18	4	6	59	12	111	575	81	1733	32484	4619
1C14	26	1	9	2	4	37	10	172	267	38	1101	2050	362
1C15	28	1	6	2	3	37	10	55	409	79	393	1270	331
1C16	29	1	6	2	5	32	11	72	226	62	1028	3122	821
1C17	29	3	12	3	3	53	12	149	936	209	733	3267	828
1C18	29	4	15	4	11	69	21	181	259	48	849	1667	377
1C19	25	0	3	1	4	32	9	211	326	50	832	1177	200

4.3.2 **Gamma Deck**

The middle level or Gamma Deck houses:

- Hot Cell
- Hot Cell Work Room (Room105)
- Locker Room (Room 106)
- Hot Chemical Lab (Room 108)
- Medium Chemical Lab (Room 104)
- Activation Analysis Lab (Room109)
- Lower Fuel Vault (Room 101A)
- Chemical Lab (Room 103)
- Airlock (Room 101) for access between the administrative wing and containment

This level also has a Truck Lock that provides vehicular and equipment access to the outside. Prior to 1979, this level housed the waste compaction system. Wastes generated at BMRC and under the site-wide radiological program were collected and compacted for volume reduction in this system.

Most rooms on the Gamma Deck were being used to store radioactive material which makes it difficult to determine the presence of residual radioactive materials on building surfaces Additional measurements will be collected on building surfaces in these areas following waste removal to verify residual radioactivity levels. All items inside the Hot Cell, including the Hot Cell, should be considered radioactive waste.

Table 4-5 Containment Wing Gamma Deck Survey Results Summary

		Removable Alpha Measurement Results (DPM/100cm^2)			Removable Beta Measurement Results (DPM/100cm^2)			Static Alpha Measurement Results (DPM/100cm^2)			Static Beta Measurement Results (DPM/100cm^2)		
Survey Unit	**#**	**Mean**	**Max**	**Std Dev**	**Mean**	**Max**	**Std Dev**	**Mean**	**Max**	**Std Dev**	**Mean**	**Max**	**Std Dev**
1G1	44	2	9	3	2	21	5	96	208	42	188	407	125
1G2	30	2	9	2	2	59	11	62	134	33	694	3074	1068
1G3	38	2	6	3	7	64	16	50	169	35	204	2947	518
1G4	38	1	6	2	6	43	13	7	60	15	4851	33706	6862
1G5	38	3	15	4	3	43	9	124	192	34	324	2011	407
1G6	40	2	12	3	7	64	14	155	259	45	1802	3849	942
1G7	38	2	6	2	11	69	20	56	209	51	1124	7763	1632
1G8	39	3	15	4	20	165	39	91	217	44	6727	32540	8022
1G9	42	2	12	3	2	27	6	93	167	37	10227	151217	25043

4.3.3 Neutron Deck

The Neutron Deck (lower level) is divided by caging into two sections. The open area includes access to the Patient Treatment Room which is lined with wood. The Patient Treatment Room was included in the original design of the BMRC to be a location to provide neutron treatments to cancer patients. There have been no indications that it was ever used for this purpose. The enclosed area includes the rerouted Primary Coolant Piping and access to the seven Beam Storage Tubes. The Beam Storage Tubes extend 10 feet through the Containment Building wall into the surrounding soil. They were used as storage-for-decay of high dose reactor components or experiments. At the point where they exit the containment building wall, they are approximately 10 feet underground. The balance of the neutron deck is open area. The Neutron Deck is the area where the majority of radioactive waste materials awaiting disposal were being stored. No removable radioactive material was found during the characterization survey. Upon removal of the stored radioactive materials, a more detailed investigation will be performed.

Table 4-6 Containment Wing Neutron Deck Survey Results Summary

		Removable Alpha Measurement Results (DPM/100cm^2)			Removable Beta Measurement Results (DPM/100cm^2)			Static Alpha Measurement Results (DPM/100cm^2)			Static Beta Measurement Results (DPM/100cm^2)		
Survey Unit	#	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev	Mean	Max	Std Dev
1N1	19	1	6	2	12	48	18	139	334	87	2092	14488	3389
1N9	35	2	9	3	8	53	14	336	593	89	11993	167287	35497
1N10	38	0	6	1	2	21	6	0	0	0	21152	200489	45808
1N11	13	0	0	0	1	5	2	1	10	3	192605	1888420	512482
1N12	41	5	15	4	18	122	29	86	198	53	181	1801	429
1N13	38	1	6	2	7	48	13	258	443	79	3216	22262	4683
1N14	38	1	6	2	17	122	27	109	308	68	674	3202	962

4.4 Concrete Cores

Concrete core samples were collected to assess volumetric activity in the neutron deck floor and the bioshield. In addition, the concrete cores in the neutron deck were used to determine if a preferential pathway existed between the neutron deck floor and the bedrock. The concrete cores were field scanned for gross gamma radioactivity prior to being cut into approximate 1 inch thick cross-sections for radioanalysis. Cross-sections that exhibited elevated count rates were sent offsite for analysis. If no elevated count rate was detected at any cross-sectional location in the core, at a minimum, the end cross-section was submitted.

4.4.1 **Floor Cores**

Six (6) locations on the Neutron Deck floor were chosen to collect 3-inch diameter concrete cores. The locations are shown in Figure 5. Field scans of the cores indicated no residual radioactive materials were present. Floor core #1 was collected through the proposed elevator pit and was therefore not as thick as the remaining floor cores. Locations #5 through #6 were cored through the concrete floor to the bedrock. During construction, the concrete floor of the neutron deck was poured directly on the excavated bedrock to an approximate depth of 30 inches. The concrete cores proved that the concrete was poured directly on the bedrock because no sand or gravel bedding layer was found in any location. Therefore a preferential pathway for radioactive materials does not exist between the concrete floor and the bedrock. The analytical results indicate essentially background activities for the evaluated radionuclides. A summary of the laboratory analytical results for the floor cores is included in Table 4-7.

Figure 5: Floor Core Locations

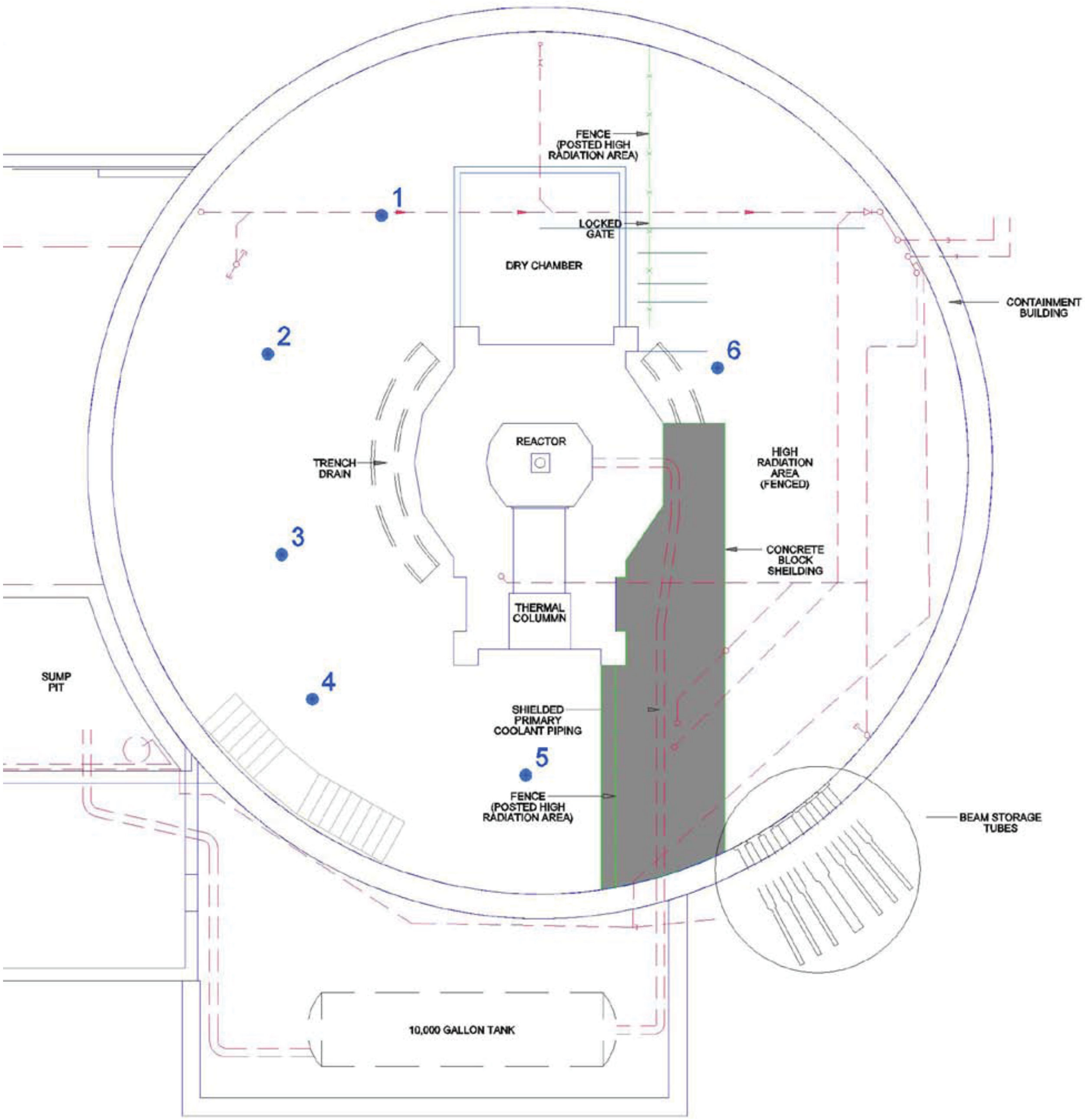


Table 4-7 Concrete Floor Cores Analytical Results Summary

Core ID>	FLOOR 1		FLOOR 2		FLOOR 3		FLOOR 4		FLOOR 5		FLOOR 6	
Radionuclide	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)
Ag-110M	-0.03	0.04	0.01	0.06	0.00	0.02	-0.01	0.02	-0.02	0.05	-0.01	0.05
Co-58	-0.04	0.06	-0.04	0.06	-0.01	0.02	0.00	0.03	0.00	0.06	-0.02	0.06
Co-60	0.09	0.06	0.00	0.07	0.01	0.03	0.00	0.02	-0.01	0.05	0.00	0.05
Cs-134	-0.03	0.05	0.01	0.06	0.00	0.03	0.00	0.03	0.02	0.05	0.00	0.05
Cs-137	0.01	0.06	0.00	0.06	0.00	0.02	0.00	0.03	0.01	0.06	0.00	0.06
Eu-152	0.08	0.15	-0.03	0.13	0.00	0.07	0.03	0.09	0.05	0.14	-0.08	0.15
Eu-154	0.02	0.09	-0.02	0.08	0.01	0.03	0.01	0.04	0.02	0.08	-0.03	0.09
H-3	3.12	0.53	3.64	0.51	2.57	0.51	2.08	0.53	5.55	0.57	2.37	0.54
K-40	5.99	0.25	6.09	0.53	3.47	0.28	3.65	0.20	4.46	0.45	2.81	0.55
La-140	-0.12	0.12	-0.04	0.20	-0.06	0.07	-0.01	0.11	-0.01	0.18	-0.01	0.20
Mn-54	0.04	0.06	-0.01	0.05	0.01	0.03	0.00	0.03	-0.02	0.05	-0.02	0.06
Ra-226	1.66	1.02			2.02	0.43			2.02	1.02		
Sb-124	0.00	0.06	-0.01	0.07	0.00	0.03	0.02	0.04	0.00	0.07	-0.02	0.07
Th-228	0.28	0.11	0.22	0.08	0.12	0.04	0.14	0.05	0.21	0.08	0.17	0.09
Th-232	0.37	0.17	0.29	0.19			0.20	0.08				

Note: Blank field indicates no reported result

4.4.2 Bioshield Cores

Six (6) locations along two (2) walls of the bioshield face (BF) on the neutron deck were chosen for core samples as shown in Figure 6. Three (3) cores were advanced in-line with the beam tubes until contact was made with the original ¼ inch aluminum liner at the inner surface of the bioshield. This resulted in a total length for each core of approximately 70 inches.

Evaluation of the data for the bioshield core samples indicates measureable activity in certain areas exceeds 12 inches in depth. The charts below show the Co-60 analytical results of the two sets of bioshield core data in a graphical manner of depth vs. activity. The Co-60 results are used as being representative of the overall results. Chart 1 presents Co-60 results vs. depth from the liner for the bioshield core samples on wall 4 and Chart 2 presents the data for bioshield core samples on wall 5. A summary of the bioshield core analytical results is presented in Table 4-8.

