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University of Michigan

Decommissioning Plan for the Ford Nuclear Reactor

Rev.	Effective Date	Revision Description	Minor Change	50.59 Implemented Change	License Amendment
00	23 June 2004	Original Issue			X
01	05 January 2006	Clarified the decommissioning cost estimate includes cost of final status survey. Added legend to organizational chart. Updated list of potential radionuclides to identify activation as source for europiums, Added specific reference to 10CFR19.12 to general employee training, Provided more detail on the generation of the dose estimate including addressing internal dose, individual maximums, and specifically state no radionuclides with high specific internal doses potentials, Clarified release of fines, sand, silts, etc., Amplified that this is a license amendment and that accidents from decommissioning and are bounded by operational accidents already in licensing basis, Changed Technical Specifications change requese to Section 6 through out to be consistent with the current Technical Specifications, Claified the Reactor Manager had approval authority for the QA and performance elements of the Final Status Survey not specifically promulgated to the review committee, and other minor wording, spelling, and grammatical corrections.			X

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ACRONYMS

AEC	airborne effluent concentration
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
Bq	Becquerel
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
Ci	curies
cpm	counts per minute
DAC	derived air concentration
DAW	dry-active waste
DCGL	derived concentration guideline level
D & D	decontamination and decommissioning
DDE	Deep-Dose Equivalent
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
dpm	disintegrations per minute
DRC	Decommissioning Review Committee
DQO	data quality objective
EDE	Eye Dose Equivalent
EPA	U.S. Environmental Protection Agency
FNR	Ford Nuclear Reactor
FSAR	final safety analysis report
FSS	final status survey
HAZWOPER	Hazardous Waste Operations and Emergency Responses
HEPA	high-efficiency particulate air
HIC	high integrity containers
HPGe	high purity germanium
HSSI	Heavy Section Steel Irradiation
HVAC	heating, ventilation, and air conditioning
JHA	job hazard analysis
LBGR	Lower Bound of the Gray Region
LDE	Lens of the Eye Dose Equivalent
LLRW	low-level radioactive waste
LSA	low specific activity
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
mCi	millicuries
MCL	Maximum Contaminant Levels
MDEQ	Michigan Department of Environmental Quality
MeV	megaelectron volts
MIOSHA	Michigan Occupational Safety and Health Act
ml	milliliters
MMPP	Michigan Memorial Phoenix Project
MPC	Maximum Permissible Concentration
mrem	millirem

ACRONYMS (cont.)

mSv	millisievert
MW	megawatts
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NUREG	NRC technical report designation (Nuclear Regulatory Commission)
NVLAP	National Voluntary Laboratory Accreditation Program
OSHA	Occupational Safety and Health Act
pCi	picocuries
PML	Phoenix Memorial Laboratory
PPE	personal protective equipment
QA	quality assurance
rem	roentgen-equivalent man
RIFLS	Reactor Irradiation Facility for Large Samples
RSO	Radiation Safety Officer
RWP	radiation work permit
SDE	Shallow Dose Equivalent
TEDE	Total Effective Dose Equivalent
TODE	Total Organ Dose Equivalent
U-Alx	uranium aluminide
UM	University of Michigan
UVa	University of Virginia
WRS	Wilcoxon Rank Sum

1.0 Summary of Plan

1.1 Introduction

1.1.1 Overview

The Ford Nuclear Reactor (FNR) is operated by the Michigan Memorial Phoenix Project (MMPP) of the University of Michigan (UM) in Ann Arbor, Michigan. The facility is located at 2301 Bonisteel Boulevard, Ann Arbor, Michigan 48109. Figure 1-1 shows the geographical location of the FNR facility. The Project was established in 1948 and reactor operations began in 1957. The reactor is a non-power generating; open pool reactor with a heterogeneous core composed of aluminum and enriched Uranium-235. The reactor is licensed (Docket 50-2, License R-28) to operate at a power level of 2 megawatts (MW) with a term of licensed operation to July 29, 2005. The objective of this decommissioning plan is to perform decontamination of the FNR and remove radiologically contaminated grounds and materials to obtain release to unrestricted use from the U.S. Nuclear Regulatory Commission (NRC) and be granted termination of the NRC license as allowed by Title 10, Code of Federal Regulations, Part 20, Section 1401 (10 CFR 20.1401). This decommissioning plan was prepared in accordance with Section 17 of NUREG-1537-PT-1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (NRC, 1996), and NUREG-1757, *Consolidated NMSS Decommissioning Guidance* (NRC, 2003).

1.1.2 Decommissioning Plan Synopsis

This decommissioning plan provides guidance on the general process and methods that will be used to decontaminate or remove radioactive materials, equipment, components, soil, and other media from the FNR in a safe, orderly manner, to prevent undue radiation exposures to the facility staff, public, and environment, in order to obtain release to unrestricted use from the NRC.

FIGURE 1-1, AREA MAP

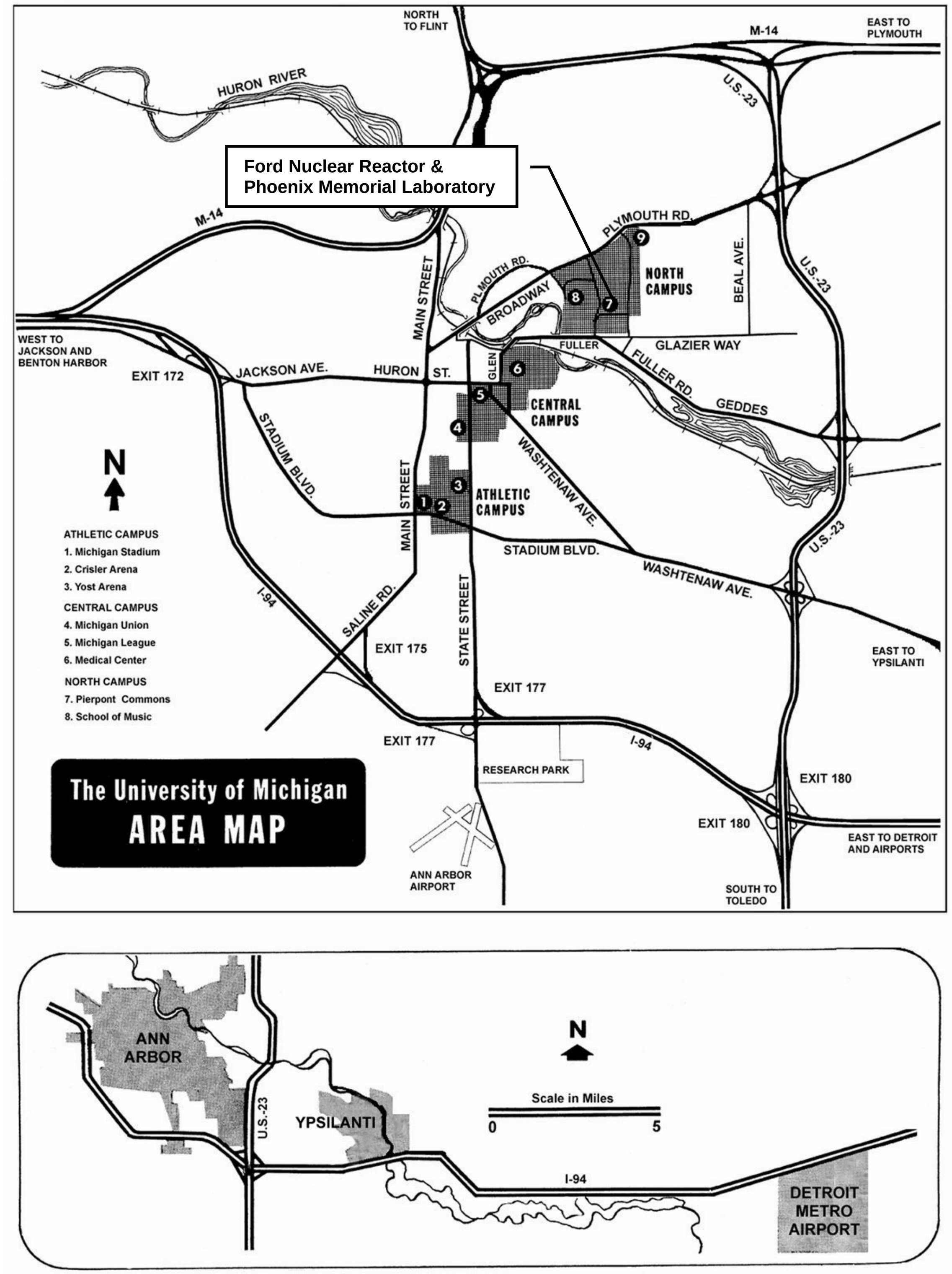


FIGURE 1-2, UM NORTH CAMPUS

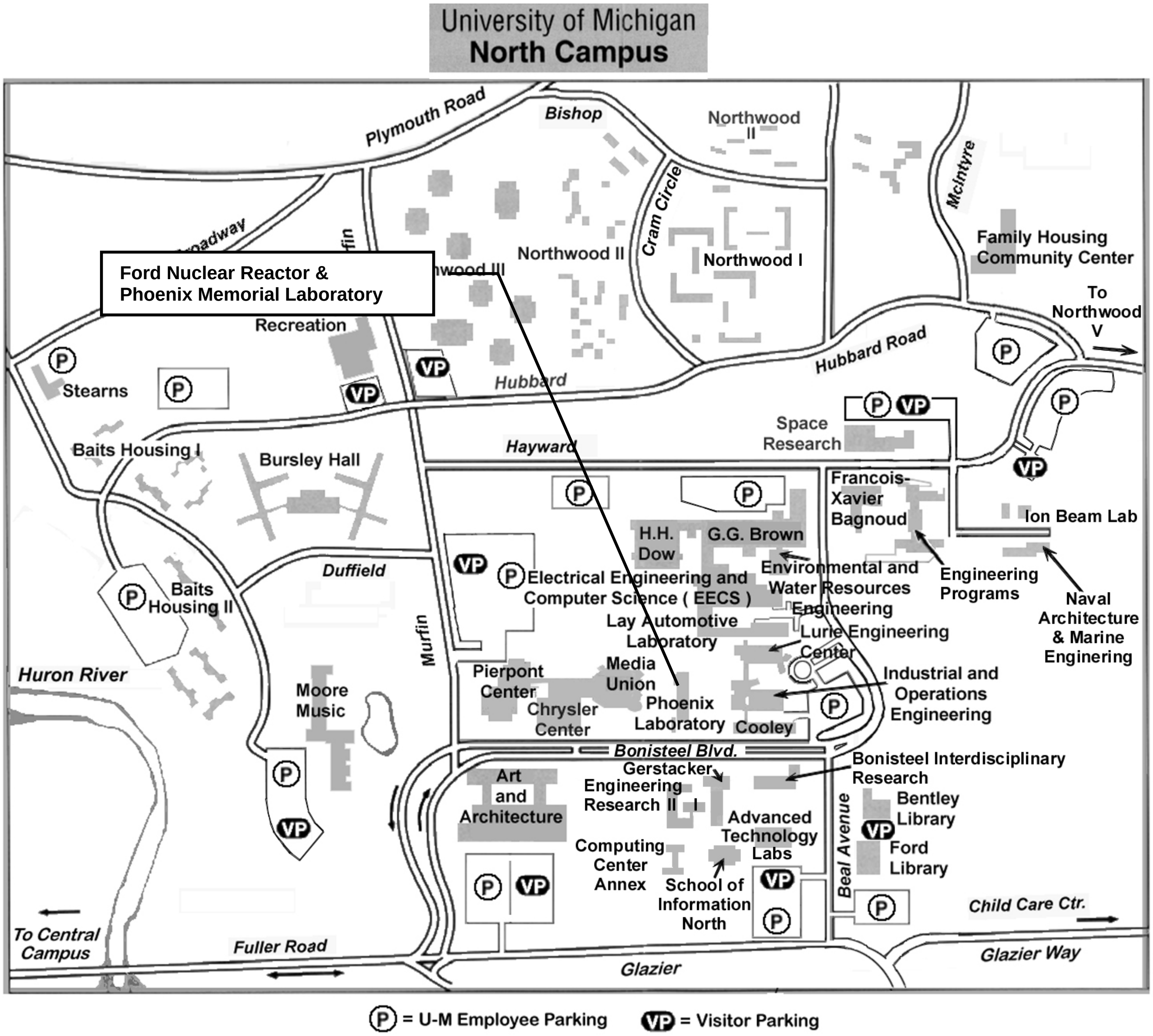
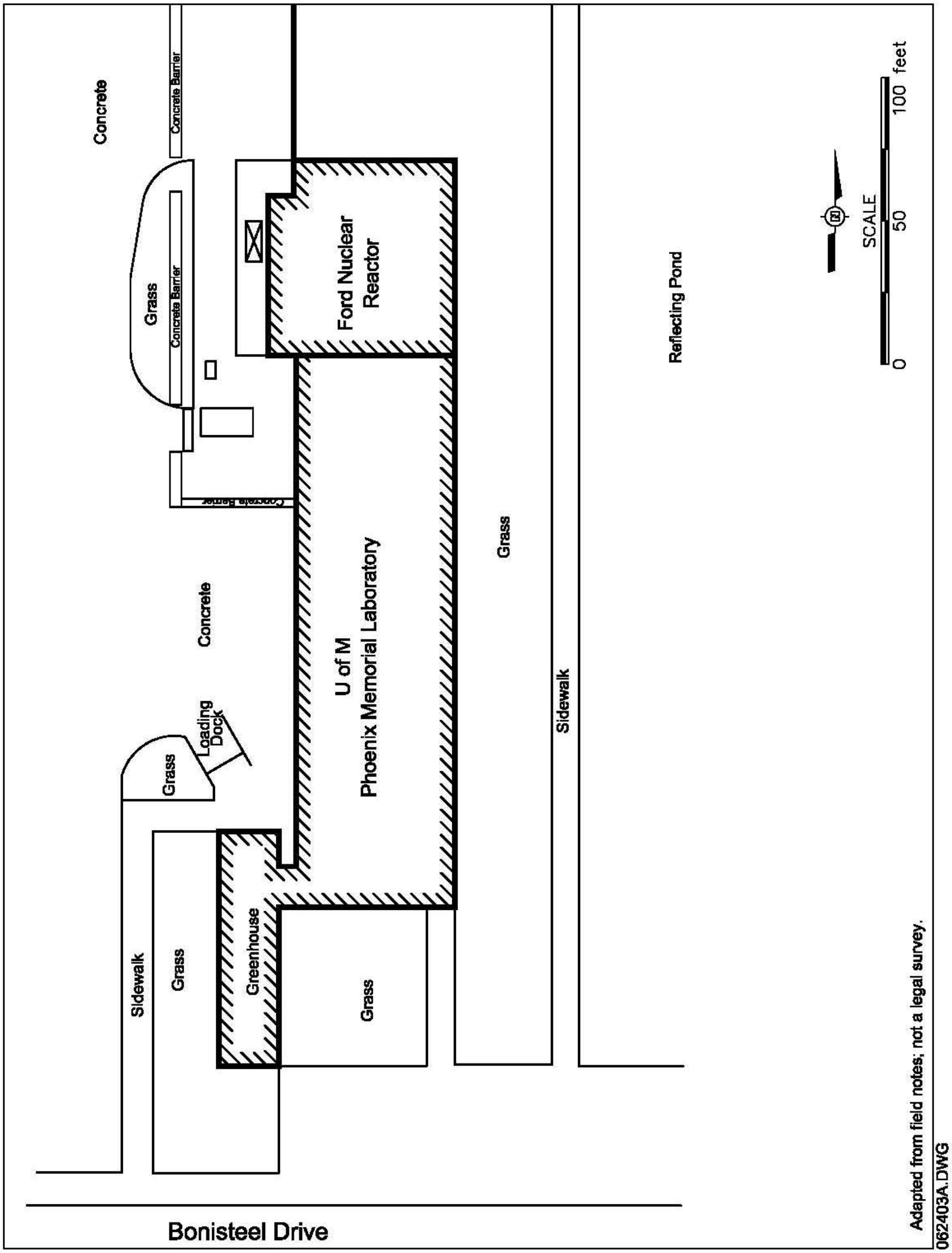


FIGURE 1-3, FORD NUCLEAR REACTOR SITE PLAN



1.2 Background

1.2.1 General

The FNR is operated by the MMPP of the UM. MMPP was established in 1948 as a memorial to students and alumni of the UM who served—and the 588 who died—in World War II. MMPP's purpose has been to encourage and support research on the peaceful uses of nuclear energy and its social implications. In 1954 the project completed the Phoenix Memorial Laboratory (PML) and the construction of FNR began in 1956. In 1957, the FNR went critical and has successfully maintained MMPP's original charter for the last 46 years.

The FNR and PML are located on the North Campus of the UM in Ann Arbor, Michigan (Figure 1-2). The North Campus of UM is a tract of 900 acres located approximately 1.25 miles northeast of the central business district of Ann Arbor (UM, 1985). Ann Arbor has a permanent population of about 114,000 (2000 Census) and a transient student population of approximately 38,000.

The reactor is a 2-MW, open pool reactor facility. The heterogeneous core is composed of aluminum and enriched uranium-235. It is suspended 20 feet beneath the surface of the pool from a moveable bridge, which is mounted on rails that lie on top of the concrete tank. The reactor is licensed to operate at a power level of 2 MW and operated under Operating License R-28, Docket 50-2. The FNR generated no electricity and was used by students, faculty, and staff of the UM and non-University institutions and entities for research, experiments, and classes. The operation of the FNR provided major assistance to a wide variety of research and educational programs. The reactor provided neutron irradiation services and neutron beam port experimental facilities for use by faculty, students, and researchers from the UM, other universities, and industrial research organizations. Reactor staff members taught classes related to nuclear reactors (and the FNR in particular) and assisted in reactor-related laboratories (UM, 1999). Additional usage included neutron activation analysis, isotope preparation, radiochemical production, gamma irradiation services, neutron radiography, testing services, and training programs.

The UM has decided to permanently shut down the reactor. It is de-fueled, and is being maintained in accordance with the license and technical specifications.

1.2.2 Phoenix Memorial Laboratory

PML, although conjoined with FNR as indicated in Figure 1-4 through Figure 1-7, is not part of the decommissioning plan. PML is a four story, reinforced concrete building supported by an integral post-and-beam structure that contains offices, wet and dry laboratories, a machine shop, two hot cells, a cobalt irradiator, and various equipment and storage rooms.

FIGURE 1-4, PHOENIX MEMORIAL LABORATORY BASEMENT - SECTION

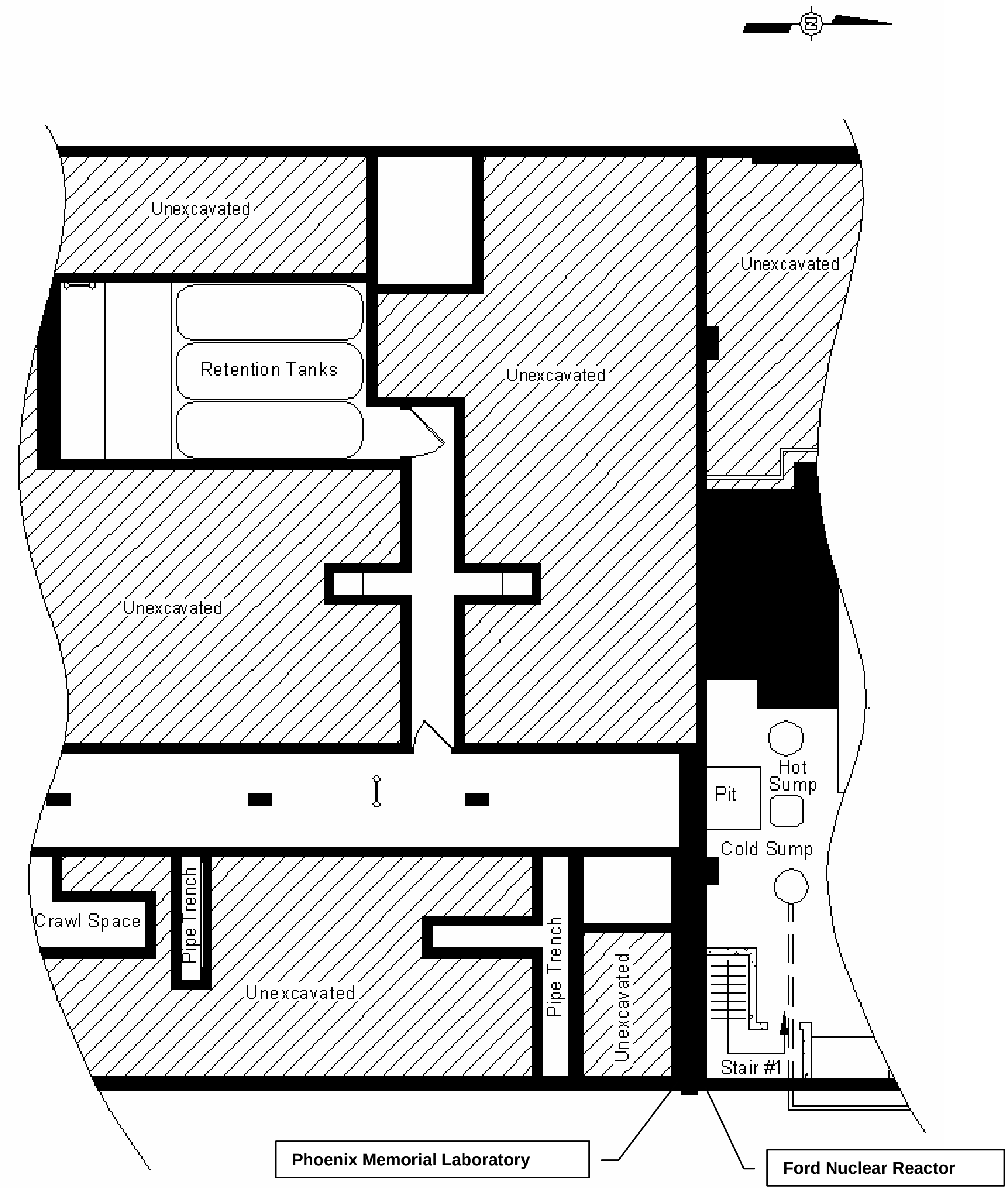


FIGURE 1-5, PHOENIX MEMORIAL LABORATORY FIRST FLOOR - SECTION

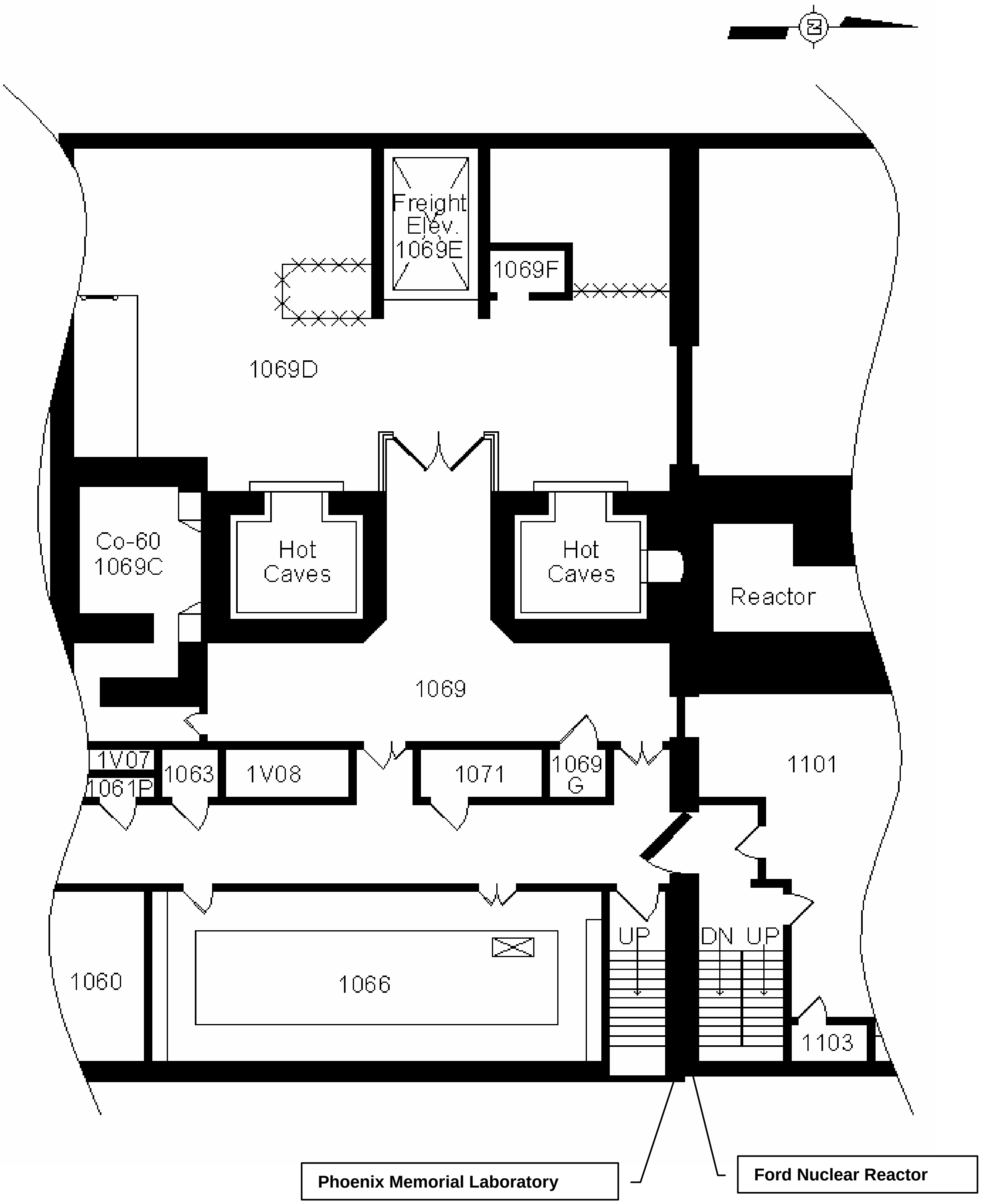


FIGURE 1-6, PHOENIX MEMORIAL LABORATORY 2ND FLOOR - SECTION

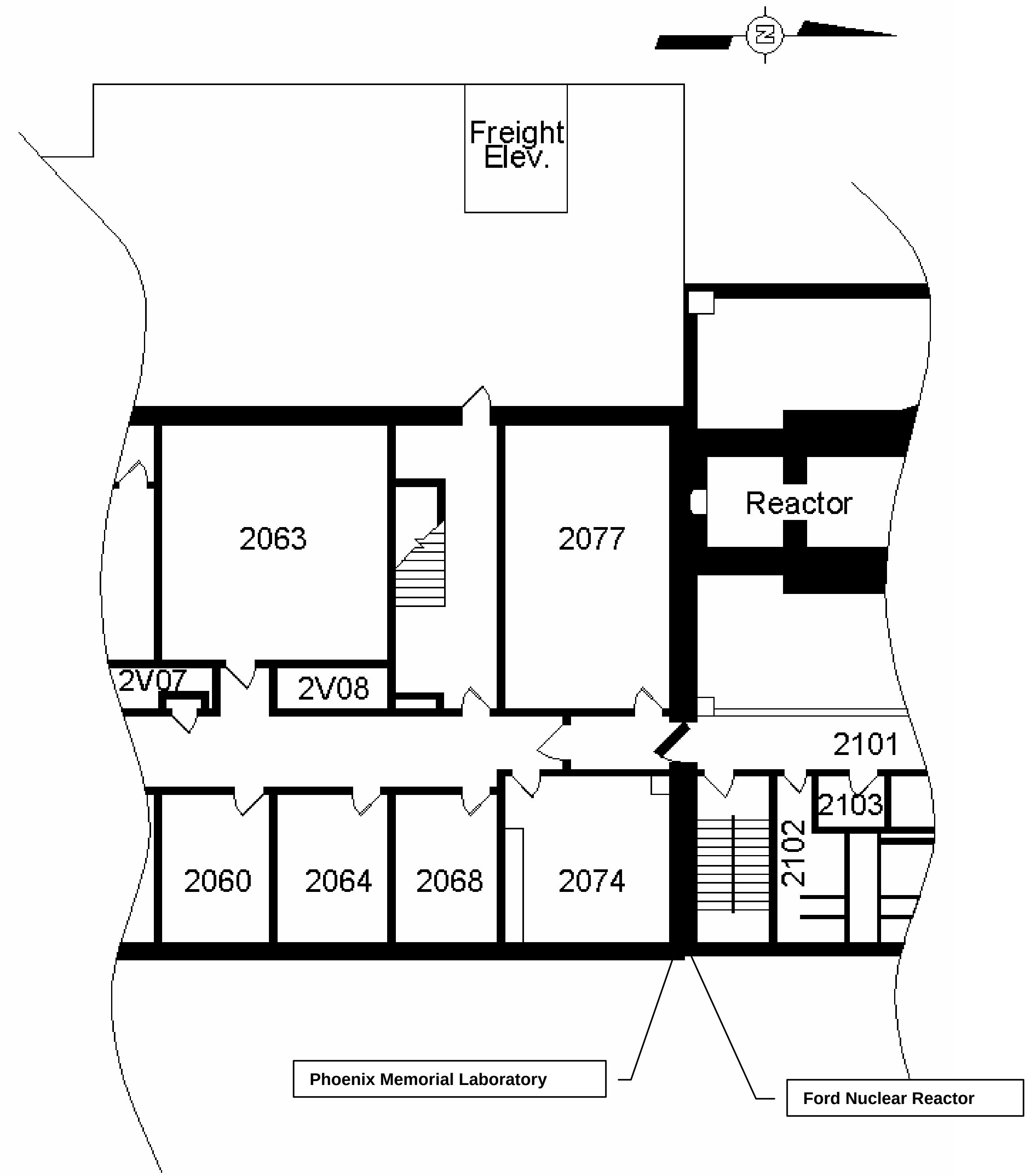
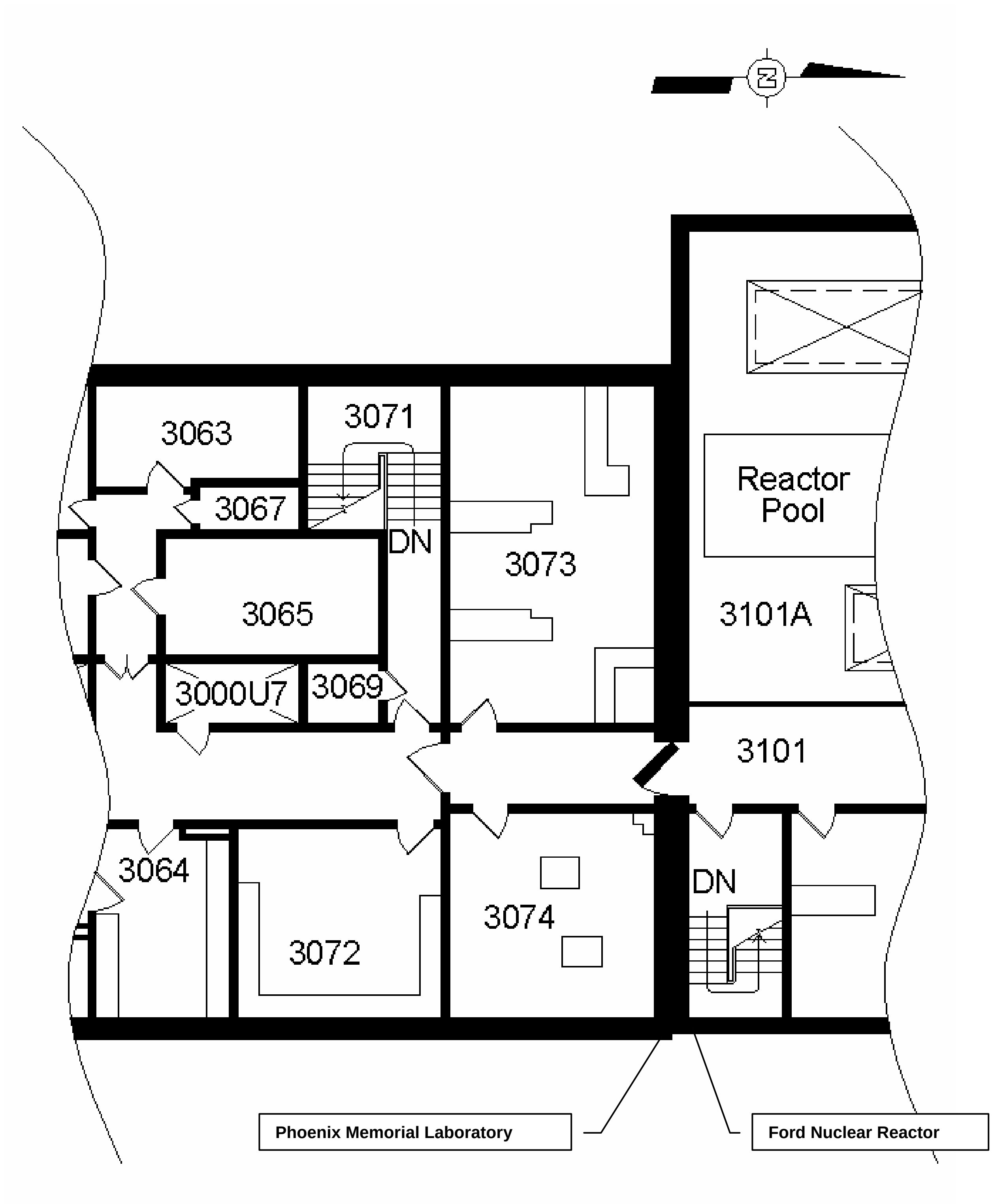


FIGURE 1-7, PHOENIX MEMORIAL LABORATORY THIRD FLOOR –SECTION



Activities involving radioactive materials within PML are covered by the following UM licenses or registrations:

- Operating License R-28, Ford Nuclear Reactor, Docket 50-2, issued by U.S. Nuclear Regulatory Commission, Washington, D.C.
- Broad Scope Materials License 21-00215-04, Docket 030-01988, issued by U.S. Nuclear Regulatory Commission, Lisle, Illinois.
- Special Nuclear Materials License SNM-179, Docket 070-192/070-01734 issued by U.S. Nuclear Regulatory Commission, Lisle, Illinois.
- Cobalt Irradiator Facility License 21-00215-06, Docket 030-06958, issued by U.S. Nuclear regulatory Commission, Lisle, Illinois.
- Radioactive Material Registration 97-10, issued by the Michigan Department of Environmental Quality (MDEQ), Drinking Water and Radiological Protection Division, Lansing, Michigan.