Figure 6: Bioshield Core Locations

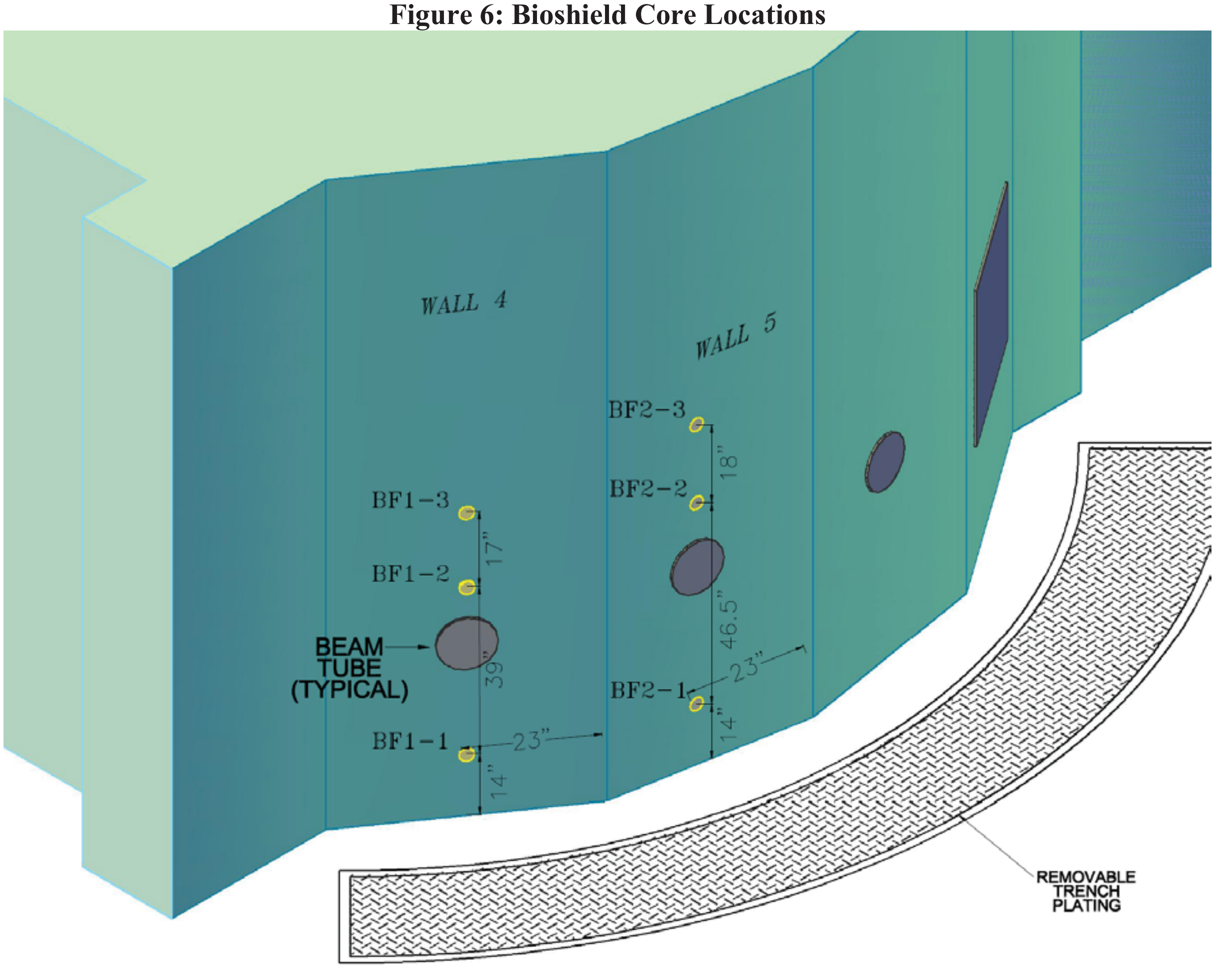


Chart 1: Bioshield Cores Wall 4

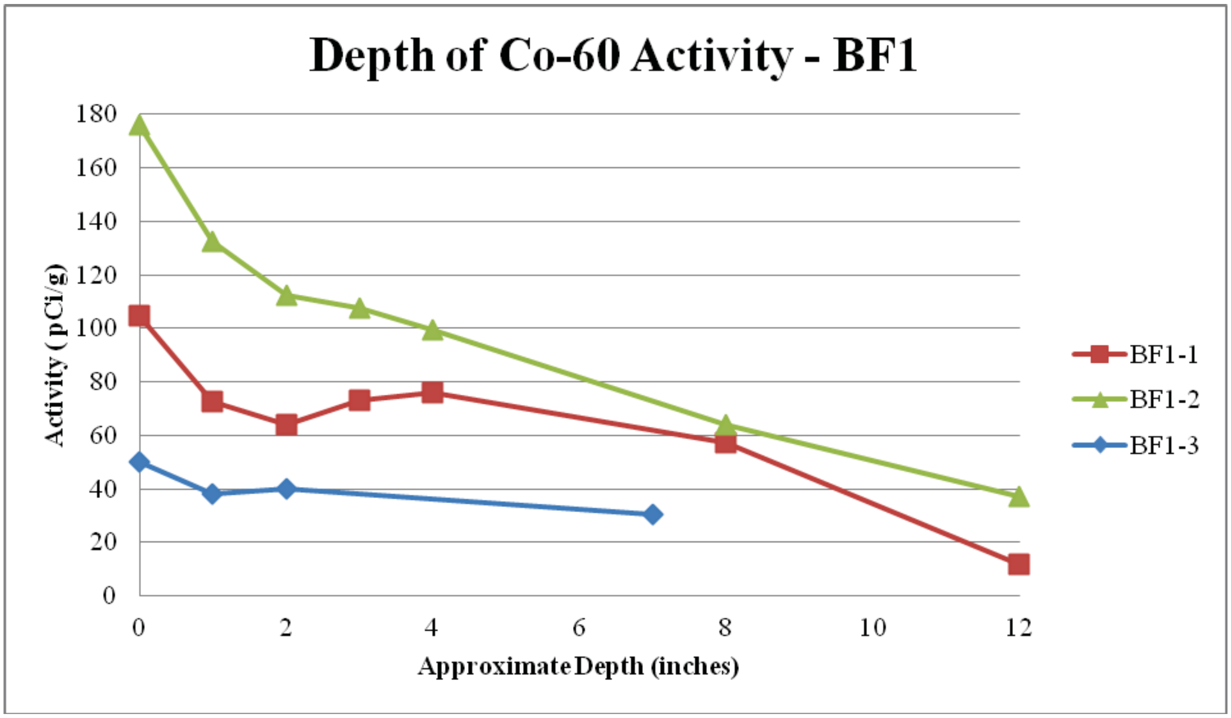


Chart 2: Bioshield Cores Wall 5

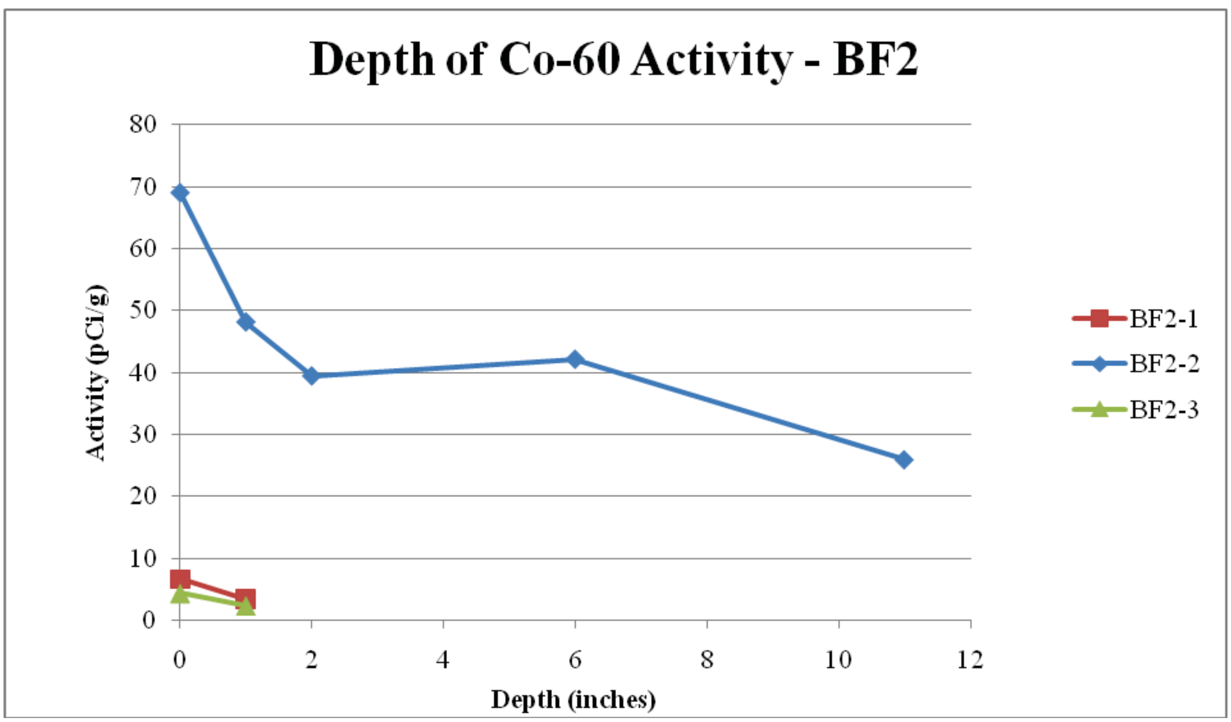


Table 4-8 Bioshield Cores Analytical Results Summary

Core ID>	BF1-1		BF1-2		BF1-3		BF2-1		BF2-2		BF2-3	
Radionuclide	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)
Ag-110M	0.02	0.24	0.08	0.28	0.06	0.16	0.00	0.05	0.03	0.18	-0.01	0.04
Co-58	0.10	0.35	0.23	0.42	0.01	0.24	0.03	0.08	0.06	0.27	0.00	0.06
Co-60	104.90	0.30	175.90	0.32	50.08	0.18	6.68	0.07	69.10	0.22	4.41	0.05
Cs-134	0.03	0.21	0.02	0.25	0.06	0.15	0.00	0.05	0.05	0.16	0.01	0.04
Cs-137	0.10	0.25	0.19	0.30	0.05	0.17	0.01	0.06	0.06	0.20	0.01	0.05
Eu-152	5.01	0.53	10.75	0.63	3.89	0.37	0.31	0.13	4.50	0.42	0.29	0.11
Eu-154	2.95	0.35	3.85	0.39	1.78	0.25	0.17	0.08	1.94	0.28	0.05	0.06
H-3	141.00	2.37	290.00	3.70	154.00	2.47	8.01	1.18	110.00	2.03	19.70	0.97
K-40							0.61	0.29	1.09	0.35	0.70	0.21
La-140	0.11	0.41	0.06	0.56	0.08	0.33	-0.02	0.13	0.07	0.34	0.01	0.10
Mn-54	0.12	0.31	0.17	0.36	0.01	0.21	0.02	0.07	0.04	0.24	0.00	0.05
Ra-226												
Sb-124	0.11	0.27	0.08	0.33	0.07	0.19	0.00	0.06	0.08	0.21	0.00	0.05
Th-228												
Th-232												

Note: Blank field indicates no reported result

4.5 Exterior Soils and Building Surfaces

4.5.1 Sub-Surface Borings

A total of 19 sub-surface soil boring locations were identified in the immediate vicinity of the building exterior including the area surrounding the liquid waste tanks as shown in Figure 7. One location (BMRC-018) was excluded because of its proximity to buried utilities.

Figure 7: Soil Bore Locations

Borings were advanced by a split spoon sampler until refusal. Samples were collected in 2-foot intervals. All samples were field scanned at the time of collection for gross gamma radioactivity. At a minimum the first (surface) and last (refusal) samples from each boring were sent for laboratory analysis. None of the collected samples indicated presence of groundwater.

The soil boring identified as BMRC-019 was advanced at the location of the former cooling tower which was known to have leaked low levels of radioactive liquids. The analyzed sample results only show the presence of naturally occurring radioactive materials.

Examining the analytical data for the soil samples shows three (3) radionuclides as detectable. These radionuclides are Cs-137, H-3, and Ni-63. Of these radionuclides, Ni-63 appears in the highest concentration with a maximum value of 11.9 pCi/g which is less than 1% of the NRC screening level and approximately 1.4% of the scaled NY State screening level. In terms of percent of screening level, H-3 appears in one sample at approximately 4% of the NRC screening level which is approximately 9% of the scaled NY State screening level. A summary of the analytical results is provided in Table 4-9 on the following page.

4.5.2 **Surface Water and Groundwater Samples**

No surface water or groundwater samples were collected during characterization. As identified in the NRC's Safety Evaluation Report, the SUNY UB campus is covered with a dense glacial clay overburden that is generally 10 to 20 meters thick. Below the overburden is bedrock consisting of a combination of two limestone formations above five dolomite formations on top of a shale layer. The bedrock layers are approximately 1000 feet in total depth. Additionally, the NRC Safety Evaluation Report for the BMRC describes the water movement in the overburden as extremely slow with percolation only in the top foot of soil.

During the characterization effort, borings advanced to the upper surface of bedrock returned no groundwater. The sample analyses showed only low levels of radionuclide concentrations which may be attributable to the facility.

There are no surface water features in the immediate vicinity of the reactor. The nearest surface water feature is a small pond 0.6 miles south of the BMRC located in McCarthy Park.

Table 4-9 Soil Borings Analytical Results Summary

Bore ID>	BMRC-001		BMRC-002		BMRC-003		BMRC-004		BMRC-005		BMRC-006	
Radionuclide	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)
Ac-228	3.23	0.20	1.47	0.20	2.07	0.24	1.65	0.19			2.08	0.17
Ag-108M	0.01	0.04	0.02	0.05	0.00	0.04	0.00	0.04	0.02	0.10	0.00	0.04
Ag-110M	0.00	0.07	-0.01	0.08	0.03	0.08	0.01	0.07	0.02	0.16	-0.01	0.06
Co-58	0.01	0.16	-0.03	0.21	-0.04	0.20	0.01	0.18	-0.02	0.38	0.04	0.17
Co-60	0.03	0.06	0.02	0.06	0.03	0.07	0.02	0.06	0.02	0.11	-0.01	0.04
Cs-137	0.09	0.06	0.08	0.07	0.07	0.08	0.03	0.06	-0.01	0.12	0.07	0.05
Eu-152	0.04	0.16	0.00	0.15	0.04	0.16	0.10	0.14	0.08	0.30	0.06	0.13
Eu-154	0.03	0.10	0.03	0.10	-0.02	0.10	0.01	0.08	0.05	0.14	0.00	0.07
H-3	0.09	0.89	2.91	0.89	-0.31	0.88	3.72	0.88	-0.39	0.86	-0.69	0.89
K-40	19.46	0.39	17.55	0.46	15.92	0.47	17.77	0.40	19.73	1.25	15.46	0.40
La-140	10.06	67.08	10.81	63.54	40.84	90.58	27.19	87.05	10.64	202.10	8.39	74.83
Mn-54	0.02	0.07	0.03	0.08	-0.01	0.08	0.01	0.07	0.03	0.15	0.03	0.07
Ni-63	9.17	0.83	11.90	0.92	0.25	0.60	0.34	0.66	-0.15	0.87	0.05	0.89
Ra-226	2.24	1.08	1.99	1.18	1.22	0.73	1.82	1.02	1.63	1.17	1.94	0.89
Sb-124	0.02	0.23	0.05	0.25	-0.01	0.25	0.01	0.22	-0.02	0.51	0.10	0.20
Th-228	0.99	0.10	0.79	0.11	0.85	0.10	0.98	0.08	0.65	0.17	0.81	0.08
Th-232	0.73	0.19	0.72	0.22	0.62	0.23	0.73	0.17	0.67	0.23	0.66	0.16
GR-A	7.27	4.89	8.46	4.30	5.88	4.29	4.83	3.87	11.50	4.95	9.43	2.85
GR-B	40.00	3.00	36.80	2.92	34.70	2.91	28.60	2.64	35.60	3.00	38.00	2.86

Note: Blank field indicates no reported result

Table 4-9 Soil Borings Analytical Results Summary (Cont.)

Bore ID>	BMRC-007		BMRC-007A		BMRC-008		BMRC-009		BMRC-010		BMRC-011		BMRC-012	
Radionuclide	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)
Ac-228	1.00	0.12	1.98	0.17	2.73	0.52			1.54	0.14	1.19	0.18		
Ag-108M	0.01	0.03	0.00	0.05	0.03	0.12	0.01	0.04	0.01	0.05	0.00	0.05	0.00	0.05
Ag-110M	0.02	0.05	0.03	0.08	0.02	0.20	0.03	0.07	0.02	0.08	0.03	0.07	0.00	0.07
Co-58	0.02	0.12	0.02	0.24	0.02	0.47	0.02	0.17	0.02	0.18	-0.04	0.18	-0.03	0.20
Co-60	0.00	0.04	0.03	0.07	0.02	0.11	0.02	0.05	0.04	0.06	0.03	0.06	0.00	0.06
Cs-137	0.00	0.04	-0.02	0.07	0.09	0.16	0.05	0.05	0.08	0.06	0.05	0.06	0.04	0.06
Eu-152	0.03	0.09	0.00	0.18	0.10	0.39	0.06	0.13	0.05	0.17	0.01	0.14	0.04	0.16
Eu-154	0.01	0.06	0.00	0.09	0.03	0.21	0.00	0.07	0.04	0.10	-0.01	0.09	0.01	0.10
H-3	-0.46	0.78	4.12	0.88	-0.18	0.86	-0.24	0.81	-0.58	0.83	-0.55	0.84	-0.40	0.87
K-40	14.85	0.27	15.10	0.65	17.93	1.40	17.13	0.43	15.22	0.46	19.14	0.45	15.87	0.50
La-140	-23.60	53.96	5.54	115.00	28.14	283.90	13.33	85.31	19.13	79.09	-0.65	107.00	-11.06	99.19
Mn-54	0.02	0.05	0.01	0.09	0.04	0.19	0.01	0.07	0.00	0.08	0.03	0.08	0.03	0.08
Ni-63	1.67	0.93	0.46	0.65	0.18	0.66	0.25	0.68	0.36	0.66	0.51	0.70	0.60	0.71
Ra-226							1.70	0.93	1.41	0.84	1.70	1.10	1.95	1.12
Sb-124	0.10	0.16	0.04	0.30	-0.02	0.66	0.09	0.20	0.00	0.25	0.14	0.28	0.04	0.26
Th-228	0.59	0.06	0.75	0.10	0.92	0.28	0.81	0.08	0.76	0.10	0.88	0.10	0.80	0.11
Th-232	0.52	0.12	0.61	0.23	0.68	0.16	0.69	0.17	0.62	0.18	0.73	0.18	0.79	0.21
GR-A	4.92	2.81	7.56	2.71	5.48	2.72	7.60	3.06	5.55	2.89	7.72	2.91	9.77	3.27
GR-B	30.00	2.86	36.20	2.82	25.20	2.82	34.30	2.88	27.30	2.87	35.20	2.87	27.00	2.88

Note: Blank field indicates no reported result
Note: BMRC-007 was moved due to concrete wall of N-16 tank room, the new hole was labeled BMRC-007A

Table 4-9 Soil Borings Analytical Results Summary (Cont.)

Bore ID>	BMRC-013		BMRC-014		BMRC-015		BMRC-016		BMRC-017		BMRC-019	
Radionuclide	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)	Max Activity (pCi/g)	Max MDC (pCi/g)
AC-228	1.48	0.21	2.59	0.21	1.84	0.21	2.58	0.22	1.44	0.21	1.31	0.17
AG-108M	0.02	0.05	0.01	0.04	0.02	0.05	-0.01	0.04	-0.01	0.05	0.01	0.05
AG-110M	0.06	0.07	0.03	0.07	0.02	0.07	0.01	0.08	0.02	0.07	-0.01	0.06
CO-58	-0.03	0.18	-0.04	0.18	0.00	0.18	-0.01	0.18	0.01	0.19	0.01	0.15
CO-60	0.01	0.05	0.02	0.06	0.02	0.06	-0.01	0.05	0.02	0.07	0.01	0.05
CS-137	0.07	0.07	0.01	0.06	0.07	0.07	0.10	0.06	0.08	0.06	0.01	0.05
EU-152	0.02	0.15	-0.01	0.15	0.03	0.15	0.06	0.15	0.00	0.15	0.07	0.14
EU-154	0.02	0.09	0.02	0.09	-0.01	0.09	0.01	0.09	0.01	0.09	0.02	0.08
H-3	-0.48	0.83	0.38	0.97	0.17	0.99	0.19	0.97	0.29	0.95	0.16	1.07
K-40	16.27	0.46	15.37	0.41	17.76	0.40	15.34	0.40	18.29	0.50	18.38	0.38
LA-140	40.73	110.30	46.46	99.35	9.87	85.15	-8.27	76.33	12.88	79.46	15.03	75.23
MN-54	0.02	0.07	0.03	0.07	0.01	0.07	0.01	0.07	0.02	0.07	0.02	0.06
NI-63	0.31	0.66	-0.12	1.53	-0.39	1.53	0.00	1.55	-0.21	1.60	-0.24	1.57
RA-226			2.17	1.06	2.17	0.93	1.68	0.88	2.23	1.10	1.69	0.77
SB-124	0.02	0.24	0.01	0.24	0.03	0.24	-0.02	0.24	0.08	0.25	0.17	0.23
TH-228	0.79	0.09	0.75	0.13	0.85	0.10	0.91	0.09	0.86	0.10	0.73	0.09
TH-232	0.59	0.20	0.70	0.19	0.86	0.19	0.66	0.20	0.78	0.20	0.63	0.16
GR-A	5.98	3.12	6.21	2.86	8.14	2.88	7.59	3.45	7.29	3.23	8.62	3.01
GR-B	31.10	2.88	29.90	2.87	37.50	2.87	35.10	2.88	29.90	2.88	28.90	2.88

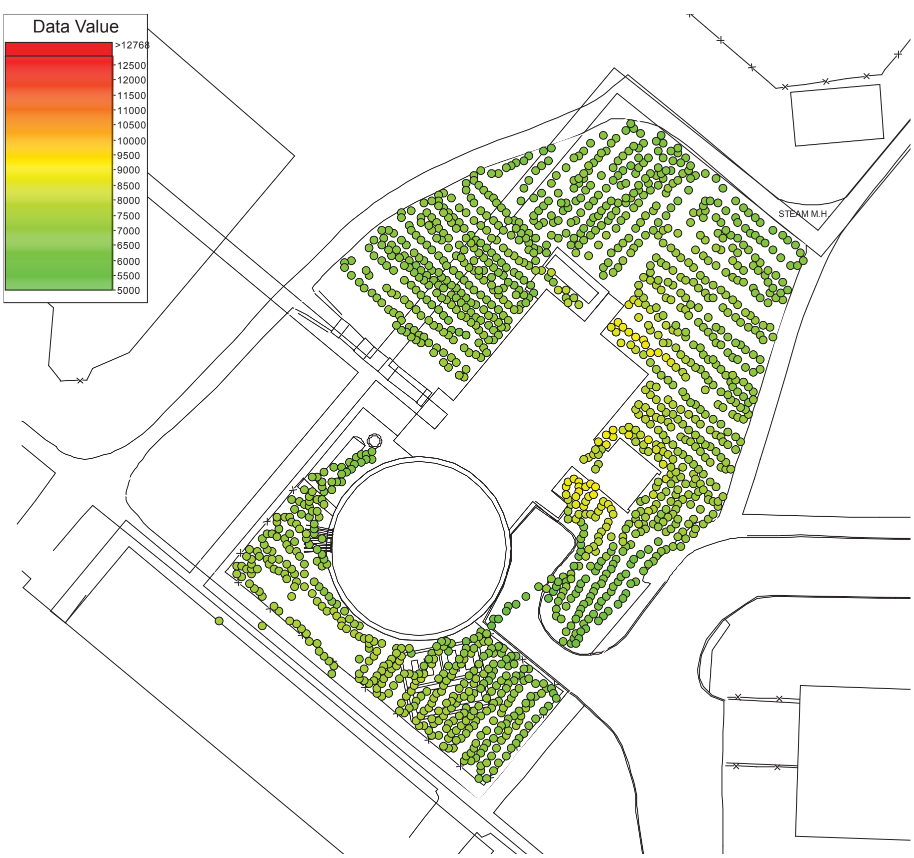
Note: Blank field indicates no reported result
Note: BMRC-018 was excluded due to proximity to buried utilities

4.5.3 Gamma Radiation Scan

A Gamma Radiation Scan was performed on the surface soils immediately surrounding the BMRC using a 2x2 Sodium Iodide (NaI) detector connected to a Ludlum Model 2221. The ratemeter reading was paired with its latitude/longitude coordinate and logged using a Trimble GPS unit. Additionally, an area several hundred yards to the northeast was selected for background radiation scans.

The average background reading was 6384 cpm with a maximum reading of 7311 cpm. The average reading for the surface soils near the BMRC was 6998 cpm with a maximum reading of 8909 cpm. The highest measurements recorded are adjacent to the current Vertical 10K Storage Tank and next to the northeast corner of the building. The elevated readings can be attributed to background interference from the Vertical 10K tank and the Cs-137 calibration source in Room 115A, respectively.

Figure 8: Gamma Walkover



4.5.4 **BMRC Rooftops**

The rooftop of the BMRC Containment Building and the Administrative Wing were surveyed for loose alpha/beta, fixed beta and loose beta. Additionally, the fixtures of the building exhaust systems were surveyed. Fourteen (14) random locations were plotted using the VSP software and an additional 6 locations were chosen based on professional judgment which included the Containment Exhaust Vent, the Tank Farm Vent pipe, and four vents from the Administrative Wing. All smear results were less than MDC. The static Beta measurements, without material specific background subtraction, had an average of 1369 dpm/100cm^2, with a maximum result of 2403 dpm/100cm^2 and a standard deviation of 456 dpm/100cm^2.

4.5.5 **Tritium and Carbon-14 Smears**

A tritium and Carbon-14 smear was collected from every room, sink, and hood throughout the BMRC Administrative Wing and Containment Buildings. A total of 83 smears (60 floor locations, 11 hoods, and 12 sinks) were analyzed by the offsite laboratory for loose tritium and loose C-14 with an MDC values at approximately 9 dpm/100 cm^2 and 6 dpm/100 cm^2, respectively. All locations in the Administrative Wing were less than MDC for both radionuclides. Eight (8) locations in the Containment Building indicated a potential presence of tritium with an average result of 30 dpm/100 cm^2 and a maximum value of 100 dpm/100 cm^2. Additionally, three (3) locations indicated the potential presence of C-14 with a maximum result of 35 dpm/100 cm^2.

When compared to the NRC Building Surfaces screening values for tritium and C-14, the maximum results are approximately 0.00005% and 0.0009% of the screening values, respectively.

4.6 Tank Farm

The Tank Farm lies in the subsurface adjacent to the containment building and extends towards the southwest of the containment building. The Tank Farm consists of two 250-gallon stainless steel tanks (Tanks 4 and 5) and two 600-gallon stainless steel tanks (Tanks 2 and 3) in connected concrete enclosures. There is also a 10,000-gallon carbon steel tank (Tank 1) in a separate enclosure to the southeast as shown in the HSA. The 250-gallon tanks were connected to the Isotope Processing Labs and the Hot Cell. The 600-gallon tanks supported the drains from the lower level labs and the floor drains. The tanks were interconnected by a manifold and pump system that allowed any combination of water transfer. The normal movement was to pump the small tanks to the 10,000-gallon tank for blending prior to discharges into the sewer system. Samples from the all of the tanks were taken and sent for analysis. The results are contained in Table 4-10 and Figure 9 provides the orientation of the tanks in the tank farm.

As discussed in the HSA, the tanks were placed in wet layup by pumping them at least half-full of water to minimize oxygen degradation in 1985. This method of storage was recommended by personnel at the DOE's Hanford Site. Sampling ports known as Tell-Tale sampling ports were installed in the original configuration to sample any liquids that may have entered the enclosures and concrete berm to verify tank and piping integrity. BMRC personnel collect samples on a quarterly basis and no indications of loss of integrity have been detected.

Figure 9: Tank Farm Layout

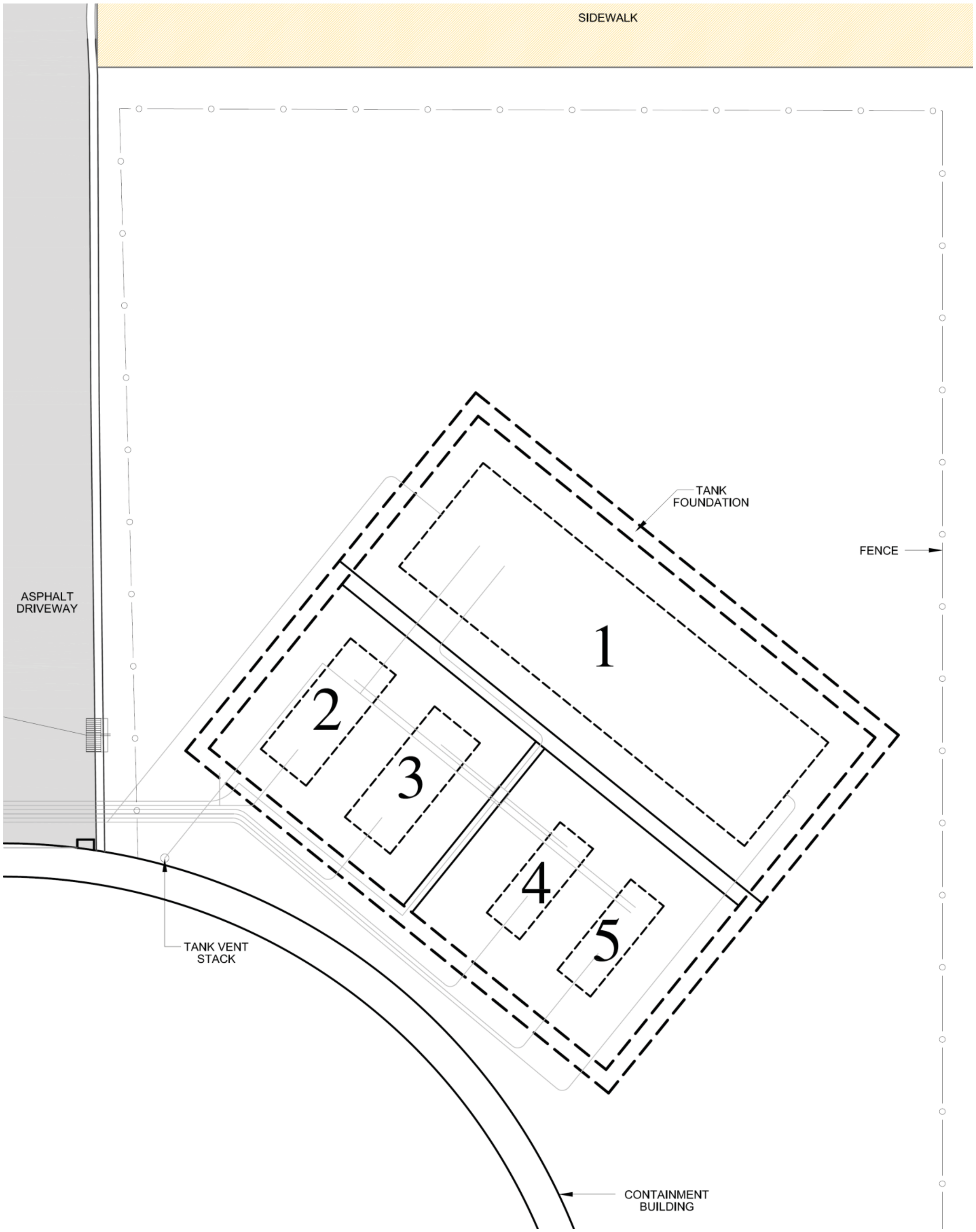


Table 4-10 Tank Farm Analytical Sample Results Summary

	BMRC-TF-001		BMRC-TF-002		BMRC-TF-003		BMRC-TF-004		BMRC-TF-005	
Radionuclide	Max Activity (pCi/L)	Max MDC (pCi/L)	Max Activity (pCi/L)	Max MDC (pCi/L)	Max Activity (pCi/L)	Max MDC (pCi/L)	Max Activity (pCi/L)	Max MDC (pCi/L)	Max Activity (pCi/L)	Max MDC (pCi/L)
Ag-108M	116	3	2456	18	244	8	59	7	660	9
Ag-110M	2	3	377	27	-1	9	1	9	2	8
Am-241 (AS)	0	0								
C-14	422	36	1510	25	1030	25	87	29	85	19
Cm-242 (AS)	0	0								
Cm-243/244 (AS)	0	0								
Co-58	0	3	-4	28	-1	12	-5	12	3	11
Co-60	128	3	10490	14	1654	8	2358	7	943	7
Cs-134	-1	3	1	19	-5	8	2	8	-7	8
Cs-137	80	3	437	22	221	10	130	9	105	9
Eu-152	1	9	-53	51	-10	25	0	21	-3	25
Eu-154	1	6	-8	24	-5	14	-4	10	6	13
Fe-55	0	216								
GR-A	11	3	8	3	9	3	3	3	2	3
GR-B	10500	21	4780	15	5410	16	3180	12	2240	10
H-3	4320	3550	5810	3580	4260	3560	3880	3620	3970	3610
I-129	-30	77								
La-140	0	6	-4	25	-13	14	-2	18	3	16
Mn-54	1	3	-2	26	3	10	2	11	-2	10
Ni-59	-47	73								
Ni-63	388	14	10400	13	25500	13	1650	13	467	13
Pu-238 (AS)	0	0								
Pu-239/240 (AS)	0	0								
Pu-241	66	69								
Sb-124	0	4	-8	22	-5	10	-5	10	2	10
Sr-89	13800	69	4220	64	5430	62	3720	63	2960	70
Sr-90	4310	18	1330	16	2300	15	1150	16	895	12
Tc-99	4	18								
U-233/234 (AS)	1	0								
U-235 (AS)	0	0								
U-238 (AS)	0	0								

Note: Blank field indicates no reported result. (AS) = Alpha Spectroscopy.

4.7 Ventilation

Surveys of the ventilation systems consisted of removable activity surveys at select access points and were performed in the course of the characterization surveys and are reported for the survey unit to which the system is associated. Table 4-11 provides a summary of the ventilation systems by survey unit.

Table 4-11 Ventilation Systems Survey Results Summary

	Removable Alpha Activity ($DPM/100cm^2$)		Removable Beta activity ($DPM/100cm^2$)	
Survey Unit	**Average**	**Max**	**Average**	**Max**
1C3	3	3	43	74
1G12	0	0	0	0
1G8	6	12	3	5
1G9	0	0	9	27
2C1	1	3	27	90
2G1	0	3	12	74

4.8 Miscellaneous Sample Areas

Additional samples have been collected from the N16 tank room soil and from the sediment material in the 1K tank located in Room N03.

4.8.1 N16 Tank Vault

The N16 tank is located in a vault in the sub-basement accessible through a wall hatch in Room N03. This room is unfinished and has concrete walls and a dirt floor. The tank is mounted on concrete piers. Piping exits the neutron level of containment and passes into this room.

Four (4) soil samples were collected as part of the characterization as shown in Figure 10. The results indicate impacts greater than the NRC and NYS Screening values for Co-60. Additionally, Ag-108m was elevated and has a high potential to be greater than the release criteria developed as part of the DP. A summary of the analytical sample results is presented in Table 4-12 on the following page.

Figure 10: N16 Tank Vault Sample Locations

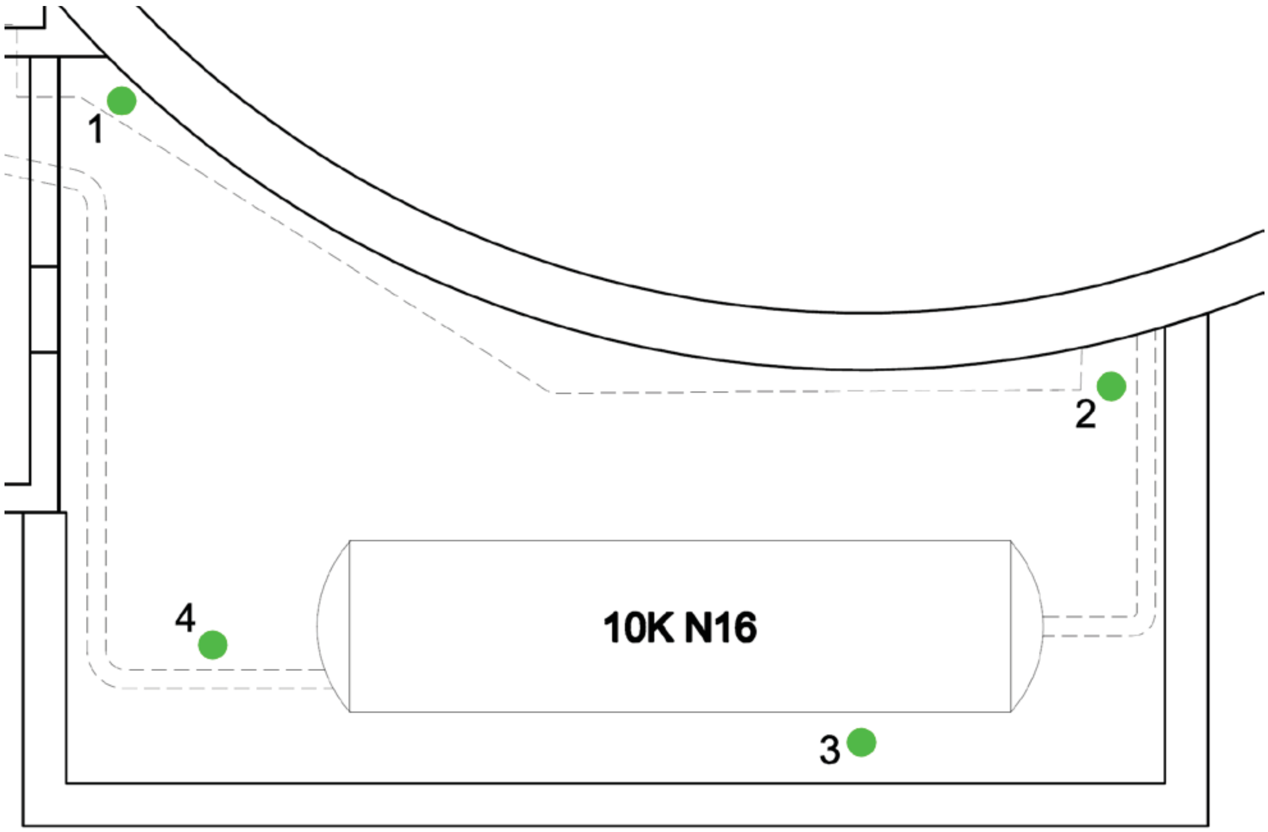


Table 4-12 N16 Tank Vault Analytical Results Summary

Radionuclide	Maximum Activity (pCi/g)	Maximum MDC (pCi/g)	Radionuclide	Maximum Activity (pCi/g)	Maximum MDC (pCi/g)
Ac-228	3.95E+00	4.71E-01	Fe-55	2.61E+00	9.81E+00
Ag-108M	9.99E+01	1.94E-01	H-3	7.19E+00	1.18E+00
Ag-110M	5.98E-01	1.42E-01	K-40	1.12E+01	4.60E-01
Co-57	9.13E-01	1.04E-01	La-140	1.84E-01	2.52E-01
Co-58	2.84E-02	1.60E-01	Mn-54	2.31E-02	1.29E-01
Co-60	1.26E+01	7.03E-02	Ni-63	7.69E+01	6.26E-01
Cs-134	3.03E-02	1.46E-01	Sb-124	2.50E-02	1.95E-01
Cs-137	4.65E+00	1.42E-01	Sr-89	5.66E-02	1.79E-01
Eu-152	2.00E+00	5.65E-01	Sr-90	3.61E-01	6.76E-02
Eu-154	9.50E-01	2.94E-01	Th-228	5.38E-01	3.28E-01
GR-A	8.36E+00	2.64E+00	Th-232	6.12E-01	4.82E-01
GR-B	8.27E+01	2.60E+00			

4.8.2 1K Tank

The 1K tank is located in a pit in Room N03. Water is collected in this tank from all sumps and drains in the containment building prior to entering the 10K tank and ultimately being discharged through the NPDES permitted sanitary system. The tank sediments were sampled and contained elevated levels of the site specific ROCs with the primary radionuclides being Co-60, Ag-108m, Ni-63, and Sr-90. The sediment sample collected from the tank had a dose rate of 1.5 mR/hr on contact. The water in the tanks was sampled and contained only Ni-63.

Table 4-13 on the following page provides a summary of the analytical results of the tank sediment.

Table 4-13 1K Tank Analytical Results Summary

	Sediments		Liquid	
Radionuclide	**Activity (pCi/g)**	**Uncertainty 2 Sigma (pCi/g)**	**Activity (pCi/l)**	**Uncertainty 2 Sigma (pCi/l)**
C-14	2.23E+01	6.70E+00	<MDC	<MDC
Ni-59	1.76E+02	1.03E+02	<MDC	<MDC
Ni-63	9.92E+03	9.94E+01	2.96E+03	4.29E+02
Sr-89	1.38E+02	2.03E+01	<MDC	<MDC
Sr-90	1.99E+02	1.37E+01	<MDC	<MDC
Ag-108m	1.03E+04	5.97E+01	<MDC	<MDC
Co-60	1.51E+03	3.57E+01	<MDC	<MDC
Cs-137	4.90E+01	3.12E+01	<MDC	<MDC
Am-241	3.90E+00	9.07E-01	<MDC	<MDC
Pu-238	1.71E+00	4.28E-01	<MDC	<MDC
Pu-239/240	5.74E+00	8.30E-01	<MDC	<MDC
U-233/234	1.74E+00	4.80E-01	<MDC	<MDC
U-238	1.09E+00	3.51E-01	<MDC	<MDC

5. HAZARDOUS MATERIALS

As part of the facility characterization, a hazardous materials assessment was also performed. The goal of the assessment was to identify known and potential hazardous materials present at the facility. The assessment was based on site observations and document reviews.