PML provides a variety of services to FNR through several systems or interconnects and receives non-radioactive, demineralized water from FNR, however, the two structures operate independently.

1.2.2.1 Stack 2 Exhaust System

PML has an exhaust system that draws from various rooms or facilities located on the west side of the first floor (both hot cells, the accelerator room, the cobalt irradiator, the retention tank area, and the processing and storage area behind the hot cells). This system also draws some exhaust from FNR via a bolted flange to two circular ducts that are sleeved through the north wall of PML and the south wall of FNR. The Stack 2 exhaust system discharges through the roof via a stack located on the northern end of PML.

1.2.2.2 Radioactive Liquid Collection and Retention Tank System

PML has a system to collect radioactive liquids in three radioactive liquid retention tanks in the basement of PML. Sinks and floor drains in areas of PML where radioactive materials are used are connected to these retention tanks. These activities in PML are licensed under a variety of radioactive materials licenses or registrations (in addition to those held under the reactor license). The FNR hot and cold sumps also pump radioactive liquids through a single common line to the free standing retention tanks in the basement of PML. The FNR hot and cold sumps collect radioactive liquids from the reactor pool, all FNR floor drains, the laboratory sink on the FNR third floor, the first and third floor FNR janitor's closets, and various other equipment drains in FNR.

Water collected in the retention tanks is typically used to refill the reactor pool or fill the transfer chute between the reactor pool and the north hot cell in PML after being treated by a series of filters, and demineralizer column(s).

1.2.2.3 Waste Handling

Solid, liquid and gaseous radioactive waste generated in PML, FNR and throughout the UM under a variety of radioactive materials licenses or registrations (in addition to the reactor license) is collected, processed and stored in PML. The majority of the radioactive waste is directly picked up for processing or disposal by licensed waste handlers directly from PML. Some lower activity radioactive waste is picked up for processing and subsequent disposal by

licensed UM waste handlers. All radioactive materials are processed or disposed of as authorized by applicable federal or state authority.

1.2.2.4 Reactor Building Service Air

Reactor building service air is provided by two compressors located on the second floor of PML. These compressors can also be cross connected to the PML building service air system when the need arises.

1.2.2.5 Demineralized Water

Demineralized water from demineralizer columns connected to a potable water supply in FNR is supplied to PML to maintain the demineralized water system for the labs in PML.

1.2.2.6 Electrical Distribution System

The primary electrical supply to the FNR is through three breakers in the distribution panels located in an equipment room on the first floor of PML. One breaker supplies the motor control center located in the reactor building basement, a second breaker supplies the instrument console located in the control room, and a third breaker provides the normal supply to an automatic bus transfer that powers a 208/120 Vac distribution panel on the first floor of the reactor building, commonly referred to as the "Y" panel. The "Y" panel feeds loads in the reactor building such as some radiation monitoring equipment and a limited amount of building lighting.

In the event of a loss of power to the distribution panel or a loss of power to the motor control center on the second floor of PML, a standby generator provides power to the motor control center on the second floor through an automatic bus transfer. This motor control center powers the PML Stack 2 exhaust fan, the sample pump for the Stack 2 air particulate monitor and provides the alternate power to the automatic bus transfer for the "Y" panel described above. A backup air compressor for the reactor building service air system and a fan for cooling the room containing the generator are powered directly from the generator.

1.2.2.7 Steam and Condensate

Steam is supplied from PML to service the heating coil in the reactor building ventilation system, the heat exchanger for the FNR perimeter heating system, a line to the reactor primary coolant heat exchanger, and capped lines on the third floor of the reactor building near the reactor pool. Condensate from the steam lines and these components is collected in the reactor building basement and pumped back to PML.

1.2.3 Ford Nuclear Reactor (Figures 1-8 through 1-12)

The reactor building is a windowless, four story, reinforced concrete building supported by an integral post-and-beam structure. The 12 inch exterior walls of the reactor building are integral with the footings and foundation mats. The building is approximately 70 feet wide. 68 feet long, and 69 feet high with 55 feet and 46 feet of the structure exposed above grade on the east and west respectively. The reactor building is immediately north of PML and is its own structure, separated from PML by a ½ inch flexcell joint (SH&G, 1955).

The building was designed to restrict leakage and is equipped with a general ventilation system that provides the primary heating for the building and exhausts through a stack on the roof. Supply air and primary exhaust air are through air handling equipment in Room 2111, where all supply air enters the building and most exhaust air is pushed to the stack on the roof. Both

the supply and exhaust can be isolated for confinement with interconnected, air operated isolation dampers. A second exhaust system connected to PML Stack 2 (see Section 1.2.2.1) draws from a trunk in the basement that collects exhaust from the blowers (2) for the pneumatic transfer system, draws from trunks (6) on the first floor located around the pool, and the storage ports in the west wall of the first floor. This exhaust duct can be isolated for confinement by an air operated isolation damper. A third exhaust system, also connected to PML Stack 2, draws on the fume hood located in Room 3104. This exhaust duct can be isolated for confinement by an air operated isolation damper.

There are a limited number of openings in the reactor building:

Doors or Accesses

- One door on each of the first, second, and third floors
- A door for the first floor to the hot cell operating face
- A freight door for the first floor to the rear of the hot cells and the elevator
- A door to the cooling tower area on the fourth floor
- A hatch in the ceiling of the first floor to outside the facility
- A door in the north stairwell between the first and second floors to outside the facility (sealed)

Ventilation Penetrations

- An exhaust duct connected to the roof stack
- A filtered duct from exhaust trunks in the basement and first floor and exhaust lines from the wall storage ports on the first floor to the Stack 2 exhaust in PML
- An intake duct from outside
- A filtered duct from a fume hood on the third floor to the Stack 2 exhaust in PML

Other Penetrations

- A capped line for the foundation's exterior drain tiles in the cold sump
- 8 penetrations in north basement wall originally for the pneumatic transfer system
- A line for transferring water collected in the hot and cold sumps to the retention tank system in PML
- A low pressure steam line supplying FNR from PML
- A 10″ water lock between the reactor pool and the north hot cell in PML
- Two conduits to the motor control center in the basement used to provide electrical power from PML
- A line supplying compressed air from PML
- A capped line used to provide gaseous nitrogen from PML
- A capped second line for transferring water collected in the hot sump to the retention tank system in PML
- A line supplying reclaimed water from the retention tank system in PML
- A conduit to the control room to provide electrical power from PML
- A condensate return line for the return of condensate from FNR
- A line supplying non-radioactive, demineralized water from FNR to PML

FIGURE 1-8, FORD NUCLEAR REACTOR BASEMENT

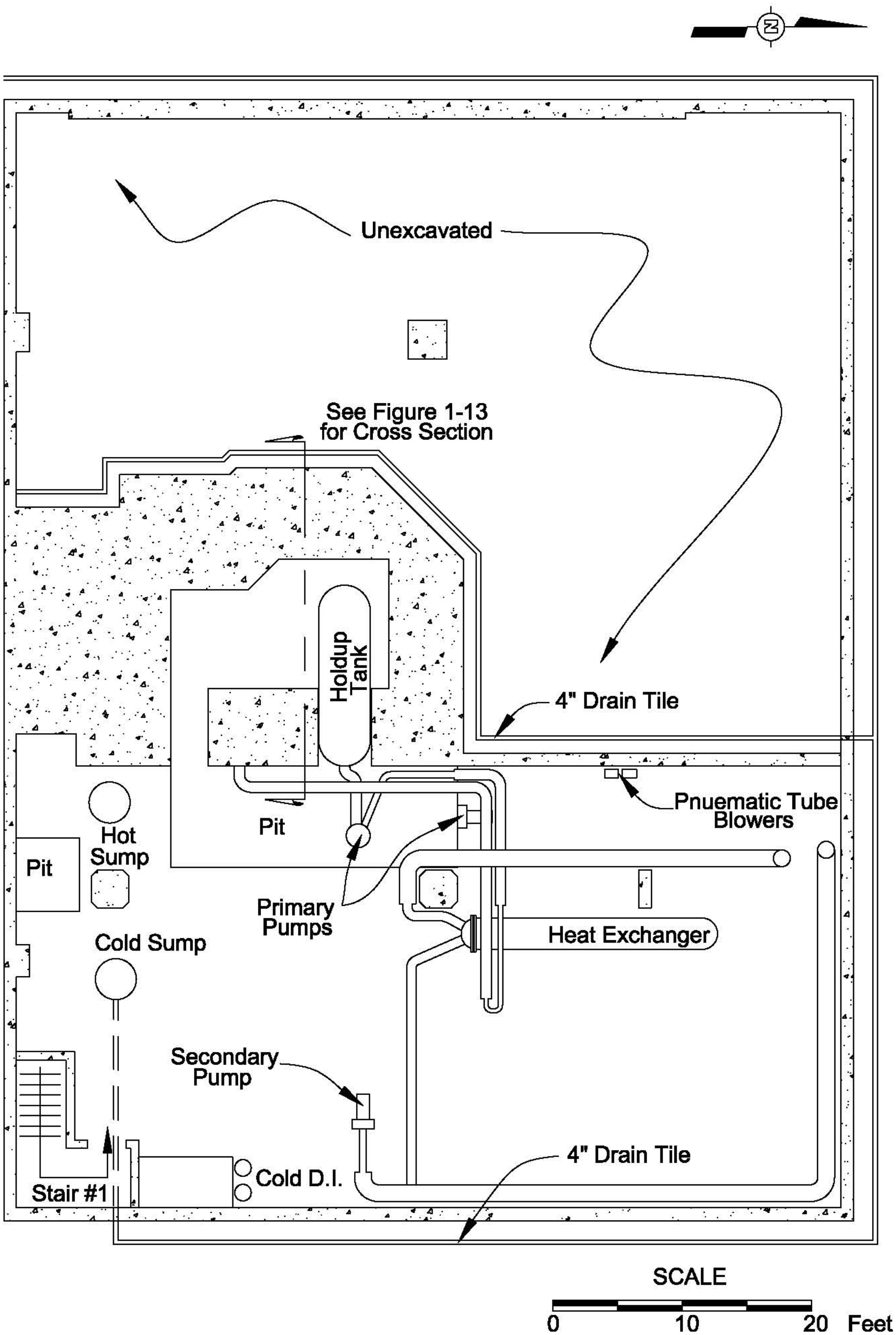


FIGURE 1-9, FORD NUCLEAR REACTOR FIRST FLOOR

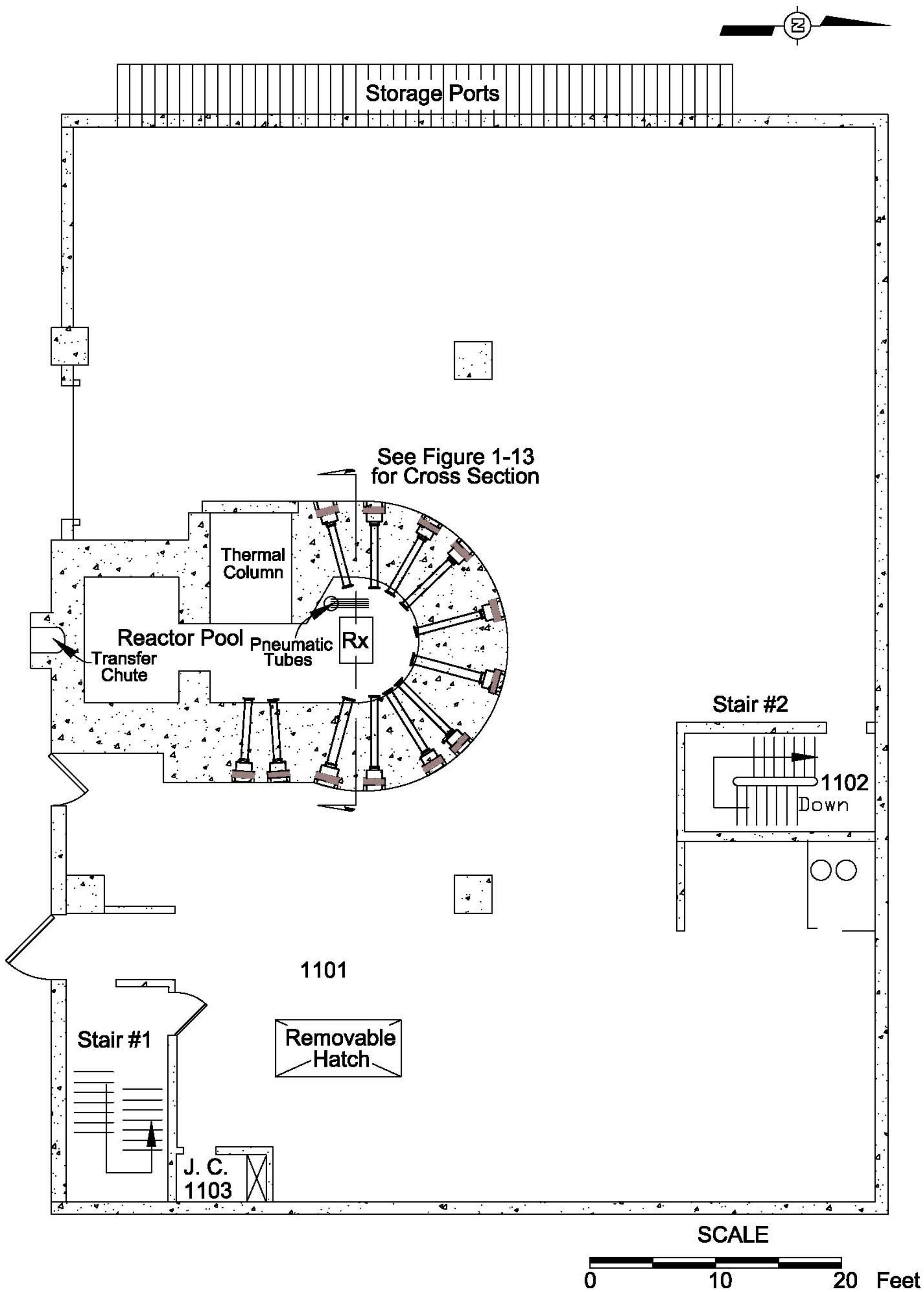


FIGURE 1-10, FORD NUCLEAR REACTOR 2ND FLOOR

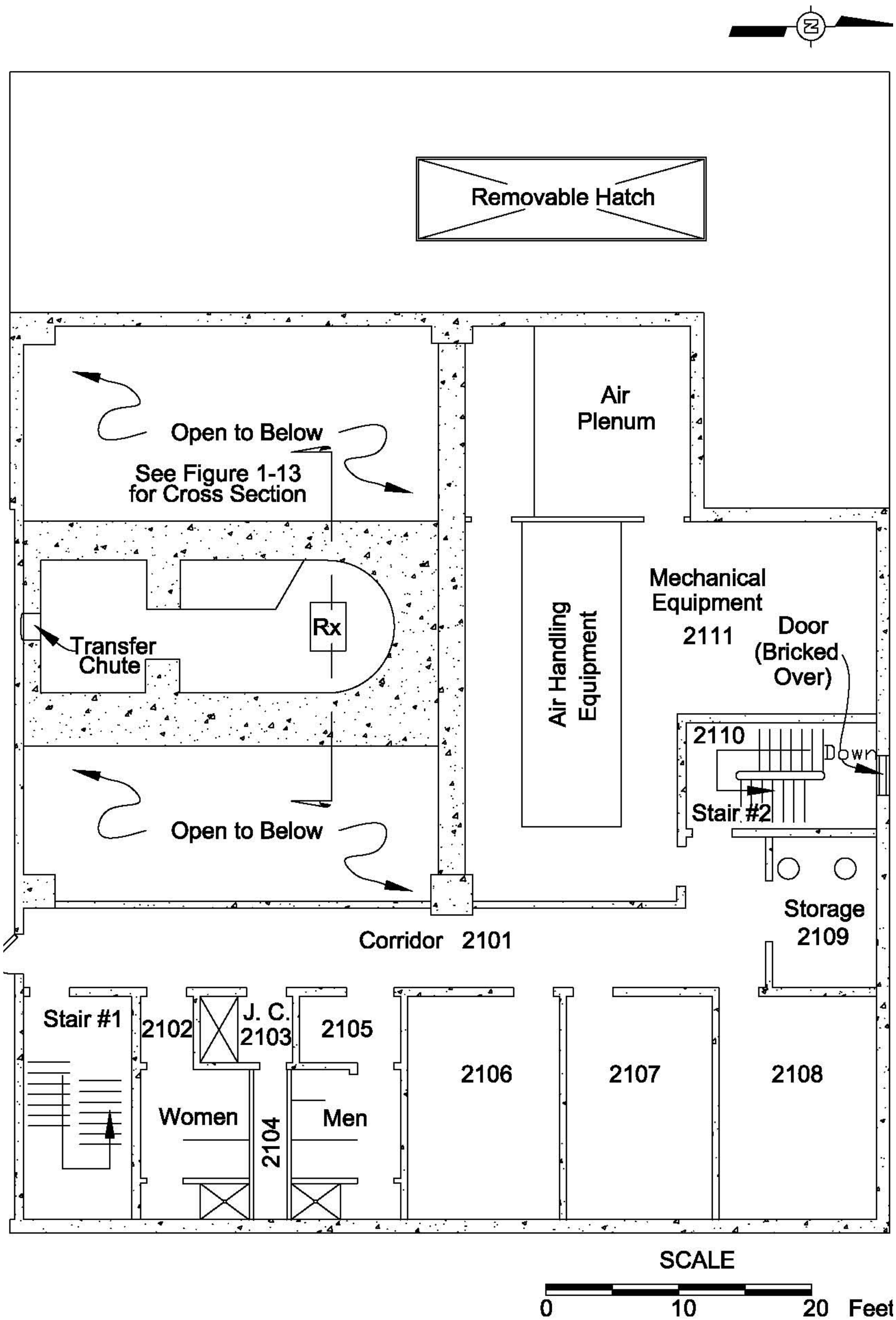


FIGURE 1-11, FORD NUCLEAR REACTOR THIRD FLOOR

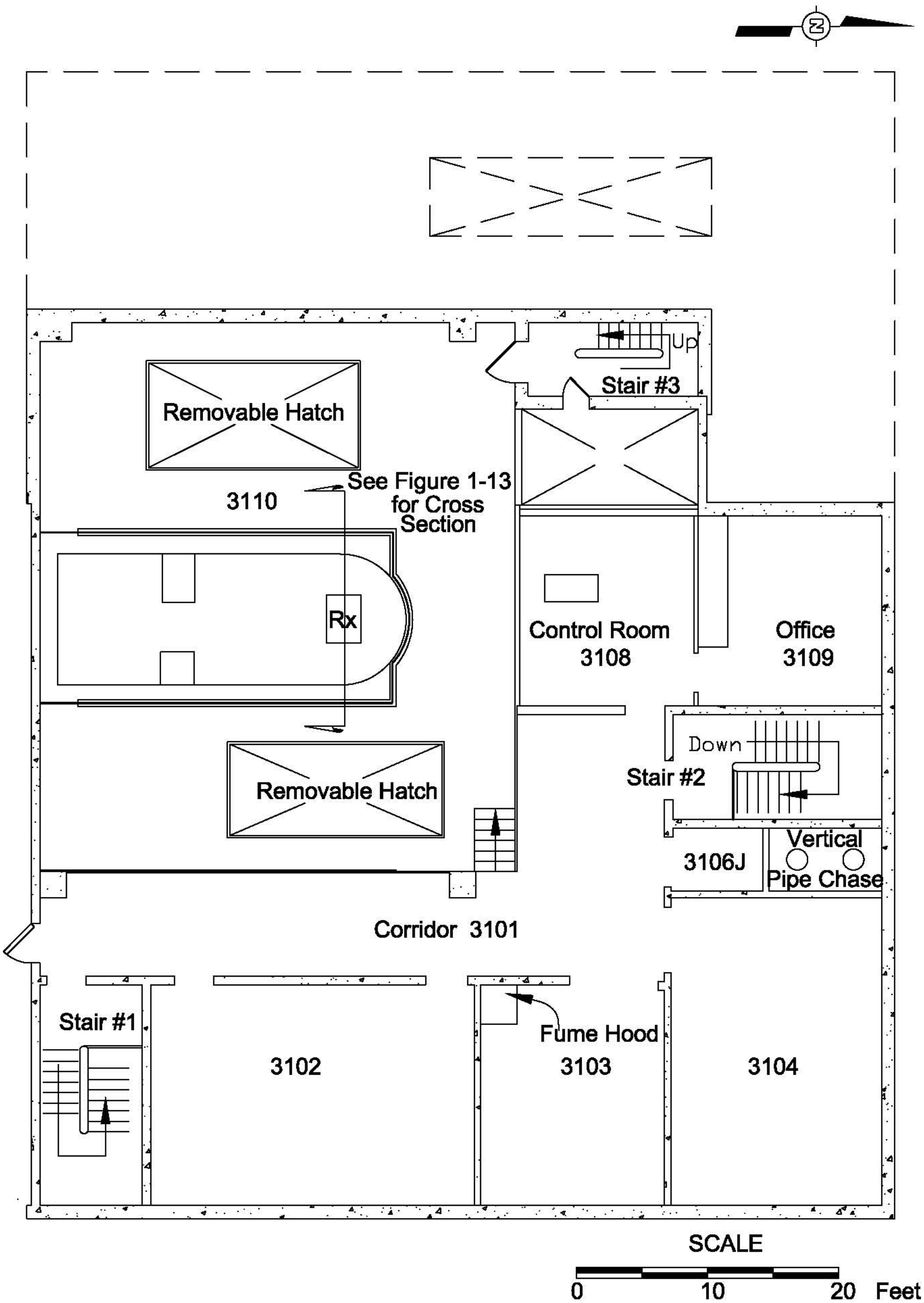
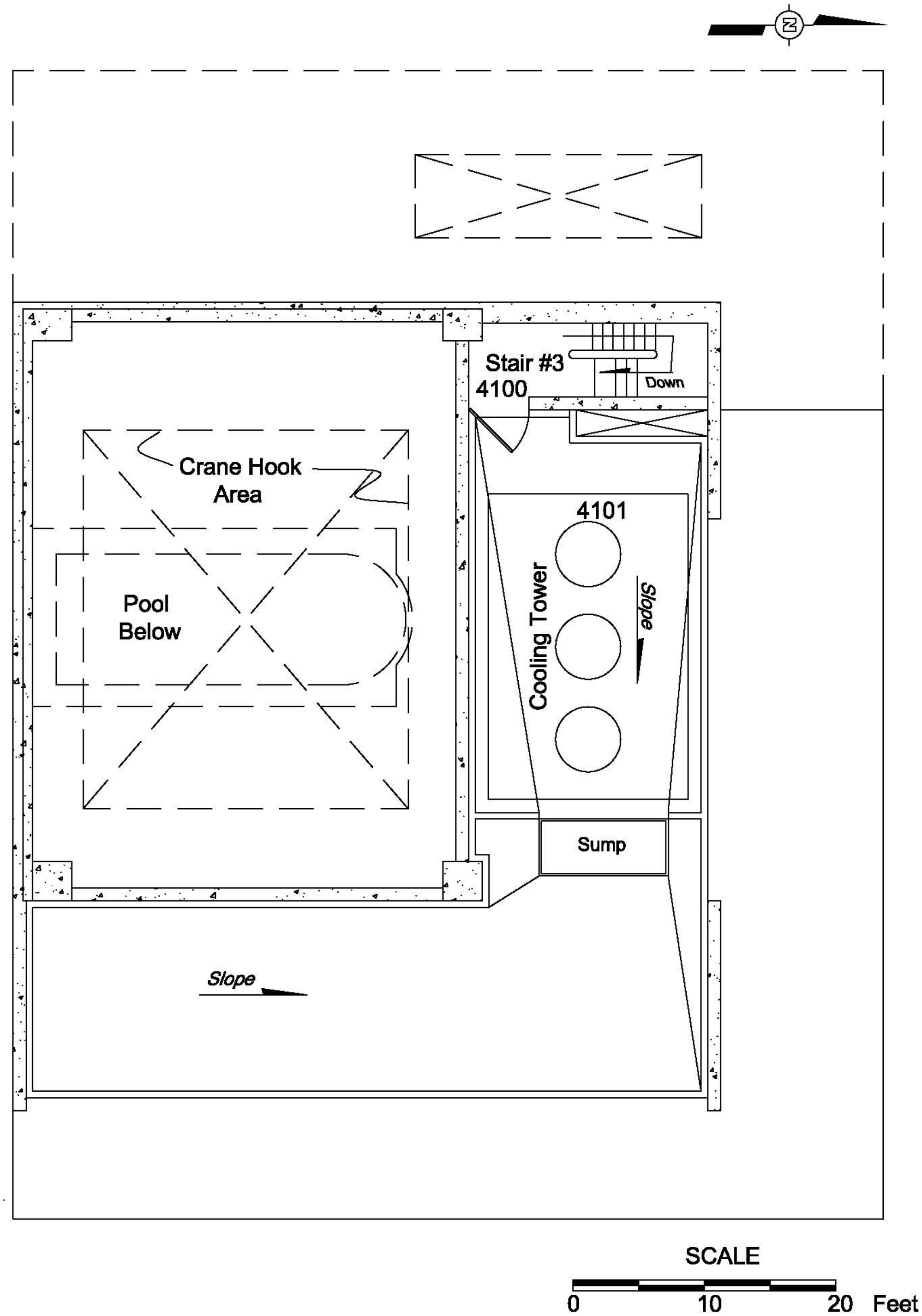


FIGURE 1-12, FORD NUCLEAR REACTOR 4TH FLOOR



The ceiling on the west side of the first floor contains four removable slabs which, when lifted out, provide a 21 ½ foot by 6 foot opening to the parking area. These removable slabs facilitate the removal of large items from the reactor building.
The third floor contains two sets of four removable plates (on the east and west sides of the reactor pool) which, when removed, provide a 16 foot by 7 foot opening to the first floor area. The first floor contains an opening that when removed provides a 10 foot by 4 ½ foot opening to the basement.

1.2.4 Reactor Pool

The U-shaped reactor pool is approximately 10 feet wide by 27 feet long by 27 feet deep which, when full, contains approximately 50,000 gallons of water. It can be divided into two sections through the insertion of a vertical gate into square grooves in two abutments, or islands, extending from the sides towards the center of the reactor pool.

The reactor pool is constructed of barytes concrete to a height of 16 feet; the remainder is ordinary concrete. The barytes concrete provides 6 ½ feet of biological shielding in the lower 16 feet of the reactor pool walls. The reactor pool is a free standing structure in the center of the building. The pool consists of three major pieces (see Figure 1-13). The pool floor and foundation was constructed early in the construction of the building. The foundation is 9 feet thick and contains an accessible void 2 feet below the pool floor and directly under the reactor for the pneumatic tube system and reactor primary cooling system penetrations. Since the reactor core was suspended in the water at a depth of 20 feet, the lower 16 feet of the pool wall is made from barytes concrete (242 cubic yards, density 219.2 pound per cubic foot, 3970 pounds per square inch) which is separate from and sits on the foundation. The upper 11 feet of the pool wall is separate from and rests on the lower 16 feet of the pool wall. Because of material procurement delays on embedded equipment, it was found expedient to enclose the reactor building completely *before* pouring the reactor pool, See Figure 1-14 (Luckow & Mesler 1957).

The pool interior is lined with white ceramic tile sealed with white cement. The tile protects the concrete from spalling, aids visibility, and is more easily decontaminated than a concrete surface.

Spent fuel was stored in racks along the walls of the reactor pool. The storage racks in the pool were used for depleted fuel, which was awaiting shipment to the U.S. Department of Energy's (DOE's) Savannah River Site, and for partially depleted fuel, which could be reused in the reactor core.

The reactor pool contains water that serves as a radiation shield for the remaining radioactive components in the reactor pool.

1.2.4.1 Bridge, Suspension Frame and Grid Plate

The bridge, a welded framework of carbon-steel structural members, supports the reactor suspension frame.

The suspension frame is an aluminum structural framework welded together to form a rigid assembly. This framework is suspended from the bridge and supports the grid plate, the ionization chambers, the fission chamber guides, and portions of the forced-cooling system.