During the on-site characterization, several potential hazardous materials were identified. These included caustics and acids (water treatment chemicals) and lubricating oils. There appeared to be no evidence of spill or leaks of these materials that would have impacted the facility. Because the facility housed many experiments, there may also be containers of laboratory chemicals stored about the facility that were not observed during the radiological site characterization activities. It is expected that containers of hazardous materials will be disposed of according to current practices and procedures prior to radiological decontamination and demolition.

UB does not have a site asbestos management plan, however, the facility has been surveyed for Asbestos Containing Materials (ACM) with ACM locations identified on piping. The floor tile in the administration wing is also known to be ACM. There is a high potential that the mastic under the tile is also ACM. The University plans on issuing a request for proposal (RFP) for asbestos removal in the second half of 2011. A new ACM survey will be performed before an RFP is issued for the ACM interference removal project.

Lead is present throughout the facility in the form of lead bricks and lead sheets. The University is in the process of issuing a contract to dispose of all loose legacy materials in the facility which includes all lead bricks and sheets. While a lead paint survey was not conducted, because of the age of the facility, it is expected that lead-based paints potentially were used throughout the facility. A lead paint survey will either be conducted at the same time a new ACM survey is conducted or will be conducted prior to an RFP being developed for building remediation and demolition.

Mercury chloride was used in the activation analysis lab on the gamma deck in the containment wing. The University required mercury monitoring during radiological surveys in the lab. Additionally, the fluorescent light ballasts and any old thermostats will likely contain hazardous materials.

6. QUALITY ASSURANCE AND QUALITY CONTROL

ENERCON ensured that quality and regulatory requirements were satisfied during the characterization effort. Activities were controlled by procedures and the Site Characterization Plan. These documents included the following Quality Control (QC) measures as an integral part of the survey process.

6.1 General Provisions

6.1.1 Written Procedures

Survey tasks were controlled by characterization survey instructions contained in each survey package, ENERCON procedures, and the BMRC Characterization Plan. The following procedures were used and referenced in the characterization survey instruction during the characterization project:

- TS 5.4.6, *Chain of Custody*
- TS 5.4.7, *Documentation of Radiological Surveys*
- TS 5.4.8, *Instrument Source and Response Check*
- TS 5.4.10, *Sampling Materials for Radiological Characterization*
- TS 5.4.11, *Sample Handling*
- TS 5.4.12, *Radiological Scans and Measurements*
- TS 5.4.13, *Technician Training and Qualification*
- SUNY-OP-01, *BMRC Reactor Bioshield Core Sampling*
- SUNY-OP-02, *Neutron Deck Coring and Subgrade Sampling*
- SUNY-OP-03, *Concrete Core Sample Processing*
- SUNY-OP-04, *Soil Sampling*

6.1.2 Instrumentation Selection, Calibration, and Use

ENERCON selected instruments that are proven to reliably detect the radionuclides present at the BMRC. Instruments were calibrated by qualified vendors under approved procedures using calibration sources traceable to the National Institute of Standards and Technology (NIST). All detectors were subject to daily response checks when in use per guidance established in American National Standard Institute (ANSI) standard ANSI N323-1997.

6.1.3 Chain of Custody

The Chain of Custody procedure established responsibility for the custody of samples from the time of collection until results were obtained. All samples shipped off site for analysis were accompanied by a chain-of-custody record to track each sample.

6.1.4 Independent Review of Survey Results

The survey package and survey data from each area received an independent review to verify all documentation is complete and accurate.

6.2 Training

All project personnel received site specific training to identify the specific hazards present in the survey areas in addition to the characterization process. The training included a briefing and review of the BMRC Characterization Plan, ENERCON procedures, and the Site Safety and Health Plan. Personnel were also required to read each document. Copies of all training records were maintained on site through the duration of the onsite activities.

6.3 Sample Analysis

ENERCON utilized a third party vendor, Teledyne Brown Engineering (TBE) in Knoxville, TN, for all radioanalytical services during the characterization project. All samples sent to TBE were accompanied by a Chain of Custody with their receipt acknowledge via email. TBE is used extensively in the nuclear industry for environmental, bioassay, and waste characterization analyses. TBE provides QC with all laboratory reports including blanks, spikes and duplicates.

7. LIMITATIONS

7.1 High Background Areas

As noted throughout this report, there were several areas where the background radiation was elevated due to the presence of stored radioactive materials. The status of these rooms at the time of the survey is documented in the photographs in Appendix A. In these areas, surveys may not be complete, but will be verified following the pre-decommissioning cleanout of loose legacy materials. Following the disposal campaign, radiological surveys will be performed in the areas and the results documented. The results will be analyzed to ensure the information submitted in the DP remains valid. A revised DP will be submitted to the NRC if a revision is required.

7.2 Internal Activity

The internal surfaces of various components potentially containing radioactive materials were not surveyed. These include the primary coolant pipes, the nitrogen-16 decay tanks, the primary coolant pump, the vertical 10K taste tank, and the reactor tank. However, external dose rates were measured for these items and were documented in the radiation surveys. A dose rate survey was conducted in the reactor tanks near all activated components. This information was used by WMG for the Component Activation Analysis report.

8. REFERENCES

8.1 BMRC Historical Site Assessment, Rev. 0, March 17 2010, ENERCON

8.2 BMRC Characterization Plan, November 24, 2010, ENERCON

8.3 BMRC Site Health and Safety Plan for the SUNY-UB Characterization, November 30, 2010, ENERCON

8.4 NUREG-1537, Part 1, Chapter 17, *Guidelines for Preparing and Reviewing Applications for Licensing of Non-Power Reactors*

8.5 NUREG-1757, *Consolidated Decommissioning Guidance*

8.6 NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*.

8.7 10CFR20, *Standards for the protection against Radiation*

8.8 DSHM-RAD-05-01, *Cleanup Guidelines for Soils Contaminated with Radioactive Materials*, New York State Department of Environmental Conservation.

8.9 ANSI N323-1997, *Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments*, American National Standard Institute (ANSI) Standard

8.10 NUREG-0982, *Safety Evaluation Report related to the renewal of the operating license for the Research Reactor at the State University of New York at Buffalo*, May 1983, NRC

8.11 NUREG/CR-3474, *Long-Lived Activation Products in Reactor Materials*.

8.12 NUREG/CR-4289, *Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants*.

Appendix A

BMRC Characterization Photographs

Administrative Wing – Control Deck Level

Photograph 1 Building Exterior Main Entrance

Photograph 2 OTB (Vertical Accelerator Room)

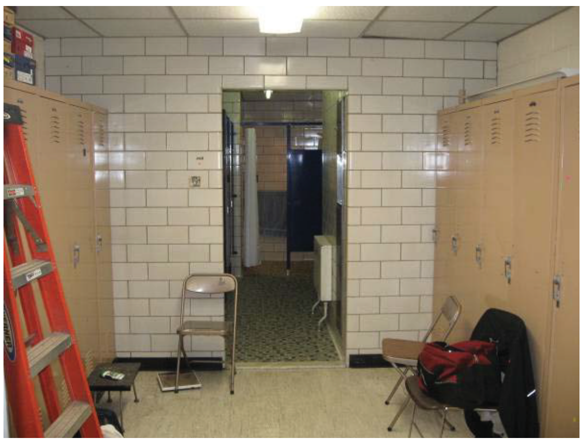
Photograph 3 Room 215

Photograph 4 Room 213

Photograph 5 Stairwell (S240)

Photograph 6 Room 221 (Women's Restroom)

Photograph 7 Main Control Deck Hallway (C241)



Photograph 8 Room 209 (Men's Restroom)

Photograph 9 Room 216

Photograph 10 Room 214

Photograph 11 Room 210/212

Photograph 12 Room 208

Photograph 13 Room 208A

Administrative Wing – Gamma Deck Level

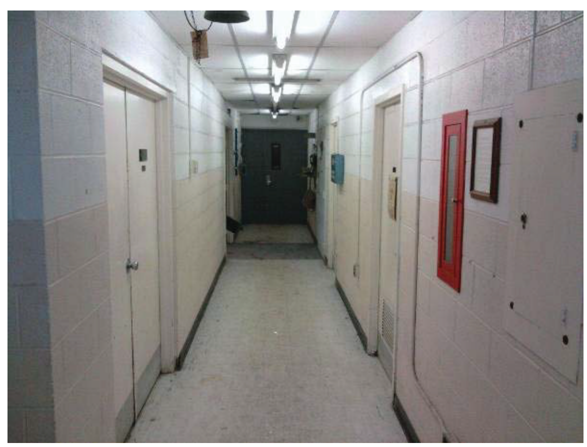
Photograph 14 Room 115A (Vertical Accelerator Target)

Photograph 17 Room 113 (Machine Shop)

Photograph 18 Room 111

Photograph 15 Room 115

Photograph 16 Stairwell (S122)



Photograph 19 Main Gamma Deck (C119)

Photograph 20 Room 118

Photograph 22 Room 114

Photograph 21 Room 117

Photograph 23 Room 110

Administrative Wing – Neutron Deck Level (Sub-Basement)

Photograph 24 10K Tank Room

Photograph 27 Room N02

Photograph 25 Room N05

Photograph 28 Room N04

Photograph 26 Room SN06 (Stairwell)

Photograph 29 Room N03

Photograph 30 Room N03 1K Tank Pit

Photograph 31 N16 Tank Room

Containment Wing – Control Deck Level

Photograph 32 Room 201A

Photograph 33 Room 202

Photograph 34 Room 203

Photograph 35 Room 204 (Fan Room)

Photograph 36 Room 205 (Control Room)

Photograph 37 Room 206

Photograph 38 Room 207

Photograph 39 Room 200

Photograph 40 Main Floor Control Deck

Photograph 41 Main Floor Control Deck

Photograph 42 Main Floor Control Deck

Photograph 43 Main Floor Control Deck

Containment Wing – Gamma Deck Level

Photograph 44 Room 101

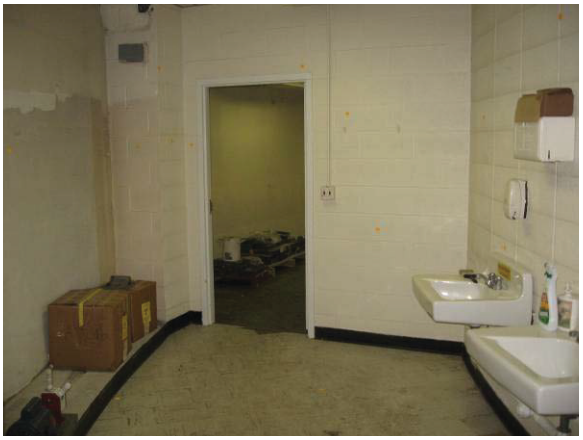
Photograph 45 Room 103

Photograph 46 Room 104

Photograph 47 Room 109

Photograph 48 Room 108

Photograph 49 Room 107



Photograph 50 Room 106

Photograph 51 Room 105

Photograph 52 Hotcell

Photograph 53 Main Floor Gamma Deck

Photograph 54 Main Floor Gamma Deck

Photograph 55 Main Floor Gamma Deck

Photograph 56 Main Floor Gamma Deck

Photograph 57 Main Floor Gamma Deck

Containment Wing – Neutron Deck Level

Photograph 58 Main Floor Neutron Level

Photograph 59 Main Floor Neutron Level

Photograph 60 Main Floor Neutron Level

Photograph 61 Main Floor Neutron Level

Photograph 62 Main Floor Neutron Level

Photograph 63 Main Floor Neutron Level

Photograph 64 Main Floor Neutron Level

Photograph 65 Beam Port Storage Tubes

Photograph 66 High Radiation Area Neutron Level

Photograph 67 Thermal Column

Photograph 68 Trench System

APPENDIX C – DECOMMISSIONING ENVIRONMENTAL REPORT

Decommissioning Environmental Report

Buffalo Materials Research Center

Prepared for:

University at Buffalo
The State University of New York

Buffalo Material Research Center
Office of Environment, Health, and Safety Services

Completed by:

ENERCON
Excellence—Every project. Every day.

4490 Old William Penn Highway
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December 2011

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LIST OF FIGURES

Acronyms and Abbreviations

ACM	Asbestos Containing Material
AST	aboveground storage tank
BLM	Bureau of Land Management
BMRC	Buffalo Materials Research Center
BSFR	Bulk Survey for Release
CAM	Continuous Air Monitor
CFR	Code of Federal Regulations
D&D	Decontamination and Decommissioning
DCGL	Derived Concentration Guideline Levels
DEC	Department of Environmental Conservation (State of New York)
DECON-A	complete decontamination and structure demolition option
DECON-B	complete decontamination and release of the structure option
DOC	Design and Oversight Contractor
DOE	United States Department of Energy
DOT	Department of Transportation
ECL	Environmental Conservation Law
EH&S	Environment, Health and Safety
ENTOMB	Entombment option
EPA	Environmental Protection Agency
ER	Environmental Report
F	Fahrenheit
FSS	Final Status Survey
HEPA	High-efficiency particulate air
HSA	Historical Site Assessment
LLRW	Low-level radioactive waste
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MGD	Million Gallons per Day
mrem	Millirem
MTR	Materials Testing Reactor
MW	Megawatt
MWt	Megawatt thermal
N-16	Nitrogen 16
NEPA	National Environmental Policy Act

NMSS	NRC Office of Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
NSTC	Nuclear Science and Technology Center
NYCRR	New York State Codes, Rules, and Regulations
NYS	New York State
NYSDOH	New York State Department of Health
PuBe	Plutonium-Beryllium
PULSTAR	Pulse Training Assembled Reactor
RCRA	Resource Conservation and Recovery Act
RFT	resilient floor tile
SAFSTOR	No-Action Alternative
SCR	Site Characterization Report (ENERCON initial draft)
SNM	Special Nuclear Material
Sr-90	Strontium-90
SUNY	State University of New York
TLD	Thermoluminescent dosimeter
UB	University at Buffalo
UST	underground storage tank
WMP	Waste Management Plan

1.0 INTRODUCTION

This Environmental Report (ER) describes the environmental effects related to the decommissioning of the University at Buffalo (UB) Material Research Center (BMRC) Research and Test Reactor Facility. The ER was prepared in accordance with the guidance provided in Chapter 6.0 of the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Material Safety and Safeguards (NMSS) NUREG-1748, Environmental Review Guidance for Licensing Actions Associated with NMSS Programs (NRC 2003b). This ER is designed to be used by the NRC in conducting its environmental assessment in accordance with the National Environmental Policy Act (NEPA) of 1969. NEPA requires federal agencies, as part of their decision-making process, to consider the environmental impacts of actions under their jurisdiction. The NRC's NEPA requirements are provided in Title 10 Code of Federal Regulations (CFR) Part 51.

1.1 Purpose and Need for Action

The reactor at the BMRC was operated from 1961 to 1963 in support of training and education for the university and other national users. In 1964, the reactor was shut down and the core and control systems were modified so that the reactor could operate with Pulse Training Assembled Reactor (PULSTAR)-type fuel at power levels up to 2 megawatts thermal (MWt). The BMRC continued in support of training and education for the university from 1964 until reactor activities were suspended in 1994. During the period of operation, the BMRC has had multiple uses including:

- Training and education
- Transient fuel performance testing
- Materials radiation damage research
- Isotope production
- Neutron interrogation through activation analysis, radiography and delayed fission assay

The unit has been in Possession Only status since June 6, 1997. All of the unused fuel was shipped to North Carolina State University in 1998 and all the spent fuel was removed from the site in 2005 and shipped to Idaho National Engineering and Environmental Laboratory. The BMRC administrative/laboratory wing (administrative wing) is currently used infrequently for two purposes. One purpose is an ion chamber calibration facility that is located in the basement of the administrative wing. The second purpose is for training of the State University of New York (SUNY) at Buffalo (UB) maintenance staff in the proper care and cleaning of asbestos-containing resilient floor tile (RFT). The top floor of the administrative wing has been determined to contain asbestos-containing RFT. Certain regulatory statutes mandate a standard of care for maintenance and custodial workers when activities affect asbestos-containing RFT. The UB Maintenance Department utilizes this space to train personnel on how to properly strip and wax the RFT. The containment building is only accessed for routine Technical Specification inspections.

1.2 The Proposed Action

UB has already removed the BMRC from service and now plans to dismantle the reactor and its ancillary support systems, remove all residual radioactive material from the BMRC facility, and demolish the entire facility. Radioactivity levels will be reduced to levels that will permit release of the licensed area for unrestricted use and allow termination of NRC Possession License Only R-77.

1.3 Applicable Regulatory Requirements, Permits, and Required Consultations

Decommissioning of the BMRC requires adherence to numerous federal, state, regional, local, and UB regulations. Guidance for determining many of the applicable federal, state, regional, and local requirements is identified in the following subsections. The information provided below is intended as a broad overview of applicable regulations and is not intended to be all-inclusive. The licensee or owner, UB in this case, is ultimately responsible for compliance with applicable federal, state, local, and internal regulations and is responsible for the costs associated with acquiring applicable permits and for the costs of implementing the necessary compliance programs during decommissioning activities.

1.3.1 Federal Requirements

Decommissioning activities that are subject to federal regulations, permits, licenses, notifications, approvals, or acknowledgements include:

- Handling, packaging, and shipment of radioactive waste
- Worker radiation protection
- License termination and final site release
- Worker, contractor, and the general public's health and safety
- Liquid effluent releases
- Hazardous waste generation and disposition
- Handling, removal, and proper disposal of asbestos-containing materials
- Handling, removal, and proper disposal of lead-containing paint and lead-based paint
- Handling, removal and proper disposal of underground and aboveground storage tanks (UST and AST, respectively)

Nuclear Regulatory Commission

The majority of radiological activities fall under Title 10 CFR and are administered by the NRC. Applicable portions of Title 10 regulations are included within the following parts:

- Part 20 – "Standards for Protection Against Radiation"
- Part 50 – "Domestic Licensing of Production and Utilization Facilities" including decommissioning activities
- Part 51 – "Environmental Protection Regulations For Domestic Licensing and Related Regulatory Functions"
- Part 61 – "Licensing Requirements for Land Disposal of Radioactive Waste"
- Part 71 – "Packaging and Transportation of Radioactive Material"

Many of the decommissioning requirements that involve activities for site control, characterization, and final status surveys (FSS) are found within the following parts of Title 10 of the CFR and are administered by the NRC. The parts include:

- Part 20.1401 – "General provisions and scope"
- Part 20.1402 – "Radiological criteria for unrestricted use"
- Part 20.1403 – "Criteria for license termination under restricted conditions"
- Part 20.1404 – "Alternate criteria for license termination"
- Part 20.1405 – "Public notification and public participation"
- Part 20.1406 – "Minimization of contamination"
- Part 20 Subpart F – "Surveys and Monitoring"
- Part 30.36 – "Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas"
- Part 40.42 – "Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas"
- Part 70.38 – "Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas"
- Part 72.54 – "Expiration and termination of licenses and decommissioning of sites and separate buildings or outdoor areas"

Department of Transportation

Radioactive material transportation activities fall under Title 49 of the CFR and are administered by the Department of Transportation (DOT). Applicable portions of Title 49 regulations are included within the following parts:

- Subtitle B – "Other Regulations Relating to Transportation," Parts 100 to 185 – as applicable

Environmental Protection Agency

The Environmental Protection Agency (EPA) provides the federal environmental requirements. The State of New York is under the jurisdiction of the EPA Region II office. The EPA regulations outlined in Title 40 of the CFR apply as follows:

- Part 61 – "National Emission Standards for Hazardous Air Pollutants"
- Part 61 Subpart M – "National Emission Standard for Asbestos" pertaining to asbestos handling, removal, and disposal
- Part 129 to 132 – Clean Water Act
- Part 190 – "Environmental Radiation Protection Standards for Nuclear Power Operations"
- Parts 260-272- Hazardous waste disposal and solid waste disposal as included in the Resource Conservation and Recovery Act (RCRA)

1.3.2 State of New York

Environmental regulations and policies in the State of New York are codified by the New York State Legislature under Environmental Conservation Law (ECL) and enforced by the Department of Environmental Conservation (DEC). Portions of the ECL may apply as follows:

- Article 1– General Provisions
  - Title 1-Declaration of Policy
- Article 3 – Department of Environmental Conservation; General Functions, Powers, Duties, and Jurisdiction
- Article 15 – Water Resources
  - Title 1 – Short Title; Statement of Policy; Definitions; General Provisions
  - Title 5 – Protection of Water
  - Title 19 - Drainage
  - Title 31 – Groundwater Protection and Remediation Program
- Article 17 – Water Pollution Control
  - Title 1 – General Provisions and Public Policy
  - Title 5 – Prohibitions
  - Title 7 – Permits and Certificates
  - Title 8 – State Pollution Discharge Elimination System
  - Title 14 – Nonpoint Source Water Pollution Control
- Article 19 – Air Pollution Control
  - Short Title; Declarations of Policy and Purpose; Definitions
- Article 27 – Collection, Treatment and Disposal of Refuse and Other Solid Waste

- Title 1 – Solid and Hazardous Waste Management Policy and Planning
- Title 3 – Waste Transporter Permits
  - Part 364 Waste Transporter Permits
- Title 7 – Solid Waste Management and Resource Recovery Facilities
- Title 9 – Industrial Hazardous Waste Management
- Title 17 – Lead-Acid Battery Recycling
- Title 24 – Environmental Tests Reporting Requirements

- Article 28 – Pollution Prevention
- Article 37 – Substances Hazardous or Acutely Hazardous to Public Health, Safety, or the Environment
  - Title 1 – Substances Hazardous to the Environment
  - Title 2 – Hazardous Packing
- Article 54 – Environmental Protection Act
  - Title 1 – General Provisions
- Article 72 – Environmental Regulatory Program Fees
  - Title 1 – Declaration of Policy; Definitions
  - Title 4 – Hazardous Waste Program Fee

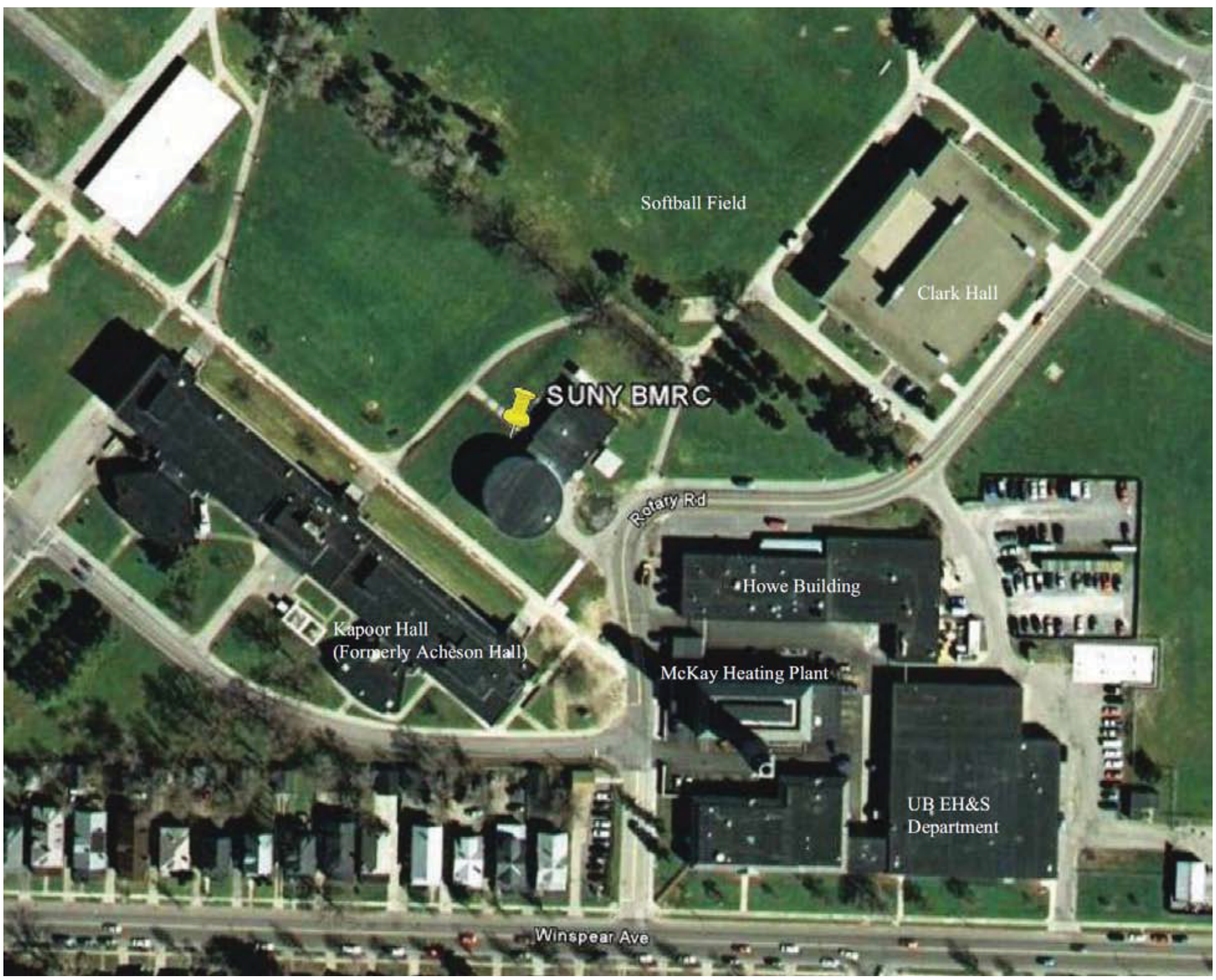
Worker protection laws in the State of New York are regulated and enforced by the State of New York Department of Labor. Portions of 12 New York State Codes, Rules, and Regulations (NYCRR) may apply as follows:

- Asbestos – Part 56 of Title 12 of the Official Compilation of Codes, Rules, and Regulations of the State of New York (cited as 12 NYCRR Part 56)

2.0 FACILITY DESCRIPTION

The BMRC is owned by UB and is located on the south edge of the South Campus of UB in the City of Buffalo, New York. See Figure 2.1 for the location of the BMRC as it is positioned on the UB campus. The facility consists of a cylindrical vapor containment building and the attached rectangular administrative wing that is oriented on the long axis to the northeast of the containment building. The reactor is housed within the containment building. The containment building is only accessed for routine technical specification inspections. An ion chamber calibration facility is located in the basement of the administrative wing and the top floor of the administrative wing is utilized as an area for training of UB maintenance and custodial staff.

Figure 1: The BMRC and Adjacent Buildings



The Containment Building

The Reactor

The containment building is a poured reinforced concrete right cylinder, 70 feet in diameter and 52 feet in height. The reinforced concrete walls and roof (supported by concrete beams) are 2 feet thick and 0.33 feet thick, respectively. The building walls and the first level are laid on bedrock. The containment building is attached to a laboratory complex formerly dedicated primarily to nuclear science-related research and instruction. The reactor pool is a reinforced high density concrete structure located mostly below grade.

The reactor facility, including the control and safety systems, was designed and built by American Machine and Foundry. The reactor first began operation in June 1961 and was originally referred to as the Western New York Nuclear Research Center and subsequently the Nuclear Science and Technology Center (NSTC) of SUNY UB. The reactor was originally fueled with materials-testing-reactor (MTR)-type fuel elements operating at a maximum steady state power level of 1 MWt. The reactor operated in this mode from 1961 to 1964. The reactor was shut down in 1964 so that core and control systems could be modified. After the modification, the reactor operated with PULSTAR-type fuel at power levels up to 2 MWt. The original core grid plate was retained and the MTR fuel elements were replaced with PULSTAR pin-type fuel clusters which were designed to utilize the existing grid space. On May 12, 1965, the reactor was additionally licensed to operate in the pulse mode with a routine energy per pulse of up to 35 MW per second and a maximum size pulse of 44 MW per second. The BMRC operated with PULSTAR fuel since 1965, with one major re-fueling in 1978.

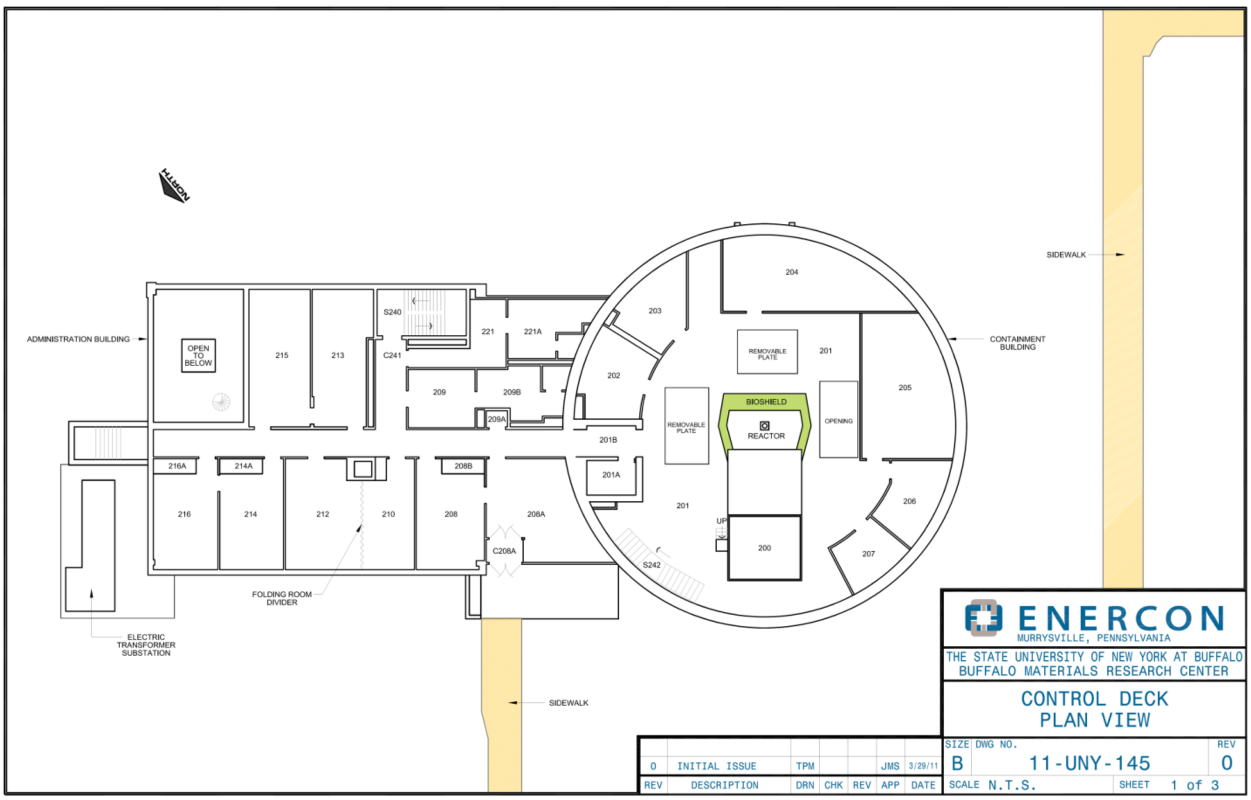
The reactor is a PULSTAR heterogeneous open-pool type water-cooled reactor using solid 6 percent enriched uranium fuel similar to nuclear power reactor fuel. The core was cooled by forced convective cooling at high power levels and by natural convection at lower power levels. The coolant/moderator is light water, and the reflector may be either water or graphite. The core is immersed in a 13,000-gallon, aluminum-lined reinforced concrete pool. The coolant was circulated through external systems for heat removal and for purification. Former reactor experimental facilities included in-core irradiation positions, a thermal column, beam tubes, pneumatic sample transport systems, a dry gamma chamber, and a gamma irradiation facility.

The core was immersed in a 13,000-gallon, aluminum lined tank pool surrounded by both high density and normal density concrete. It was cooled by a 5,000-gallon circulating water system exiting from the reactor pool bottom via pumps to a subsurface 5,000-gallon hold-up tank for N-16 decay. The water then went through a heat exchanger connected to an external cooling tower located on the southeast side of the administrative wing. The cooled water was then circulated back to the top of the pool. After a leak was detected in 1977 where the coolant piping penetrated the bottom of the reactor pool, modifications were made to the cavity lower liner and the piping was rerouted to exit out of the side of the pool and through the lower wall of the Neutron Deck of the containment building. After a leak was detected in 1989 in the liner, a new liner was installed over the original liner in the lower section of the tank. All the spent and unused fuel has been shipped off-site. The reactor and all its associated components remain in the pool in the normal operational configuration as they were during operation.

The Control Deck

The Control Deck (top level) houses a dual-hook overhead crane with a ten-ton and a two-ton hook on tracks at the ceiling level. Crane access to the lower levels was via floor plates on the Control Deck and a webbing cover on a floor penetration from the Gamma Deck to the Neutron Deck. In addition to the reactor pool, the Control Deck houses the Reactor Control Room (Room 205), the Superintendent's Office (Room 207), two labs (Rooms 202 and 203), the Fan Room (Room 204), another office (Room 206), and an airlock (Room 201B) for access to the upper or first floor of the administrative wing. The New Fuel Storage Room (Room 201A) is located between the inner and outer door of the airlock. Figure 2 below provides the layout of the Control Deck.

Figure 2: Control Deck Layout

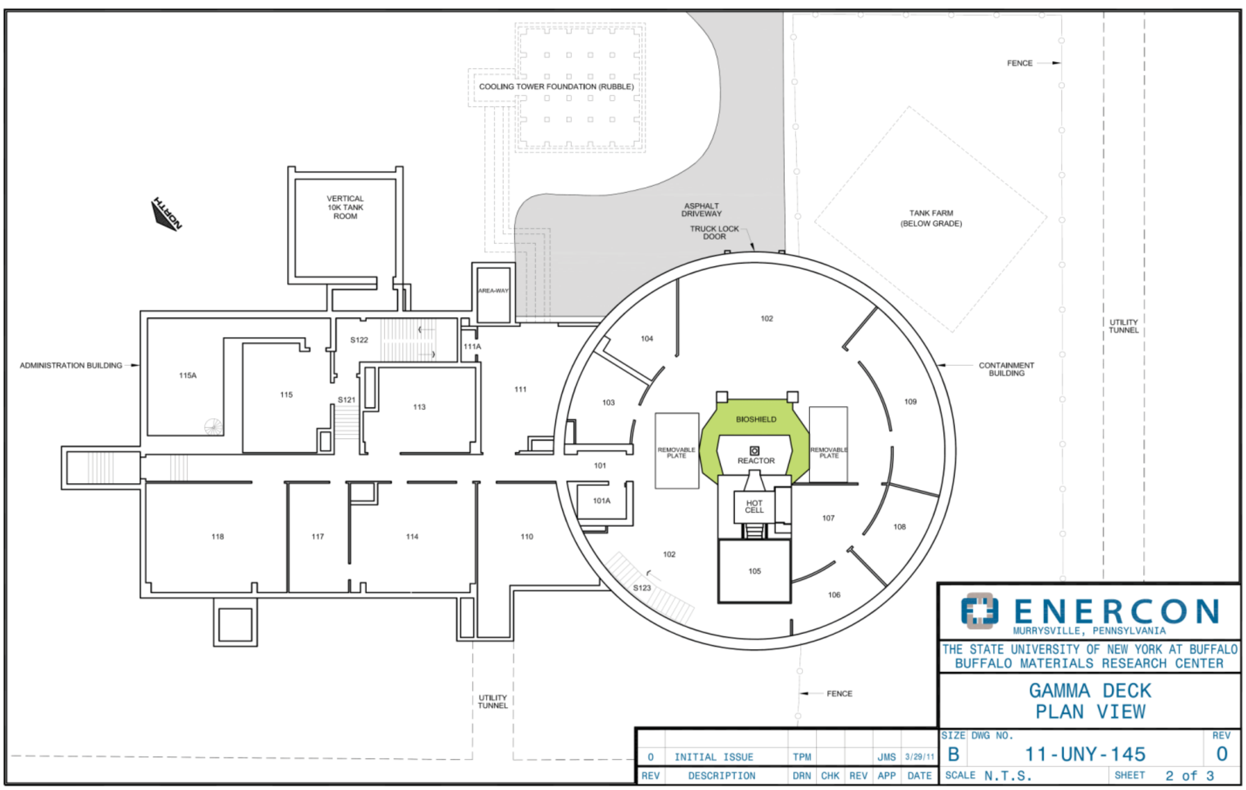


The Gamma Deck

The middle level of the Containment Building is called the Gamma Deck. A plan view of the Gamma Deck is shown on Figure 1-4. The Hot Cell, Hot Cell Work Room (Room 105), Locker Room (Room 106), Room 107, Hot Chemical Lab (Room 108), Medium Chemical Lab (Room 104), Activation Analysis Lab (Room 109), Lower Fuel Vault Room (Room 101A), and Chemical Lab (Room 103) are located on the Gamma Deck. This level also has an Airlock (Room 101) for access to the bottom level of the Administration Building and a truck door for access to the asphalt drive located on the southern side of the Containment Building.