FIGURE 1-13, EAST – WEST CROSS SECTION OF THE REACTOR POOL

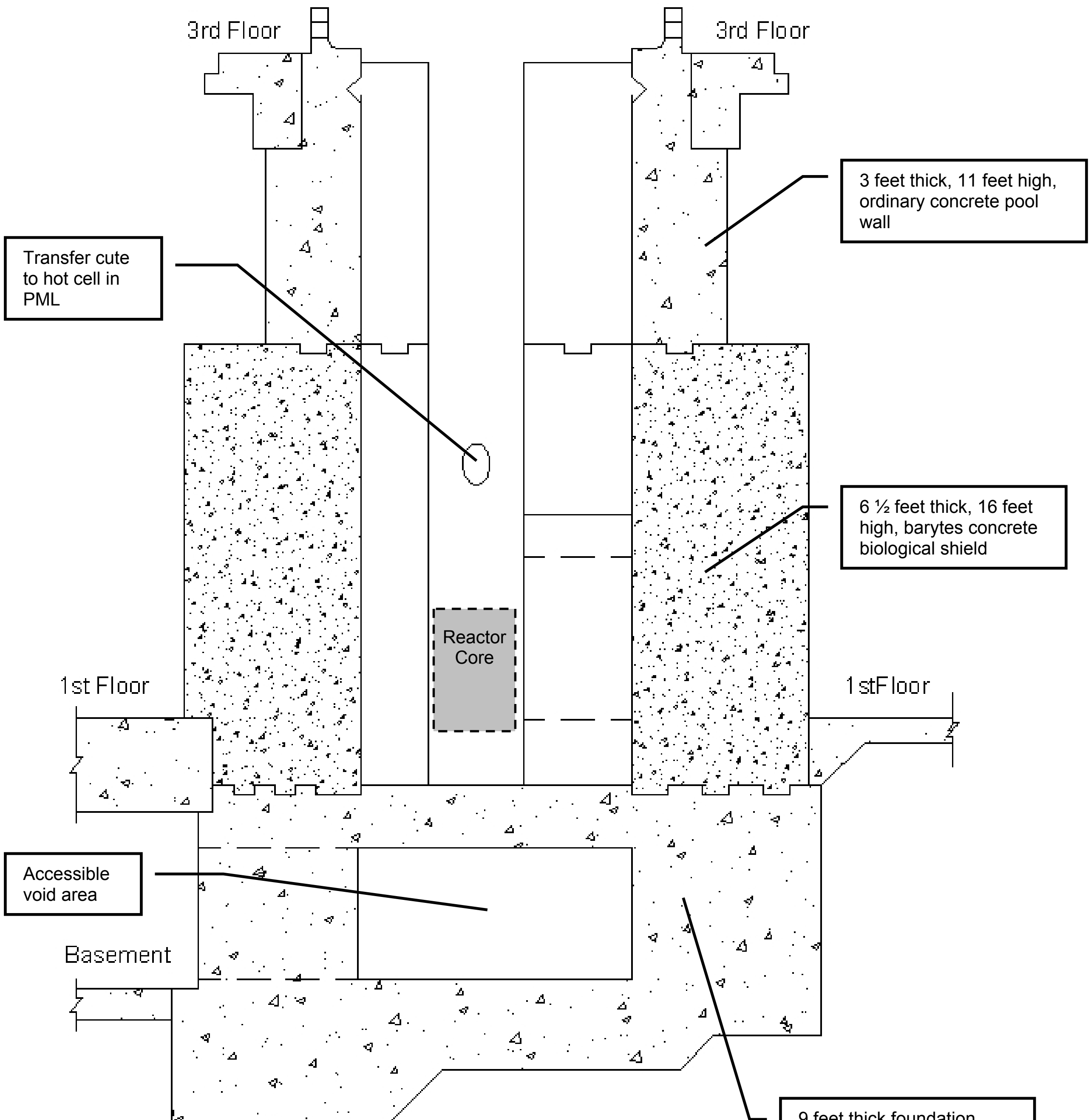
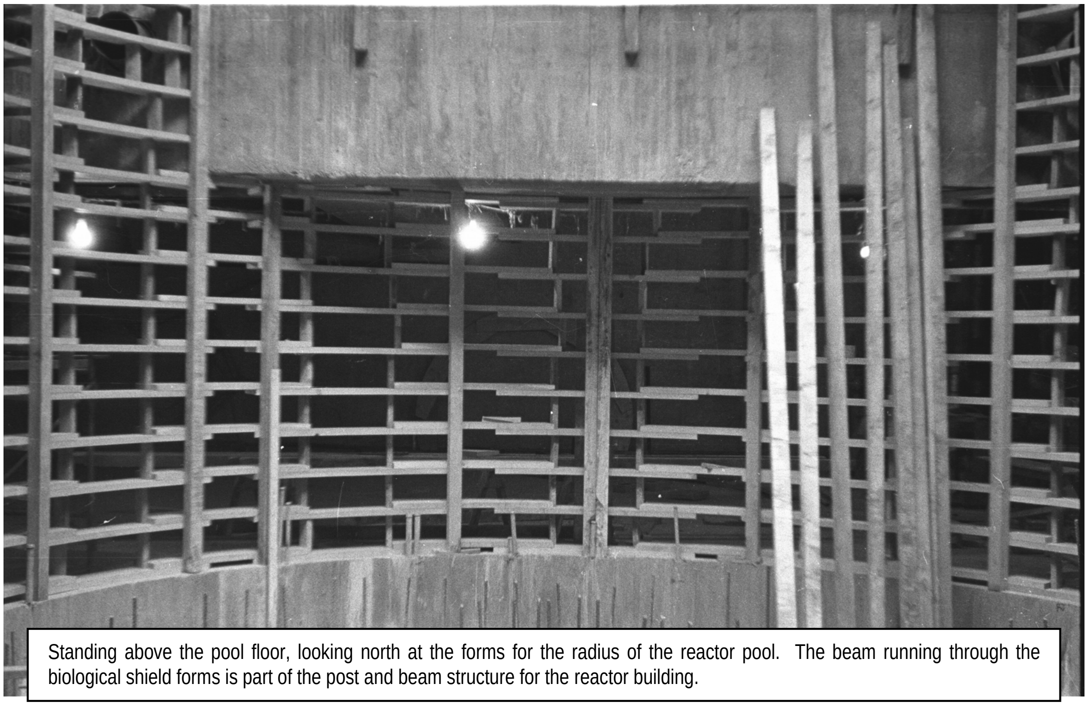
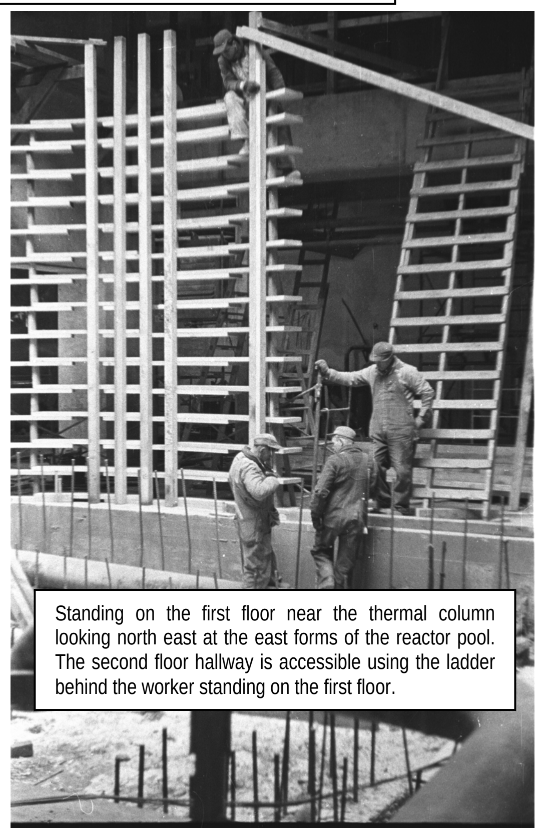


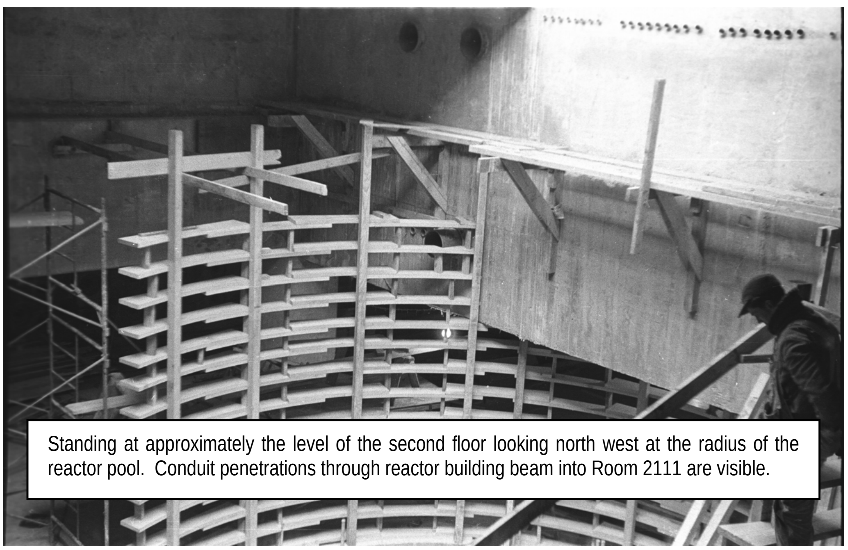
FIGURE 1-14, PHOTOGRAPHS OF THE FORM CONSTRUCTION FOR THE REACTOR POOL



Standing above the pool floor, looking north at the forms for the radius of the reactor pool. The beam running through the biological shield forms is part of the post and beam structure for the reactor building.



Standing on the first floor near the thermal column looking north east at the east forms of the reactor pool. The second floor hallway is accessible using the ladder behind the worker standing on the first floor.



Standing at approximately the level of the second floor looking north west at the radius of the reactor pool. Conduit penetrations through reactor building beam into Room 2111 are visible.

The fuel elements, reflectors, and experiment holders making up the reactor core were assembled on the grid or matrix plate (see Figure 2-2) attached to the end of the suspension frame. The grid plate is drilled with eighty holes, in a 8 by 10 array, into which the ends of the fuel elements, reflectors, or experiment holders were inserted. The grid plate was machined from a 5 inch aluminum slab and has a ¼ inch stainless steel pin for each grid location, which is two or more rows in from the edges of the grid plate. A hopper, or funnel, hangs from the bottom of the reactor grid to mate with the header for the primary coolant system.

1.2.4.2 Reactor Fuel

All fuel elements have been removed from the facility.

The FNR utilized MTR type fuel elements composed of three main components: 1) the curved fuel plates, 2) the aluminum side plates, and 3) the nose cones. Aluminum was the predominant material of construction. The standard fuel element contained 18 curved fuel plates inserted in slots in the side plates to form a box approximately 3 inches by 3 inches. The fuel plates were fabricated in a sandwich fashion, utilizing a technique similar to that used to make fuel plates for the U.S. Naval Reactors program. A mixture of aluminum powder and uranium aluminide (U-Alx) is laid out in a dog-bone shape inside an aluminum "picture frame" onto a 0.015 inch aluminum clad plate, the second 0.015 inch clad plate is laid on top to complete the "sandwich" and then entire assembly is fused into a single plate under extreme pressure. The result is a fuel plate where, upon the exposure of individual grains of U-Alx to the reactor coolant, only the small amount of uranium and fission products associated with the exposed grains would be released.

1.2.4.3 Heavy Water Reflector

A heavy water reflector, containing approximately 50 gallons, is located over the 4 northern most rows of the grid plate. The heavy water tank provided an enhanced thermal neutron spectrum for the beam ports and acted as a startup source for the reactor. Two standpipes extend from the top of the tank to the surface of the reactor pool. These standpipes provide the necessary connections to fill and drain the tank. The heavy water level is maintained below the level of the reactor pool to ensure that leakage will be from the pool into the reflector. The heavy water reflector is currently estimated to contain 227 curies (Ci) of tritium (July 3, 2003). The heavy water is on loan from DOE and will be returned to the Savannah River Site for cleanup and processing when removed from the reflector tank.

1.2.4.4 Transfer Chute

The south end of the pool has a 10 inch diameter water lock approximately 10 feet below the water surface that connects to the north hot cell on the first floor of PML. The water lock allows shielded transfer of highly radioactive material such as experiments and fuel elements between the reactor pool and hot cell.

1.2.4.5 Beam Ports

Twelve aluminum beam ports penetrate the north (rounded) end of the reactor pool at points 5 to 6 feet above the floor of the pool. Since the pool bottom is 3 feet below the first floor of the reactor building, the beam port openings are 2 to 3 feet above the first floor. Eight of the beam ports are set radially in the pool wall and focus on the center of the reactor grid. The remaining four ports form two "through-ports" which pass through the pool just north of the reactor grid. Six of the radial ports and the two through-ports are 6 inches in diameter and two of the radial ports are 8 inches in diameter. Each beam port consists of an assembly in the pool wall with an

opening outside the pool and a flange just inside the pool, and a beam port extension. The radial beam port extensions have a flange at their open end and a beveled end to fit the face contour of the reactor at the other. A flange clamp or split clamp holds the beam port extension to the assembly in the pool wall. The through-port beam port extensions are flanged at both ends to connect to the pool wall assemblies on the opposite sides of the pool.

Two ports, with a diameter of 6 inches, penetrate the east wall of the reactor pool directly across from the thermal column. These two beam ports do not have beam port extensions but have blind flanges held on the end of the pool wall assemblies by a flange clamps.

1.2.4.6 Thermal Column

A thermal column, 6 feet square, penetrates the west wall of the reactor pool. The casing of the thermal column is a telescopic design (increasing the dimensions in steps from the inside surface of the pool wall) filled with square graphite blocks. The thermal column was intended for radiation and exponential experiments using thermalized neutrons. In the late 1960's/early 1970's the thermal column was opened and approximately half of the graphite was removed to allow for the injection of sealant around the casing of the thermal column to minimize water leakage from the reactor pool. The graphite was not reinstalled. Shielding is provided by a 12 3/4 inch lead and steel door on the first floor side of the thermal column.

1.2.5 Pneumatic Tube System

A pneumatic tube system permitted quick irradiation of samples by placing and removing samples between the western edge of the reactor core and a select loading/unloading station through an aluminum tube. Originally there were four separate systems consisting of two aluminum tubes each, one for the rabbit the other for the air motive force. Pneumatic tube service to the PML was discontinued in 1993 due to leaks in three of the tube pairs in the reactor pool, which are plugged. The pneumatic tube system served only the load/unloading station in the fume hood in room 3103 at the time the reactor was shutdown. The pneumatic tube system enters the reactor pool through the floor of the reactor pool through a specially designed flanged access opening that seals the pool water from the system.

The motive force for the pneumatic tube system was provided from one of two blowers located in the FNR basement. The blowers exhaust to PML Stack 2 as described in Section 1.2.2.1 and Section 1.2.3.

1.2.6 Cooling System

The reactor core or reactor pool, depending on the mode of operation for the reactor, utilized a forced circulation cooling system to remove heat. A movable header is connected to the suction of the primary pump(s) through a penetration in the floor of the reactor pool. When raised, the movable header provides forced cooling of the reactor by mating with a hopper attached to the bottom of the reactor grid. Water from the reactor pool passes through a holdup tank, piping, pump, and a heat exchanger located in the reactor building basement before being returned to the reactor pool through another penetration in the pool floor. The heat exchanger is a shell and tube design with the reactor pool water on the shell side. Heat is removed from the reactor pool water when secondary cooling water passes through the U-shaped tubes inside the heat exchanger.

The secondary cooling system pumps water through the U-shaped tubes in the heat exchanger to a cooling tower on the roof of the reactor building. The cooling tower sump is located approximately 15 feet above the surface of the reactor pool to maintain the secondary cooling

system at a higher pressure than the pool water passing through the heat exchanger. Makeup for the secondary cooling system is supplied from potable water.

During reactor operations, a demineralizer system drew water from the pool cooling system for passage through filters, piping, pumps, and demineralizer columns located in the basement for the purpose of maintaining the required low conductivity of the pool water. The demineralized water was returned to the reactor pool as the process cycled.

1.2.7 Emergency Makeup Water

Emergency makeup water for the reactor pool is provided by a 4 inch water main to the reactor building. Following manual initiation, this line is capable of supplying water to the reactor pool at a rate greater than the loss rate through a ruptured pneumatic tube. The emergency makeup water could keep the reactor fuel covered with water in the event of a loss of coolant accident.

1.2.8 Storage Ports

Fifty storage ports which extend through the west wall of the first floor of the reactor were designed and used to store beam port plugs, collimators or other experiments from the beam ports. The storage ports were also used to store items with elevated dose readings. Storage Port No. 1 was used for a number of years to secure and store two large $Pu^{238}Be$ sources (since relocated external to the reactor building). These 10 ½ feet long, schedule 40 steel pipes are capped outside the building and supported by a concrete wall at the far end. Each storage port has an off-gas vent connected to the exhaust system on the first floor that is connected to the Stack 2 exhaust system in PML.

1.2.9 Building Crane

A 15 ton overhead gantry crane spans the pool area on the third floor of the reactor building. This crane traverses east-west on rails supported by the post and beam structure for the reactor building. Reactor staff routinely utilizes the crane for moving loads on the first and third floors, as well as loads within the reactor pool. This crane was used routinely by the reactor staff to insert and remove the BMI-I shipping cask from the reactor pool during shipments of irradiated reactor fuel. There were 13 shipments in the past 5 years.

1.2.10 Fire Alarm and Protection Systems

The fire protection system for the FNR consists of a dedicated fire alarm system, and a fire service main with an independent water supply, as well as a dedicated dry-pipe, sprinkler system for the cooling tower area. Connected to the fire main are single fire hose connection stations on the first and second floors and two fire hose connections on the third floor.

The FNR fire alarm system is manually activated when a fire exists which is more than minor. The FNR fire alarm system is initiated automatically if the dry-pipe cooling tower sprinkler system actuates. Sounding of this alarm initiates the ringing of fire bells throughout the FNR and PML facility and notifies the UM Department of Public Safety Communication Center, which is manned 24 hours a day.

Multipurpose fire extinguishers are available at various locations in the basement, first, second, and third floors of the FNR building.

1.2.11 Foundation Drain Tile

There is a 4 inch drain tile running externally along the footer of the outside walls of the reactor building which are not contiguous with walls of the PML (approximate elevation 825 feet). Additionally, this 4 inch drain tile runs along the west and north edges of the foundation for the reactor pool (approximate elevation 825 feet) . This 4 inch drain tile system originally drained to the cold sump in the reactor basement (approximate elevation 823 feet), but this connection was plugged in 1993 following the backflow of approximately 7,500 gallons of pool water into the drain line (see Section 2.1.1 for details).

1.3 Reactor Decommissioning Overview

The objective of the UM FNR decommissioning activities is to decontaminate or remove radioactive materials, equipment, components, soil, and other media as necessary to obtain release to unrestricted use for the FNR building from the NRC and be granted termination of the NRC reactor license.

1.3.1 Decommissioning Alternative

The three alternatives considered for decommissioning the FNR were Safe Storage, Entombment, and Decontamination. A summary of each alternative is described below.

In **Safe Storage**, the facility would be placed and maintained in a condition that allows it to be safely stored and decontaminated sometime in the future to a level permitting release to unrestricted use. Safe Storage was rejected as an alternative even though it has essentially the same risks and environmental impacts as the Decontamination option, but during a much longer period of time. This alternative would necessitate continued surveillance and maintenance of the facility for a long period of time, during which the risk of environmental contamination would continue to exist. Moreover, Safe Storage would not allow the UM the option of using the remaining structure for other activities.

In **Entombment,** radioactive materials essentially would be placed in long-term storage in structurally long-lived material, such as concrete. The entombed structure would be maintained under continual surveillance until radioactivity decayed to a level permitting release to unrestricted use. Entombment was rejected as an alternative because it would require continued surveillance and maintenance of the facility for a substantial period of time. The risk of environmental contamination would exist during this period and the UM would not be able to use the facility for other activities if this alternative had been chosen.

In **Decontamination,** all radioactive materials associated with the site, such as the equipment, structures, portions of the facility, and site, would be removed, or decontaminated to levels consistent with the NRC release criteria. Decontamination, recommended by the NRC for non-power reactors, was the alternative chosen by the UM, and will be conducted using methods designed to minimize radioactive waste. Materials, equipment, soils, and portions of building found to be radiologically contaminated or activated will be processed, as necessary, to meet release criteria. A thorough final remediation and contamination survey will be performed to demonstrate to the NRC that remediation efforts meet the release criteria for unrestricted use permitting for termination of the reactor license.

The UM plans to remove the FNR from service and reduce the radioactivity associated with the facility to a level that will permit release of the property to unrestricted use and allow termination of the NRC license.

With Decontamination selected as the preferred alternative, the following are the major decommissioning tasks necessary in implementing this alternative (see Section 2.2.2 for a detailed description of each item below):

- Remove loose equipment
- Isolate, remove, or inactivate systems formerly important to safety

  Standby generator, heavy water tank, spent fuel storage racks, parts of the pneumatic tube system, secondary cooling system, emergency cooling system, control console, exhaust for the fume hood in Room 3103, exhaust on the first floor, beam port extensions, etc.
- Isolate, remove, or inactivate other systems
- Remove asbestos
- Install temporary systems
- Remove reactor structures and activated materials from the reactor pool
- Drain the pool and associated systems
- Characterize previously inaccessible areas (pool, hold-up tank, etc.)
- Remove activated portions of the concrete pool and biological shield and removing or decontaminating other areas of contaminated concrete inside and outside of the building
- Decontaminate or remove embedded piping (i.e., piping embedded in concrete)
- Sample surface and subsurface soil and remediate as appropriate
- Remove or decontaminate fixed equipment, components, and piping
- Decontaminate portions of PML

  Retention tanks and retention tank pit, storage ports and drawers in Room 1069D, hot cells, Room 1069D, Room 1069, Stack 2 ductwork, etc.
- Survey PML and update other licenses
- Conduct final status surveys of all affected areas after decontamination to verify that radioactive material has been removed to below the license termination criteria
- Submit reports to the NRC that demonstrate compliance with the license termination requirements.

The final status survey plan will be developed following the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) guidelines (NRC, 2000b), a multi-agency consensus document that was developed collaboratively by four federal agencies having authority and control over radioactive materials, including the NRC and the U.S. Environmental Protection Agency (EPA).

The planning phase and site preparation will last approximately 4 months. Then decontamination and dismantling activities will begin and last approximately 9 months. It is estimated that the total FNR Decommissioning Project will be completed in approximately 13 to 15 months.

Approximately 312 m^3 (11,000 cubic feet) of radioactive waste will be generated during the FNR Decommissioning Project. It will be sent offsite either to a licensed processor for decontamination, incineration, or volume reduction, and then sent to a licensed disposal facility or directly to a licensed disposal facility.

The total radiation dose estimated to be received by workers from decommissioning the FNR is approximately 4.8 person-rem. The estimated radiation doses to transportation workers and the public along transportation routes from transporting radioactive waste from FNR decommissioning are estimated to be less than 0.1 person-rem. The greatest radiation exposure to the UM community, or general public would occur during an accident which results in the release of significant quantities of airborne radioactive material. The limiting hypothetical accidents relevant to the FNR and the dose associated with those accidents are described in previous license submittals or in Section 3.4.

1.3.2 Estimated Cost

An optimistic or low estimate, best or expected estimate, and a ceiling or high estimate for each major cost associated with the decommissioning of the FNR was developed. The intent was to provide an full and accurate examination of the probable range of decommissioning costs for the FNR. The expected cost estimate is based upon the decommissioning approach described in Section 2.3, the radiological conditions as described in Section 2.1.2, and present costs for disposal of radioactive materials as described in Section 3.2. The high cost estimate allows for contingencies, removes costs savings identified in the expected estimate, and makes a best effort assessment of the maximum variability in the individual costs. These estimates are presented in Table 1-1. The factors utilized in these cost estimates were based upon a detailed cost estimate provided by CH2M HILL, under contract to the UM, based upon the results of the characterization results to date and was heavily based upon their recent experience in performing similar activities at the University of Virginia (UVa) and Georgia Tech, and combined with their ongoing experience at Rocky Flats, Hanford, etc. During the actual decommissioning planning phase, a detailed engineering cost estimate will be prepared.

1.3.3 Availability of Funds

The UM Regents have approved a request of the Executive Vice President/Chief Financial Officer to proceed with the “Ford Nuclear Reactor Decommissioning Project” (UM, 2004). The Regents specifically approved the expenditure of funds from investment proceeds sufficient to cover the “High” cost estimate contained in Table 1-1. The University is committed to providing funding for decommissioning of the FNR. This commitment satisfies the requirements of 10 CFR 50.75(e)(iv).

1.3.4 Program Quality Assurance

1.3.4.1 Overview

This section provides a brief description of the quality assurance programs utilized during decommissioning.

- A quality assurance program is applied to the design, fabrication, construction, and testing of structures, systems, and components of the facility. These quality assurance requirements would apply to the remediation activities conducted.
- A quality assurance program, which may or may not be the same as the above mentioned program, is applied to the design, purchase, fabrication, handling, shipping,

storing, cleaning, assembly, inspection, testing operations, maintenance, repair and modification of components of a packaging used in the transportation of licensed material.

- Additional quality assurance requirements are applied to the final status survey and associated documentation (e.g. characterization information used in the design of the final status survey) to ensure that data and the analysis of the data provided to the NRC in the final status survey report is accurate and complete.

TABLE 1-1, FORD NUCLEAR REACTOR DECOMMISSIONING COST ESTIMATE

	Low	Expected	High
Site Characterization and Decommissioning Plan			
Contract Specialists	$ 352,690	$ 352,690	$ 373,343
Decommissioning Contracted Expenses			
HIC/Liner Waste Burial	*$ 224,056* [1]	$500,000	$750,000
Low Level Waste Burial	$1,353,460	*$1,353,460*	$2,128,734
Contractor Effort	$1,204,517	*$1,324,969*	$1,656,211
Sub Contract Specialists			
Metal Segmentation	$227,273	*$250,000*	$312,500
Pool Segmentation [2]	$1,211,095	*$1,332,205*	$1,665,256
Laboratory Analysis	$7,985	*$8,783*	$10,979
Equipment	$228,160	*$250,976*	$313,720
Consumables	$192,782	*$212,060*	$265,075
Travel	$32,039	*$35,243*	$70,486
UM Decommissioning Costs			
Project Management Costs	$ 142,376	$158,195	$259,548
Professional Fees and Consultants	$ 152,583	$192,583	$ 240,729
Planned Contingency Costs			
Pool Water Disposal-Incineration	N/A	N/A	$300,000
Unallocated Contingency (5%)	$234,068	$263,385	$358,648
Unplanned Contingency Costs (15%)	$702,205	$790,154	$1,075,944
Total Cost	**$6,265,290**	**$7,024,703**	**$9,781,173**

Note: 1 Numbers in italics are from the draft cost estimate prepared by CH2M HILL and supplied to the UM with the Ford Nuclear Reactor Decommissioning Plan (Draft), October 2003.

2 Estimate is based on pool wall removal, the most expensive and conservative decommissioning option. However, the UM may choose pool decontamination or a combination of partial pool removal and decontamination if maintaining radiation exposure as low as reasonably achievable (ALARA), safety, structural, cost/schedule, and future use considerations warrant.

3 Estimate includes the costs of shipping and disposing of radioactive waste and of the final status survey." 1

1.3.4.2 Quality Assurance for Design, Construction, Testing, Modification, and Maintenance

FNR has a quality assurance program that provides the requirement for establishing and executing a quality assurance program for the design, construction, testing, modification, and maintenance of a research reactor which satisfies the requirements of 10 CFR 50.34. The descriptions of the managerial and administrative controls are described in Section 2.4, and will result in a revision to the current quality assurance program. FNR will continue to maintain this quality assurance program for the design, construction, testing, modification, and maintenance (including remediation activities).

The UM will continue to require that all contractors and subcontractors participating in design, construction, testing, modification, and maintenance (including remediation) activities follow the established quality assurance program. Contractors and subcontractors may recommend or

request changes to the quality assurance program. The UM may or may not make the change to the quality assurance program after review against applicable guidance or standards (NRC, 1977).

Changes to the quality assurance program will be approved as discussed in Section 2.4.

1.3.4.3 Quality Assurance for Packaging, Preparation for Shipment, and Transportation of Licensed Material

10 CFR 71, Subpart H provides the requirements for packaging, preparation for shipment, and transportation of licensed material. The managerial and administrative controls the FNR has established to satisfy the requirements of this subpart are described in Section 2.4 and differ slightly from those previously utilized. The current FNR quality assurance program (described above) has been approved by the NRC as required by 10 CFR 71.101(c) (NRC, 2000d). The existing quality assurance program will be followed and maintained through timely renewal, as necessary, to support packaging, preparation for shipment, and transportation of licensed material during remediation activities.

The UM will continue to require that all contractors and subcontractors participating in packaging, preparation for shipment, and transportation of licensed material follow the approved quality assurance program. Contractors and subcontractors may recommend or request changes to the quality assurance program. UM may or may not make the change to the quality assurance program after review against the requirements of 10 CFR 71, Subpart H. Revisions to the quality assurance program shall be submitted to the NRC for approval as required by 10 CFR 17.101 (c), prior to implementation and use for the packaging, preparation for shipment, and transportation of licensed materials.

Changes to the quality assurance program will be made as discussed in Section 2.4.

The UM may elect to utilize a contractor's or subcontractor's quality assurance program to fulfill the requirements contained in 10 CFR 71, Subpart H after verification that the contractor's or subcontractor's quality assurance program is acceptable to the UM and has been approved by the NRC.

1.3.4.4 Quality Assurance for Final Status Survey and Associated Documentation

Additional quality assurance requirements are applied to the final status survey and associated documentation (e.g. characterization information used in the design of the final status survey) to ensure that adequate controls are in place. These controls are applicable to:

- Methods that provide sufficient data about pertinent structures, systems, components, equipment, the site, and the environment included in the final status survey report.
- Types, calibrations, and operating conditions of instruments used in the final status survey and associated documentation.
- Methods used to obtain and analyze data, including methodology selected to translate instrument readings or sample analysis results into appropriate units used in the final status survey and associated documentation.
- Comparisons with preoperational radiation survey results and other data on background radiation used in the finals status surveys and associated documentation.
- Methods for auditing and verifying data used in the final status survey and associated documentation.

- Quantitative error or statistical analyses of data and results used in the final status survey and associated documentation.
- Bases and methods for making statistical inferences from data selected to ensure that all significant residual sources of radiation are found and quantified in the final status survey and associated documentation.

The quality assurance program for the final status survey and associated documentation is described in Section 4.2

The UM will develop the elements of the quality assurance requirements for the items described above. These elements will be reviewed and approved as discussed in Section 2.4.

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2.0 Decommissioning Activities

The objective of the UM FNR decommissioning activities is to remove licensed radioactive material from the facility and the surrounding grounds necessary to obtain NRC approval for release to unrestricted use of the facility 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. Upon termination of the NRC license, the building will be fully renovated to continue to support instructional and research activities for the UM.

2.1 Facility Radiological Status

2.1.1 Facility Operating History

The FNR operated for 17,868 MW days between 1957 and 2003. In 1966, a continuous operating cycle for the FNR was adopted at its licensed power level. The cycle consisted of approximately 25 days at full power followed by 3 days of shutdown for maintenance. In 1975, a reduced operating cycle was adopted consisting of 10 days at full power level followed by 4 days of shutdown for maintenance. A typical week consisted of 120 full-power operating hours. In 1983, the reactor operating schedule was changed to Monday through Friday at licensed power, with weekend shutdowns. Periodic maintenance weeks were scheduled during the year. In 1985, a cycle was established consisting of 4 days (or 96 full-power operating hours per week) at licensed power level, followed by 3 days of shutdown for maintenance. This was done to eliminate the periodic maintenance weeks needed in the previous cycle. Beginning July 1, 1987, the reactor operating cycle returned to 10-day operation at full power level followed by 4 days of shutdown for maintenance (UM, 1999). Operation of the reactor ceased on July 3rd, 2003.

All fuel was removed from the facility in December 2003.

Historical and characterization information indicates the following potential causes of radioactive contamination in the reactor building from normal operations and routine activities:

- Samples irradiated in the reactor were routinely unloaded on the third floor near the reactor pool and transferred to room 3103 for processing or to room 3104 for analysis.
- Samples irradiated in the reactor were routinely unloaded and stored in a holding area on the west side of the reactor pool at the south end of the third floor.
- Samples irradiated in the pneumatic tube system were routinely loaded and unloaded in the fume hood located in room 3103.
- At 2 MW, portions of the lower section of the pool walls near the beam ports and the reactor pool floor were exposed to thermal neutron flux of approximately 3×10^7 neutrons per cm^2 per sec and a fast neutron flux of approximately 3×10^6 neutrons per cm^2 per sec based upon measurements made in June 2003.
- The potentially contaminated water from the drains in the basement, first and third floors of the reactor building, the sink in Room 3103, the overflow in the reactor pool, and the drains for each beam port, and the drains from the janitors closets on the first and third floors is collected in the hot and cold sumps located in the reactor basement.

- Long contaminated or activated items removed from the reactor pool were routinely hung on the western side of the third floor south wall.
- A pool cover, used to limit the evaporation rate and heat loss from the reactor pool when the reactor was shutdown, was routinely hung on the third floor south wall upon removal.
- Prior to 1991, the demineralizer for the pool cleanup system, located in the pit at the south end of the reactor basement, was routinely recharged by the operating crew. After 1991, the use of disposable resin columns eliminated this activity. The resin within these disposable columns was routinely transferred to a licensed disposal facility.
- The storage ports in the west wall of the first floor of the reactor building were routinely used to store irradiated collimators and other irradiated components.
- The FNR Technical Specifications permitted the discharge of airborne radioactive materials from the facility's exhaust stacks at higher concentrations than specified in 10 CFR 20 while maintaining exposure of the UM community and the public within all applicable regulatory limits.
- The FNR Technical Specifications permitted the discharge of readily soluble or readily dispersible biological radioactive materials from the facility's liquid waste system to the city of Ann Arbor sanitary sewerage system at higher concentrations than specified in 10 CFR 20 while maintaining exposure of the UM community and the public within all applicable regulatory limits. Liquid discharges to the sanitary sewer were suspended in 1991.
- Singly encapsulated materials irradiated by the FNR were limited by the FNR Technical Specifications such that a release of all the gaseous, particulate, or volatile materials of the target material could not lead to an exposure exceeding 10 percent of the equivalent annual exposures stated in 10 CFR 20.
- Doubly encapsulated materials irradiated by the FNR were limited by the FNR Technical Specifications such that a release of all the gaseous, particulate, or volatile materials of the target material could not lead to an exposure exceeding the equivalent annual exposures stated in 10 CFR 20.
- Prior to 1992, tritium-loaded heavy water was routinely transferred out of the heavy water reflector and refilled with fresh heavy water to keep the total tritium inventory of the heavy water reflector to less than 50 Ci.
- Irradiated reactor fuel elements were routinely removed from the reactor, packaged and shipped to DOEs' Savannah River facility.
- Except for a period in the 1980's when radioactive waste disposal options were limited or non-existent, radioactive waste was routinely packaged and transferred to licensed disposal facilities. This was predominantly dry-active-waste (DAW) generated by reactor operations, but also included activated and contaminated materials from maintenance, modifications, and reactor experiments.
- From 1972 to 2003, two plutonium-238/beryllium neutron sources, totaling 84 Ci were stored in one of the storage ports in the west wall of the reactor building. These sources have been relocated to a UM facility outside of the reactor building.

- For the entire operating history of the reactor, none of the tubes in the shell and tube heat exchanger has ever leaked or been plugged, thereby maintaining the separation between the primary and secondary cooling water.

Throughout the operating history there were no major releases of radioactive materials to the environment as verified by the onsite and offsite environmental monitoring program. During nearly 50 years of operation there have been relatively few instances that resulted in a significant impact on the radiological status of the facility. A summary description of these instances is provided here to support a discussion of the current radiological status of the facility and to support the eventual conclusion that the activities proposed in this decommissioning plan are appropriate and support termination of the reactor license in compliance with current NRC criteria for license termination.

An examination of the history of the facility showed that the staff identified problems and followed-up this identification with proper and complete corrective measures to ensure that the spread of radioactivity was minimized and contained within the reactor building or to ensure that the release of radioactive materials to the environment or surroundings was kept as low as possible. This has been generally confirmed by the lack of or low level radioactive contamination found on surfaces throughout the facility and the lack of radioactive contamination found in the soil and groundwater sampling conducted by CH2MHILL, under contract with the UM, and described in the Characterization of the Ford Nuclear Reactor (Appendix A).

Historical and characterization information indicate the following potential causes of radioactive contamination in the reactor building from non-routine occurrences, operations, accidents or spills. The impact of these non-routine occurrences, accidents, or spills will be fully remediated and assessed to ensure that the facility can be unconditionally released and the reactor license terminated as allowed by state and federal regulations.

- May 1959 – Approximately 3,600 gallons leaked from the reactor pool through beam port D (north east side of the reactor pool) onto the first floor when a solid shield plug was inserted into the port and penetrated the reactor face of the beam port extension (UM 1960).

- May 1961 – A large portion of the first floor of the reactor building, the second floor hallway and the control room were contaminated with Silver-110m when a researcher disassembled part of an experimental setup on an unknown beam line. The researcher was not aware of loose contamination on the experiment until after he had spread the contamination and discovered contamination on his hands and feet (UM, 1961).

- June 1963 – The collimator installed in beam port J (southern-most beam port on the east side of the reactor pool) leaked locally, contaminating the first floor (UM, 1963).

- March 1966 – An estimated 5 to 10 milliliters of contaminated heavy water leaked from the packing of the transfer pump onto the third floor (immediately north of the reactor pool) during routine transfer of heavy water with the heavy water reflector (UM, 1966).

- September 1967 – The level of radioactivity measured by the third floor continuous air monitor peaked at approximately 2 percent of the Maximum Permissible Concentration (MPC) allowed by the regulations. Daughter products of fission product gases, Cs-138 and Rb-88, were identified on the filter media of the continuous air samplers. Fuel element No. 47 was identified as the source of the fission product gases and removed

from service. Efforts were made to determine the presence of other fission products in the pool water and none were detected (UM, 1967a, b, & c).

- October 1967 – The high-efficiency particulate air (HEPA) filter in the duct servicing the first floor of the reactor building, the storage ports and the blowers for the pneumatic tube system was reading approximately 10 millirem (mrem) per hour of gamma activity. A gamma analysis was conducted on samples taken from the HEPA filter. Co-60 peaks were identified, but other peaks were not identifiable and radon daughters were not suspected. The source of the contamination was postulated to originate from a portion of an old, blackened, poly rabbit (reading > 25 R/hr) that had been removed from the core end of the PL-2 irradiation location almost a month later. Gamma analysis of the blackened, poly rabbit was consistent with the gamma spectrum of the samples taken from the HEPA filter (UM, 1967d).

- May 1969 – During routine fuel handling procedures a fuel element was dislodged from the reactor grid and dropped to the floor of the reactor pool. The element was recovered and placed in a storage rack. Almost immediately the third floor continuous air monitor alarmed. Subsequent gamma analysis of the filter media for this continuous air monitor identified fission product gases. It was believed that a single, short burst of activity may have occurred when the element was raised from the pool bottom and the water pressure reduced on the element, leading to the release of a small gas bubble from a pinhole or scratch on the element (UM, 1969).

- October 1971 – The gaseous activity detector monitoring the duct servicing the first floor of the reactor building, the storage ports and the blowers for the pneumatic tube system alarmed as the result of the release of Argon-41 when an 8 inch beam port was flooded (UM, 1971).

- January 1977 – Fission products, identified by gamma analysis of the filter media, caused the continuous air monitor for the duct servicing the first floor of the reactor building, the storage ports and the blowers for the pneumatic tube system to alarm with levels of 17,000 counts per minute (cpm). The source of the fission products was determined to be from the pneumatic tube PL-1. No uranium samples had been irradiated in PL-1 for the 12 months prior to the occurrence. The uranium was believed to have been residue from some previously irradiated sample (UM, 1977a).

- December 1977 – Operation of a new pneumatic tube system designed to exhaust directly to the FNR building exhaust plenum near the exhaust air radiation monitor (located in the air handler in room 2111) generated a release of airborne radioactivity below 10 CFR 20 release limits but sufficient to cause the Building Exhaust Radiation Exhaust Air Monitor to Alarm and initiated isolation or confinement (UM, 1977b).

- April 1987 – An estimated two pints of contaminated heavy water containing 182 millicuries (mCi) of tritium was spilled onto the third floor (immediately north of the reactor pool) during a routine heavy water transfer to the heavy water reflector (UM, 1987).

- January 1989 – The identification of fission products, Iodine-131 (2.76×10^{-6} microcuries per ml) and Xenon-133 (1.39×10^{-5} microcurie per ml) in the reactor pool water was followed by the confirmation of the presence of small but detectable quantities of Rubidium-88 and Cesium-138 in high volume air samples taken above the reactor core.

Fuel element No. 204 was identified as the source of the activity and retired from service (UM, 1989).

- July 1991 – NRC conducted a special investigation to review FNR's liquid radioactive discharges to the sanitary sewer in response to the identification of low-levels of certain radioactive isotopes in sludge generated by the Ann Arbor waste water treatment plant. The inspection concluded that the liquid releases for the previous 18 months were in compliance with the requirements of 10 CFR 20.303 for releases to the sanitary sewer system (NRC, 1991b).

- November 1991 – The investigation following the identification of fission products, Cesium-138 (2.70×10^{-5} microcuries per ml), Iodine-133 (8.28×10^{-6} microcurie per ml), Krypton-85m (5.24×10^{-6} microcuries per ml), Molybdenum-99 (5.59×10^{-6} microcuries per ml) and Xenon-135 (1.70×10^{-6} microcuries per ml) in continuous air samples taken from above the reactor core identified that fuel element no. 224 as the source of the activity. The fuel element was retired from service (UM, 1991).

- July 1993 – Approximately 7,500 gallons of low-level radioactive water was released from the reactor building through the drain tiles around the foundation of the building over approximately 36 days. No radioactivity from the reactor was identified in the soil near the reactor building at the time of the occurrence or in the recently collected soil samples from under and near the reactor building. Although this release of low-level radioactive water did result in detectable levels of tritium in the ground water near the reactor building in 1993, the levels of tritium were below levels of regulatory concern and recent sampling of the ground water near the reactor build provide no indication of tritium from the release in 1993 nor any current release of tritium from the facility.

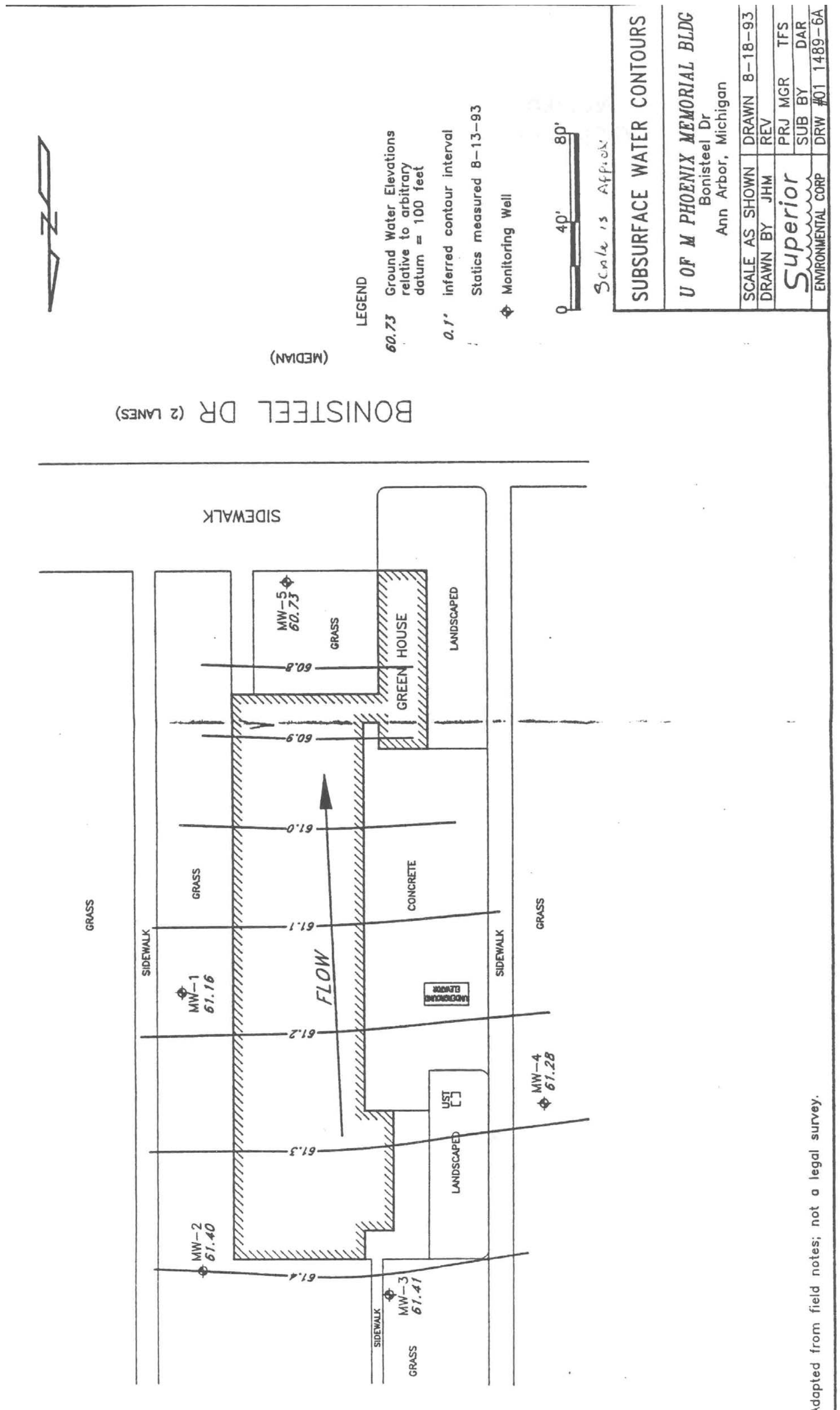
  The estimate of the total radioactivity released after seven days of decay is given in Table 2-1.

TABLE 2-1, ESTIMATED RADIOACTIVITY RELEASED AFTER 7 DAYS OF DECAY

Nuclide	Total Activity (microcuries)
Sodium-24	1.83
Tungsten-187	2.23
Chromium-51	407
Silver-110m	177
Antimony-122	7.55
Iron-55	1,740
Tritium (Hydrogen-3)	141,000

In response, in August 1993, five ground water monitoring wells, MW-1 through 5, were installed around the reactor building as shown in Figure 2-1 to monitor for radioactivity in the groundwater. Soil samples were collected during the installation of the monitoring wells. The samples from the saturated zone were counted for 2 hours in bags laid directly on top of a high purity germanium (HPGe) detector. No radioactivity normally associated with water from the reactor pool was detected in these samples. Soil samples from above the saturated zone were counted in a similar manner and each showed no radioactivity normally associated with water from the reactor pool.

Figure 2-1, Subsurface Water Contours – August 1993



Tritium was detected from the monitoring well in the south east corner of the reactor building (MW-1) at approximately 2.1 x 10^{-4} microcuries per ml (7 percent of the 10 CFR 20 maximum permissible concentration for an unrestricted area). By September 1993 samples taken from this well peaked at 2.4 x 10^{-3} microcuries per ml (80 percent of the 10 CFR 20 maximum permissible concentration for an unrestricted area) followed by a gradual decrease. This behavior in the concentration of tritium in the groundwater indicated that the tritium concentration in the ground water was from the release of approximately 7,500 gallons of low-level radioactive water through the foundation drain tiles and not from a sustained leak in the reactor pool.

The tritium from the 7,500 gallons of low-level radioactive water was detected in the monitoring well immediately south of the PML (MW-5) and peaked at 9.54 x 10^{-6} microcuries per ml (EPA's Maximum Contaminant Level (MCL) is 20 x 10^{-6} microcurie per ml) in February 1994.

A pathway analysis of the tritium plume produced by this release of low-level radioactive water was performed using the PAGAN and GENII low level waste performance assessment modeling codes. PAGAN, a one dimensional transport code which includes advection and dispersion, was employed to model the tritium concentration in the contaminant plume detected by the monitoring well in the south east corner of the reactor building (MW-1). No retardation was assumed for the tritium transport. Although no drinking water wells are located within 1000 meters from the FNR, GENII, a dose assessment code, was then used to estimate the radiation exposure for the drinking water pathway when all drinking water was assumed to be obtained from a contaminated well. For a well 10 meters from the reactor building, the maximally exposed individual was estimated to receive 130 mrem per year in the first three months of the year following the release. For a well located 1000 meters from the release point, the maximally exposed individual was estimated to receive 0.1 mrem per year. (Bullen, 1994).

The connection to the drain tiles through which the low-level radioactive water was released was mechanically plugged shortly after discovery of the release.

- May1998 – Approximately 75 gallons of pool water leaked from a hole in the housing of a fiberglass resin column onto the basement floor (near pit at the south end). The water was collected by the facility's drain system (UM, 1998).
- March 2001 – The third floor (west and north sides of the reactor pool) was contaminated after a flux measurement wire was removed from a thin cadmium sleeve following irradiation. Contamination levels as high as 13,000 counts per minute were measured. The area was fully decontaminated (UM, 2001).

As shown by the above listing, during nearly 50 years of operation of the FNR there were instances which lead to the radiological contamination of the facility. However the prompt response and follow-on cleanup activities by the facilities staff produced only limited contamination in areas not expected to be contaminated by routine operations. Additionally, the facility's practice of periodic monitoring and maintaining contamination levels between 3 and 10 times above background, see Section 2.1.4, resulted in only a limited number of areas where contamination levels above the anticipated release criteria are known to exist as detailed in Appendix A, Section 5.2.2.

2.1.2 Current Radiological Status of the Facility

Many areas identified in the Historical Site Assessment (CH2M HILL, 2003) as potentially impacted have been shown to be free of contamination. This allows future characterization activities to concentrate on known or potentially impacted surfaces and support planning for license termination (final status) surveys.

Radiological surveys performed on accessible areas have demonstrated that the major portion of the FNR structure is not contaminated as a result of reactor operations and activities, and will not require remedial activities to achieve license termination. Contamination was identified predominantly in locations that historic use would suggest such contamination.

The following are locations of structure contamination:

- Third Floor – Floor drains, floor near pool, south wall above pool, Room 3103
- Second Floor – None identified
- First Floor – Floor drains, floor trench around pool wall, pool wall west and north walls, storage ports in the west wall.
- Basement – Floor drains, sumps, floor

  Note: No contamination was identified on the fourth floor/cooling tower.

Because the reactor was fueled and operational at the time of the above surveys, elevated ambient radiation levels prevented direct surveys for surface contamination throughout the basement and at some locations on the first and third floors. Access to reactor systems and sampling that could affect the integrity of systems or structure were not possible because of the status of the reactor. The following systems and surfaces therefore will be addressed by continuing characterization, after the reactor is permanently shutdown, fuel removed, and systems can be accessed:

<u>Third Floor</u>

Reactor pool and contents

South wall of reactor room

Floor in vicinity of reactor pool

Drains

Exhaust ventilation for the fume hood in Room 3103

Janitors' closet

<u>First Floor</u>

Pool wall

Drains

Source storage ports and surrounding soil

Floor near pool wall

Soil beneath floor near pool wall and thermal column trench

Storage port and local exhaust ventilation

Basement

Drains

Sumps

All structure surfaces

Soil beneath reactor and around sumps

2.1.2.1 Principal Radioactive Components

Estimates of the radioactivity inventory can be determined by considering the constituent elements of the material in question and calculating the duration of exposure to the neutron flux and energies of the incident neutrons (Erdman 1976). These calculations provide a first order estimate of the radioactivity and can also be made or refined from direct measurements which can include, but are not limited to:

- Limited sampling to establish ratios of radionuclides present in a structure or component
- Direct analysis using sodium iodide (NaI), HPGe, or other detectors to analyze the gamma spectrum being emitted to identify specific isotopes, establish ratios of isotopes, or to fully quantify isotopes
- Direct measurement of dose rates to support computational methodologies for the determination of radionuclides (e.g. MicroShield or hand calculations)
- Direct measurement of similar items for extrapolation via computational methods for inaccessible components or structures

Direct measurements, however, are generally more reliable and will be used during actual removal or dismantlement of components. The results of these direct measurements can also be used to specify the necessary safety measures and procedures for various dismantling, removal, decontamination, waste packaging and storage operations so that exposure to personnel is maintained as low as reasonably achievable (ALARA).

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2.1.2.2 Reactor Pool

2.1.2.2.1. Pool Water

The reactor pool and the primary cooling system contain approximately 50,000 gallons of water requiring removal. The water was supplied from potable water through filters and demineralizers. The cleanliness of the water was maintained by a system of filters and H-OH demineralizers were used as necessary to maintain the conductivity less than 5 micromho per centimeter. Chemical additions to the water were not required to maintain the pH between 4.5 and 7.5. The levels of radioactivity in the pool water as of March 2004 are listed in Table 2-2.

TABLE 2-2, RADIOACTIVITY OF THE REACTOR POOL WATER (MARCH 17, 2004)

Gross Alpha	<7.18 pCi/l
Gross Beta	699 pCi/l
Tritium	1,110,000 pCi/l
Silver-108m	66.8 pCi/l
Silver-110m	1,150 pCi/l
Zinc-65	645 pCi/l

pCi- picocurie, l - liter

2.1.2.2.2. Bridge Suspension Frame and Grid Plate

The reactor grid plate has overall dimensions of approximately 25 inches by 33 inches by 6 inches thick. The results of a recent survey of the reactor grid, using a Eberline RO-7 with high range probe in the underwater housing are shown in Figure 2-2. A large contribution to the dose rates for the interior positions of the reactor grid, two rows in from each edge, is believed to be from the ¼ inch by 1 inch stainless steel alignment pins at these locations. Additionally there are 18 stainless steel bolts (¼ inch by 5-3/8 inch) around the outer edge of the reactor grid, which were used to suspend the hopper from the bottom of the reactor grid.

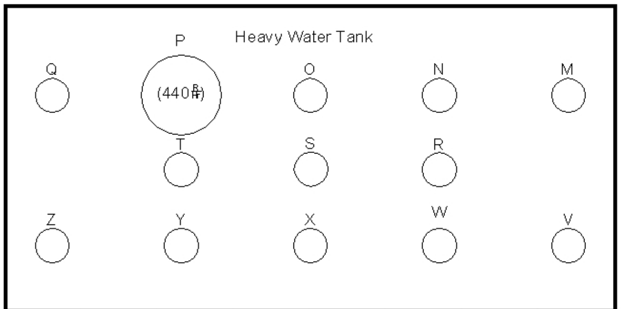
2.1.2.2.3. Reactor Fuel

All reactor fuel has been removed from the facility.

2.1.2.2.4. Heavy Water Reflector

The last transfer of heavy water to the heavy water reflector occurred in February 1992. After that, heavy water transfers to keep the tritium content of the heavy water reflector below 50 Ci were no longer required because of the removal of the 50 Ci limit, through Amendment 36 to the Technical Specifications. For calculation purposes it will be assumed that the reflector contained the maximum allowed activity of 50 Ci of tritium prior to the transfer and contained 44.6 Ci of tritium after the 5 gallon transfer was completed. Tritium buildup in the heavy water reflector was calculated on an annual basis using an average tritium production rate and the power history for each year and accounting for decay. The tritium inventory in the heavy water reflector at the end of March 2004 was evaluated to be 217 Ci. The heavy water is on loan from DOE and will be returned to the Savannah River Site.

FIGURE 2-2, RADIATION LEVELS (R/HR) ON THE REACTOR GRID PLATE – APRIL 2004



75 (170 R/Hr)	65 (450 R/Hr)	55 (540 R/Hr)	45 (540 R/Hr)	35 (550 R/Hr)	25 (440 R/Hr)	15 (280 R/Hr)	5 (210 R/Hr)
76 (190 R/Hr)	66	56 (560 R/Hr)	46 (570 R/Hr)	36 (580 R/Hr)	26 (510 R/Hr)	16	6
77 (200 R/Hr)	67 (480 R/Hr)	57 (550 R/Hr)	47 (560 R/Hr)	37 (560 R/Hr)	27 (480 R/Hr)	17 (290 R/Hr)	7 (220 R/Hr)
78 (190 R/Hr)	68	58 (440 R/Hr)	48 (430 R/Hr)	38 (440 R/Hr)	28 (370 R/Hr)	18	8
79 (130 R/Hr)	69 (250 R/Hr)	59 (290 R/Hr)	49 (290 R/Hr)	39 (300 R/Hr)	29 (260 R/Hr)	19 (220 R/Hr)	9 (190 R/Hr)
80 (140 R/Hr)	70 (150 R/Hr)	60 (210 R/Hr)	50 (220 R/Hr)	40 (220 R/Hr)	30 (210 R/Hr)	20 (180 R/Hr)	10 (220 R/Hr)

Date Performed: April 7, 2004

2.1.2.2.5. Beam Ports

The collimators or experiment plugs that were installed in most of the 6 inch and 8 inch beam ports have been or will be removed by the reactor staff with the following exceptions:

- The upper 6 inch through port, running east west, which is believed to contain shielding materials and a ¾ inch aluminum tube that runs through this shielding material and is open on the east and west ends of the through port. The contents of this through port will need detailed characterization before removal, processing and disposal.
- The beam port closest to the thermal column is believed to contain a collimator that extends the full length of the beam port, i.e. from the opening to the reactor core. This collimator is believed to contain a ½ inch stainless steel box in a taper, small at the reactor end and wide at the opening, around which lead and polyethylene shielding was attached. The contents of this beam port will need detailed characterization before removal, processing and disposal. Dose rates of several R/hr or higher are expected from the reactor end of this collimator. A dose rate of approximately 35 mrem per hr is present at the open end of the collimator, which extends from the beam port opening.

Each 6 inch beam port opening is shielded by a 500 pound lead door or shutter and each 8 inch beam port opening is shielded by a 630 pound lead door. These doors can be raised or lowered across the beam port opening (for the lower 6 inch through port, the door moves side to side rather than up and down).

The contents of all remaining beam ports have been or will be removed and concrete port plugs reinstalled. Minimal contamination and little or no activation are expected from these items based upon past experience.

2.1.2.2.6. Thermal Column

The thermal column has recently been opened and compared with the facility drawings. This investigation revealed that half of the graphite was removed in the late 1960s to early 1970s. Surveys of the graphite blocks showed elevated levels of contamination in areas where water had calcified, but radiation exposure levels were indistinguishable from background. Small samples of some of the graphite material were taken from the graphite blocks in the approximate center of the column; results of the analysis of these samples are pending. From facility drawings, the thermal column contains the volumes of materials listed in Table 2-3.

TABLE 2-3, ESTIMATED MATERIAL VOLUMES FOR THE THERMAL COLUMN

Material	Volume (ft^3) or Mass (lbs)
Cadmium – sheet	0.8 ft^3
Boral (Boron Carbide between Al cladding)	1.0 ft^3
Graphite block	96 ft^3
Lead shot	3,752 lbs
Lead Block	16,747 lbs
Other Lead (caulk & thin strip)	50 lbs

ft^3 – cubic feet, lbs – pounds

To what extent the materials in the thermal column are surface contaminated or activated is not known. These materials will require detailed characterization before and after removal to determine the options for disposal.

2.1.2.2.7. Pneumatic Tube System

Six of the eight tubes to the irradiation stations on the west side of the reactor grid are currently plugged at the point where the tubes penetrate the floor of the reactor pool. Detailed characterization will be required before removal, processing, and disposal.

2.1.2.2.8. Other Items in the Reactor Pool

This section is based on process knowledge and direct measurements. The following are some of the other, higher level, radioactive components to be handled and processed during the FNR decommissioning:

- Reactor Irradiation Facility for Large Samples (RIFLS) reading as much as 50 R/hr
- Heavy Section Steel Irradiation (HSSI) experiment reading about 11,200 R/hr
- Reactor control/shim rods reading about 2,500 R/hr

2.1.2.3 Sanitary Sewer Lines

From the opening of the PML in 1954 until the summer of 1991, low-level radioactive liquids were discharged from the retention tanks in PML through the sanitary sewer line in PML following sampling to verify that applicable regulatory limits and license conditions were satisfied. The sanitary sewer line runs south along the western side of PML, turns west and follows Bonisteel Blvd towards the UM hospital at which point it turns and runs with the river to the Ann Arbor sewage treatment plant. A swab of the internal pipe surface of the sewer line at the point it exits PML was taken. Attempts to obtain sludge (solids) from several locations along this pathway, however, found little sludge available at most locations. The samples,

therefore, mainly consisted of liquid. There were no detectable radionuclides from PML or other licensed radionuclides (see Appendix A, Section 5.2.3.3).

2.1.2.4 Soil Beneath the Reactor Building

Because of the possibility of pool water leakage and the 1993 loss of approximately 7,500 gallons of low-level radioactive water (see Section 2.1.1), an investigation was made of radionuclides from pool water in the soil beneath, and around the reactor building during the characterization study (see Appendix A). During the investigation, three soil borings were made: 1) immediately north of the reactor pool taken through the first floor into the unexcavated area, 2) through the basement floor near the point where the foundation drain tile connects to the cold sump (the source of the 7,500 gallon leak in 1993), and 3) immediately east of the drain tile line just outside the reactor building. This investigation detected only radionuclides normally present in background soil and tritium (14.5 picocuries [pCi] per gram of soil) in the topmost foot of soil taken from immediately north of the reactor pool (see Appendix A, Section 5.2.3.1). This tritium concentration is well below the EPA trigger level of 228 picocurie per gram called out in the memorandum of understanding between the NRC and the EPA (EPA, 2002) .

2.1.2.5 Groundwater

Because of the 7,500 gallons of low-level radioactive water in 1993, and the possibility of leaks from the reactor pool itself, an investigation was made of radionuclides in the groundwater near the reactor building. Since a previous monitoring well in this location was decommissioned because it dried up, a new ground water monitoring well was established in April 2003 immediately south of PML. Sampling of this well found, with the exception of 333 pCi per liter of tritium (EPA's MCL is 20,000 pCi per liter), there were no detectable radionuclides from pool water other than those present in background water samples (see Appendix A, Section 5.2.3.6).

2.1.3 Radionuclides

There have been numerous potential radionuclides associated with the FNR since operations began in 1957. The potential radionuclides are a direct result of reactor operations as well as experiments performed over the years and are listed below. Several of these radioisotopes have short half-lives.

The potential radionuclides are shown in Table 2-4 and have been collected through research of FNR historical documents and interviews with knowledgeable personnel.

Sampling of material from accessible areas in the FNR has identified Cobalt-60 and Cesium 137 as the dominant contaminants with smaller amounts of numerous other activation and fission products. There does not appear to be a uniform radionuclide mix. At the time decommissioning is expected to begin, the radionuclides remaining are expected to be: Antimony-125 (Sb-125), Carbon-14 (C-14), Cesium-137 (Cs-137), Cobalt-60 (Co-60), Europium-152 (Eu-152), Europium-154 (Eu-154), Iron-55 (Fe-55), Manganese-54 (Mn-54), Nickel-63 (Ni-63), Silver-110m (Ag-110m), and Zinc-65 (Zn-65). Co-60 and Cs-137 are expected to be the dominant radionuclide mix at the start of decommissioning because of their current levels and longer half-lives.

TABLE 2-4, LIST OF POTENTIAL RADIONUCLIDES

Nuclide	Half-Life (yr)	Decay Mode	Notes
Antimony-125 (Sb-125)	2.8	β^{-}, γ	AP; from n-activation of materials containing tin
Bismuth-210m (Bi-210m)	3.0×10^{6}	α, γ	AP; from n-activation of SS hardware
Cadmium-109 (Cd-109)	1.26	ε, γ	AP; from n-activation of cadmium metal or materials containing cadmium
Carbon-14 (C-14)	5.73×10^{3}	β^{-}	AP; from n-activation of graphite or materials containing carbon
Cesium-134 (Cs-134)	2.1	β^{-}, γ	AP; from n-activation of cesium, FP; minor FP inventory constituent
Cesium-137 (Cs-137)	30.2	β^{-}, γ	FP; expected to be predominant FP species present
Cobalt-60 (Co-60)	5.3	ε, β^{-}, β^{+}, γ	AP; from n-activation of SS hardware; expected to be predominant AP species present
Europium-152 (Eu-152)	13.5	β^{-}, γ	AP; from n-activation of europium, FP
Europium-154 (Eu-154)	8.5	β^{-}, γ	AP; from n-activation of europium, FP 1
Iron-55 (Fe-55)	2.7	ε	AP; from n-activation of SS hardware or materials containing iron
Manganese-54 (Mn-54)	0.86	ε, γ	AP; from n-activation of SS hardware
Nickel-59 (Ni-59)	7.5×10^{4}	ε, γ	AP; from n-activation of SS hardware
Nickel-63 (Ni-63)	100	β^{-}	AP; from n-activation of SS hardware
Scandium-46 (Sc-46)	0.23	β^{-}, γ	AP; from n-activation of materials used in testing/experiments
Silver-108m (Ag-108m)	127	ε, γ	AP; from n-activation of materials containing silver
Silver-110m (Ag-110m)	0.68	β^{-}, γ	AP; from n-activation of materials containing silver
Tritium (H-3)	12.3	β^{-}	AP; from n-activation of water and from shield tank
Zinc-65 (Zn-65)	0.67	ε, β^{+}, γ	AP; from n-activation of SS hardware

Symbols/Abbreviations: β^{-} = Beta, β^{+} = Positron, ε = Electron Capture, γ = Gamma-Ray, AP = Activation Product, FP = Fission Product, SS = Stainless Steel

Note: The list of potential radionuclides provided above is based on the assumption that operations of the FNR have resulted in the neutron activation of reactor core components and other integral hardware or structural members that were situated adjacent to, or in close proximity to, the reactor core during operations. Specific items, which are considered to have been exposed to neutron activation, include materials composed of aluminum, steel, stainless steel, graphite, cadmium, lead, concrete, and possibly others. Neutron activation of materials beyond the concrete liner/biological shield structure (i.e., into surrounding soil volumes) is not expected for the FNR based on earlier studies, experience from similar research reactor decommissioning projects, reactor-specific calculations that considered measured values for neutron leakage fluence, integrated operating power histories, reactor core/pool structural configurations, and material composition of pool structures.

2.1.4 Cleanup or Decontamination Already Completed

Throughout the operational history of the facility, FNR personnel have followed policies on decontaminating the facility on an ongoing basis to maintain contamination levels as low as possible. Surveys of the reactor building were conducted regularly to identify areas where low levels of radioactive contamination existed. The standard practice is to recommend the decontamination of areas where smearable contamination greater than three times background (200 disintegrations per minute [dpm] above background measured on a liquid scintillation counter) is identified and to require the decontamination of areas where smearable contamination greater than 10 times background (667 dpm above background measured on a liquid scintillation counter) is identified. Drain lines were flushed periodically to the sumps to remove accumulations of dirt and materials that could collect in the lines. Sediment from the sumps was removed periodically to reduce the potential for contamination spread.