The Truck Door provides vehicular and equipment access to the outside. Prior to 1979, the Truck Door area housed a waste compaction system. Wastes generated at the BMRC (NSTC at the time) and under the site-wide radiological program were collected and compacted for volume reduction in this system. Figure 3 provides the layout of the Gamma Deck.

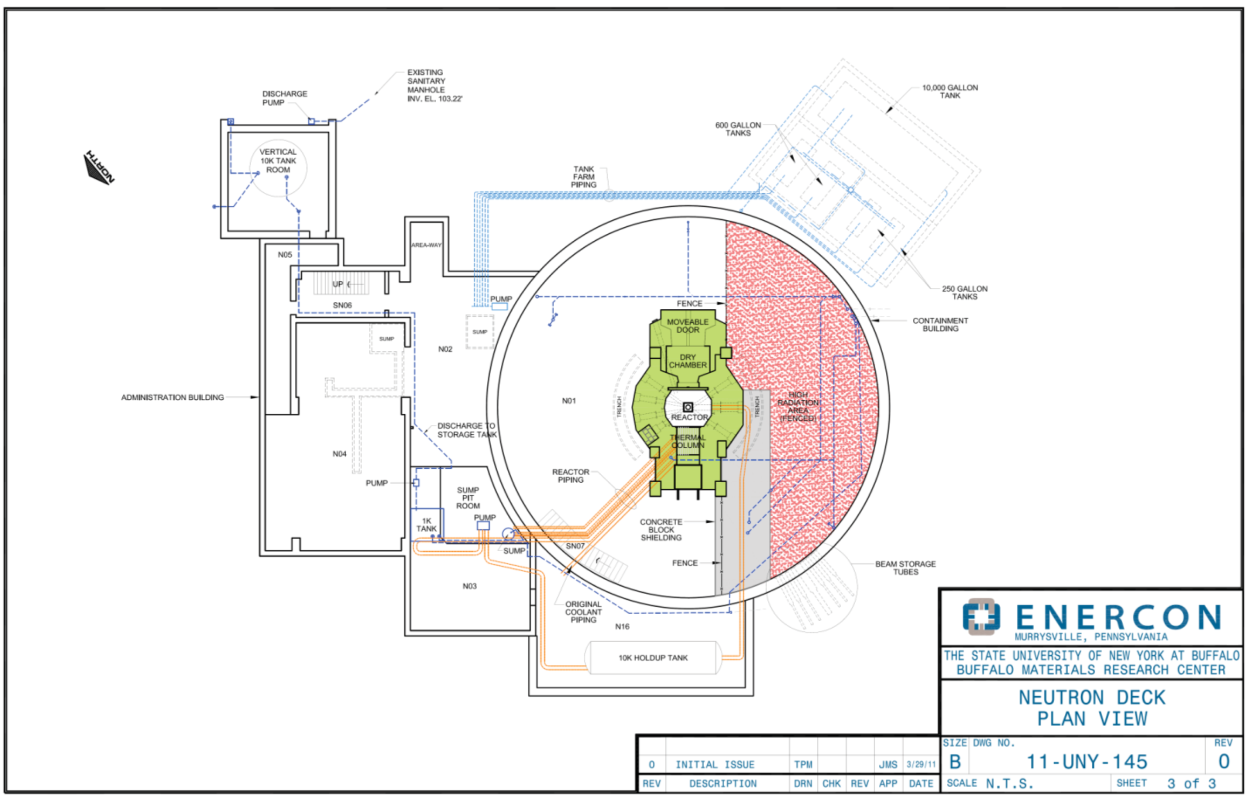
Figure 3: Gamma Deck Layout



The Neutron Deck

The Neutron Deck (lower level) is divided by caging into two sections. The open area includes access to the "Patient Treatment Room" which is lined with wood. The Patient Treatment Room was included in the original design of the BMRC to be a location to provide neutron treatments to cancer patients. No history that it was ever used for this purpose was found. The enclosed area includes the rerouted Primary Coolant Piping and access to the seven Beam Storage Tubes. The Beam Storage Tubes extend 10 feet through the containment building wall into the surrounding soil. They were originally designed to hold the beam tube plugs but were used as storage for decay of high dose reactor components or experiments. At the point where they exit the containment building wall, they are approximately 10 feet underground. The balance of the Neutron Deck is open area where the majority of radioactive contaminated materials awaiting disposal are being stored. Figure 4 provides the Neutron Deck layout.

Figure 4: Neutron Deck Layout



The Administrative/Laboratory Wing

The Upper Floor

The administrative wing housed the Locker Rooms (Rooms 209, 221, and 221A), Conference Room (Room 212), offices (Rooms 208, 210, 214, and 216), the Reception Area (Room 208A), Upper Air Conditioning Equipment Room (Room 209B), the Janitors Closet (Room 209A) and two low level labs (Rooms 213 and 215) on the upper floor.

The Basement Floor

On the lower or basement floor there was a Horizontal Accelerator (Room 118) with an adjacent cave area. This area has supported multiple functions; including a lab, a Class Room (Room 114), a Health Physicist Office (Room 117), Counting Room, The Vertical Accelerator Room (Rooms 115 and 115A), and a Machine Shop (Room 113). The Electrical Service Room (Room 110) is also located on this level. There is a tunnel off the Electrical Service Room (Room 110) that connects to the Campus Service Tunnel system and is locked to prevent access.

Subbasement

The space below the basement floor level is the area designated as the subbasement. This area has rooms housing the N16 Vault (Room N16), the Cooling Water Equipment Room and the Facility Sumps (Room N03), the lower Air Conditioning Equipment Room (Room N02), the Air Handling Duct Room (Room N04), and access to the 10,000-gallon tank enclosure (Room N05). These rooms house the decay, filtration, and cooling systems for the reactor. Room N03 also contains the 1,000-gallon waste holding tank.

3.0 ALTERNATIVES

There are four alternatives available to, and considered by, the UB: 1) the No-Action alternative (SAFSTOR); 2) the entombment option (ENTOMB); 3) complete decontamination and structure demolition (DECON-A); and 4) complete decontamination and release of the structure (DECON-B). The DECON options are recommended by the NRC for non-power reactors. The selected alternative is DECON-A. The four alternatives are summarized in the following sections.

3.1.1 SAFSTOR Alternative

The SAFSTOR alternative would involve the same potential risks and environmental impacts as the proposed action, but for a much greater time period. The alternative would require that the UB maintain current radiological controls, site security, required licensing, a reactor administrator, and the utilities until the DECON option was implemented. The UB will have to incur all of the expenses associated with maintaining the facility. This alternative would require that the UB apply for and obtain an extension to the current NRC operating license. The SAFSTOR alternative would leave the site as unusable space that will delay the beneficial reuse and the potential for environmental contamination would still exist. This alternative is not environmentally preferable.

3.1.2 ENTOMB Alternative

The ENTOMB alternative would require the same as the SAFSTOR alternative with the additional requirement that the radioactive material is placed in a long-lived structure, i.e., concrete, or the current structure proven adequate for long-term storage. This alternative delays the impacts of the proposed action and the potential for environmental contamination would continue to exist. This alternative is not environmentally preferable.

3.1.3 DECON-A Alternative

The facility records and current facility characterization reveal minimal facility contamination from past reactor operations. This alternative requires the site to be released and restored for unrestricted use. The reactor, the containment building, and the administrative building under this option will be disassembled and the radioactive material removed to meet the release criteria. The BMRC facility is located in the center of a growing university and the land area could be reused for future construction; therefore, complete decontamination and demolition (D&D) of the BMRC facility, DECON-A, is the preferred option. This alternative poses minimal risk and impacts to the environment as described in Section 8.0.

3.1.4 DECON-B Alternative

The DECON-B alternative is similar to DECON-A, however, the structure would not be removed after remediation. The risk and impacts to the environment are the same as DECON-A for this alternative, but the UB has determined that the reuse of the building is not in its best interests. Additionally, a final status survey of the soils and bed rock is more complicated and requires a significantly greater level of surveying and sampling than releasing an open excavation. This alternative is not preferred by the UB.

3.2 Decommissioning of the Reactor, License Termination, and Site Reuse

3.2.1 Decommissioning of the Reactor

UB plans to decommission the reactor which will require the removal of all radioactive materials from the BMRC facility; dismantling the reactor, the reactor peripheral support systems, and support buildings; and removal of the UST and associated piping, materials associated with the previously demolished cooling tower, and any legacy waste.

Some of the reactor components and systems are either activated or contaminated and will need to be segregated from non-radiological components and surfaces. Components and systems that are designated as activated or impacted will be disposed of as low-level radioactive waste (LLRW). Building materials, such as the reactor tanks, will need to be evaluated for radiological activity and removed and disposed of according to their radiological status, as necessary.

The following are decommissioning tasks, which are necessary for site release. The sequence in which these tasks occur may vary:

- Pre-Decommissioning Cleanout - removal of loose legacy equipment and materials Isolation and removal of inactive systems
- Removal of hazardous materials (i.e., lead) and abatement of asbestos containing material (ACM)
- Perform supplementary characterization
- Installation of temporary systems and preparation of the facility for decommissioning operations
- Removal of the irradiated reactor components
- Removal of and disposition of water in the reactor
- Removal of and disposition of the reactor liner
- Decontamination of concrete associated with reactor, bioshield, and hot and dry cell
- Removal of the reactor, bioshield, and hot and dry cell
- Segregation, packaging and shipment of materials according to radioactivity levels
- Removal of auxiliary systems (rabbit system, water purification, ventilation)
- Decontamination of building surfaces
- Interior release survey using Reg Guide 1.86 limits
- Demolition of the BMRC facility
- Perform the FSS on BMRC footprint

- Submit required reports that demonstrates to the NRC that the facility meets the release requirements
- Request license R-77 termination
- Restore the site by backfilling the void created by removing below grade structures for future use by UB.

The BMRC has been in Possession Only status since June 1997 and the spent fuel was removed from the site in 2005. The on-site decommissioning tasks are expected to start date in the second half of 2012.

The FSS will be developed by the Design and Oversight Contractor (DOC) using the criteria provided in NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NRC 2000). Radiological measurements identified in the Characterization Report indicated that the typical levels of residual radioactive materials on the building surfaces are less than release criteria established by the NRC in Regulatory Guide 1.86. Residual radioactive materials were only found in areas indicated by the Historical Site Assessment. Soil samples indicated that only limited remediation of soils will be required outside of the facility. Radionuclides above derived concentration guideline levels (DCGL) were detected in the N-16 Vault, which does not have a concrete floor covering the bedrock. The N16 Vault is identified on Figure 4. The structures and components that contain the majority of the residual radioactive materials are the activated reactor components and bioshield, the former and current liquid waste tanks, and the reactor water cooling system. Residual radioactive materials were also identified in the Hot Cell and laboratory fume hoods. Since all BMRC structures, systems, and components will be removed from the site, the FSS will only need to cover the exposed surface soils.

3.2.2 Licenses

The BMRC maintains NRC license No. R-77 for Possession Only. Recently in 2011, the BMRC's Special Nuclear Material (SNM) License No. SNM-273, for possession of a Plutonium-Beryllium (PuBe) neutron source was terminated and the PuBe source transferred to the University's New York State Department of Health (NYSDOH) radioactive material license, No. 1051. The scope of the NYSDOH license is for possession of sources and for decommissioning.

The R-77 license is to be terminated as part of the decommissioning. The MARSSIM NUREG-1575 provides guidance to assemble a statistically accurate FSS plan to support the termination of the NRC license. The NYSDOH license will have all sources transferred to another NYSDOH license at UB. The SNM license will have a request for license termination submitted after the NYSDOH license is amended in order to transfer the PuBe source.

3.2.3 Reuse of Site

The site is expected to be reused as either green space or a future road.

3.3 Cumulative Effects

The cumulative effects of the implementation of the proposed action will result in short-term cumulative impacts as discussed in Section 5.0 of this ER.

3.4 Comparison of the Predicted Environmental Effects

The following table provides the predicted environmental effects and impacts of the alternatives that were evaluated for this ER. More detailed information on the expected impacts can be found in Section 5.0 of this ER.

Affected Environment	No Action	Proposed Action
Land Use	Possible Impact	Possible Impact
Transportation	No Impacts	Possible Impact
Geology and Soils	Possible Impact	Possible Impact
Water Resources	No Impacts	No Impacts
Ecological Resources	No Impacts	No Impacts
Meteorology, Climatology and Air Quality	No Impacts	Possible Impact
Noise	No Impacts	Possible Impact
Cultural and Historic Resources	No Impacts	No Impacts
Visual/Scenic Resources	No Impacts	No Impacts
Socioeconomic	No Impacts	No Impacts
Environmental Justice	No Impacts	No Impacts
Public and Occupational Health	No Impacts	Possible Impact
Waste Management	No Impacts	Possible Impact

4.0 DESCRIPTION OF THE AFFECTED ENVIRONMENT

The following sections describe specific areas of the environment that may be affected because of the decommissioning activities.

4.1 Land Use

The BMRC is a stand-alone structure located on the south edge of the South Campus of UB. The structure is comprised of two distinctively different architectural components, the containment building and the attached administrative wing.

The containment building is a three-story right cylinder constructed of concrete, 70 feet in diameter with a height of 52 feet. The reactor is located within the three-story cylindrical building. The former core was immersed in a 13,000-gallon, aluminum-lined tank pool surrounded by both high density and normal concrete. The cylindrical building walls and the lower level of the containment building rest directly on bedrock, the exterior walls are poured concrete finished with paint, and the roof of the cylindrical building is four-inch thick reinforced concrete supported by concrete beams. Interior floors are poured concrete with the Neutron Deck formed of two courses of concrete with a finish course of about 1.25 inch thickness, and the Gamma and Control Decks are monolithic. Select floors in the administrative building are formed of terrazzo, mosaic tile, or resilient floor tiles. Ceilings in select areas may be formed of acoustic tiles of varying sizes resting on a metal grid system suspended with metal hangers.

The administrative wing is comprised of three levels, the Upper Floor, the Basement Floor, and the Foundation Level. The administrative wing was originally constructed for office, laboratory, utility rooms, and classrooms for activities conducted with and/or for the reactor. The exterior walls are formed of load-bearing concrete masonry blocks and are finished with a brick façade. Interior floors primarily consist of nine-inch by nine-inch vinyl asbestos floor tiles adhered to the slab with adhesive. Walls are formed of painted masonry block and finished with paint.

<u>The Tank Farm</u>

The Tank Farm lies in the subsurface area south of the containment building and extends towards the southeast of the containment building. The Tank Farm consists of two 250-gallon stainless steel tanks and two 600-gallon stainless steel tanks in connected concrete enclosures. There is also a 10,000-gallon carbon steel tank outside of the enclosures to the east. The 250-gallon tanks were connected to the Isotope Processing Labs and the Hot Cell and the 600-gallon tanks supported the drains from the lower level labs and the floor drains. The tanks were interconnected by a manifold and pump system that allowed any combination of water transfer. The normal movement was to pump the small tanks to the 10,000-gallon tank prior to discharging to the sewer system. The 10,000-gallon tank sits on concrete saddles inside a concrete bermed area. The enclosure and the 10,000-gallon tank are beneath 3-4 feet of soil.

4.2 Transportation

The BMRC is located in a relatively moderate to high traffic area in the northeastern section of the city limits of Buffalo, New York. Surrounding areas are largely residential. A recent traffic congestion study by INRIX, as reported by a local Buffalo television station, indicated that while traffic congestion in Buffalo is better than the national average, it has increased from 2009 to 2010 faster than the national average. In Section 3.0, Highway System Profile of a Niagara Frontier Urban Area Freight Transportation Truck study indicated that the region has a total of 3,675 miles of highway network, including major interstates, state routes, and local arterial roads. The major interstate highways that service Erie and Niagara counties included Interstate 90, Interstate 190, Interstate 290, and Interstate 990. The local streets will be utilized to transport radioactive waste and construction debris associated with the decommissioning and demolition of the BMRC. Interstates 90 and 190 are less than five miles from the site and will likely be utilized as transport routes to the eventual disposal destinations.

4.3 Geology and Soils

The City of Buffalo is located within the Niagara Region, a section of a great plain that extends north to south from the Laurentian Highlands (Canadian Shield) approximately 161 kilometers north of Toronto, Ontario to the Southern Allegheny Plateau. This plain is a small portion of the Great Lakes low lands within which all of the Great Lakes are located. Within the Niagara Region is located the Niagara Escarpment, the weathered edge of an ancient sea bottom extending roughly from Watertown, New York westerly through Illinois and Wisconsin. The Escarpment is neither a fault line nor a rift line, but is instead an erosional feature classified as a cuesta, a ridge with a gentle slope on one side and a cliff on the other. (NUREG-0982)

A smaller escarpment, the Onondaga Escarpment, often referred to as the Onondaga Cuesta, trends east to west located along the northern shore of Lake Erie. The escarpment is capped with the Onondaga limestone. The SUNY campus is located on the extreme edge of the Onondaga Cuesta. The cuesta faces the northwest and has a local relief of 80 feet. The southeastern portion of the campus is located on the dip slope of the cuesta. In general, the area within which the City of Buffalo is located is very flat with the exception of the Onondaga Cuesta. As a result, local streams are slow moving and widely spaced. (NUREG-0982)

The bedrock of the area consists of Silurian and Middle Devonian marine shale, dolomite and limestone with a southerly drop of 50 feet per mile. The rock formation is a joint system consisting of two joint sets arranged in a north-south east-west configuration intersecting at approximately 90 degrees. This is underlain by a similar marine series of Ordovician and Cambrian age to a depth of 3,000 feet that rests on a pre-Cambrian deposition. (NUREG-0982)

The overburden from the bedrock to the surface averages about 15 feet in the area of the BMRC. The soil proper occupies the first 14 inches with subsoil, which is made of dense boulder and clay extending uniformly to the bedrock. The soil from the surface to a depth of nine inches consists of moderately compacted, brown-gray, heavy, silty clay loam. Beneath this is a roughly five-inch stratum of compact dull yellowish-brown silty clay with a considerable amount of fine pebbles. The layer below this is made up of dull brown clastic, heavily compacted, silty clay containing crystalline pebbles dispersed throughout. This clay material extends to bedrock without any significant change. (NUREG-0982)

The data in this section was collected during initial licensing and design of the BMRC. The primary sources of information for this section is the reactor's Safety Analysis Report and NUREG-0982, *Safety Evaluation Report related to the renewal of the operating license for the Research Reactor at the State University of New York at Buffalo*. More recent drilling on the south campus in 2008 and 2009, supports the above description of the area geology and soils.