Cleanup of the facility began in 1998 with the implementation of a strategic plan to revitalize the facility. Old experiments, excess equipment, and materials were decontaminated or disposed of if no longer required by the facility. This concentrated effort to renovate the facility included:

- Decontamination and painting of the east wall of the reactor pool
- Cleanup and renovation of the offices and hallway on the second floor
- Power washing and scrubbing of the reactor basement, the first floor, Room 2111, the pool floor, and all the stair wells
- Disassembly and removal of the experiment hut from the first floor around the reactor pool
- Removal of all old experimental facilities from inside the beam ports on the east side of the first floor as well as the associated equipment and shielding
- Removal of sediment from many of the sumps and basins
- Elimination of the storage shelves and work benches on the third floor around the reactor pool

Clean up activities resulted in the processing, packaging and shipment of approximately 96,000 lbs of dry active waste, including resins, irradiated hardware, mixed waste, sealed sources, construction materials, etc. in 531 containers to a licensed waste processor or licensed waste disposal facility. Also, with the aid of a licensed waste broker, the facility loaded and shipped for direct burial a high integrity container of mixed waste (irradiated beryllium reflectors) and a liner containing 1,300 pounds of irradiated hardware exhibiting higher levels of radioactivity to Barnwell, South Carolina. Additionally, the facility staff successfully loaded and shipped, without the assistance of outside contractors or consultants, large quantities of high level waste (irradiated reactor fuel) to the Department of Energy's Savannah River facility." Since the FNR
ceased operating in July 2003, the staff has worked diligently to cleanup and learn more about 1
the facility. These efforts have included:

- Unloading of several collimators from the beam ports
- Opening and inspecting the thermal column
- Removing all of the experimental facilities from the first floor of the reactor building
- Segregating and characterizing many of the experimental facilities and irradiated hardware in the reactor pool towards the goal of obtaining approval for direct burial at the Barnwell, South Carolina facility
- Eliminating, to the maximum extent possible, flammable materials and hazardous materials from the reactor building
- Characterizing the lead content of the painted surfaces in critical areas of the facility
- Transferring equipment to other licensees including other research reactors
- Processing, packaging and shipping low level radioactive waste from operations and activities listed above to a licensed waste disposal facility

Activities such as these continue.

2.1.5 Remediation Criteria

The decontamination decommissioning alternative that has been proposed in this decommissioning plan does not require the dismantling of the FNR building. The results of the site and facility radiological characterization indicate that the building structure may be directly releasable without the need for extensive decontamination. This section provides the specific radiological criteria that will be applicable for unrestricted release of the FNR building and termination of NRC license R-28.

2.1.5.1 General

To terminate the license the residual radioactivity from licensed materials must be reduced to levels shown to demonstrate that the site meets the criteria contained in 10 CFR 20, Subpart E, *Radiological Criteria for License Termination*. The criteria for residual radioactivity from licensed material for an unrestricted use are: 1) The Total Effective Dose Equivalent (TEDE) from residual radioactivity that is distinguishable from background radiation must not be greater than 25 mrem per year, and 2) residual radioactivity levels must be as low as reasonably achievable (ALARA).

This section communicates the cleanup criteria the UM intends to use during remediation activities under the decommissioning plan. The actual release criteria for license termination will be detailed in the Final Status Survey Plan, which will be submitted to the NRC in a separate license amendment request during remediation activities. UM may revise these cleanup criteria as necessary to ensure that the 10 CFR 20, Subpart E for both the 25 mrem per year criterion and the ALARA criterion can be achieved and verified by the Final Status Survey. These revisions may become necessary as full knowledge of the radionuclide mixture, levels of contamination, and levels of activation are not known for all areas of the facility. The complete methodology that the UM will utilize in satisfying 10 CFR 20 Subpart E will be detailed through an amendment requesting the approval of the Final Status Survey Plan (see Section 4.0 for a preliminary description of the Final Status Survey Plan).

2.1.5.2 Surfaces

For remediation activities, the criteria for residual radioactive material contamination on FNR facility surfaces and in facility soil, also referred to as derived concentration guideline levels (DCGLs), are selected from the tables of NRC default screening values (refer to NUREG-1757). The screening values for total structure surface contamination are listed in Table 2-5; guideline levels for removable activity are 10 percent of the values in that table. These default screening levels have been conservatively evaluated by the NRC as satisfying the goal that estimated doses to facility occupants and the public during future facility use do not exceed 25 mrem annually; default screening criteria are based on conservative exposure scenario and pathway parameters and are generally regarded as providing a high level of confidence that the annual dose limits will not be exceeded. These screening values are applicable where it can be demonstrated that the contaminant is surface only and non-volumetric (<10 mm in depth).

Characterization surveys have identified multiple radionuclide contaminants on surfaces and in various media at FNR. Predominant contaminants anticipated at the time of license termination are Co-60 and Cs-137; however, additional fission and activation products are present on some surfaces – generally at lower concentrations and at spotty distributions. Variable radionuclide mixtures are also present on many surfaces. For surfaces, concentrations of specific significant contaminants and ratios to their respective DCGLs will be determined to demonstrate satisfying the Unity Rule as described in section 4.3.3 of MARSSIM (NRC 2004); gross beta measurements will be used to demonstrate compliance with surface activity guidelines, and the gross beta

DCGL will be based on measurements of surrogate contaminants with known relationships to the total contamination mix.

The criteria described in this section are net (above background) concentrations and activity levels of radionuclides; appropriate adjustments for instrument background levels and naturally occurring radionuclide concentrations in various media will be made to survey data before comparing data to the respective criteria.

Because of the conservatism used in the development of the default screening values, further evaluations and actions relative to demonstrating that the final conditions satisfy ALARA are not required.

TABLE 2-5, ACCEPTABLE LICENSE TERMINATION SCREENING VALUES OF COMMON RADIONUCLIDES FOR SURFACE STRUCTURES (NRC 1998) [a]

Radionuclide	Symbol	Acceptable Screening Levels[1,2] for Unrestricted Release (dpm/100 cm^2)[3]
Hydrogen-3 (Tritium)	^{3}H	1.2E+08
Carbon-14	^{14}C	3.7E+06
Sodium-22	^{22}Na	9.5E+03
Sulfur-35	^{35}S	1.3E+07
Chlorine-36	^{36}Cl	5.0E+05
Manganese-54	^{54}Mn	3.2E+04
Iron-55	^{55}Fe	4.5E+06
Cobalt-60	^{60}Co	7.1E+03
Nickel-63	^{63}Ni	1.8E+06
Strontium-90	^{90}Sr	8.7E+03
Technetium-99	^{99}Tc	1.3E+06
Iodine-129	^{129}I	3.5E+04
Cesium-137	^{137}Cs	2.8E+04
Iridium-192	^{192}Ir	7.4E+04

Notes:

1 Screening levels presented here are taken from *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination*. NRC (1998). Site specific screening levels will be developed for the project in the manner described in that reference

2 Screening levels are based on the assumption that the fraction of removable surface contamination is equal to 0.1. For cases when the fraction of removable contamination is undetermined or higher than 0.1, users may assume for screening purposes that 100 percent of the surface contamination is removable, and therefore the screening levels should be decreased by a factor of 10. Users may calculate site-specific levels using available data on the fraction of removable contamination and DandD version 2.

3 Units are disintegrations per minute (dpm) per 100 square centimeters (dpm/100 cm^2). One dpm is equivalent to 0.0167 Becquerel (Bq). Therefore, to convert to units of Bq/m^2, multiply each value by 1.67. The screening values represent surface concentrations of individual radionuclides that would be deemed in compliance with the 0.25 mSv/yr (25 mrem/yr) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies (see Part 20, Appendix B, Note 4).

Symbols/Abbreviations: dpm - disintegrations per minute, cm^2 - square centimeter, Bq – Becquerel, m^2 - square meter, mSv – millisievert, mrem – millirem, yr - year.

2.1.5.3 Surface Soil and Sediment

For remediation activities, the criteria for residual radioactive material contamination in surface soil (top 15 cm of soil) under or near the FNR facility or sediments, DCGLs, are selected from the tables of NRC default screening values (refer to NUREG-1757). The screening values for contaminants in soil are listed in Table 2-6. These default screening levels are based on assurance that estimated doses to facility occupants and the public during future facility use do not exceed 25 mrem annually; default screening criteria are based on conservative exposure scenario and pathway parameters and are generally regarded as providing a high level of confidence that the annual dose limits will not be exceeded.

Characterization surveys have identified multiple radionuclide contaminants on surfaces and in various media at FNR. Predominant contaminants anticipated at the time of proposed license termination are Co-60 and Cs-137; however, additional fission and activation products are present in some media – generally at lower concentrations and at spotty distributions. Variable radionuclide mixtures are also present for different media. Concentrations of specific significant contaminants and ratios to their respective DCGLs will be determined in a manner satisfying the Unity Rule, as described in Section 4.3.3 of MARSSIM (NRC, 2000b).

The criteria described in this section are net (above background) concentrations and activity levels of radionuclides; appropriate adjustments for instrument background levels and naturally occurring radionuclide concentrations in various media will be made to survey data before comparing data to the respective criteria.

Because of the conservatism used in establishing the default screening values, further evaluations and actions relative to demonstrating the final conditions satisfy ALARA are not required.

TABLE 2-6, ACCEPTABLE LICENSE TERMINATION SCREENING VALUES OF COMMON RADIONUCLIDES FOR SURFACE SOIL (2 PAGES)

Radionuclide	Symbol	Surface Soil Screening Values[1,2]
Hydrogen-3	^{3}H	1.1E+02
Carbon-14	^{14}C	1.2E+01
Sodium-22	^{22}Na	4.3E+00
Sulfur-35	^{35}S	2.7E+02
Chlorine-36	$^{36}C1$	3.6E-01
Calcium-45	^{45}Ca	5.7E+01
Scandium-46	^{46}Sc	1.5E+01
Manganese-54	^{54}Mn	1.5E+01
Iron-55	^{55}Fe	1.0E+04
Cobalt-57	^{57}Co	1.5E+02
Cobalt-60	^{60}Co	3.8E+00
Nickel-59	^{59}Ni	5.5E+03
Nickel-63	^{63}Ni	2.1E+03
Strontium-90	^{90}Sr	1.7E+00
Niobium-94	^{94}Nb	5.8E+00
Technetium-99	^{99}Tc	1.9E+01
Iodine-129	^{129}I	5.0E-01
Cesium-134	^{134}Cs	5.7E+00
Cesium-137	^{137}Cs	1.1E+01

TABLE 2-6, ACCEPTABLE LICENSE TERMINATION SCREENING VALUES OF COMMON RADIONUCLIDES FOR SURFACE SOIL (2 PAGES)

Radionuclide	Symbol	Surface Soil Screening Values[1,2]
Europium-152	^{152}Eu	8.7E+00
Europium-154	^{154}Eu	8.0E+00
Iridium-192	^{192}Ir	4.1E+01
Lead-210	^{210}Pb	9.0E-01
Radium-226	^{226}Ra	7.0E-01
Radium-226+C[3]	^{226}Ra+C	6.0E-01
Actinium-227	^{227}Ac	5.0E-01
Actinium-227+C	^{227}Ac+C	5.0E-01
Thorium-228	^{228}Th	4.7E+00
Thorium-228+C	^{228}Th+C	4.7E+00
Thorium-230	^{230}Th	1.8E+00
Thorium-230+C	^{230}Th+C	6.0E-01
Thorium-232	^{232}Th	1.1E+00
Thorium-232+C	^{232}Th+C	1.1E+00
Protactinium-231	^{231}Pa	3.0E-01
Protactinium-231+C	^{231}Pa+C	3.0E-01
Uranium-234	^{234}U	1.3E+01
Uranium-235	^{235}U	8.0E+00
Uranium-235+C	^{235}U+C	2.9E-01
Uranium-238	^{238}U	1.4E+01
Uranium-238+C	^{238}U+C	5.0E-01
Plutonium-238	^{238}Pu	2.5E+00
Plutonium-239	^{239}Pu	2.3E+00
Plutonium-241	^{241}Pu	7.2E+01
Americium-241	^{241}Am	2.1E+00
Curium-242	^{242}Cm	1.6E+02
Curium-243	^{243}Cm	3.2E+00

Notes:

1 These values represent surficial surface soil concentrations of individual radionuclides that would be deemed in compliance with the 0.25 mSv/yr (25 mrem/yr) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies; see Part 20, Appendix B, Note 4.

2 Screening values are in units of (pCi/g) equivalent to 0.25 mSv/yr (25 mrem/yr). To convert from pCi/g to units of Becquerel per kilogram (Bq/kg), divide each value by 0.027. These values were derived using DandD screening methodology (NUREG/CR-5512, Volume 3, *Residual Radioactive Contamination for Decommissioning* [NRC, 1992b]). They were derived based on selection of the 90th percentile of the output dose distribution *for each specific radionuclide* (or radionuclide with the specific decay chain). Behavioral parameters were set at "Standard Man" or at the mean of the distribution for an average human.

3 "Plus Chain (+C)" indicates a value for a radionuclide with its decay progeny present in equilibrium. The values are concentrations of the parent radionuclide but account for contributions from the complete chain of progeny in equilibrium with the parent radionuclide (NUREG/CR-5512, Volumes 1, 2, and 3, *Residual Radioactive Contamination for Decommissioning* [NRC, 1992b]).

Symbols/Abbreviations: Bq – Becquerel, g – gram, mSv – millisievert, mrem – millirem, pCi – picocuries, yr - year.

2.1.5.4 Subsurface and Inaccessible Structures

The criteria for residual radioactive contamination on FNR facility surfaces discussed in Section 2.1.5.2 are not applicable for surfaces where the contaminant is non-surface (> 10 mm in depth), activated surfaces, surfaces where the contamination is volumetric, inaccessible areas besides buried pipes, etc. The criteria for residual radioactive material contamination of these items which remain following remediation must be developed, based upon characterization results not yet obtained because of limitations caused by the past and current status of the facility, utilizing RESRAD-BUILD (C. Yu et. al. 2003) or equivalent methodology. The criteria shall be developed to ensure that estimated doses to facility occupants and the public during future facility use do not exceed 25 mrem annually. The criteria shall be developed based on a conservative exposure scenario and pathway parameters which will provide a high level of confidence that the 25 mrem annual dose limit will not be exceeded.

Characterization surveys have identified multiple radionuclide contaminants on surfaces and in various media at FNR. Predominant contaminants anticipated at the time of proposed license termination are Co-60 and Cs-137; however, additional fission and activation products are present in some media – generally at lower concentrations and at spotty distributions. Variable radionuclide mixtures are also present for different media. Concentrations of specific significant contaminants and ratios to their respective DCGLs will be determined in a manner satisfying the Unity Rule, as described in Section 4.3.3 of MARSSIM (NRC, 2000b).

10 CFR 20, Subpart E, *Radiological Criteria for License Termination* also requires that the level of residual radioactivity due to licensed material for release to unrestricted use must be as low as reasonably achievable. The criteria for residual radioactive material contamination of subsurface structures or components within the physical structure of the FNR facility (left after remediation or decontamination) may need to be further reduced to satisfy the ALARA requirement. Reduction of the cleanup criteria for subsurface and inaccessible structures will follow an examination of the reduction in the estimated dose to the facility occupants and the public using the RESRAD-BUILD (C. Yu et. al. 2003) combined with an examination of the costs associated with achieving these reduced levels of residual radioactivity. This evaluation shall be documented in the final report to the NRC.

The criteria described in this section should be net (above background) concentrations and activity levels of radionuclides; appropriate adjustments for instrument background levels and naturally occurring radionuclide concentrations in various media will be made to survey data before comparing data to the respective criteria.

2.1.5.5 Groundwater

For remediation, the criteria for the residual levels of a single licensed material in groundwater will be the radionuclide MCLs established under the Safe Drinking Water Act. If more than one licensed material is identified in the groundwater, then the Unity Rule will be applied to the contaminant levels to ensure compliance with 40 CFR 101 and the Memorandum of Understanding between the NRC and the EPA (EPA, 2002).

The criteria described in this section are net (above background) concentrations and activity levels of radionuclides; appropriate adjustments for instrument background levels and naturally occurring radionuclide concentrations in various media will be made to survey data before comparing data to the respective criteria.

No remediation of groundwater is anticipated.

2.2 Decommissioning Tasks

2.2.1 Decommissioning Preparation

The general dismantling and decontamination activities are discussed below and may not follow the sequence shown for ALARA, safety, accessibility, or scheduling reasons.

2.2.1.1 Characterization Surveys

Characterization studies have been conducted as part of the planning activities for the decommissioning plan. The type, quantity, condition, and location of radioactive and/or hazardous materials that are or may be present in the FNR have been identified. Extensive surveys of accessible areas of the FNR were conducted in September 2002 and April 2003 by CH2M HILL under contract with the UM. Results of these surveys as summarized in the Characterization Report are provided in Appendix A. Additional surveys will be performed in conjunction with the activities discussed below, as previously inaccessible areas are made accessible.

2.2.2 Dismantling and Decontaminating

Upon receiving the license amendment approving the decommissioning plan, the dismantling and decontaminating activities can begin.

Dismantling and decontamination will be required to remove materials that were activated or radiologically contaminated during operation of the FNR in order to meet the unconditional release criteria for license termination. Standard industry dismantling and decontamination techniques using tools such as wire saw, high pressure/ultra-high pressure water, needle guns, jack hammers, torches/plasma arc torches, hydraulic cutters, and hand tools will be used, following approved procedures or work packages.

2.2.2.1 Isolate or Remove Inactive Systems Formerly Important to Safety

All inactive systems or systems not currently required by technical specifications or later decommissioning activities but formerly identified in the Safety Analysis may be inactivated (de-energized and isolated) or removed. Several systems, structures or components are specifically identified here in this section to identify their removal from the facility and to facilitate their removal under the change process proposed in Section 9.0.

2.2.2.1.1. Standby Generator

The standby generator will be isolated and removed from the facility. The standby generator is not required nor is it a safety-related system. The standby generator is a convenient source of backup electrical power that maintains limited lighting, ventilation to the fume hoods, hot cells, and other areas on the first floor of PML, service air to PML and FNR, and other loads. The standby generator is not required to ensure that any safety system, structure, or component performs its intended safety function. The standby generator may be isolated, inactivated or removed as determined by the UM.

2.2.2.1.2. Heavy Water Reflector

The heavy water will be transferred from the heavy water reflector to shipping containers for shipment to the DOE's Savannah River Site. Because of the large quantity of tritium in the heavy water reflector, the quality assurance requirements identified in Section 1.3.4.2 will be applied to the process and equipment utilized in this transfer. The heavy water reflector will be flushed with demineralized water before being processed for disposal.

2.2.2.1.3. Spent Fuel Storage Racks

The spent fuel storage racks are no longer required to hold fuel and will be prepared for disposal.

2.2.2.1.4. Pneumatic Tube System – External to the Reactor Pool

The portions of the pneumatic tube system external to the reactor pool will be isolated, removed and surveyed for release or prepared for disposal. The termination of the lines from the reactor pool will be subject to the quality assurance requirements identified in Section 1.3.4.2.

2.2.2.1.5. Secondary Cooling System

The cooling tower will be isolated, dismantled, and surveyed for release. The transite panels on the sides of the cooling tower, not expected to be contaminated, will be disposed through an appropriate waste handler. The secondary cooling system piping will be isolated mechanically from the primary cooling system (possibly utilizing blank flanges at the inlet of the heat exchanger since the valves are known to leak) subject to the quality assurance requirements identified in Section 1.3.4.2. The secondary cooling system penetrations in the reactor building envelope will also be plugged subject to the quality assurance requirements identified in Section 1.3.4.2 to ensure confinement is maintained until it is no longer needed. Once isolated the secondary cooling system piping will be removed and surveyed for release or disposal. The staff of the FNR is currently working with another research reactor to receive the heat exchanger.

2.2.2.1.6. Emergency Cooling System

The 4-inch water main to the reactor pool, intended for emergency fill of the reactor pool following a loss-of-coolant accident will be isolated, removed, and surveyed for release or disposal. Upon the removal of a portion of the shield water to the reactor pool, the reactor pool can be refilled using the fire hose stations through out the building or the normal makeup water systems. The removal of the emergency cooling system line will be subject to the quality assurance requirements identified in Section 1.3.4.2.

2.2.2.1.7. Control Console

The control console supports the rod control system, the reactor annunciator system, the SCRAM function and the automatic rod insertion function. These systems are required only to operate the reactor (currently prohibited by the possession only license). Unless authorized for removal by an earlier license amendment, the reactor console will be isolated and removed by the reactor staff. Power to systems required by the technical specifications, the current licensing basis, or later decommissioning activities that are currently fed from the control console will be relocated to facilitate the removal of the control console. The UM may restore the console for a historical display.

2.2.2.1.8. Exhaust for Hood in Room 3103

The exhaust for the fume hood in Room 3103 is no longer required and will be removed. The exhaust line penetration in the building envelope into PML will be plugged subject to the quality assurance requirements identified in Section 1.3.4.2 to ensure confinement is maintained until it is no longer needed. The fume hood in Room 3103 and all associated ducts to the circular duct sleeved through the wall between the reactor building and PML will be removed and surveyed for release or disposal.

2.2.2.1.9. Exhaust for Pneumatic Blowers, First Floor Trunks around Pool, and Storage Ports

The exhaust duct for the trunk to the pneumatic blowers in the reactor basement, the trunks on the first floor located around the pool and the storage ports will be removed when no longer needed. Exhaust from the pneumatic tube system is no longer required as the system does not produce or contain airborne radioactive materials. The exhaust trunks around the reactor pool were in support of experiments utilizing the beam ports and the storage ports in the west wall contain only activated solids, mostly metal, which do not generate gaseous or airborne radioactivity. The exhaust line penetration in the building envelope into PML will be plugged subject to the quality assurance requirements identified in Section 1.3.4.2 to ensure confinement is maintained until it is no longer needed. Following this isolation, all associated ducts in the system up to the circular duct sleeved through the wall between the reactor building and PML will be removed and surveyed for release or disposal.

2.2.2.1.10. Beam Port Extensions

As discussed previously, the beam ports were built in two sections connected by a split clamp. The split clamps will be loosened and removed; then the beam port extension can be removed and a blank flange installed. A port plug, using an o-ring seal to maintain pool integrity, will be installed in the beam port. The removal of the port extensions will allow for their characterization and processing for disposal. This activity will be subject to the quality assurance requirements identified in Section 1.3.4.2.

2.2.2.2 Isolate, Remove or Inactivate Other Systems

All inactive systems or systems not previously required by technical specifications, the current licensing basis, or later decommissioning activities may be inactivated (de-energized and isolated) or removed. Examples of systems that may be isolated include potable water line, drain lines to the hot or cold sump, reactor air to miscellaneous supplies, gaseous nitrogen supply lines, and the demineralized water supply to PML. Systems interfacing with the contiguous wall of the PML will be isolated on the PML side of the interface, when practical. The quality assurance requirements identified in Section 1.3.4.2 will be applied when required.

2.2.2.3 Asbestos Removal

Radioactively contaminated asbestos-containing materials will be removed, packaged, and disposed of in accordance with applicable regulations. Non-contaminated asbestos-containing materials may also be removed, surveyed and disposed of in accordance with applicable regulations.

2.2.2.4 Temporary Systems

Temporary systems may need to be installed to support decommissioning activities. These may include additional electrical outlets for temporary ventilation or decontamination equipment, water purification system to purify or decontaminate liquids, openings in the reactor building

for equipment access or waste removal, waste storage and handling systems or equipment, service air, potable waste, fire detection, fire hose stations, etc.

2.2.2.5 Items and Materials in the Reactor Pool

2.2.2.5.1. Characterize

The radioactivity associated with the high dose items (the reactor grid, shim and control rods, beam port extensions, etc.) needs to be estimated when these items become accessible. Estimates will be determined as described in Section 2.1.2.1.

2.2.2.5.2. Segregate, Reduce and Load

The reactor grid plate, shim and control rods, heavy water reflector, pneumatic tubes, RIFLS, HSSI experiment, and remaining miscellaneous high-dose items will require size reduction to facilitate loading into high integrity containers (HIC) or, for inherently stable items, a liner. Long-reach tools, remotely operated equipment, human divers, or a combination of these techniques may be used to size-reduce and load the HIC or liner.

The water in the reactor pool is planned for use as shielding and contamination control during high-dose item size reduction and removal activities. However, it may be necessary to lower the water level or drain the pool to remove items such as the pneumatic tube bundle penetration that could, upon removal, introduce a potential pool drainage pathway. If the pool water levels are lowered, shielding or remote size reduction techniques may be required to maintain personnel exposure ALARA.

High dose items such as the shim and control rods, RIFLS, HSSI experiment, etc. may be transferred to the hot cells in PML for size reduction. High dose items may also be transferred and loaded dry into the HIC or liner using shields.

Once the high dose items are loaded into the HIC or liner, the HIC or liner is then placed in an approved, shielded shipping cask for transport to an approved disposal site. The HIC or liner should be directly loaded into a shipping cask submerged in the reactor pool (similar to the way staff loaded and shipped irradiated fuel elements) whenever the size of the cask permits. The HIC or liner may require indirect loading using a shielded transfer cask if the size of the cask or other factors prohibits loading in the reactor pool.

2.2.2.5.3. Drain Pool and Piping

The water in the reactor pool will be disposed of when no longer useful as a radiological shield. Liquid from the pool and piping will be filtered and treated --as necessary to meet discharge requirements of the technical specifications, federal, state, and local laws-- then discharged to the City of Ann Arbor sanitary sewer using approved procedures. Liquid not meeting release criteria will be treated, stabilized, and packaged to meet waste acceptance criteria at an approved disposal site.

2.2.2.5.4. Characterize Reactor Pool

Following draining of the pool, the structure will be characterized to determine the extent and depth of activation and contamination in the reactor pool floor, pool walls, and embedded beam port tubes. The characterization results may be used by the UM to select either the pool removal or pool decontamination pathway for decommissioning based on ALARA, safety, structural, cost, schedule and future use considerations.

2.2.2.5.5. Pool Removal/Decontamination

Future plans for the reactor building require the elimination of the reactor pool from the building. Considering the success Battelle Memorial Institute achieved during the removal of a similar reactor pool from their reactor facility in West Jefferson, Ohio, and the lessons learned by UVa during the remediation of their reactor pool, the UM has elected to remove those portions of the reactor pool that may not be readily remediated. Contingent upon the results of the reactor pool characterization, the reactor pool walls and possible portions of the reactor pool floor will be cut into large blocks, prepared and disposed as radioactive waste. Materials embedded in the concrete (beam port tubes, drain pipes, conduit, tile, etc) may not be removed unless it is necessary to meet transportation requirements and the disposal site waste acceptance criteria.

If decontamination of the reactor pool or a portion of the reactor pool is elected for the decommissioning pathway, then pool surfaces will be decontaminated and the activated concrete necessary to achieve termination of the license will be removed. Core bore samples will be taken to evaluate subsurface contamination. Subsurface contamination identified through examination of surface cracks or voids or located by the core bores will be remediated. Contaminated or activated embedded pipes will be decontaminated or removed. Contaminated waste will be packaged and shipped to a licensed disposal site.

2.2.2.6 Decontaminate/Remove Embedded Pipes

Contaminated pipes, drains, and conduit embedded in concrete will be decontaminated or removed. Sludge, scale and other waste generated will be treated or stabilized and packaged to meet the disposal site waste acceptance criteria. Decontamination liquid will be discharged to the sanitary sewer if it meets the technical specifications, federal, state and local requirements for discharge to the sewer. If it does not meet discharge requirements then it will be handled like sludge and treated or stabilized for disposal.

2.2.2.7 Surface and Subsurface Soil Sampling

Sufficient soil samples from unexcavated areas beneath and west of the pool will be taken to determine if an unknown leak in the pool contaminated the soil surrounding the pool. Any holes drilled through the concrete will be sealed or plugged to prevent the hole from becoming a potential pathway to the environment.

2.2.2.8 Decommission and Decontaminate Contaminated Equipment

Contaminated equipment from each floor of the FNR will be removed or decontaminated. Equipment; piping; heating, ventilation, and air conditioning (HVAC); and electrical and instrument systems interfacing with PML or FNR systems remaining in place will be isolated to reduce the potential for accidental releases of water or energy. Contaminated systems interfacing with the contiguous wall of the PML will be removed or decontaminated on the PML side of the interface. The following are examples of equipment that may need to be decontaminated or removed:

- Basement - Primary coolant piping and instrumentation, hold-up tank, primary pump and motor, ion exchange piping and system, and hot and cold sump pumps and motors.
- First Floor – HVAC ducts, source storage ports, transfer chute, thermal column and thermal column door, drain lines and piping not embedded in concrete.
- Second Floor – HVAC equipment, ducts and butter-fly valves.

- Third Floor – Reactor bridge, remaining reactor suspension frame, pool and reactor instrumentation, heavy water reflector support equipment, HVAC, drain lines and piping, pool filter/vacuum system, and any miscellaneous low dose items in or attached to the pool.

  Note: The UM is looking to restore the reactor bridge for a historical display.

- Fourth Floor – Crane over the pool, HVAC (all not expected to be contaminated).

2.2.2.9 Decontaminate and Survey Remaining Areas

Any remaining contaminated areas within the FNR will be decontaminated or removed and surveyed to confirm the area has been decontaminated to levels that will meet unconditional release criteria. Areas that may require decontamination, in the basement, include the concrete floor, hot and cold sumps, hold-up tank pit, ion exchange pit, and the walls. On the first floor, remaining areas that may require decontamination are the floor, wall by the source storage ports, and thermal column door trench. On the third floor, contaminated areas may include the laboratories, floor around the pool, and the south wall. Decontamination is not expected to be required on the second and fourth floors.

Waste generated during this activity will be packaged and disposed at a licensed disposal site.

2.2.2.10 Soil and Buried Pipe Remediation

If contaminated soil is identified and the source of the contamination is from the FNR, it will be evaluated against the release criteria and ALARA requirements. If contamination levels require removal, it will be removed, packaged, and disposed of at an approved disposal site. If buried pipes (e.g., drain tiles) are found to be radiologically contaminated and cannot be decontaminated to meet final release criteria, they will be remediated, packaged, and disposed of at a licensed disposal site.

Final release samples will be taken after the soil or buried pipes have been remediated. The excavation will remain open to permit the NRC to perform confirmatory surveys or sampling. Split samples will be taken before backfilling the excavation and saved or provided to the NRC if the excavation must be backfilled for safety reasons. The NRC will also be notified of the expected completion date of the remediation so that a representative is afforded the opportunity to be present to obtain verification samples. Once NRC concurrence is received, the excavation will be backfilled to reduce any potential safety hazard.

The assumption that neutron activation of the soil beneath the reactor pool did not occur will be confirmed by examining the activation of the concrete floor in the void directly beneath the reactor core which is accessible from the reactor basement (See Figure 1-8 and Figure 1-13).

2.2.2.11 Decontaminate Portions of PML

Upon termination of the reactor license, the areas in PML where radioactive materials were introduced by reactor operation and could remain will be regulated by one of the NRC material licenses currently held by the UM. Such areas in PML where radioactive contamination is potentially greater than 20 times background will be decontaminated in preparation for transfer to these licenses. Additional areas requiring decontamination may be identified by the UM Radiation Safety Officer (RSO). These areas include, but are not limited to:

- Retention tanks and retention tank pit

- Storage ports and drawers in the north west corner of room 1069D;

- North and south hot cells
- Room 1069D including the freight elevator and freight elevator pit
- Room 1069
- Stack 2 exhaust system

Every reasonable effort will be made to reduce the removable radioactive contamination in these areas to 20 times background (this is the action level specified in the UM Broad Scope Materials License 21-00215-01, Docket 030-01988). Areas where the UM RSO determines the residual radioactive contamination cannot be reduced to at or below these levels will remain a restricted area under the applicable NRC license(s) as defined by 10 CFR 20.

2.2.2.12 Survey and Update UM Licenses

A survey and inventory of decontaminated areas of PML will be performed to demonstrate that the radiological condition and radioactive materials present in PML are within the limits of one of the NRC materials licenses held by the UM. If the quantities and isotopes of materials identified by the survey and inventory are not fully encompassed by one of the NRC licenses, then decontamination will continue. Alternatively, an amendment to the applicable license will be sought from Region III of the NRC.

2.2.2.13 Final Radiation Survey

Following decontamination and remediation activities of the FNR, a final radiation survey will be performed covering the entire FNR facility. The final radiation survey, executed according to the approved Final Status Survey Plan, will document that release criteria have been met.

2.2.2.14 Final Release Report

Once all decontamination has been performed and verified through final radiation surveys, a final release report will be developed. This report will record the decontamination and remediation activities performed and document the final radiological status of the FNR facility and associated grounds. This final report will be used in part as the basis of the application for license termination.

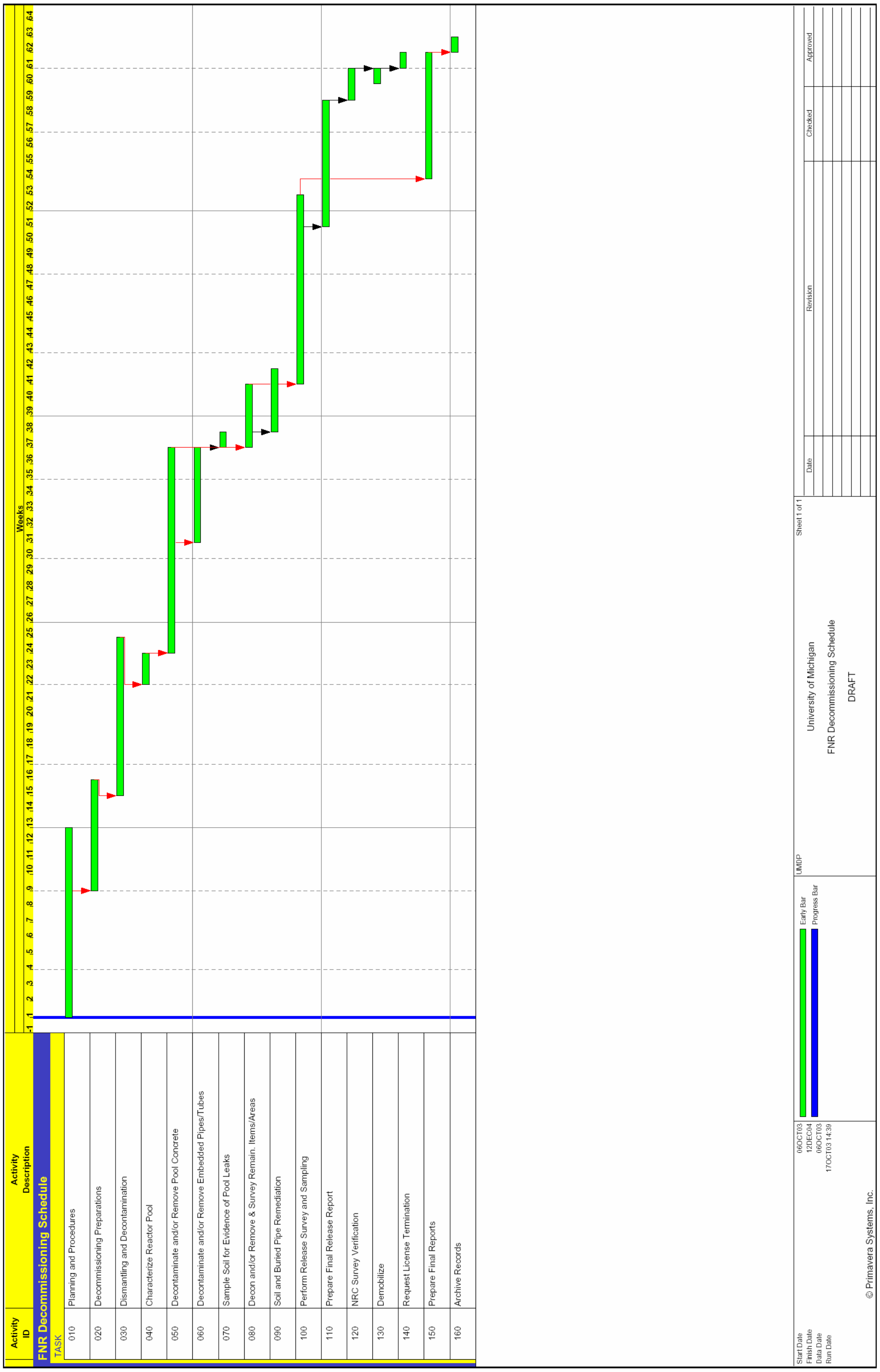
2.2.2.15 Demobilization

On completion of any verification sampling by the NRC or its representative, the site will be demobilized, including back filling of open excavations and removal of temporary systems (e.g., temporary power), trailers, equipment, and storage. Project records will be transferred to the appropriate department or agency and archived.

2.3 Schedule

Figure 2-3 presents the proposed project schedule. The scheduled time, from regulatory approval of the FNR decommissioning plan to submittal for release of the site for unrestricted use, is 15 months. Changes to the schedule may be made at UM's discretion as a result of resource allocation, availability of a radioactive waste burial site, interference with ongoing UM activities, ALARA considerations, further characterization measurements and/or temporary onsite radioactive waste storage operations.

FIGURE 2-3, FNR DECOMMISSIONING SCHEDULE



This project schedule reflects and is consistent with the actual performance of similar activities at the UVa. The decommissioning of the UVa research reactor began in April 2002 and completed remediation in May 2003. The final status survey has been performed and is awaiting NRC's approval.

2.4 Decommissioning Organization and Responsibilities

The Regents of the UM are responsible for overall planning, managing, and financing the decommissioning of the FNR. The Regents are committed to, and retain ultimate responsibility for, full compliance with all applicable licenses or registrations held by the UM and with compliance to applicable federal and state regulatory requirements.

For the nearly 50 years of licensed activities at FNR, the Director of MMPP was responsible for the facility's operating license. However, the bylaws of the Board of Regents specifically assign responsibility for the decommissioning to the Executive Vice President and Chief Financial Officer. The Executive Vice President has established, through the Associate Vice President for Facilities and Operation, a project organization to oversee the decommissioning of FNR as shown in Figure 2-4. The FNR Project Staff is lead by the Director of Occupational Safety and Environmental Health, who will be responsible for the facility's license and shall authorize the expenditure of funds on decommissioning activities. The Reactor Manager remains responsible for decommissioning FNR and assuring that all activities are conducted in a safe manner within the limitations of the facility's license and in compliance with applicable federal and state regulations. The RSO, who is organizationally independent of the Reactor Manager, his support staff or any decommissioning contractors or sub-contractors, remains responsible for radiological safety at the facility and shall be responsible for safeguarding the UM community, the public, and personnel involved in decommissioning from undue radiation exposures. A review committee, chaired by a representative of the Vice President for Research, will monitor decommissioning activities to ensure they are being performed safely, economically, and according to all applicable licenses or registrations held by the UM and in compliance with applicable federal and state regulatory requirements.

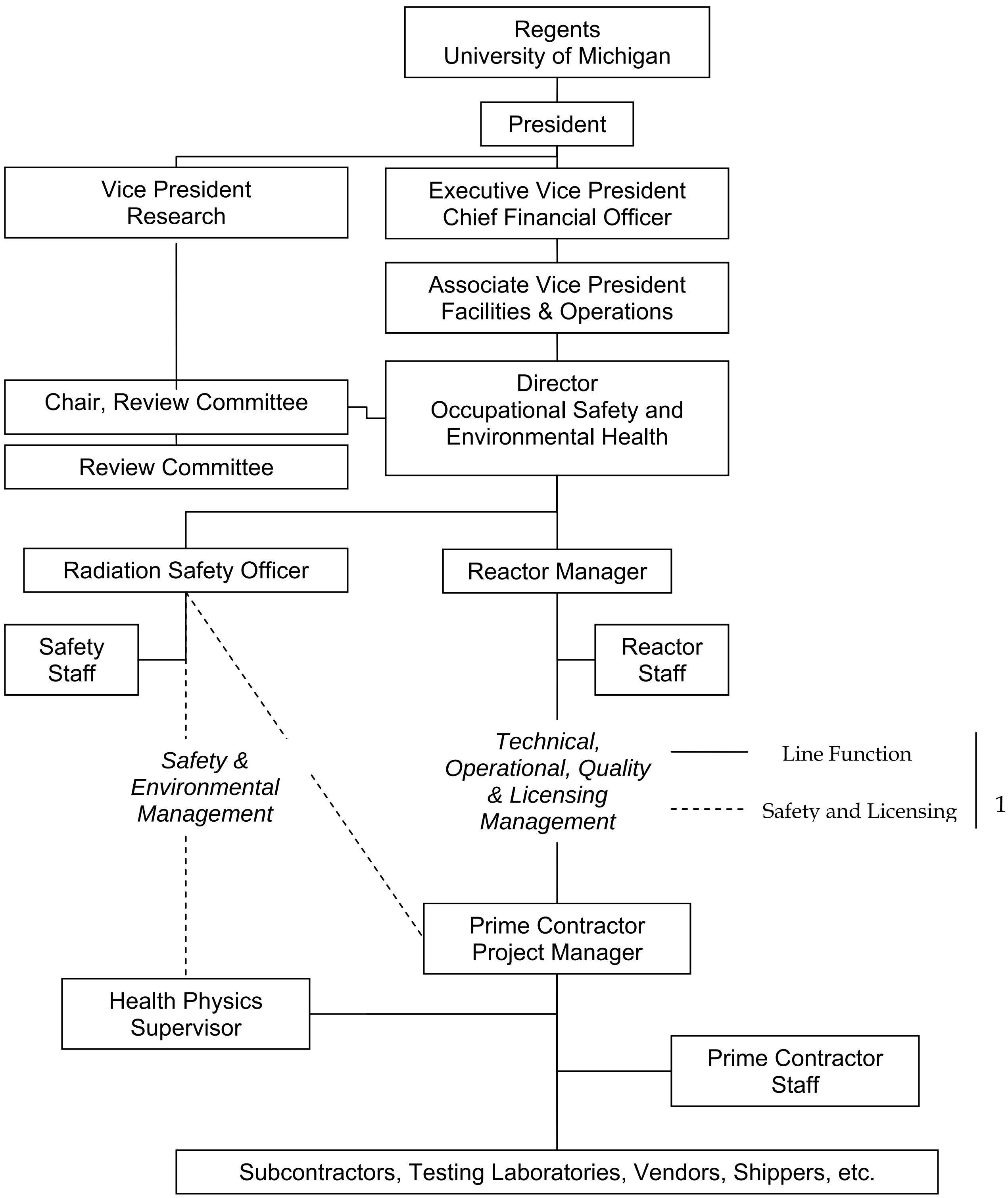
Note: This section of the decommissioning plan forms the basis of the revision to Section 5.1, *Organization*, Section 5.2, *Review*, Section 5.3, *Audit* and Section 5.5, *Procedures* of the Technical Specification amendment request contained in Section 5.2. This description of the facility organization replaces the facility organization description contained in Section 12, *Conduct of Operations* of the Safety Analysis Report and amended by license amendments 33, 34, 43, and 45.

2.4.1 Director of Occupational Safety and Environmental Health

The Director of Occupational Safety and Environmental Health (Director) (Level 1) has oversight authority and is responsible for:

- The facility's license (compliance and amendments)
- Successful completion of decommissioning activities
- Authorizing the expenditure of funds for decommissioning
- Requesting termination of the license for FNR
- Approval of contractors, subcontractors, and consultants
- Approval of budgets and schedules
- Serving as technical spokesman for the UM on decommissioning activities
- Resolving conflicts between the Reactor Manager, RSO and review committee
- Ensuring that the conduct of decommissioning activities complies with all applicable licenses and registrations held by the University and with compliance to applicable federal and state regulatory requirements

FIGURE 2-4, ORGANIZATION CHART FOR THE FNR DECOMMISSIONING PROJECT



The Director shall be advised by a review committee that will review, monitor and audit decommissioning activities and approve changes to the facility's license or the decommissioning plan as described in Section 9.0.

The Director places authority for managing decommissioning activities and directing contractor oversight through his management team consisting of two direct reports, the Reactor Manager and the RSO. The Director may utilize personnel from Architecture & Engineering Services, in a non-license role, to manage contracts, substantial purchasing activities, and project tracking or management.

At the time of appointment to the position, the Director shall receive briefings sufficient to provide an understanding of the decommissioning and licensing aspects of the facility.

2.4.2 Nuclear Reactor Laboratory Manager

The Reactor Manager has responsibility for:

- Controlling and maintaining safety during decommissioning activities and protection of the environment.
- Determining facility staffing and organization.
- Assuring performance to cost and schedule.
- Reporting performance to the Director and the review committee.
- Approving changes to the facility which satisfy the equivalent requirements of 10 CFR 50.59 contained in the license.
- Providing licensing interface with the NRC, EPA (if required), MDEQ, and other regulatory agencies.
- Providing technical oversight and guidance.
- Review of work procedures, radiation work permits (RWPs), and job hazard analyses (JHAs).
- Ensuring that shipments of hazardous materials are prepared and transported safely and in accordance with all applicable regulations and requirements of the receiver.
- Acting as interface between contractor, subcontractors, or consultants and the Director or review committee.
- Coordinating contractor, subcontractor, or consultant activities.
- Resolving facility or site issues.
- Drawing upon other UM engineering, technical, or skilled trade resources as needed.
- Providing technical support to the Director and review committee.
- Ensuring all quality assurance program(s) requirements are effectively implemented by all staff, contractors, and other UM staff supporting decommissioning.
- Approval of those quality assurance and performance elements of the Final Status Survey including but not limited to plans, specifications, designs, procedures, data, and reports not specifically promulgated for approval by the review committee specified in Section 2.4.5. 1
- Investigating adverse monitoring or audit findings, scheduling corrective action, including measures to prevent recurrence of significant conditions adverse to quality, and notifying the Director and each review committee member of action taken or planned or to be taken.
- Assisting the Director in ensuring that decommissioning activities comply with all applicable licenses or registrations held by the UM and with compliance to applicable federal and state regulatory requirements.

The Reactor Manager shall have the authority to enforce safe performance of decommissioning activities and to shut down or suspend any operations or activities because of safety, environmental, licensing or regulatory issues, if immediate corrective action is not taken. Resumption of any activity shut down or suspended by the Reactor Manager shall require the approval of the Director or Reactor Manager.

At the time of appointment to the position, the Reactor Manager shall have a minimum of 6 years of nuclear experience. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job related may be substituted for a degree on a case-by-case basis. The degree may fulfill 4 years of the 6 years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate facility specific training based upon a comparison of the individual's background and abilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required.

UM reserves the right for the Reactor Manager or any facility staff supporting decommissioning to be a contractor or consultant under direct contract with the UM. If the UM elects to use a contractor or consultant as the Reactor Manager, the UM shall determine that the individual selected does not have a conflict of interest with any other contractor, subcontractor or consultant involved in activities for the decommissioning before appointing the contractor or consultant as the Reactor Manager.

2.4.3 Radiation Safety Officer

The RSO is responsible for

- Maintaining the radiation safety and health aspects of programs or procedures and ensuring compliance with programs or procedures.
- Determining facility radiation safety staffing and organization.
- Reviewing work procedures, RWPs, and JHAs where potential radiation exposure or safety could be affected.
- Providing technical support to the Director and review committee.
- Ensuring procedures and practices are established to ensure ALARA is applied to radiation exposures to the public and facility personnel.
- Identifying locations, operations or conditions that have the potential for significant exposures to radiation or radioactive materials and initiating actions to minimize or eliminate unnecessary exposures.
- Monitoring contractor and subcontractor health physics coverage of decontamination and decommissioning activities.
- Monitoring collective dose for decommissioning activities.
- Ensuring the implementation of an industrial safety, industrial hygiene; and environmental protection program which satisfies all applicable licenses, permits, or registrations held by the UM and complies with all applicable federal and state regulatory requirements.

The RSO and any facility radiation safety staff shall have the authority to enforce safe performance of decommissioning and to shut down or suspend operation or activities because of either safety or environmental issues, if immediate corrective action is not taken. Resumption of any activity shut down or suspended by the RSO or a facility radiation safety staffer shall require the approval of the Director or RSO.

At the time of appointment, the RSO shall have a minimum of 6 years of radiation safety experience. The individual shall have a recognized baccalaureate or higher degree in health physics, nuclear engineering, or scientific field. Education or experience that is job related may be substituted for a degree on a case-by-case basis. The degree may fulfill 4 years of the 6 years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate facility-specific training based upon a comparison of the individual's background and abilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required.

UM reserves the right for the RSO or any facility radiation safety staff member to be a contractor or consultant under direct contract with the UM. If the UM elects to use a contractor or consultant as the RSO or a facility radiation safety staff member, the UM shall determine that the individual selected does not have a conflict of interest with any other contractor, subcontractor or consultant involved in activities for the decommissioning before appointing the contractor or consultant as the RSO or a facility radiation safety staff member.

2.4.4 Prime Contractor

The UM shall select a prime contractor to manage and supervise all or part of the FNR Decommissioning Project. In selecting a contractor, the UM will produce a request for proposal to define the qualifications and experience necessary for prospective decommissioning contractors and subcontractors. Prior history and performance of prospective decommissioning contractors and subcontractors on non-power reactor decommissioning projects will be key to assisting the UM. The selected prime contractor will manage and supervise operations and services such as characterization, dismantlement, decontamination, waste handling, quality assurance, etc. The prime contractor shall establish and maintain a Project Manager who will serve as the overall project manager and be a vital member of the project team. The prime contractor shall also establish and maintain a Health Physics Supervisor to be responsible for providing basic radiation safety support for contractor and subcontractors activities.

The UM will select the Prime Contractor through an evaluation of (but not limited to) the following criteria:

- Ability of the firm to perform the required task as demonstrated by the quality of information provided in a Statement of Qualification Package.
- Qualifications of key individuals, including but not limited to the key contractor individuals identified in this section, based upon resume and license.
- Record of the contractor and identified key subcontractors past performance with respect to compliance with all federal, state and local regulations.
- Safety record of the contractor and key subcontractors based on information submitted and a review of past projects.
- Relevant experience of contractor and key subcontractors, particularly with decommissioning of research reactors.
- References from owners and federal, state and local authorities on previous decommissioning projects for which the contractor and key subcontractors participated.
- Review of example work product (RWPs, JHAs, characterization studies, work packages, quality assurance procedures, etc.) provided by the contractor and key subcontractors.
- Financial stability of the contractor and key subcontractors to complete the project and ability to meet the minimum insurance requirements.

The prime contractor may not assign or transfer responsibility for performing all or part of the FNR Decommissioning Project without prior approval of the Director. The prime contractor may retain subcontractors or hire consultants to help in the performance of all or part of the FNR Decommissioning Project with the prior approval of the Director.

2.4.4.1 Project Manager

The Project manager shall:

- Supervise the day-to-day operations of the prime contractor, subcontractors and consultants.
- Provide technical support to the Director, Reactor Manager and review committee.
- Be responsible for ensuring that all contractor personnel involved in decommissioning activities are trained as required by Section 2.5.
- Be responsible for ensuring the safety and health of the prime contractor's employees and the employees of any subcontractors or consultants they retain.
- Ensure that all employees of the prime contractor and any subcontractors or consults they retain comply with the requirements of all applicable licenses or registrations held by the UM and with compliance to applicable federal, state, and local regulatory requirements.

All these efforts will be subject to the review, approval, and authority of the Director, Reactor Manager, or the RSO.

The Project Manager will report to the Reactor Manager.

The Project Manager shall be subject to the approval of the Director.

At the time of appointment to the position, the Project Manager shall have a minimum of 6 years of experience in nuclear power or decommissioning. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job related may be substituted for a degree on a case-by-case basis. The degree may fulfill 4 years of the 6 years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate project specific training based upon a comparison of the individual's background and abilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required.

2.4.4.2 Health Physics Supervisor

The Health Physics Supervisor shall:

- Review work procedures, RWPs, JHAs where potential radiation exposure and safety could be affected.
- Ensure that procedures and practices are established to apply ALARA to radiation exposures to the public and facility personnel.
- Identify locations, operations or conditions that have the potential for significant exposures to radiation or radioactive materials and initiating actions to minimize or eliminate unnecessary exposures.
- Monitor contractor and subcontractor health physics coverage of decontamination and decommissioning activities.
- Provide support resources necessary for implementing and maintaining exposure records.
- Monitor collective dose for decommissioning activities.

- Ensure that the radiation protection staff, organization, and supporting services are adequate.
- Provide technical guidance to the Project Manager and RSO.

The Health Physics Supervisor shall report operationally to the Project Manager and shall also possess a degree of organizational independence to facilitate direct communication with the RSO as necessary to support the RSO's responsibilities.

The Health Physics Supervisor shall be shall be subject to the approval of the Director.

At the time of appointment, the health physics supervisor shall have a minimum of 10 years of radiation safety experience in nuclear power or decommissioning. The individual shall have a recognized baccalaureate or higher degree in health physics, nuclear engineering, or another scientific field. Education or experience that is job related may be substituted for a degree on a case-by-case basis. The degree may fulfill 4 years of the 10 years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate project specific training based upon a comparison of the individual's background and abilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required.

2.4.5 Review Committee

The UM will establish a review committee to review decommissioning activities and advise the Director in matters relating to the health and safety of the UM community, the public and the safety of decommissioning activities.

2.4.5.1 Composition and Qualification

The review committee shall be composed of a minimum of three members and an unspecified number of alternates, of which only a minority shall be from the FNR Decommissioning Project staff.

The members and alternates shall be appointed by the Vice President for Research. The review committee chair shall be appointed from the UM tenured faculty with a degree in engineering or a scientific field. The review committee chair shall receive, at the time of appointment, briefings sufficient to provide an understanding of the decommissioning project. The remaining members of the review committee and alternates shall collectively represent a broad spectrum of expertise appropriate for the decommissioning of FNR and may be either from within or outside the UM. Alternates may attend and vote on matters, regardless of the absence of regular members.

The review committee shall meet at least semiannually through the completion of the final status survey. After the completion of the final status survey the review committee shall meet as necessary to review or approve such matters as desired by the committee chair, the Director, Reactor Manager or the RSO.

A quorum shall consist of not less than one-half the regular review committee membership, not including alternates(where the FNR Decommissioning Project staff does not constitute a majority), and a representative of UM management at the Associate or Assistant Vice President level or higher. Approval of items by the review committee must be by a majority of the full review committee membership. Approval of items by the review committee may be cast at meetings or via individual polling of the regular review committee members.

The review committee chair may appoint subcommittees to facilitate targeted reviews or audits. The subcommittee chair shall be a regular committee member or alternate and shall not be a member of the FNR Project Staff. The subcommittee shall forward items to the review committee chairman with recommendations. The full review committee shall approve all products of the subcommittee.

The minutes of the review committee shall be distributed to the Director, Reactor Manager, RSO, Project Manager, Health Physics Supervisor, the regular members of the review committee, and such others as the chairman may designate.

The review committee shall approve:

- Proposed changes in the license or technical specifications.
- Proposed changes to the facility that can be implemented without the prior approval of the NRC as authorized by the license conditions implementing 10 CFR 50.59.
- Proposed changes in the Decommissioning Plan that can be implemented without the prior approval of the NRC as described in the *Decommissioning Plan*, Section 9.0 and authorized by license condition.
- New procedures and proposed changes to the procedures for the following activities which shall be in effect and followed:
    1. Normal operation of all systems structures or components described in these technical specifications or which are important to safety.
    2. Actions for responding to emergency conditions involving the potential or actual release of radioactivity, including provisions for evacuation, reentry, recovery, and medical support.
    3. Actions to be taken to correct specific and foreseen malfunctions of systems, structures or components described in these technical specifications or which are important to safety.
    4. Activities performed to satisfy a surveillance requirement contained in these technical specifications.
    5. Radiation and radioactive contamination control.
    6. Physical security of the facility.
    7. Implementation of the quality assurance for the calibration and response testing of radiation instrumentation utilized for direct measurement in support of characterization, release, final status survey, or other quality assurance activities.

    These procedures shall be appropriate to protect the UM community, the public, and personnel involved in decommissioning and to implement the quality assurance necessary to support a request for the termination of the license. Substantive changes to these procedures shall be made only with the approval of the review committee. Non-substantive changes to these procedures may be made with the approval of the Reactor Manager. All non-substantive changes made to procedures shall be documented and subsequently reviewed by the review committee.

The review committee, as a review function, shall review:

- Violations of technical specifications and reportable occurrences made pursuant to the requirements of the technical specifications.
- Audit reports issued by a member or subcommittee of the review committee made to satisfy any requirement of the review committee's audit function.

- Plans for the following decommissioning activities prior to their implementation:
  1. Any activity which could compromise the structure and integrity of the reactor pool or the primary coolant system while pool water is relied upon for shielding of irradiated reactor components.
  2. The dismantlement of the irradiated reactor components in preparation for disposal.
  3. The movement of any heavy objects, greater than 5 tons in weight.
  4. Any activity that could compromise the structural integrity of the post and beam structure which supports the reactor building.
  5. Any activity which will result in the direct release of radioactivity from the facility to the sanitary sewer or a navigable waterway.
  6. The draining of the reactor pool.
  7. The decontamination or dismantlement of the reactor pool structure.
  8. Any activity for which it is estimated that the cumulative radiation exposure for the activity will exceed 1 person-rem, or an individual radiation exposure to either an occupationally exposed person or a member of the public will exceed 20 percent of any applicable exposure limits of 10 CFR 20.
  9. Any activity, known or anticipated by the review committee, which the review committee requests to review, subject to the approval of the Director.

The review committee, as an audit function, shall ensure that the following are independently monitored or audited:

1. Decommissioning operations to ensure they are being performed safely and in accordance with all applicable licenses and registrations held by the UM and in compliance with applicable federal and state regulatory requirements (Radiological Protection Plan, Environmental Safety and Health Plan, etc.).
2. The quality assurance program to verify that performance criteria are met as well as to determine the effectiveness of the program in satisfying the quality assurance requirements of the decommissioning plan and 10 CFR Part 71.

Each monitoring or audit report shall identify the monitor(s) or auditor(s), describe the scope of the review, identify persons contacted, summarize audit results (including a statement on the effectiveness of the elements monitored or audited) and describe each reported adverse finding. Each monitoring or audit report shall be distributed to the Director, Reactor Manager, all review committee members, and others at the direction of the Director.

Monitoring or audits shall be performed annually as a minimum and should be scheduled by the Chair of the review committee, in a manner to provide coverage and coordination with ongoing activities, based on the status and importance of activities. Scheduled monitoring or audits should be supplemented by additional monitoring or audit of specific subjects when necessary to provide adequate coverage.

The lead auditor and the audit team, generally from one to three in number, including the lead auditor, shall be:

- Appointed by the Chair of the review committee.
- Not directly associated with decommissioning activities and not a member of the FNR Decommissioning Project Team.
- Familiar with quality assurance requirements applicable to the decommissioning of nuclear facilities.

2.5 Training Program

Decommissioning activities are much different from typical FNR operations, and as such, will require special training for the existing FNR operations staff and the decommissioning personnel. Individuals (employees, contractors, and visitors) who require access to the work areas or radiologically restricted area will receive training commensurate with the potential hazards to which they may be exposed. Individuals will also receive continued training, as necessary, to ensure that job proficiency is maintained.

Personnel will be qualified for their assigned duties before working or will be under the direct supervision of a qualified employee. Personnel performing special processes will be qualified according to specific codes and standards and/or in accordance with national consensus documents. Qualification will include proficiency demonstrated by each individual, both initially and then periodically. Qualification also will be demonstrated when required by the designated codes or standards.

Training records will be maintained and will include the trainee's name, dates of training, types of training, test results, protective equipment use authorizations, and instructors' name.

Care will be taken to ensure that properly qualified instructors conduct all training. As the primary criteria, persons responsible for presentation of training should have knowledge and experience in the process or subject matter. It is desirable that trainers also have the presentation skills or classroom conduct appropriate to the level of the training being presented. For those with limited background in training, early instruction should be monitored and feedback should be provided.

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

- General Employee Training – provides general training for emergency response, spill response, alarms, alarm response, communication systems and channels, waste management, and waste minimization and will satisfy the instruction requirements 1
  of 10CFR19.12, Instruction to Workers.
- Radiation Safety Training
  - General Radiological Training – training for personnel who are required to enter radiological restricted areas, with the exception of visitors and infrequent support personnel, but are not authorized to perform hands-on radiological work.
  - Radiological Worker Training – training for personnel who require unescorted access to radiological restricted areas and who are authorized to perform radiological job functions.

  These trainings will consist of core training and site-specific training. Core training may be accomplished under any program that meets basic requirements. Site specific training will be given to all personnel. Refresher training will be given annually to all personnel.
- Hazardous Waste Operations and Emergency Responses (HAZWOPER) - training for personnel engaged in hazardous substance removal or other activities that potentially expose them to hazardous substances and health hazards, which satisfies 29 CFR 1910.120.

- Respirator Training and Fit Testing - training, medical qualification, and fit-testing for each person who wears a tightly fitting respirator which satisfies the requirements of 10 CFR 20, Subpart H and Regulatory Guide 8.15, .*Acceptable Programs for Respiratory Protection* (NRC 1976)

- Department of Transportation (DOT) Hazmat Employee Training – all personnel involved in the loading, unloading, or handling of hazardous materials, preparation of hazardous materials for transportation (including packaging and preparation of manifests), responsible for the transportation of radioactive materials, or operation of a vehicle used to transport hazardous materials (49 CFR 171.8) shall be provided training as required by 49 CFR 172, Subpart H.

- Hazardous Materials: Security Requirements for Offerors and Transporters of Hazardous Materials – all personnel involved in the offering of placarded quantities shipments of hazardous materials will be trained in the facilities security plan which satisfies the requirements of 49 CFR 172.

- Hazard Communication Training - all personnel will be trained on the hazardous chemicals in their work area, as required by 29 CFR 1910.1200(h). Personnel will be provided update training whenever a new physical or health hazard is introduced into their work area. This training, at a minimum, shall include the proper use of the materials, the required personal protective equipment (PPE), and the emergency procedures associated with these materials.

- Hearing Conservation Training – training on the effects of noise on hearing and the purposes, advantages, disadvantages, and attenuation of various types of hearing protective devices.

- Permit-Required Confined Space Entry Training – training for personnel if entry into confined spaces is to be performed.

- Lockout/Tagout Training – training for hazardous energy control.

- Trenching and Excavation Training – training for the purpose of determining the safety and stability of excavations.

- Fire Watch Training – training on the proper selection, use, and application of extinguishing agents; characteristics and classification of fires.

- Asbestos Abatement Training – training on requirements, potential health effects, and controls for asbestos abatement.

- Torch/Plasma Arc Cutting, Welding, and Open Flame Trainings- training in the use of, and understanding the reasons for, protective clothing and equipment, including the need for flame-resistant clothing.

- Tailgate Training – routine, short training given, usually at the beginning or end of a regular work force briefing, intended to provide a brief review of a safety or programmatic topic, which is applicable to current work activities.

- Other Specific Mandated Training - any other training that may be required by the Michigan Occupational Safety and Health Act (MIOSHA) specific standards or applicable standards before initiating work that may fall within the scope of work.

2.6 Decontamination and Decommissioning Documents and Guides

The FNR decommissioning plan has been written using the guidance and format specified in Chapter 17 of NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (NRC, 1996). The radiological criteria for license termination to allow unrestricted use will be as set forth in 10 CFR Part 20, Subpart E, *Radiological Criteria for License Termination*. Activities necessary to reach the radiological criteria for license termination will be guided by the following regulations, regulatory guides, and standards:

NOTE: The documents listed below are not considered part of the current licensing basis described in Section 9.0. In addition, the listing of a document below should not be taken as a commitment by the licensee to requirements in the listed documents, excepting federal or state regulations.

Code of Federal Regulations:

10 CFR Part 19	"Notices, Instructions and Reports to Workers; Inspections"
10 CFR Part 20	"Standards for Protection Against Radiation"
10 CFR Part 30	"Rules of General Applicability to Domestic Licensing of Byproduct Material"
10 CFR Part 50	"Domestic Licensing of Production and Utilization Facilities"
10 CFR Part 51	"Licensing and Regulatory Policy and Procedures for Environmental Protection"
10 CFR Part 61	"Licensing Requirements for Land Disposal of Radioactive Waste"
10 CFR Part 71	"Packaging of Radioactive Material for Transport and Transportation of Radioactive Material under Certain Conditions"
10 CFR Part 140	"Financial Protection Requirements and Indemnity Agreements"
29 CFR Part 1910	"Occupational Safety and Health Standards"
29 CFR Part 1926	"Occupational Safety and Health Standards for Construction"
49 CFR Parts 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, Test, and Experiments"
8.2	"Guide for Administrative Practices in Radiation Monitoring"
8.4	"Direct-Reading and Indirect-Reading Pocket Dosimeters"
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 Exposure As Low As Is Reasonably Achievable"
8.13	"Instruction Concerning Prenatal Radiation Exposure"
8.15	"Acceptable Programs for Respiratory Protection"

Other Regulatory Documents:

NUREG-1505	"A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys"
NUREG-1757	"Consolidated NMSS Decommissioning Guidance"
NUREG-1507	"Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions"
NUREG-1549	"Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination, Draft"
NUREG-1575	"Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)"
NUREG/CR-1756	"Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors"
NUREG-1727	"NMSS Decommissioning Standard Review Plan"
NUREG/CR-5849	"Manual for Conducting Radiological Surveys in Support of License Termination".