4.4 Water Resources

The City of Buffalo is located along the eastern end of Lake Erie. Water comes from Lake Erie into an intake out in the lake and flows through a 12-foot by 12-foot conduit to the Colonel Ward Water Treatment Plant and out to the city through 800 miles of pipe controlled by 25,000 valves. The system supplies an average of 99 million gallons per day to 82,000 service connections and 7,600 fire hydrants. A peak supply of 127.8 MGD was reached in 2001. This capacity is adequate to satisfy foreseeable future demands. Water quality is good and complies with all regulations.

Four miles east of the campus at Williamsville is Elliot Creek, which flows north to an unnamed creek that has its headwaters approximately two miles north of UB campus. There are no other streams in close proximity to the BMRC. The nearest water access is Lake Erie, which is five miles to the southwest of the BMRC site. The Niagara River, on which Niagara Falls lie, is 4.6 miles to the southwest while Lake Ontario is approximately 30 miles from the facility. These bodies of water are not anticipated to be impacted by any activities at the BMRC.

The UB campus is located along the edge of the Onondaga Cuesta, which trends from the southwest to the northeast. The southeastern portion of the campus is positioned on the down slope side of the cuesta.

With the exception of the cuesta, the general area is relatively flat. As a result, streams are slow moving and widely spaced apart. In the immediate area of the BMRC, stormwater run-off flows generally towards the south where it is collected by a stormwater management system and eventually routed to the sewage treatment plant on campus. There are no streams in the immediate vicinity of the BMRC. The slope of the adjoining land and the drainage path near the building will not permit accumulation of water around the building. There are no credible paths for reactor pool water to get into the campus water system, however surface water run-off is routed towards the UB sanitary sewer system. There are no surface water features in the immediate vicinity of the reactor. The nearest surface water feature is a small pond 0.6 miles south of the BMRC located in McCarthy Park.

The density of the clay soils does not allow percolation of water into the sub-surface. For all practical purposes, surface water penetration ceases below a depth of 13 inches. Water will move through the saturated and undisturbed material in the first eight inches of topsoil at an average rate of 0.02 to 0.2 inches per hour. In the next five-inch layer, the penetration decreases sharply to zero. The underlying Onondaga and substrata are not known to contain water in the immediate vicinity, but it is not impervious. Based on observations several miles east of the campus, the water was presumed to move to the escarpment edge. Water movement within the overburden including the soil proper is extremely slow. Only within the first 13 inches water percolation takes place very slowly, 0.2 inches per hour. In the next 15 feet, permeability ceases for all practical purposes. Surface water flows in a southerly direction into established storm drains. (NUREG-0982)

4.5 Ecological Resources

The BMRC is located within a well-developed area where virtually all flora consist of species introduced for landscaping of residential, commercial, and educational facilities. The likelihood of endangered native species of plants is very small. Any native fauna would consist primarily of small, nocturnal animals not likely to appear on any endangered species list. The DEC website for endangered species did not list any insects as endangered or threatened in Buffalo, New York.

4.6 Meteorology, Climatology, and Air Quality

Buffalo experiences a fairly humid, continental- type climate, but with a definite "maritime" flavor due to strong modification from the Great Lakes. Lake Erie lies to the southwest of the facility and functions to moderate the temperatures. The average temperatures measured at the Buffalo International Airport range from 18 degrees to 80 degrees Fahrenheit (F) with extremes recorded of -20 degrees and 99 degrees F. Annual rainfall averages 38.5 inches in the Buffalo area and snowfall averages 93.3 inches per year.

Winters in Western New York are generally cloudy, cold, and snowy but are changeable and include frequent thaws and rain as well. Over half of the annual snowfall comes from the "lake-effect" process and is localized. Due to the prevailing winds, areas south of Buffalo receive much more lake-effect snow than locations to the north.

As the prevailing airflow is mostly southwesterly, areas near the lake are often as much as 20 degrees colder than inland locations. The cool lake waters act as a strong stabilizing influence, so areas near the lakeshore, which includes the City of Buffalo, experience more sunshine and fewer thunderstorms than inland areas.

Summer is also influenced by the stabilizing effect of Lake Erie. It inhibits thunderstorms and enhances sunshine in the immediate Buffalo area. The lake also modifies the extreme heat that approaches from the Ohio Valley. Tornados are rare in Buffalo; the Weather Channel website reports that only 19 tornados have struck the vicinity of Buffalo from the period 1950 to 2009.

4.7 Noise

The BMRC is located within an academic setting and noise is typically at a low level. At the time of the drafting of this document, renovation activities were occurring at a nearby structure. The Environment, Health and Safety Department (EH&S) of UB SUNY has issued guidance for noise in their Contractor Safety Guidebook, Document Number SA-023-A, August 25, 2009. Section 9.0 instructs contractors to keep work areas as quiet as possible when the work is proximal to buildings within which class activities are occurring. The UB SUNY project manager or the EH&S can temporarily halt work activities that cause noise at a level that affects classrooms activities.

The Common Council of the City of Buffalo has issued an ordinance to regulate the level of noise within the city under Chapter 293, Noise, August 13, 1990. Section 293-4 states the following regarding construction:

I. Construction activity.

> (1) The performance or engagement in construction work, building, excavating, hoisting, grading, demolishing, dredging or pneumatic hammering within the limits of the city between the hours of 9:00 p.m. and 7:00 a.m. that causes sound which annoys or disturbs a reasonable person of normal sensitivities in a residential real property zone, except for emergency work of public service utilities or as otherwise provided in Subsection I(b) herein [is prohibited].

4.8 Cultural and Historical Resources

The Buffalo Preservation Board maintains a website with detailed information on various historic districts within the City of Buffalo. The Board is an agency of the City of Buffalo and is administered by the City of Buffalo Office of Strategic Planning. Nine historic districts are listed as either Certified Local Preservation District or National Register Historic Districts. Maps for each of the districts are available for viewing on the website. After viewing the maps, it was determined that the UB South Campus is not located in any of the historic districts. The nearest historic district was the Hamlin Park Local Preservation District.

The historic district known as Hamlin Park is located approximately two miles south-southwest of the BMRC. Hamlin Park developed during the first 25 years of the 19th century as a textbook zone of emergence. The name of the area is derived from a former horse racing park, Hamlin's Driving Park, which opened in 1858. The owner of the track, Cicero Hamlin, once touted a race as the Kentucky Derby of the North. Following Cicero Hamlin's death in 1905, Hamlin's racing park lay dormant. After several fires and years of neglect, this three hundred acre parcel of scruffy and largely vacant land lay empty and unused until 1912, when it was bought by a Toronto real estate developer. Homes were developed for the middle class and quickly sold. By the late 1950's and early 60's, Hamlin Park had changed. A transition had occurred as middle-class African-Americans moved in and in the process created the best and the nicest African-American neighborhood in Buffalo. In 1999, the area was officially designated as a Historic District.

Several buildings of historical significance are located within the city limits of Buffalo. A select few worthy of mention include Buffalo's City Hall which opened in 1931; St. Paul's Episcopal Cathedral, originally built in the mid-19th century and re-built 40 years later after a fire; and the Theodore Roosevelt National Historic Site, the site of the emergency swearing in as president of Theodore Roosevelt following the assassination of President William McKinley in Buffalo in 1901. The Buffalo Preservation Board lists 89 local Buffalo Landmarks, and 77 state and regional registers within the Buffalo city limits. There were no sites listed that correlated with addresses or buildings located on or near the South Campus

The UB South Campus, also known as the Main Street campus, is the former grounds of the Erie County Almshouse and Insane Asylum, of which four buildings remain (Hayes Hall, the former insane asylum; Wende Hall, a former maternity hospital; Hayes D; and Townsend Hall, a former nurses' quarters). The four buildings were constructed during the mid to late 1880s. The college was designed by architect E.B. Green in 1910 and was originally intended to resemble Trinity College in Dublin, Ireland. Although these structures are of a sufficient age and historical significance, they are not listed on any documents for local, state, or national historical recognition.

4.9 Visual/Scenic Resources

The BMRC is located in the southeastern section of the UB South Campus. The campus is within an area of heavy residential neighborhood to the south, west, and north. The Veterans Administration Hospital is to the southeast and a golf course is located to the northeast. Some commercial development is present north of the South Campus.

The BMRC is located within what would be classified as a foreground-middle ground zone by the BLM Visual Resource Inventory and Evaluation System. Local residents would rarely glimpse the structure as it is positioned such that the view from the south, southwest, and southeast is obscured by other campus buildings. The population most likely to view the BMRC on a regular basis is the student population. It is unlikely that most students would feel a negative aesthetic impact during the demolition of the BMRC or after the structure is removed.

4.10 Socioeconomic

Buffalo is the second most populous city in New York State with an estimated 261,310 residents (2010 Census). The city has seen a steady decline in population numbers since peaking at around 580,132 residents in 1950. The decline has come about because of the loss of industrial base but researchers of the population indicate that the population decline may have finally stabilized. Based on the 2005-2007 American Community Survey Estimates, the population was 53% white, 41% black or African American, 1% American Indian and Alaska Native, 2% Asian, 4.5% of another race and 2.5% from two or more races. Buffalo has sizeable populations of Irish, Italian, Polish, German, Jewish, Greek, Arab, African American, Indian, and Puerto Rican descent.

The nearest private residences that are not under UB control are approximately 450 feet south of the BMRC. The nearest private residences in the direction of the prevailing southwest to northeast winds that are not under UB control are approximately 3,700 feet northeast of the BMRC. There are no heavy industrial facilities near the South Campus. The nearest airport, Buffalo Niagara International Airport is approximately 4.5 miles east-southeast of the campus. The intersection of Interstates 290 and 90, major highways in the area, are approximately 2.5 miles east of the campus. An Amtrak passenger station is located approximately 5 miles south-southwest of the BMRC. Freight by rail is offered by CSX, Norfolk Southern, Canadian Pacific, and Canadian National in the Buffalo area. The United States Coast Guard

Auxiliary, Flotilla 3-3, 9th District, Eastern Region is located 6.25 miles southwest of the campus. There are no other major military bases or facilities in the Buffalo region.

A May 1983 Safety Evaluation Report related to the renewal of the operating license for the Research Reactor at the State University of New York at Buffalo drafted by the United States Nuclear Regulatory Commission was reviewed as part of this report. The Report indicated that no heavy industry, heavy air or ground traffic would constitute an external threat to the integrity of the reactor facility. The report concluded that there was no significant risk from accidents to the reactor because of activities related to the military, industry, or heavy transportation traffic. Since the drafting of the 1983 report, the major interstates I-90 and I-290 have been added to the region to the east, northeast, and southeast of the BMRC. No other significant changes have occurred since 1983 that would alter the conclusions of the report. The addition of the interstates to the area is not considered a significant risk to the reactor at the BMRC.

4.11 Public and Occupational Health

Public and Occupational Health activities associated with the current operations of the BMRC include the following:

Containment Building/Reactor

All of the spent fuel was shipped to the Department of Energy (DOE) in September 2005 and the unused fuel was sent to North Carolina State University in July 1998. Radiation exposures from the former reactor core are reduced to acceptable levels by water and concrete shielding. The Primary Coolant heat exchanger was removed and disposed of as radioactive waste in 1994. The containment building had a significant amount of legacy waste on all three levels until the Pre-decommissioning Cleanout project in the summer of 2011. All of the reactor components remain in the pool and the components from the 1991 reactor rebuild were stored under shielding on the Neutron Deck of the building until they were sent for burial at the EnergySolutions' facility in Clive, UT during the Pre-decommissioning Cleanout Project. There are no regularly scheduled activities within the containment building.

ENERCON conducted a Historical Site Assessment (HSA), and released a report with findings on March 17, 2010. The HSA reported that events and operations have transpired that have impacted the containment building. The roof, interior and all the systems and structures in the building are designated as Impacted Areas. These areas and/or components have been designated as impacted due to elevated levels of residual radioactive materials or the potential for the presence of residual radioactive materials and are expected to require, in part or whole, remediation and/or disposal at an NRC licensed disposal facility. The HSA reported that the primary contaminants of concern for the entire BMRC site were H^{3}, C^{14}, Mn^{54}, Co^{58}, Co^{60}, Eu^{152}, Eu^{154}, Ag^{108m}, Ag^{110m}, Sb^{124}, La^{140}, and Ni^{63}. Since the plant has been shutdown since June 23, 1994, the Co^{58}, Sb^{124}, Ag^{110m} and the La^{140} have decayed significantly due to their short half-life. The HSA concluded with a recommendation that scoping/characterization surveys be conducted for the containment building.

ENERCON conducted a site characterization study and released a Site Characterization Report (SCR), on May 30, 2011. The study did not include the activation of reactor components and the pool liner. The SCR offered the following findings:

- Control Deck:
    - Many areas were being used to store legacy radioactive wastes. The elevated background from the radioactive material storage areas made it difficult to determine the presence of residual radioactive materials on building surfaces in these areas. However, the limited radiological measurements collected in the areas of the Control Deck did not identify removal or fixed radiological activity. Therefore, it seems reasonable to conclude that the building surfaces in these areas are not contaminated. Additional testing for confirmation of this conclusion will be performed when all legacy radioactive wastes are removed from the area.
    - Room 202 was the location of a strontium-90 (Sr-90) release that contaminated areas of the walls adjacent to the California hood. Sections of Plexiglas have been affixed to the walls to afford temporary shielding from the strontium.
- Gamma Deck
    - Most rooms on the Gamma Deck are currently being used to store radioactive waste, making characterization of building surfaces difficult. Additional testing of building surfaces will be performed subsequent to the removal of all items from the area. All items inside the Hot Cell, including the Hot Cell, should be considered radioactive waste.
- Neutron Deck
    - The Neutron Deck is the area where the majority of radioactive waste materials awaiting disposal are currently stored. Subsequent to the removal of all radioactive material, a more detailed investigation of the building surfaces will be performed.
    - Concrete cores of the floor were collected at six locations. Coring confirmed that the concrete was originally poured directly onto bedrock. Because a sand or gravel layer was not placed on top of the bedrock prior to pouring of the concrete, no preferential pathway exists between the Neutron Deck and the bedrock. The analytical results of the testing indicated background levels for the evaluated radionuclides.
    - Concrete cores of the bioshield were collected at six locations. The data from the testing indicates measureable radioactivity due to neutron activation in certain areas exceeds 12 inches in depth.

Administrative/Laboratory Wing

The administrative wing also has radioactive materials stored in various areas. In addition, the UB radiological instrument calibration facility is located in the Laboratory Wing. The administrative wing is used infrequently for two purposes. One purpose is an ion chamber calibration facility that is located in the basement of the administrative wing. Secondly, the top floor has asbestos-containing floor tile in some areas. The UB Maintenance Department uses this floor to train maintenance and custodial personnel how to safely strip and wax this type of floor tile.

Based on the HSA, dated March 17, 2010, events and operations have transpired that have impacted the administrative wing. The roof, interior and portions of the systems and structures in the building are designated as Impacted Areas. These areas and/or components have been designated as Impacted due to elevated levels of residual radioactive materials or the potential for the presence of residual radioactive materials and limited portions are expected to require remediation and/or disposal at an NRC licensed disposal facility.

The ENERCON SCR offered the following findings:

- Control Deck
    - Rooms 208, 208A, 209, 209A, 209B, 210, 212, 213, 214, 215, and 216 showed no elevated measurements of residual radioactive materials.
- Gamma Deck
    - Rooms 110, 111, and 113 showed no elevated measurements of residual radioactive materials.
    - Rooms 114, 115, 115A, 117, and 118 contain radioactive materials. These spaces may receive additional surveys subsequent to the removal of radioactive materials.
- Sub-Basement
    - Room NO3 contains a 1,000-gallon tank containing six inches of sludge with elevated levels of radioactive materials.
    - Room N16 contains a 10,000-gallon tank internally contaminated with radioactive materials.

Exterior Sub-Surface

- Soil samples characterized for radionuclides outside the facility indicated that remediation of soil will not be necessary.

Tank Farm

- The laboratory analysis of samples collected from each of the five tanks indicated that elevated levels of radionuclides are present in the waste tanks. The liquids will require proper handling to ensure radioactive materials are not released to the environment. The water volume in each tank is 75-100% of their maximum capacity.
- The tank farm has sample tubes that extend down outside of the tank to the bottom of the enclosure and the bottom of the concrete berm. These sample tubes, known as "tell-tale" samples, have not indicated any major leaks from the tanks.

N 16 Tank Vault

- Testing of dirt and debris on the bedrock in the N16 Tank room indicated impacts greater than the NRC and New York State (NYS) screening values for Co-60. Ag-108m was elevated and has a high potential to be greater than the DCGL's stated in the decommissioning plan.

1,000 Gallon Tank

- Water and sediments were collected for characterization from the 1,000-gallon tank located in Room N03. The tank sediments contained elevated levels of the radionuclides Co-60, Ag-108m, Ni-63, and Sr-90. The water sample contained Ni-63.

Extraneous Sources

Sources of radiation within the BMRC were associated primarily with legacy wastes stored in a variety of areas in both the administrative wing and the containment Building. The reactor is inoperable and all fuel and spent fuel have been shipped from the site. There are currently no regularly scheduled activities within the BMRC except for occasional training and routine inspection. UB has developed written policies that ensure that authorized persons only are allowed into the BMRC. An intrusion alarm system has been installed and is operational any time the BMRC is unoccupied. The alarm system is checked on a weekly basis.

UB has developed a written monthly and quarterly check sheet to ensure that failsafe systems are operational within the BMRC. Systems inspected and calibrated include, but are not necessarily limited to, the Building Gas electronics, Building Particulate electronics, the Control Deck Continuous Air Monitor (CAM), and the emergency damper closure time in response to a manual damper scram. Groundwater samples are collected from the underground wastewater vaults on a monthly basis to test for radionuclides.

Routine Monitoring

According to UB EH&S staff, approximately ten to twelve UB employees are required to wear personal Thermoluminescent dosimeters (TLD) during activities in the BMRC. The personnel badges are analyzed on a quarterly basis. The EH&S staff report that personal exposures are less than the minimum reportable dose of 10 mrem per sampling period. TLDs are also placed at various interior and exterior locations to monitor the general areas of the BMRC. The TLDs are submitted to a testing laboratory for analysis on a monthly basis. The exterior TLDs read 0 to 4 mrem above background levels per reporting period. The analytical results of the interior TLDs typically range from 0 to 2 mrem per month, however, the TLD positioned on the bridge over the reactor pool yields the highest readings of 3 to 7 mrem per period.