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3.0 Protection of the Health and Safety of Radiation Workers and the Public

The Radiation Protection Program involves more than simply following procedures. It involves active participation by all employees in constantly striving to make improvements. No one knows any job better than the person doing that job. No one knows the condition of the equipment, potential problems with the procedures, and the work environment of a job better than the person doing that job. Therefore, it is up to all personnel to be constantly aware of changes in the work environment, and to bring any potentially harmful conditions to the attention of the decommissioning team as soon as possible. It is the responsibility of the decommissioning team to promptly and effectively respond to employee concerns about safety and health.

3.1 Radiation Protection

This section describes the FNR Decommissioning Project ALARA program and enhancements to the existing health physics program that will be in effect during decontamination and decommissioning (D & D) of the FNR. The existing FNR health physics program will remain in effect during decommissioning. Enhancements and revision to this program will be required to prepare for and support decommissioning activities and adjust for the intense contractor involvement not previously accounted for by the program. UM retains responsibility for the health physics program and will approve all changes to the written program as discussed in Section 2.4.

A radiation protection program will remain under the cognizance of the RSO and the review committee as discussed in Section 2.4. The program will be uniformly applied to all UM and contractor personnel. Radiation safety personnel will be present at the site when decommissioning activities are in progress to provide complete support and health physics supervision. These services include, but are not limited to, implementing ALARA principles; providing radiation worker training; establishing occupational and public dose limits; monitoring personnel for occupational exposures; controlling exposure, waste disposal; providing radiation monitoring equipment; performing station area and environmental surveys; and maintaining records and generating of reports as necessary to comply with NRC and license requirements.

3.1.1 Ensuring ALARA Radiation Exposures for Decommissioning Activities

Work control for FNR decommissioning activities shall be performed in accordance with the enhanced requirements of a health physics program and will incorporate provisions for reducing individual and collective radiological exposures to ALARA as discussed in subsequent subsections.

NOTE: The ALARA requirement of 10 CFR 20.1402, which applies to exposures following termination of the license, is briefly discussed in Section 2.1.5 and will be fully discussed in the Final Status Survey Plan, when submitted at a later date (see Section 4.0).

3.1.1.1 Activity Work Control

Activity work control for tasks with significant or potentially significant radiation exposures will include a formal plan which shall require use of procedures and engineering controls that reduce the exposures as low as reasonably achievable. In developing work plans, the potential radiological exposures will be examined for the workers, the UM community, and members of the public and specific actions taken to minimize the exposure of the individual workers, the collective dose of the entire decommissioning work force, the UM community, and the public, whenever reasonably achievable. Process or other engineering controls will be the preferred methods for maintaining exposures to radiation and radioactive materials as low as reasonably achievable. This would include incorporation of shielding (radiation reduction), containment/confinement structures (radioactive material isolation), and controlled ventilation (airborne radioactive material reduction). Other controls that are evaluated for inclusion into the work control include the following:

- Control of access to the radioactive sources, including remote handling.
- Techniques to reduce exposure times.
- Techniques to increase distance between the individual and the source.
- Use of specialized PPE and respiratory protection equipment.

In addition to dose reduction for performing work activities, certain pre-task activities, such as source reduction (that is, decontamination) and/or source removal, should be considered during the ALARA planning for an activity.

Before implementation, any procedure and control considered for ALARA purposes will be evaluated to ensure the following:

- Cost associated with implementation is justifiable.
- Implementation results in an overall risk reduction and not simply a risk transference.
- Overall risk reduction is justifiable in the context of overall task or project objectives.

Work control plans will be approved by the Reactor Manager, who may delegate this approval authority to any member of the reactor staff satisfying the education and training requirements for the Reactor Manager. Work control plans and procedures for the decommissioning will be maintained and subject to the approval requirements discussed in Section 2.4.

3.1.1.2 ALARA Evaluation or Review

A documented ALARA evaluation will be required for each specific task satisfying the following:

- A conservative estimate that 5percent of the applicable dose limit for the TEDE, the Eye Dose Equivalent (EDE), the Shallow Dose Equivalent (SDE), or the sum of the Deep-Dose Equivalent (DDE) and the Committed Dose Equivalent (CDE) to any individual organ or tissue other than the lens of the eye could be exceeded.
- A conservative estimate determines that the effluent averaged over one year is expected to exceed 20 percent of the applicable concentration in 10 CFR 20, Appendix B, Table 2, Columns 1 and 2.

The existing FNR health physics program will be revised to add an ALARA review procedure subject to the approval requirements discussed in Section 2.4. ALARA evaluations for the decommissioning will be maintained.

3.1.1.3 Radiation Work Permits

RWPs will be used for administrative control of personnel entering or working in restricted areas (as defined in 10 CFR 20.1003 – *Restricted Area)*. The existing FNR health physics program will be revised to add a RWP procedure subject to the approval requirements discussed in Section 2.4. The RWP program allows the project staff, the radiation safety staff, the contractor, and the contract health physics staff to specify, to the extent practical, the controls, protections, process, or other controls (e.g., time, distance, shielding, remote handling, hot cells, localized containment, localized ventilation) to maintain the exposure of individual workers, the collective dose of the entire decommissioning work force, the UM community, and the public ALARA. RWPs will not replace work procedures, or JHAs (described later), but will act in concert with these programs to protect the overall health and safety of individual workers, the entire decommissioning work force, the UM community, and the public.

The RWP program will follow the guidance contained in NCRP Report No. 127, Section 6.2.3, *Radiation Work Permits*.

RWPs for activities with low exposure levels will be approved at the health physics technician level (UM or contractor) or the health physics supervisor level (UM or contractor) while RWPs for jobs with potentially high dose commitment or significant radiological hazards will be approved by the RSO. In the absence of the health physics technician level (UM or contractor) or the health physics supervisor level (UM or contractor) for low exposure level activities or in the absence of the project RSO for jobs with potentially high dose commitment or significant radiological hazards, then a UM staff member satisfying the educational and training requirements of the project RSO may approve the RWP.

3.1.1.4 Respiratory Protection and Controls to Restrict Internal Exposure

To the extent practical, process or other engineering controls shall be used to control the concentration of radioactive materials in the air. Engineering controls may include, but are not limited to the following:

- Tents or other confinements around work areas with ventilation systems that provide HEPA filtration.
- Confinement structures around a work task that isolate the task from the individual (for example, glove bags, remote manipulation of materials).
- Close capture ventilation (if necessary) or use of HEPA vacuums at the work location that moves potential airborne activity away from the individuals.
- Use of surfactant materials for fixing removable contamination to surfaces before handling/sampling.

When it is not practical to employ process or other engineered controls, or when these controls are not sufficient to maintain the airborne contamination levels or potential levels below the values that define an airborne radioactive material area, the actions to increase monitoring and limit intake for an individual shall be taken by one of the following means:

- Work controls that limit time.
- Access controls to the area.
- Use of respiratory protection equipment.
- Other controls.

These actions shall be taken consistent with maintaining the total exposure as low as reasonably achievable. The usage of respiratory protection is described later.

3.1.1.5 Control and Storage of Radioactive Materials

The existing FNR health physics program controls radioactive materials in a manner which:

1. Deters the inadvertent release of radioactive materials to unrestricted areas.
2. Ensures that personnel are not inadvertently exposed to licensed radioactive materials.
3. Minimizes the volume of radioactive waste generated during licensed activities.

3.1.2 Health Physics Program

The existing FNR health physics program will be updated as described above and below and will remain under the control and authority of the UM. The health physics program will be revised as necessary to ensure that it will continue to satisfy the following radiation protection program commitments during decommissioning:

1. Ensure radiological safety of the public, occupationally exposed personnel, and the environment.
2. Monitor radiation level and radioactive materials.
3. Control distribution and releases of radioactive materials.
4. Maintain potential exposures to the public and occupational radiation exposure to individuals within the limits of 10 CFR Part 20 and at levels ALARA.

3.1.2.1 Radiation Exposure

UM management is committed to minimizing exposure of individuals to radiation or radioactive materials as low as reasonably achievable. To support this commitment, individuals conducting decommissioning activities will be subject to administrative controls for radiation exposure, which will be based the requirements contained in 10 CFR 20 and may be used to ensure compliance with the annual dose limits and for maintaining exposures ALARA.

Provisions for exceeding these administrative limits will be defined in writing and approved as described in Section 2.4 before authorizing exposure in excess of these limits. Prior authorization to exceed administrative limits for any radiation worker will be obtained, in writing, from the RSO.

The administrative limits for FNR decommissioning activities are as follows:

Adult Employees

- TEDE — less than or equal to 2.0 rem/year
- Total Organ Dose Equivalent (TODE) — less than or equal to 2.0 rem/year
- Lens of the Eye Dose Equivalent (LDE) — less than or equal to 2.0 rem/year
- SDE — less than or equal to 2.0 rem/year

Embryo/Fetus (Declared Pregnant Employee Exposure)

- TEDE less than 0.1 rem over the duration of the pregnancy

Adult Visitor, Member of the UM community, and Member of the Public

- TEDE less than 0.05 rem/year

Personnel monitoring of occupational radiation exposure from external sources will be performed through the use of individual monitoring devices as required by 10 CFR 20.1502. At a minimum, on an annual basis, or whenever changes in worker exposures warrant, an external exposure evaluation will be performed to ensure the personnel monitoring of occupational radiation exposure from external sources is in compliance with 10 CFR 20.1502(a). Dosimeters that require processing (e.g. thermoluminescent or OSL dosimeters) will be provided by the UM and shall be processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited dosimetry processor.

Monitoring of occupational exposure from licensed radioactive materials internal to an individual will be determined through monitoring of the quantities of licensed materials in the air collected through air samples, *in vitro* or *in vivo* bioassay techniques, or a combination of air monitoring and bioassay as allowed by 10 CFR 20.1204 and required by 10 CFR 20.1502 (b). If respiratory protection equipment is used for protection against airborne radioactive material, air monitoring, or bioassays will be performed to evaluate the actual intakes as allowed by 10 CFR 20.1204. To ensure compliance with 10 CFR 1502(b), bioassay for intakes of licensed materials may be performed for the personnel with the greatest potential for intake at a sample frequency appropriate for the pulmonary retention class (days, weeks, years).

When exiting restricted areas that have known removable contamination or exiting restricted areas that have the potential for removable contamination, personnel shall monitor their hands and feet for contamination. If contamination is detected, then a check of the exposed areas of the body and clothing should be made. Personnel leaving potentially contaminated areas periodically monitor their hands and feet for contamination, consistent with the nature and quantity of the radioactive materials present.

The concentrations of radioactive material released from the facility in gaseous effluents from the facility will continue to be measured. The dilution factor of 400, taken from previous safety analyses submitted to the NRC and contained in the Technical Specifications, continues to apply to the FNR exhaust and the PML stack exhausts. UM may also utilize the other options for showing compliance with the annual dose limit to an individual member of the public from concentrations of radioactive material released from the facility in gaseous effluents from the facility, as allowed by 10 CFR 20.1302.

To ensure compliance with the requirements of 10 CFR 20, the concentrations of radioactive material released from the facility in liquid effluents will continue to be measured. UM may also utilize the other options for showing compliance with the annual dose limit to an individual member of the public from concentrations of radioactive material released from the facility in liquid effluents as allowed by 10 CFR 20.1302.

3.1.2.2 Surveys and Monitoring

Radiation surveys and monitoring will be performed in accordance with the existing radiation protection program and as necessary to support work activities in areas where there is potential for exposure to radiation or radioactive materials. The effectiveness of controls to minimize or eliminate that exposure will be assessed in the following two ways:

- Direct measurement of the external radiation or the radioactive material intake an individual receives.
- Measurement of the radiological conditions in the area(s) occupied by the individual.

Levels and extent of direct radiation and radioactive materials in any work area will be measured and assessed in accordance with the health physics program. These measurements will include, as a minimum:

- Direct dose rate measurements.
- Surface contamination measurements (fixed and removable).
- Airborne radioactive material measurements.

All instruments and equipment used for these measurements shall be calibrated for the radiation type to be measured on frequencies as listed in Section 3.1.2.4

3.1.2.3 Exposure Control

Restricted areas are defined on the basis of the known or suspected hazard potential from radiation sources that have been defined from measurement or inferred from process knowledge. Exposure to an individual entering such an area may be from any combination of the following:

- Direct radiation
- Surface contamination (fixed and removable)
- Airborne contamination

3.1.2.3.1. Exposure to Direct Radiation

Control of exposure to individuals from direct radiation is based on two elements:

- Measurement and assessment of the location and strength of the radiation sources
- Control of the individual’s access to those radiation sources

Routine monitoring of the levels and extent of radiation and radioactive materials is a key part of the health physics program. Levels and extent of direct radiation and radioactive materials in work areas are also measured and assessed under the health physics program. These measurements include direct dose rate measurements, surface contamination measurements,

and airborne radioactive material measurements. The first of these elements is discussed in Section 3.1.2.2.

Before defining control requirements for limiting direct radiation exposure to individuals, the location of the radiation sources and the magnitude of the radiation will be determined. Direct radiation exposure measurements will be made at the time of decommissioning, concentrating on areas identified as having a worker exposure potential. This survey work also will include specific areas or systems identified during work planning before project startup.

Based on this measurement and data assessment, shielding, or barriers that restrict access to sources of radiation will be established. Posting at access points through those barriers will be established based on the potential exposures that an individual could receive upon entry through the access points or along external surfaces of the barrier, in accordance with regulatory requirements.

3.1.2.3.2. Exposure to Surface Contamination

Surface contamination may result in radiation exposure to an individual in several possible ways:

- Direct exposure from the contaminated surface.
- Transfer of "smearable" contamination from the surface in question to the surface (for example, skin and clothing) of the individual.
- Suspension in air of the radioactive contamination from the surface(s) because of the activities of the personnel or equipment.

Controlling exposure to individuals from surfaces contaminated with radioactive material may be accomplished either by prior decontamination or by using protective equipment for personnel to minimize or limit exposure to the surface material.

Prior decontamination for planned work activities is the preferred method of contamination control. However, this will be evaluated for ALARA considerations to ensure that exposures resulting from the decontamination/removal do not offset exposure savings for the planned work activities.

Controlled surface contamination areas may need to be established. Because the FNR contaminants are beta-gamma emitting activation and fission products, the administrative control postings for "contamination areas" and "high contamination areas" are as follows:

- Contamination Area—An area where surface contamination levels exceed the requirements for unrestricted release of a surface, but are less than 100 times the surface values in Table 3-1.
- High Contamination Area—An area where surface contamination levels exceed 100 times the surface values in Table 3-1.

TABLE 3-1, SURFACE CONTAMINATION VALUES

Nuclide	Removable	Average	Maximum
U-nat, U-235, U-238, and associated decay products	1,000 dpm α/100 cm^2	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2
Transuranics, Th-230, Th-228, Pa-231, Ac-227, I-125, and I-129	20 dpm/100 cm^2	100 dpm/100 cm^2	300 dpm/100 cm^2
Th-232, Sr-90, U-232, I-126, and I-131	200 dpm/100 cm^2	1000 dpm/100 cm^2	3,000 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	1,000 beta-gamma dpm/100 cm^2	5,000 beta-gamma dpm/100 cm^2	15,000 beta-gamma dpm/100 cm^2

Notes for Table 3-1:

1. Some radionuclides listed are not anticipated based on the historical review and the current characterization. Their listing is taken from the original source of the above table, Regulatory Guide 1.86, *Termination of Operating License for Nuclear Reactors,* Table 1, *Acceptable Surface Contamination Levels* (NRC 1974), and they are maintained for consistency and in case these nuclides should be identified during the survey.
2. Where surface contamination by both alpha and beta-gamma emitting radionuclides exists, the limits established for alpha and beta-gamma-emitting nuclides should be applied independently.
3. 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 the qualifying detector accounting for background, efficiency, and geometric factors associated with the instrumentation.

When decontamination is impractical or ineffective, protection of the individual from surface contamination is provided by PPE. In determining the appropriate PPE, the following will be considered:

- Radiological conditions (for example, amount of contamination, radionuclides, potential for the contamination to transfer).
- Type of work to be performed (for example, strenuous inspection work, cutting, welding, grinding).
- Potentially stressful environmental conditions (for example, heat, cold, humidity).
- Physical condition of surfaces (for example, wet versus dry).
- Duration of the activity.

Radiation workers may be required to don PPE that may include Tyvek coveralls, booties, and gloves. If the potential for being exposed to airborne contamination in excess of 12 derived air concentration (DAC)-hours in a workweek of the most limiting DAC is encountered, workers will be required to don full-face respirators if work must be performed in these areas.

Contamination control measures that will be employed include, as appropriate but not limited to, the following:

- Local containment barriers such as designed barriers, glove bags, containers, and plastic bags will be used to prevent the spread of radioactive material.
- Physical barriers such as Herculite sheeting, strippable paint, tacky-mat step off pads, absorbent pads, and drip funnels will be used to limit contamination spread.

3.1.2.3.3. Exposure to Airborne Contamination

If air monitoring results indicate levels of airborne radioactive materials in excess of NRC-prescribed levels, access points to this airborne activity will be posted "Airborne Radioactivity Area." as per 10 CFR 20.1003. Airborne Radioactivity Area means a room, enclosure, or area in which airborne radioactive materials composed wholly or partly of licensed material exist in concentrations that are in excess of the DAC specified in Appendix B to 10 CFR 20.1001-20.2401 or to such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake, or 12 DAC-hours.

When it is not practical to employ the engineering controls described previously, or when these controls are not sufficient to maintain the airborne contamination levels or potential levels below those identified for an airborne radioactivity area above the use of respiratory protection equipment may be necessary. When respiratory protection is required, it will be as described in a respiratory protection program satisfying the requirements of 10 CFR 20, Subpart H. This program will include worker training and medical qualification requirements for use and descriptions of the following:

- Respiratory equipment to be used
- Air monitoring requirements to support the use
- Bioassay program to evaluate the effectiveness of use
- Equipment cleaning, testing, and maintenance requirements

No workers will be assigned to work tasks that require the use of respirators unless they are medically qualified.

Workers who use respiratory protection equipment will provide bioassay samples at a minimum, prior to performing work in a radiologically controlled area and at the end of their assignment to the Decommissioning Project.

3.1.2.4 Radiation Monitoring Equipment

A sufficient inventory and variety of instrumentation will be maintained onsite to facilitate effective measurement of radiological conditions and control of worker exposure consistent with ALARA and to evaluate suitability of materials for the release of materials for unrestricted use. Radiation monitoring equipment will be capable of measuring the range of dose rates and radioactivity concentrations expected to be encountered during remediation and decontamination activities to the minimum values required for release or materials for unrestricted release.

Radiation monitoring equipment that is inoperable or out-of-calibration will be clearly identified and removed from service. Radiation monitoring equipment will be calibrated at manufacturer-prescribed intervals (if frequency shorter than annual) or annually or prior to use using standards that are traceable to National Institute of Standards and Technology (NIST) or an equivalent stands organization. Survey instruments and equipment will be operationally tested daily when is use. Instruments will be calibrated at a minimum frequency as established in Table 3-2.

The requirements for instruments and equipment for the Final Status Survey are discussed in Section 4.0.

TABLE 3-2, RADIATION MONITORING EQUIPMENT CALIBRATION FREQUENCY

Instrument Type	Application	Calibration Frequency
Count rate meters	Personnel monitoring and surface contamination measurements	Annually or greater frequency as specified by the manufacturer
Exposure or dose equivalent rate meters	Determine exposure or dose equivalent rates	Annually or greater frequency as specified by the manufacturer
Gross alpha or beta scaler	Quantify radioactive material on air samples or smears	Annually or greater frequency as specified by the manufacturer
Air samplers	Collect airborne radioactive material samples	Six months or greater frequency as specified by the manufacturer
Continuous air monitors	Monitor the concentration of radioactive material in the air	Six months or greater frequency as specified by the manufacturer
Effluent monitors	Monitor the quantity or concentration of radioactive material in site effluents	Annually or greater frequency as specified by the manufacturer
Microprocessor based contamination monitors	Detect and quantify radioactive material on personnel, items, or equipment	Annually or greater frequency as specified by the manufacturer
Spectroscopy, liquid scintillation, gas flow proportional, and other laboratory equipment	Quantify radioactive material in samples	Annually or greater frequency as specified by the manufacturer

3.1.3 Control of Radioactive Materials

The existing FNR health physics program establishes radioactive materials controls that ensure the following:

- Prevention of the inadvertent release of licensed radioactive material to uncontrolled areas
- Assurance that personnel are not inadvertently exposed to radiation from licensed radioactive materials
- The amount of radioactive material generated by the licensee during decommissioning will be minimized.

All materials leaving a restricted area will be surveyed to ensure that licensed radioactive materials are not removed. These surveys incorporate the guidance in NRC Circular No, 81-07, *Control of Radioactively Contaminated Material,* and Information Notice No. 85-92, *Surveys of* 1
Wastes Before Disposal from Nuclear Reactor Facilities (NRC, 1981 and NRC, 1985).

For items where the contaminants are beta-gamma emitting activation and fission products the following survey methods will be used (General Atomics, 1999):

- Materials and Equipment – direct frisking with a portable Geiger-Mueller detector (e.g., Ludlum Model 44-9, Eberline Model HP-210 or equivalent) having a minimum level of detection above background of less than or equal to 5,000 dpm per 100 cm^2.
- Smear Samples – analysis with a Geiger-Mueller detector (e.g. Ludlum Model 44-9, Eberline Model HP-210 or equivalent) having a minimum detection level above background of less than or equal to 1,000 dpm per 100 cm^2.

- Sand, Soil, Concrete Dust, Silts, Fine Sediments, etc. – analysis of representative sample(s) with a high resolution gamma spectroscopy system having a lower limit of detection above background of less than or equal to 0.18 pCi per gram for Cs-137 (e.g. ≤ 1
180 pCi per kilogram) (NRC 1982). Background equivalent gamma activity, an unshielded gamma ray dose measured 1 meter from any surface, measured with a microR meter shall not exceed 5 microrem per hour above background. 1

Additional methods for release of surface contaminated materials may be developed and are subject to the minimum detection levels in Table 3-3. Detection sensitivities of instruments and techniques may be determined using the guidance contained in the *Multi-Agency Radiological Survey and Site Investigation Manual* (MARSSIM) (NRC, 2000b) and *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions* (NRC, 1997b). Equipment and materials may be relocated to areas of lower ambient background for the conduct of release surveys.

TABLE 3-3, ACCEPTABLE LICENSED MATERIAL MINIMUM SURFACE DETECTION LEVELS FOR RELEASE OF MATERIALS

	Minimum Detection Level, above background	
Radionuclides [a]	Fixed	Removable
U-nat, U-235, U-238, and associated decay products	5,000 dpm per 100 cm^2	1,000 dpm per 100 cm^2
Transuranics, Th-230, Th-228, Pa-231, Ac-227, I-125, and I-129	100 dpm per 100 cm^2	20 dpm per 100 cm^2
Th-232, Sr-90, U-232, I-126, and I-131	1,000 dpm per 100 cm^2	200 dpm per 100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except for Sr-90 and others noted above	5,000 dpm per 100 cm^2	1,000 dpm per 100 cm^2

Notes for Table 3-3:

1. Some radionuclides listed are not anticipated based on the historical review and the current characterization. Their listing is taken from the original source of the above table, ANSI 15.11-1993 (ANS, 1993), and they are maintained for consistency and in case these nuclides should be identified during the survey.
2. Where surface contamination by both alpha and beta-gamma emitting radionuclides exists, the limits established for alpha and beta-gamma-emitting nuclides should be applied independently.
3. 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 the qualifying detector accounting for background, efficiency, and geometric factors associated with the instrumentation.

In evaluating equipment and materials for fixed or smearable licensed radioactive materials, 1
items painted with other than the original manufacturer's paint will not be released unless clear process knowledge demonstrates that the paint was applied to a clean surface containing no discernable radioactivity from licensed materials prior to its use in a restricted area. The project RSO may approve the release of items for which it cannot be demonstrated that paint was applied to a clean surface following the satisfactory completion of a survey, which satisfies the requirements listed above.

If the potential exists for contamination on inaccessible surfaces, the equipment will be assumed to be internally contaminated unless 1) the equipment is dismantled allowing access for surveys, 2) appropriate tool or pipe monitors are used to satisfy the survey requirements listed above are utilized to provide confidence that no licensed radioactive materials are present, or 3) it may readily be concluded that surveys from accessible areas are representative of the

inaccessible surfaces (i.e. surveying the internal surface of both ends of a pipe from a nonradioactive process system with cotton swabs would be representative of the inaccessible areas).

Materials will be released only if no discernable radioactivity above background from licensed materials is detected by a survey method identified above.

Items not releasable under the above criteria shall be controlled as licensed radioactive material and labeled as required by 10CFR20.

Licensed radioactive materials slated for disposal or disposition should be handled as described 1
in Section 3.2 "Radioactive Waste".

Licensed radioactive materials may be transferred to other locations within the control of UM as allowed by appropriate radioactive material licenses issued by the NRC. Licensed radioactive materials may be transferred to other locations outside the UM that possess the appropriate radioactive material licenses issued by the NRC, an Agreement State, or are otherwise authorized to possess such radioactive material (e.g. DOE sites, foreign research reactors).

Personal effects (e.g., notebooks, pens, flashlights) which are hand-carried into a restricted area are subject to the same survey requirements as the individual possessing the item.

The existing health physics program will be revised to ensure that the requirements above are established by a procedure subject to the approval requirements discussed in Section 2.4.

3.1.4 Dose Estimates

The total estimated occupational exposure to complete the FNR Decommissioning Project is 4.8 person-rem. The dose estimate for decommissioning of the FNR was prepared using the individual work activity durations and work crew sizes estimated by CH2M HILL, under contract with the UM, based upon the results of the characterization results to date and was heavily based upon their recent experience in performing similar activities at the UVa, and Georgia Tech, combined with their ongoing experience at Rocky Flats, Hanford, etc.

The FNR reactor structure and bioshield (pool) design and power level is similar to that of the UVa. In addition, the FNR has a thermal column similar to that of Georgia Tech. Since FNR was most similar to the UVa design and power level, the dose estimate from the UVa Decommissioning Plan was used as the basis for the FNR dose estimate. The actual dose for decommissioning of the UVa reactor was well below the 3.9 person-rem as estimated in the UVa Decommissioning Plan (UVa 2000). The similarities and differences between FNR and the UVa reactor are shown in Table 3-4

TABLE 3-4, COMPARISON OF FNR TO UVa REACTOR – RELEVANT TO THE EXTENSION OF THE DOSE ESTIMATE

Characteristic	FNR	UVa
Maximum Power, MW(t)	2	2
Fuel Type	45 MTR, LEU, Al U	64 MTR, LEU, U Si
Reflector	Heavy and Light Water	Graphite and Light Water
Operating Time, MW-days	17,868	2,559
Number of Beam Port Openings	9	3
Date of Shutdown	July 2003	July 1998

1

As characterization data for neither the FNR pool nor the FNR beamports was available to support a dose estimate, FNR was operating at the time of the characterization, the FNR dose

estimate was developed by to cross referencing the conservative UVa Decommissioning Plan's task specific dose estimates to the tasks planned as part of the FNR decommissioning. Activities accounted for in the UVa Decommissioning Plan that were not part of the tasks planned for decommissioning FNR were removed (e.g. buried waste and hot cell tanks, undistributed labor and costs, pool barrier, etc.). This provided a baseline dose estimate for FNR based upon the known conservative estimate in the UVa Decommissioning Plan.

Adjustments were made to this baseline dose estimate to account for differences in the planned activities and the anticipated exposure levels from components. Knowing that the removal of the graphite from the thermal column during the Georgia Tech reactor decommissioning and the removal of bioshield during the Georgia Tech reactor decommissioning increased the overall planned dose by approximately 39% and were the primary dose contributor for the entire project, the baseline dose estimate "Dismantling and Decontamination" was increased 39% from 1.78 person-rem to 2.5 person-rem. Similarly, the baseline dose estimate for "Decontaminate and/or Remove Pool Concrete" was increased 39% from 1.08 to 1.5 person-rem. The assumed 50/50 ratio of Cs-137 to Co-60 was not a determining factor in the development of the dose estimates for any activity.

Using these individual work activity durations, and work crew sizes, surveys results and characterization results, a check dose estimate was generated for each activity in the baseline estimate. This check dose estimate was then compared to the actual doses experienced during similar activities performed as part of the UVa and Georgia Tech decommissioning efforts to ensure that dose estimate for FNR was conservative. The dose estimates for the decommissioning activities described in this decommissioning plan are provided by those activities in Table 3-5. 1

TABLE 3-5, DOSE ESTIMATE BY ACTIVITY

Occupational Exposure Estimate for Decommissioning the FNR					
Task Name	**Hours**	**Persons**	**Estimated Exposure (Person-Rem)**	**Number of Exposed Individuals**	**Individual Maximum Exposure (Rem)**
Planning and Procedures	1,080	12	0.01	2.25	0.01
Decommissioning Preparations	2,056	7	0.01	5.71	0.01
Dismantling and Decontamination	3,912	10	2.5	6.11	0.81
Characterize Reactor Pool	360	2	0.17	4.50	0.08
Decontaminate and/or Remove Pool Concrete	2,840	12.4	1.5	5.73	0.52
Decontaminate and/or Remove Embedded Pipes/Tubes	660	6	0.28	2.75	0.20
Sample Soil for Evidence of Pool Leaks	110	1	0.01	2.75	0.01
Decontaminate and/or Remove and Survey Remaining Items/Areas	640	4	0.28	4.00	0.14
Soil and Buried Pipe Remediation	400	4	0.01	2.50	0.01
Perform Release Survey and Sampling	1,920	12	0.02	4.00	0.01
Prepare Final Release Report			0		0
NRC Survey Verification			0		0
Demobilize			0		0
Request License Termination			0		0
Prepare Final Reports			0		0
Archive Records			0		0
Total Estimated Occupational Exposure			**4.79**		

The estimate of the maximum person-rem an individual might receive during any activity was developed using the number of hours estimated for the activity by individuals estimated to receive a dose or "Hours", the number of hours per person planned for the task or "Persons" and the average dose per person per activity, task total rem divided by the task average persons. Since dose generally is not spread evenly among all involved workers, the average rem per person was doubled to determine the individual maximum dose per activity

All estimates were presented to the nearest tenth of a person-rem since they were based on a rough estimate. In addition, the dose estimates were presented as at least one significant digit less than expected actual measurements (estimates in tenths and hundredths of a person-rem versus expected actual readings in the thousandths of a person-rem) since the degree of accuracy was less than expected for the actual measurements. In two cases the UVa Decommissioning Plan baseline estimates of 0.001 person-rem were conservatively presented as 0.01 person-rem versus 0.0.

No internal exposures are included in this dose estimate. Exposures from inhalation or ingestion of radioactive materials will be mitigated by the controls specified for contaminated surfaces, see Section 3-7 and airborne contamination, see Section 3-9, the use of radiation work permits, see Section 3-3, a respiratory protection program, see Section 3-3, and the required ALARA program, see Section 3-2. Characterization surveys and historical information does not indicate the potential for transuranics which could provide a disproportionate amount of internal exposure for a small quantity of material. The lower-intensity peaks referred to in Section 5.2.4 of the Characterization Report, Appendix B, from the in situ gamma spectroscopy measurements occur at ~630 kev, ~350 kev and possibly up to four, buried in the Compton continuum between 200 and 250 kev. Identification of these peaks is limited by the low resolution of the hand held NaI based survey meter used for the analysis. None of these peaks potentially represent materials which could provide a disproportionate amount of internal 1
exposure for a small quantity of material. This conclusion is supported by the fact that no significant alpha levels, highest was 79 dpm per100 cm^2, were identified in any of the smears taken during the characterization surveys as summarized in Section 5.2.2 of the Characterization Report, Appendix B.

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 UM and because the area where decommissioning activities are taking place are fully contained within the facility (with the exception of loading and unloading of shipments of equipment and radioactive materials). This is consistent with the negligible (less that 0.1 man-rem) dose estimate provided for the "reference research reactor" in the *Final Generic Environmental Impact Statement on the Decommissioning of Nuclear Facilities* (NRC 1988).

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. Area dose rates used for this estimate are based on process knowledge and current survey maps (where available).

3.2 Radioactive Waste Management

Decommissioning will require the handling of a relatively large volume of radioactive materials to reduce the residual levels of radioactivity to a level permitting the release of the site for unrestricted use and termination of the license. Materials that are not decontaminated and released will be processed as radioactive waste. This section of the decommissioning plan

presents the programs used to manage and control the processing of solid, liquid and gaseous radioactive waste.

FNR will continue to ensure appropriate processing, packaging and monitoring of solid, liquid and gaseous wastes during decommissioning by continuing the health physics program, developing process control procedures and continuing the radiological environmental monitoring program.