A 1963 report on Environmental Monitoring Program for Western New York Nuclear Research Center Inc. and UB was available for review. The report detailed environmental monitoring of the former BMRC and surrounding area beginning on March 1, 1961. Sampling for radioactivity was performed on surface water, soil, vegetation, air, and atmospheric fallout. The report concluded that activities at the nuclear research facility had not caused any detectable increase in the radiation background of the environment.

4.12 Waste Management

A Waste Management Plan (WMP) will be implemented for the disposal of the waste generated during the BMRC D&D project. The WMP will include detailed guidance for the characterization, sampling, classification, segregation, handling, packaging, manifesting, transporting and disposal of all waste categories.

ENERCON performed a waste study for the BMRC of the legacy waste items and released a report dated December 2009, *Study to Determine Quantities, Types, and Disposition of Waste at the Buffalo Materials Research Center*. The waste study divided all of the materials that will eventually require disposal into five potential treatment categories. These are as follows:

- Items for on-site survey and unrestricted release (free release)
- Bulk Survey for Release (BSFR)
- Off-site Processing
- Lead Processing
- Direct Disposal

The waste study discusses the five potential treatment categories in depth and the reader of this report is referred to the waste study for additional detail, if needed. The legacy waste was removed during the Pre-Decommissioning Cleanout. The R-77 license authorizes the licensee to ship radioactive material.

5.0 ENVIRONMENTAL IMPACTS

The following sections describe the potential environmental impacts that may result from the decommissioning activities of the BMRC.

5.1 Land Use Impacts

Land use impacts are expected to be significant during the decommissioning activities. The construction activities involved during decommissioning will require the removal of all radioactive materials from the BMRC, dismantling the reactor and any peripheral support systems, removal of UST and associated piping, removal of materials associated with the previously demolished cooling tower, demolition of both wings and removal of any waste.

Following the decommissioning, the site will be returned to usable space by backfilling the void created to match the normal grade of the surrounding areas. Any further land use impacts are expected to be minimal.

5.2 Transportation Impacts

The primary project impacts to the environment from transportation would occur when various forms and quantities of radioactive, nonradioactive, and hazardous wastes are shipped from the BMRC facility during decommissioning activities. During transport, all materials are to be effectively packaged to prevent significant radiation or other impacts external to the truck. Thus, the primary impacts are accident risk and emissions/noise from the trucks themselves.

The dose consequence from transportation accidents occurring with radioactive waste could be higher than the contamination accident scenarios described in Section 5.6 and Section 5.12.2 because high-activity reactor components could be involved. As such, there is a potential for a moderate dose consequence of between one 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.

Local streets will be utilized initially to transport radioactive waste and construction debris associated with the decommissioning and demolition of the BMRC. Interstates 90 and 190 are less than five miles from the site and will likely be utilized as transport routes to the eventual disposal destinations. Therefore, it is expected that there will be minimal transportation impacts due to increased trucking activity during the decommissioning activities.

5.3 Geology and Soils Impacts

Based on the Site Characterization results three (3) samples collected from the N16 room in the sub-basement indicated impacts greater than the NRC Screening values for Co-60. Additionally, Ag-108m was elevated and has a high potential to be greater than the DCGL developed as part of the Decommissioning Plan. The impacted dirt and debris will be removed during the decommissioning activities. Therefore, it is expected that there will be no geology or soil impacts as a result of decommissioning of the BMRC.

5.4 Water Resources Impacts

ENERCON anticipates that there will be no impact to water resources as a result of decommissioning of the BMRC. All water generated as a result of the decommissioning of the BMRC will be contained and disposed of in accordance with all UB, federal, state and local regulations. No water other than water mist for dust control will be introduced to the subsurface environment or into the natural environment outside of the facility. No water will be discharged to the stormwater system, unless it has been sampled to verify it is within regulatory limits.

5.5 Ecological Resource Impacts

No ecological resource impacts are expected, as all work will be performed within the BMRC. Potential lay down areas for equipment will only be placed in developed areas. Decommissioning the reactor would have no impact on amphibians, reptiles, birds, fish, mammals or threatened species.

5.6 Air Quality Impacts

The decommissioning activities could minimally impact the air quality near the BMRC from both stationary and mobile source emissions.

A small increase in mobile source emissions, such as carbon monoxide and nitrogen oxides, could be released from contractor’s trucks and cars. Due to the temporary nature of the decommissioning activities, effects from mobile source emissions would be low.

Stationary emissions from the decommissioning activities could occur during demolition and soil remediation activities. The emissions from the building demolition, subsurface material excavations and general construction work are expected to be minimal. All applicable dust control measures will be implemented as part of the decommissioning construction activities. Standard asbestos abatement procedures implemented by a contractor licensed by the State of New York will be used to remove any asbestos-containing materials. Site workers would be protected during decommissioning activities through air monitoring and the use of air purifying respirators when required.

An uncontrolled release of airborne radioactivity could occur during cutting and demolition activities involving activated materials. Such activities may take place inside temporary containment structures equipped with high-efficiency particulate air (HEPA) filter ventilation systems or other systems, if applicable. The failure of a containment structure could result in the release of airborne radioactive materials into environment. Therefore, an alarming CAM will be used in the work areas to warn against the release of airborne radioactivity.

While the actual concentrations of airborne radioactive materials are unknown at this time, the dose consequence of an uncontrolled release is expected to be low (<1 mrem off-site impact and < 25 mrem to on-site workers). As such, safety management operations (standard engineering and administrative controls) are sufficient for protection against such accidents.

5.7 Noise Impacts

Noise impacts during the decommissioning of the BMRC are expected to be those associated with normal construction activities. The UB project manager or the EH&S can temporarily halt work activities that cause noise at a level that affects classrooms activities. The construction activities will take place between 7:00 am and 9:00 pm to ensure compliance with the City of Buffalo noise regulations.

5.8 Historical and Cultural Resources Impacts

The decommissioning and demolition of the reactor and support facilities is not expected to have any historical or cultural impacts.

5.9 Visual/Scenic Resource Impacts

Visual/Scenic Resource impacts associated with decommissioning are primarily visual and relate to the structures and visual attributes of the decommissioning site. The impact of decommissioning on site aesthetics (e.g., truck traffic, demolition, noise) is limited in duration. Therefore, any impacts are temporary and are expected to be minimal.

5.10 Socioeconomic Impacts

No socioeconomic impacts are expected from the decommissioning of the BMRC.

5.11 Environmental Justice

High and adverse health, economic or environmental effects to local low-income and minority populations characterize environmental justice. There is no reason to believe that low-income or minority populations would be affected by BMRC decommissioning.

5.12 Public and Occupational Health Impacts

Nonradiological and radiological impacts from decommissioning activities are expected to be minimal. The work associated with nonradiological and radiological controls will be conducted within approximately 150 feet of the UB property boundary and within 450 feet of the nearest residential area. Figure 5, on the next page, identifies the BMRC location in association with the property boundary and the nearest fulltime resident.

The following sections describe the potential nonradiological and radiological impacts associated with the decommissioning activities.

5.12.1 Nonradiological Impacts

As part of the decommissioning process nonradiological impacts will be associated with the removal of any general building materials and asbestos in the decommissioning area. The removal of asbestos containing materials will be conducted in accordance with applicable state and federal regulations. Removal of general building materials and debris will be conducted utilizing industry standard controls and applicable state guidelines.

5.12.2 Radiological Impacts

There is a potential for radiological accidents during the BMRC facility decommissioning project resulting from the uncontrolled release of radioactive materials to the work area or the environment. These releases are most likely associated with the mismanagement of contaminated liquids in the waste tanks. Uncontrolled releases of airborne contamination could also occur during the demolition of the reactor tank and segmentation of the activated bioshield. An uncontrolled release of radioactive material could also occur during a transportation accident.

The proper handling of the waste liquids by draining the tanks prior to moving them is deemed adequate to prevent uncontrolled release. The demolition of the reactor tank and the segmentation of the bioshield should be conducted inside containment with HEPA filtration to help prevent an unplanned release to the environment.

The accidental dropping of an activated reactor component was also considered as a potential accident. However, because the more highly activated components are located under water and the water is purified, the surface contamination on these parts is minimal and would not release significant quantities of radioactive materials during such an incident. Such an incident would most likely result in additional external exposures. The airborne release fraction from dropped metal or soil removal is relatively low. Very minimal amounts of soil are expected to require remediation and removal from the site.

A fire is another possible source of an uncontrolled release of radioactive materials. However, the majority of the combustibles that will be present onsite will be clean materials. Potentially contaminated combustibles may include dry active waste such as personal protective clothing, 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 will be no fissile materials located on the site that could result in a criticality incident because all nuclear fuel has been removed.

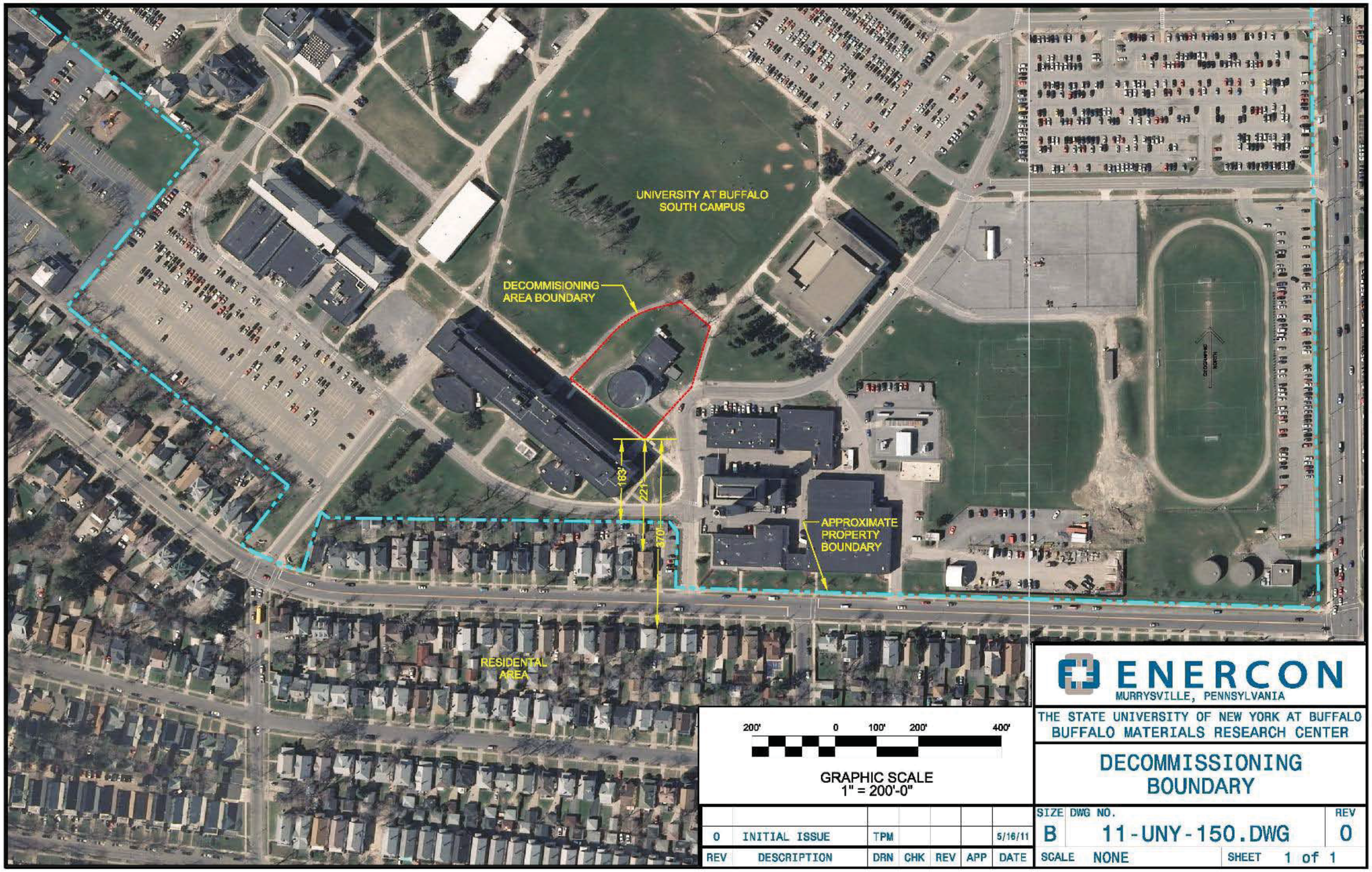
5.13 Waste Management Impacts

The BMRC decommissioning will generate soil LLRW, mixed waste (i.e., contaminated lead and cadmium), hazardous waste (i.e. ACM, oils and fluid drained from equipment). These wastes will be handled, stored and disposed of according to applicable state and federal regulations. The DOC will coordinate with the waste disposal site(s) regarding the site's waste acceptance criteria and pre-shipment processing requirements.

Waste processing may include volume reduction through compaction or segmentation, neutralization, stabilization or solidification. Due to the limited size of the facility and work area, concrete rubblization beyond that required for demolition is not expected to occur on-site. Complying with written procedures, standard work practices and operating with the limits of the NRC license will ensure safe waste processing. The decisions as to the type and degree of waste processing will primarily be based on economics that weigh the costs of additional handling and processing compared to transferring the material off-site for treatment and/or disposal.

After the characterization surveys and sampling are complete, wastes will be wrapped, bagged and/or containerized and staged in the appropriate designated area. Items and containers will be properly labeled as Radioactive Material and the label will indicate the external dose rate from the material. Radioactive wastes will be stored in properly secured radioactive materials storage areas. Waste material logs will be maintained for materials placed in disposal and shipping containers.

Figure 5: Decommissioning Boundary



5.13.1 Radiological Impacts

Prior to disposal, all waste streams will be properly characterized according to the requirements of the disposal facility. This characterization will include quantification of primary radionuclides of concern as well as hard-to-detect radionuclides. Additionally, those radionuclides that have specific limits for Class A waste will be directly quantified or estimated based on ratios to concentrations of other radionuclides.

All waste will be shipped to an acceptable waste disposal site in accordance with applicable NRC, DOT, and NYS regulations regarding waste packaging, labeling and placarding. Each LLRW shipment will be accompanied by a shipping manifest as specified in Section I of Appendix F to 10 CFR 20, “Requirements for Low-Level Waste Transfer for Disposal at Land Facilities and Manifests.” The waste will be manifested consistent with its classification. Only licensed transporters will be used to transport wastes from the BMRC facility.

Mixed wastes may be shipped to an NRC licensed processing facility or directly to an NRC licensed land disposal facility depending on the nature of the waste and the treatment options available.

The consequence levels discussed in this section are described in more detail in the DOE Standard DOE-STD-1120-2005, “Integration of Environment, Safety and Health Into Facility Disposition Activities” (DOE 2005)

6.0 MITIGATION MEASURES

The following table identifies the areas that may be impacted by the decommissioning of the BMRC and the mitigation measures:

- Land Use
- Transportation
- Geology and Soils
- Meteorology, Climatology and Air Quality
- Noise
- Public and Occupational Health
- Waste Management

Mitigation measures that will be used to reduce potential impacts in these areas will include:

Affected Environment	Mitigation Measures
Land Use	Following strict site demolition and restoration procedures.
Transportation	Proper loading and shipping of containers and/or materials going off-site.
Geology and Soils	Proper removal and disposal of soils impacted above regulatory levels.
Meteorology, Climatology and Air Quality	Build negative air pressure tents (when feasible) around work areas so that potential air contaminants are contained within work areas. Using water sprays to minimize the amount of dust generated. Following applicable construction and abatement procedures.
Noise	Limiting the time of increased noises to daylight hours, with a schedule that takes account of the UB academic calendar.
Public and Occupational Health	Constant radiological and air monitoring during decommissioning activities will ensure that the public and the occupational workers are safe.
Waste Management	Proper decontamination and segregation of waste will ensure that wastes are disposed of properly.

7.0 ENVIRONMENTAL MONITORING AND MEASUREMENT PROGRAMS

This section describes the environmental monitoring and measurement programs that will be used during the decommissioning of the BMRC.

7.1 Radiological Monitoring

Frequent radiological monitoring will be performed during the BMRC decommissioning activities. CAMs will be used to monitor the air during selected decommissioning activities. Survey stations will be set up to monitor personnel and equipment. All packaged waste materials will be surveyed prior to leaving the facility for disposal.

7.2 Physiochemical Monitoring

Monitoring for chemical/hazardous constituents such as asbestos, lead, cadmium and mercury will be performed as needed during the BMRC decommissioning activities. A thorough asbestos inspection will be performed prior to decommissioning activities and all materials suspect to contain asbestos will be characterized. Asbestos-containing materials will be removed by a trained and licensed New York Department of Labor Asbestos Abatement Contractor prior to the eventual demolition of the facility.

7.3 Ecological Monitoring

No ecological monitoring is required during the decommissioning of the BMRC.

8.0 COST BENEFIT ANALYSIS

Detailed cost estimates for the completion of the BMRC decommissioning activities are included in the Decommissioning Plan. The following are some of the benefits associated with completion of the decommissioning of the BMRC.

- Increased public health and safety;
- Decreased operating and maintenance costs;
- Free release the site for reuse;
- Termination of the NRC licenses.

9.0 SUMMARY OF ENVIRONMENTAL CONSEQUENCES

The following is a summary of the environmental consequences related to the implementation of the proposed action:

Affected Environment	Environmental Consequences
Land Use	Short Term Impacts – Construction and demolition of the BMRC buildings, excavation and removal of all associated tanks and piping and removal of all subsurface materials. Following demolition conducting site restoration to usable space.
Transportation	Short Term Impacts – Hazardous, solid, and radioactive waste transportation
Geology and Soils	Soils will be impacted by removal of structures and tanks. Fill will be introduced to the site as necessary and the area will be restored to grade.
Water Resources	No Impacts
Ecological Resources	No Impacts
Meteorology, Climatology, and Air Quality	Short Term Impacts – Possible air quality issues associated with construction activities.
Noise	Short Term Impact – Noise from heavy equipment and demolition of the structures.
Cultural and Historic Resources	No Impacts
Socioeconomic	No Impacts
Public and Occupational Health	Short Term Impact – Possible radiological exposure, possible asbestos fiber exposure, and dust above background.
Waste Management	Short Term Impact – Disposal of solid, hazardous, and radioactive waste.

There will be no adverse or long-term environmental impacts as a result of the implementation of the proposed action.

10.0 REFERENCES

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