These programs will be maintained in compliance with federal and state regulations, disposal site requirements, and any other applicable requirements. The radioactive waste program is a key part of the health physics program.

The waste stream(s) resulting from decommissioning activities will be similar to those resulting from past reactor operations and maintenance. There are no regulatory transportation issues specifically related to the decommissioning of the FNR that are not covered by existing procedures (See Section 2.0 for a description of some of these activities). 1

During decommissioning, significant resources will be expended to process and dispose of liquid and solid radioactive waste. Radioactive wastes include neutron-activated materials, contaminated materials remaining in the containment building, tools, and equipment 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.

Industry-proven methodologies will be used to ensure the separation 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 practices. The guidance to be provided for station monitoring of radioactive materials is provided by the facility's health physics procedures.

UM will continue to monitor and evaluate offsite low-level radioactive waste (LLRW) disposal and storage options prior to and during the decommissioning process. UM currently has access to two operating LLRW disposal sites - Barnwell, South Carolina and Envirocare in Clive, Utah. These two facilities are assumed to remain available to the UM for the decommissioning project.

Transportation of radioactive waste will be in accordance with applicable NRC and DOT regulations and facility procedures. Radioactive waste and material will be shipped either by truck including open and closed transport, trailer mounted shipping cask or by a combination of truck and rail. Shipments will be planned in a practical and efficient manner. The appropriate measures will be taken to ensure the shipments comply with UM policies, regulations, and the receiving site's license. Packages, packaging, and labeling for radioactive materials and waste shipment will meet all applicable regulations and requirements.

3.2.1 Fuel Removal

All irradiated reactor fuel was returned to the DOE's Savannah River Site between October and December 2003.

All un-irradiated reactor fuel was returned to the DOE through BWXT Technologies in August 2003.

3.2.2 Radioactive Waste Processing

Generally, system components will not be decontaminated onsite. Mildly contaminated items may be decontaminated onsite, if it is determined that a component or portion of a component can be safely and economically decontaminated with the onsite personnel, including staff, contractors, or specialty contractors using techniques and materials within the capabilities of those personnel as determined by the UM. Experienced offsite vendor(s) may also be used to decontaminate the components if that can be safely and economically decontaminated as determined by the UM. Currently the intent is to dismantle the contaminated piping systems and dispose of the material or to decontaminate and free release those materials.

Decommissioning of the FNR will result in the generation of solid and liquid low-level radioactive waste, mixed waste, and hazardous waste. Solid radioactive wastes include neutron-activated materials, contaminated materials remaining in the reactor building and those items necessarily contaminated onsite during the remediation activities. Little if any soil remediation that would result in solid radioactive waste is anticipated. Liquid low-level radioactive waste includes the water in the reactor pool and the associated piping as well as contaminated water generated during remediation activities. There is no gaseous radioactive waste because the reactor has been shutdown for over 9 months and all radioactive gases have decayed.

Handling, staging, and shipping of packaged radioactive waste will be performed in accordance with 10 CFR 20.2006, *Transfer for Disposal and Manifests*; 49 CFR 100-177, *Transportation of Hazardous Materials*; 10 CFR 61, *Licensing Requirements for Land Disposal of Radioactive Waste*; MDEQ regulations; disposal site waste acceptance criteria; FNR licenses and permits and the disposal or processing facility license conditions. 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 FNR decommissioning activities will be manifest in a manner consistent with its waste classification.

3.2.3 Low-Level Liquid Radioactive Waste Disposal

3.2.3.1 Sanitary Discharge

Approximately 50,000 gallons of low-level radioactively contaminated water in the reactor pool and associated piping is to be disposed of by discharge to the public sewer system operated by the City of Ann Arbor. Additional low-level radioactively contaminated water generated during remediation activities may also be disposed of if the discharged liquid can be shown to meet the requirements for sewage disposal in the Clean Water Act and in the Code of the City of Ann Arbor, Chapter 29, *Sewage and Sewage Disposal*.

The low-level radioactively contaminated water from the reactor pool and associated piping will be processed using techniques that are cost effective and meet ALARA goals. This processed water may then be discharged after it has been monitored and approved for discharge.

The liquid waste generated by remediation activities will be processed using techniques, which are cost effective and meet ALARA goals. During demolition activities, installed plant equipment used to process liquid radioactive waste may be removed. Therefore, temporary filtration units or demineralizers may be used as the primary means of treatment. Any temporary liquid treatment system necessary to ensure that disposal requirements are met will be connected to tanks for storage of processed water prior to discharge. Once it has been verified that the stored processed water meets the allowable discharge limits specified in the Technical Specifications, the water may be released. The effluent monitoring instrumentation will be used to monitor discharges of liquid effluent as required, and to demonstrate compliance with Technical Specifications and applicable regulations.

Filters will be replaced as appropriate to keep exposures as low as reasonably achievable. System components will be positioned or shielded as required to maintain dose rates to workers as low as reasonably achievable.

Makeup water used for flushing will generally be from the existing potable water supply. However, water that has been processed through a temporary system may be used if quantities and economics suggest a savings without a reduction in safety. The effluent stream(s) from such activities will be processed as above, by filtration and demineralization.

If radioactively contaminated water is disposed to the sanitary sewer, the discharge pathway will be resurveyed and remediated as necessary.

3.2.3.2 Onsite Evaporation

Low-level radioactively contaminated water may be evaporated onsite. The facilities effluent monitoring program, used to monitor airborne effluents, localized airborne monitoring, or evaluation of the radioactivity released based upon sampling of media can be used to demonstrate compliance with Technical Specifications and applicable regulations.

3.2.3.3 Transfer for Offsite Treatment

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. UM may elect to transfer all or some of the liquid radioactive waste from decommissioning to a licensed waste processor based upon its evaluation of the potential for success and associated costs of offsite treatment.

3.2.3.4 Liquid Radioactive Waste Containing Chelating Agents

There are no current plans to utilize chelating agents in any chemical decontamination activities for FNR systems or structures. Radioactive wastes containing chelating agents will be generated only if necessary, and in that case will be minimized to the fullest extent possible.

FNR will continue to monitor requirements for packaging and disposal of radioactive waste containing chelating agents. No radioactive waste containing chelating agents will be generated during FNR facility decommissioning operations that result in packaged radioactive waste that is not consistent with waste form, packaging transportation and disposal requirements existing at the time when the operations are performed.

3.2.3.5 Tritium Loaded Heavy Water

Tritium loaded heavy water, owned by the DOE, will be returned to the DOE's Savannah River Site for processing and reuse.

3.2.4 Solid Radioactive Waste Disposal

The majority of solid waste will be a direct result of the decontamination and dismantlement of activated and contaminated systems, structures, or components.

Information on the estimated curie content, volume, and waste classification for this decommissioning project is extremely limited at this time. Additional information is required to determine the waste classification. The estimates of waste volumes are conservative and do not account for any volume reduction techniques and, further, any estimates assume only direct burial rather than allowing for decontamination and possible free release.

Solid radioactive waste is expected to be primarily Class A waste.

FNR is planning a number of measures to reduce the volume of solid radioactive waste that will require disposal at a licensed burial facility. The primary components of the solid waste to be generated by the decommissioning of the FNR facility are expected to be disposed of as follows:

3.2.4.1 Irradiated Reactor Hardware

Irradiated reactor hardware may require size reduction to facilitate loading. Irradiated reactor hardware will be loaded into a HIC or liner then placed in an approved, shielded shipping cask for transport and subsequent direct burial at the licensed land disposal facility in Barnwell, South Carolina. The current estimate for the volume of irradiated reactor hardware requiring burial at Barnwell is 300 cubic feet. Activities to complete the characterization and volume determination of these items are discussed in Section 2.2.2.5.1.

Irradiated hardware with dose levels that permit processing by an offsite, licensed radioactive waste processor, generally less than 200 mrem per hour on contact, may be transferred to an offsite processor for volume reduction, recycling, or treating the radioactive metal wastes (i.e. recycling, metal melt, super-compaction, encapsulation). UM will use offsite processing for some or all of the radioactive metal waste depending on whether it reduces the total cost associated with the final disposition of the radioactive waste and has a high potential of success.

Cask shipments will comply with the quality assurance requirements discussed in Section 1.3.4.3.

3.2.4.2 Piping, Equipment and Other Metals

The contaminated systems piping and equipment will be segmented. As cuts are made, a suitable cover will be placed on open ends to preclude the spread of contamination. The small bore piping may be removed from the system, packaged, and shipped off-site to a licensed vendor offering decontamination and volume reduction services. Large bore piping may be moved directly to the packaging area. As the containers are filled, they will be moved to a staging area awaiting final preparation and loading for shipment off site to a volume reduction facility. Components and instruments will be bagged for contamination control and handled in a manner designed to minimize contamination spread as required by the health physics program. Containments will be selected by considering the proper size, contamination levels and ability to process. Material that can be economically dismantled and decontaminated will be appropriately handled onsite or sent to a vendor facility. Material that cannot be economically decontaminated will be placed in proper disposal containers (e.g. low specific

activity [LSA] containers) and sent to an appropriate processor or burial facility. 5300 cubic feet of activated or contaminated material is estimated for processing or disposal.

Offsite, licensed radioactive waste processors provide specialized services for volume reduction, recycling, or treating the majority of the radioactive metal wastes generated during decommissioning activities. These include such processes as decontamination, recycling, (such as the U.S. Navy's lead reuse program), metal melt, super-compaction, general sorting, encapsulation, and survey and release. UM may use offsite processing for some or all of the radioactive metal waste depending on whether it reduces the total cost associated with the final disposition of the radioactive waste and has a high potential of success.

The accumulation of contaminated equipment, piping, or other materials will not represent an exposure concern because of the generally low dose rates from these items.

3.2.4.3 Concrete, Concrete Rubble and Dust

Activated or contaminated concrete removed in large sections will be packaged as LSA material in approved shipping containers for direct shipment to the licensed land disposal facility operated by Envirocare of Utah, Inc. An estimated 5200 cubic feet of activated or contaminated concrete, two-thirds of the concrete making up the reactor pool will require disposal in this manner.

Activated or contaminated concrete rubble and dust may be packaged as LSA material in approved shipping containers. When feasible, this material will be used to fill void space in other radioactive waste shipping containers where allowed under the waste acceptance guidelines for the licensed waste disposal facility.

3.2.4.4 Dry Active Waste

DAW consisting of contaminated paper, plastic, coveralls, etc. will be packaged as LSA material in approved shipping containers. DAW will be shipped non-compacted to an offsite vendor for volume reduction and processing if supported by ALARA and cost considerations. When feasible, DAW will be used to fill void space in other radioactive waste shipping containers. If the preceding is not reasonable, the DAW may be shipped for direct burial. An estimated 300 cubic feet of DAW will require transfer to a licensed waste disposal facility for post-processing and disposal.

3.2.4.5 High Efficiency Particulate Air Filters

Engineering controls such as HEPA, filtered ventilation will be required to capture potential airborne contaminants. Spent HEPA filters will be changed out and treated as DAW radioactive waste. An estimated 25 cubic feet of contaminated filter media will require transfer to a licensed waste disposal facility for post processing and disposal.

3.2.4.6 Resins and Filters

Radioactive waste treatment systems will be required to process the liquid waste stream resulting from various decommissioning activities as described above. Filtration and ion exchange processing will be used to remove residual radioactivity in the water. Temporary demineralization and filtration systems may be supplied by a vendor or by FNR. The volume of spent resins and filters required to process the water is estimated to be less than 400 cubic feet.

Resins generated by water processing systems will be transferred to a licensed waste disposal facility for processing and disposal.

3.2.4.7 Asbestos

Contaminated asbestos waste is not expected but may be identified by decommissioning or preparatory activities. Asbestos material should be transferred to an offsite, licensed radioactive waste processor for compaction or for survey and release. Large items containing asbestos waste may require size reduction before transfer to the offsite, licensed radioactive waste processor.

3.2.4.8 Mixed Waste

The only known mixed waste at the FNR is from lead shielding, possibly lead paint and cadmium. There are approximately 13,000 pounds of contaminated lead, 1,600 pounds of activated lead, approximately 400 pounds of contaminated cadmium, and approximately 20 pounds of activated cadmium. These materials will be encapsulated or otherwise treated by a vendor for ultimate disposal or recycle.

Lead paint will be acceptable for burial as part of the component that was painted, since the fraction of lead available for leaching appears to be less than the maximum leached fraction allowed.

Lead paint chips will be acceptable for burial if the leaching appears less than the maximum leached fraction allowed.

UM's objective is to generate no new mixed waste during decommissioning activities. Procedures currently in place for hazardous and radiological waste management are sufficient to provide the assurance that waste will not be generated arbitrarily and that generated wastes will be disposed of properly. At this time, no processes are planned to be used during decommissioning that will create a non-treatable mixed waste.

However, in the event that a mixed waste is identified or inadvertently generated, the programs in place for hazardous waste and for radioactive waste establish the responsibilities, controls, and practices necessary to appropriately handle the waste.

3.2.5 Method of Estimating Types, Amounts and Radionuclide Concentrations of Radioactive Waste Generated During Decommissioning

The estimate of total radioactivity present in systems, structures, or components will be derived directly from field radiological measurements, supplemented by analytical data or through computational estimates. These estimates may also be made from direct measurements, which can include, but are not limited to:

- Limited sampling to establish ratios of radionuclides present in a structure or component.
- Direct analysis using NaI, HPGe, or other detectors to analyze the gamma spectrum being emitted to identify specific isotopes, establish ratios of isotopes, or to fully quantify isotopes.
- Direct measurement of dose rates to support computational methodologies for the determination of radionuclides (e.g. MicroShield [Grove Engineering 1996] or hand calculations).
- Direct measurement of similar items for extrapolation via computational methods for inaccessible components or structures.

Estimates of irradiated items may be based on the constituent elements of the material in question and by calculating the duration of exposure and the energies of the incident neutrons (Erdman, 1976).

The activity present within internally contaminated piping and on plant structures will be determined by radiological surveys.

3.3 General Industry Safety Program

The RSO, with the cooperation of the full project management team discussed in Section 2.4, will be responsible for ensuring that the occupational health and safety requirements of project personnel and the general public are met. The primary functional responsibility is to ensure compliance with the Occupational Safety and Health Act (OSHA) of 1973 and the Michigan Occupational Safety and Health Act (MIOSHA) of 1974. Specific responsibilities include establishing training requirements for project personnel in general safe work practices; reviewing plans and procedures to verify adequate coverage of industrial hygiene and safety requirements and concerns; conducting periodic inspections of work areas and activities to identify and correct any unsafe conditions and work practices; coordinating industrial hygiene services as required; and advising the Director on industrial hygiene and safety matters, and on the results of periodic safety inspections.

All personnel working on the FNR Decommissioning Project will receive health and safety training in order to recognize and understand the potential risks to personnel health and safety associated with the work at the FNR. The health and safety training also ensures compliance with the applicable requirements of the NRC (10CFR), the EPA (40CFR), OSHA (29CFR), and MIOSHA (Act 154 of 1974). Personnel will be trained on the plans, procedures, and operation of equipment to conduct themselves safely on the FNR Decommissioning Project.

The implementation of occupational health and safety requirements for activities involving significant or unfamiliar hazards will be through the use of a JHA. Each JHA will identify all hazards associated with the activity (e.g. fall protection, hot work, confined space). A procedure implementing the JHA will be prepared and subject to the approval requirements discussed in Section 2.4. The JHA allows the project management, project staff, contractor staff, and UM industrial safety personnel (through the RSO or Reactor Manager) to specify the controls, process, or other controls necessary to protect the safety of individual workers, the UM community, and the public. The JHA will act in concert with the RWP, if required, to complete the protection program. JHAs will be approved by a representative of the UM industrial safety staff, the RSO, or the reactor manager. In their absence, the RSO and the reactor manager can delegate this approval authority.

3.4 Radiological Accident Analyses

The current licensing basis for the Ford Nuclear Reactor describes the accident(s) perceived to have the highest consequences: 1) Failed experiment as described in Section 14.3 of the Ford Nuclear Reactor Safety Analysis, Revision 3, 2) Abnormal Loss of Coolant as described in Section 14.2 of the Ford Nuclear Reactor Safety Analysis and 3) Rupture of the Heavy Water Tank as described in License Amendment No. 35 and updated in License Amendment No. 36. Note: Accidents involving spent fuel shipping casks are NOT evaluated in the Safety Analysis nor are they being required in recent license renewals for non-power reactors. 1

The maximum hypothetical accident for licensed activities remains bounded by those analyzed for conditions where the reactor is operating. The accidents analyzed in the current licensing basis all state the maximum consequence, or dose, for each of the accidents listed above as 50 mem or 400 times the AEC to an individual member of the public.

The decommissioning plan accident analyses concluded that the credible postulated accidents remain bounded by the maximum hypothetical accident in the currentl licensing basis. Thus the description presented below for each of the accidents proposed shows that it is reasonable to conclude that none of the proposed accidents would replace the maximum hypothetical accident contained in the current licensing basis. 1

Potential radiological accidents during decommissioning of the FNR were evaluated by evaluating FNR areas that contain the highest inventories of radioactive material, reviewing proposed decommissioning activities, and considering combinations of these elements that could lead to a release of radioactive material. This 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
- Pool leak
- Tritium loaded heavy water spill

3.4.1 Fire

The consequences of a fire during decommissioning of the FNR were considered and are not significantly different than the consequences of a fire during reactor operations. Most materials in the FNR are metals, concrete, or similar non-combustible materials. Upon termination of reactor operation most of the combustible materials required for reactor operations were removed from the reactor building to further reduce the potential consequences of a fire. The likelihood that a fire would start or that a fire could become intense enough to release radioactive material is remote. The impact of the release of radioactive materials from a fire involving dry radioactive waste (i.e., rags, wipes, and anticontamination clothing) is presented.

Dry radioactive waste is normally collected in metal pails with lids located throughout the facility. Once full, the dry waste is normally transferred into 55 gallon drums meeting the strong-tight requirement for shipment to a licensed waste processor. Small quantities of dry radioactive waste requiring special handling or segregation are stored in plastic 5 gallon pails. This practice limits the volume of dry radioactive waste available for consumption by fire to a few pounds and lowers the potential for a fire to consume additional waste collections. Any fire in dry radioactive waste would be limited to a few microcuries of radioactivity from the radionuclides contained in the list of expected radionuclides, Table 2-4.

During a fire in dry radioactive waste the emission of airborne radioactivity up the FNR Exhaust stack would continue unless operator action is taken or upon automatic action when the radioactivity levels exceed 1 mrem per hour at the building exhaust radiation monitor (required by the Technical Specifications). For the purposes of this evaluation, no initiation is assumed, credit is taken for the FNR exhaust stack dilution factor of 400, (Technical Specifications and the current licensing basis), an emission rate of a minimum of 8, 000 cubic feet per minute (slow speed exhaust fan) up the FNR exhaust stack is used, and an 8 hour fire is assumed.

The quantities of individual radionuclides from Table 2-4 that can individually pass up the FNR stack during a fire without exceeding the airborne effluent concentration (AEC) limits for a full year as specified in 10 CFR 20, Appendix B, Table 2, *Effluent Concentrations-Air* are presented in Table 3-6.

TABLE 3-6, QUANTITIES OF INDIVIDUAL EXPECTED RADIONUCLIDES PRODUCING THE EMISSION OF THE ANNUAL EFFLUENT CONCENTRATION DURING AN 8 HOUR FIRE (2 PAGES)

Nuclide	Individual Quantity [1]
Antimony-125 (Sb-125) W class	130 mCi
Bismuth-210m (Bi-210m) D class	0.4 mCi
Cadmium-109 (Cd-109) W class	8.7 mCi
Carbon-14 C-14 (Monoxide)	86 Ci
Cesium-134 (Cs-134) D class	8.7 mCi
Cesium-137 (Cs-137) D class	8.7 mCi
Cobalt-60 (Co-60) W class	8.6 Ci
Europium-152 (Eu-152) W, All classes	390 mCi
Europium-154 (Eu-154) W, All classes	1.3 mCi
Iron-55 (Fe-55) W class	260 mCi
Manganese-54 (Mn-54) All classes	44 mCi
Nickel-59 (Ni-59) W class	430 mCi
Nickel-63 (Ni-63) W class	173 mCi
Scandium-46 (Sc-46) Y, All classes	13 mCi
Silver 108m (Ag-108m) W class	17 mCi
Silver 110m (Ag-110m) W class	13 mCi
Tritium (H-3)	4.3 Ci
Zinc-65 (Zn-65) Y, All compounds	17 mCi

Note:

[1] Activity = AEC* 28,317 cc/ft^3*8,000 cfm*60 min/hr*8 hr * 400

AEC – airborne effluent concentration, Ci – Curies, mCi - millicuries

These quantities of radionuclides are in the mCi range and are significantly greater than the levels expected in any localized, individual containers of dry radioactive waste (i.e., rags, wipes, and anticontamination clothing).

A fire in dry radioactive waste containing a single quantity of any radionuclide listed in Table 3-6 or containing a quantity by fraction of the radionuclides listed in Table 3-6, which sum to unity, would result in a maximum exposure to an individual of 50 mrem.

During a fire in dry radioactive waste where the ventilation system is secured shortly after the initiation of the fire, the exposure would be limited to those individuals who provide initial fire suppression activities, those individuals who evacuated the facility and those individuals who are required to reenter to the reactor building.

The construction of the reactor building provides three large volumes in which hot gases from a fire would collect. One of these volumes is the area above the reactor pool, the other two areas are just below the third floor on both the east and west sides of the reactor pool. This inherent design feature aids in the reduction of the concentration of radioactive materials in the breathing space near the first, second, and third floors of the reactor building, and limits the inhalation of radioactive materials of individuals who provide initial fire suppression activities, those individuals who evacuated the facility, and those individuals who are required to reenter to the reactor building.

In the event of a fire, individuals present in the facility may make a reasonable attempt to extinguish the fire using the portable extinguishers provided throughout the facility. If the fire cannot be extinguished, the Ann Arbor Fire Department is summoned, as discussed in the FNR Emergency Plan. The exposure seen by individuals during the short period while attempting to extinguish the fire in the dry radioactive waste or evacuating the area would be minimal. Fire fighting personnel responding to a fire potentially involving radioactive materials utilize a self-contained breathing apparatus, minimum protection factor of 100, which would ensure that any internal exposure would be significantly less that the 50 mrem analyzed above.

3.4.2 Pool leak

In the event of a major leak from the reactor pool all water lost would be collected by the floor drains or pass through openings in the first floor to the basement of the reactor building. The dimensions of the reactor basement are large enough to allow for the collection of all 50,000 gallons of water from the reactor pool. Loose radioactive contamination in the water's pathway to the reactor basement would be entrained, but should not cause an increase in the radioactive materials content of the pool water above the levels experienced while the reactor was operating. The resulting levels of radioactive material from the evaporation of the water spread over the basement and first floors would be limited to the tritium contained in the water and would be less than the evaporation rates experienced from the surface or the reactor pool while the reactor was operating. The other radionuclides would remain in the facility.

Note: During normal operation of the reactor pool, prior to reactor shutdown, the 240 square feet of the pool's surface was maintained between 90 degrees Fahrenheit and 116 degrees Fahrenheit with an estimated evaporative loss rate of 4 gallons per hour.

3.4.3 Tritium-Loaded Heavy Water Spill

The consequences of a spill from a 55 gallon drum of tritium-loaded heavy water would be the emission of tritium via the FNR Exhaust stack. Taking credit for the FNR exhaust stack dilution factor of 400, (Technical Specifications and the current licensing basis) and assuming that the emission of 8, 000 cubic feet per minute up the FNR exhaust stack, the emission of tritium from the facility based upon the AEC limit for a full year as specified in 10 CFR 20, Appendix B, Table 2, *Effluent Concentrations-Air* is 9.1 mCi per hour. The most concentrated tritium-loaded heavy water is contained in the heavy water reflector. At 217 Ci of tritium (April 2004) in an estimated 50 gallons, the highest estimated concentration of tritium is 1.1 mCi per ml. Given this concentration, a spill from this tank would require the evaporation rate be limited to approximately 9 ml per hour if the emission were averaged over an entire year. Any spill of tritium-loaded heavy water could be easily flushed to the floor drains for collection in the hot and cold sumps and eventual collection in the retention tanks. Conservatively, one week or less would be needed to cleanup the spill or to stop the tritium evaporation. This allows the emission rate to increase to 473 mCi per hour over one week. This equates to an evaporation rate of one pint per hour of the tritium-loaded heavy water for the entire week of cleanup

activities. The emission of tritium at a rate of 473 mCi per hour for the one week of cleanup would result in a maximum exposure to an individual of 50 mrem.

In the event of a spill of the heavy water reflector while still in the reactor pool, the dilution by the water in the pool would decrease the concentration of the tritium in the water source and result in a lower emission rate of tritium from the facility (See license amendment No. 35 and No. 46).

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4.0 Proposed Final Status Survey Plan

4.1 General Survey Approach

All factors influencing the final status survey for the FNR are not available and will not be available until more details on the facility are evaluated following additional characterization activities to be conducted following the approval of the decommissioning plan. This section provides the outline for the Final Status Survey Plan. The outline for the final status survey plan provided in this section is intended to provide information to the NRC in determining the adequacy of the licensee's understanding of the final status plan as it pertains to the goal of remediation in a manner satisfying the radiological criteria for license termination. The final status survey plan, which will be formally submitted to the NRC for approval at a later date (see Section 5.0 for requested license condition), will adequately demonstrate compliance with the radiological criteria for license termination. This is consistent with the methodology used by the NRC for the termination of the license for the UVa license (NRC, 2002). Upon approval by the NRC this section of the decommissioning plan will be replaced with the Approved Final Status Survey.

This proposed survey plan was prepared in accordance with the guidelines and recommendations presented in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual"* (MARSSIM) (NRC, 2000b). The process emphasizes the use of Data Quality Objectives (DQOs) and Data Quality Assessment, along with a quality assurance and quality control program. The graded approach concept will be followed to assure that survey efforts are maximized in those areas having the greatest potential for residual contamination or the highest potential for adverse impacts of residual contamination.

Final Status Surveys (FSS) will be performed by trained radiological control technicians, who are following standard, written procedures and using properly calibrated instruments, sensitive to the potential contaminants.

Also, designs for specific surveys for some areas, including determination of specific nuclide mixture guidelines, sampling or measurement methods, survey unit identification and classification, and data evaluation techniques, may be developed at the time of survey in accordance with the guidance presented in this proposed plan.

4.2 Final Status Survey Quality Assurance Program

4.2.1 General

The UM will be responsible for developing a Final Status Survey Quality Assurance (FSS QA) program appropriate for the final status survey and associated documentation (e.g. characterization information used in the design of the final survey). The FSS QA program will be reviewed and approved as described in Section 2.4. The FSS QA program will incorporate the appropriate regulatory requirements applicable to the planning and conduct of radiological surveys necessary for the termination of the FNR license and the release of the site for unrestricted use.

The quality assurance program presented here implements the appropriate criteria taken from 10 CFR 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*. The following sections describe the required components of the FSS QA program:

4.2.2 Organization

Written definitions of authority, duties, and responsibilities of managerial, operations, and safety personnel; a defined organizational structure; assigned responsibility for review and approval of plans, specifications, designs, procedures, data, and reports; and assigned responsibility for procurement and oversight of services (e.g., analytical laboratory) are identified in Section 2.4. Personnel assigned organizational responsibility for performing QA functions will be given the necessary independence and authority to allow them to identify quality problems; to initiate, recommend, and provide solutions; and to verify implementation of solutions.

The Reactor Manager will provide overall management and execution for implementing all aspects of the FSS QA program. The Reactor Manager will ensure that survey activities meet the requirements outlined in the FSS QA program to safeguard the decommissioning staff, the UM community and the public. The Reactor Manger will regularly review the adequacy of the FSS QA program, and provide an assessment to the Director and the review committee. The Reactor Manager will inform the appropriate UM decommissioning staff and contractors on decommissioning activities related to the FSS QA program.

The Project Manager will ensure that the contractor complies with FSS QA program, satisfies the objectives and requirements for final status survey, and that all activities are performed in a manner to permit the termination of the FNR license and the release of the site for unrestricted use. The individual(s) or organization(s) responsible for establishing and executing the FSS QA program may delegate any or all of the work to others but shall retain responsibility there-for (ASME 2001, Requirement 1).

4.2.3 Written Quality Assurance Program

A documented quality assurance program for the final status survey and associated documentation (e.g. characterization information used in the design of the final survey) shall be established at the earliest practical time, consistent with the schedule for accomplishing the activities. This quality assurance program shall be documented through written polices, procedures or instructions and shall be carried out through the conduct of activities for the final status survey and creation of associated documentation in accordance with those policies, procedures, or instructions. Activities for the final status survey and creation of associated documentation affecting quality shall be accomplished under suitably controlled conditions. Controlled conditions included the use of appropriate equipment, suitable environmental conditions for accomplishing the activity, and assurance that prerequisites for the given activity have been satisfied. The quality assurance program shall provide for any special controls, processes, survey equipment, tools, and skill to attain the required quality of activities and items and for verification of that quality.

4.2.4 Training

Personnel will be qualified for their assigned duties before working independently or will be under the direct supervision of a qualified individual. Personnel performing special processes will be qualified according to specific codes and standards or in accordance with national consensus documents. Qualification will include proficiency demonstrated by each individual,

both initially and then periodically. Qualification also will be demonstrated when required by the designated codes or standards.

Training records will be maintained and will include the trainee's name, dates of training, types of training, test results, protective equipment use authorizations, and instructors' names.

Care will be taken to ensure that properly qualified instructors conduct all training. As the primary criteria, persons responsible for presentation of training should have knowledge and experience in the process or subject matter. It is desirable that trainers also have the presentation skills or classroom conduct appropriate to the level of the training being presented. For those with limited background in training, early instruction should be monitored and feedback should be provided.

4.2.5 Quality Assurance Records

Sufficient records will be specified, prepared, reviewed, authenticated, and maintained to reflect the achievement of the required quality. Records will include documents such as operating logs, results of reviews, inspections, tests, assessments, work performance monitoring, and material or sample analyses. Records will be identifiable, available, and retrievable. The records will be reviewed to ensure their completeness and ability to serve their intended function. Requirements will be established concerning record collection, safekeeping, retention, maintenance, updating, location, storage, preservation, administration, and assigned responsibility. Requirements will be consistent with applicable regulations and the potential for impact on quality and radiation exposure to workers and the public.

Policies, procedures or instructions that specify quality requirements or prescribe activities affecting quality, such as instructions, procedures and drawings, will require control and will be identified. Policies, procedures, or instructions (including revisions) will be reviewed by qualified personnel for conformance with technical requirements, and quality system requirements and will be approved as discussed in Section 2.4. Policies, procedures, or instructions requiring control shall be kept current for use by personnel performing activities. Measures will be taken to ensure that personnel understand the document controls to be used. Obsolete or superseded documents will be identified and measures will be taken to prevent their use.

All documents related to the final survey documentation will be controlled by appropriate policies, procedures or instructions. All significant changes to such documents will be similarly controlled. This documentation normally would include a survey plan, survey packages, survey results, and a survey report.

4.2.6 Control of Measuring Equipment

Measures shall be established to assure that instruments and other measuring devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

Selection of instruments shall be based on the type, range, accuracy, and tolerance needed to accomplish the required measurements for determining conformance to specified requirements. Selection and use of instrumentation for the final status survey will also be based upon the need to ensure that the residual radioactivity remaining on site meets the release criteria. Table 4-1 lists the instrumentation intended for use for the final status survey and associated documentation (e.g. characterization information used in the design of the final survey), along

with estimated detection sensitivities. Other instruments, which are the functional equivalent of those listed, will also be acceptable.

Because radionuclides present as contaminants emit (with few exceptions) beta particles with maximum energy greater than 0.300 megaelectron volts (MeV), detector efficiencies for measuring surface activity are generally determined using Tc-99 (maximum beta energy of approximately 0.292 MeV). For situations where contaminants emit beta particles of lower energy, e.g., facilities contaminated with Ni-63, detector efficiencies are specifically determined for those contaminants.

TABLE 4-1, INSTRUMENTATION FOR FNR FINAL STATUS SURVEY

Detector	Type	Make	Meter	Application	Sensitivity (dpm/100 cm^2, except as noted)	
					Scanning	Static Count (1 minute)
43-68	Gas Proportional	Ludlum	2221	Beta scan and measurement	1200	500
43-68	Gas Proportional	Ludlum	2221	Ni-63 Beta scan and measurement	5000	2000
43-37	Floor Monitor	Ludlum	2221	Beta scan	800	N/A
43-68	Gas Proportional	Ludlum	2221	Alpha measurement	200	70
Tennelec LB5100	Gas proportional	Tennelec	N/A	Alpha smear measurement	N/A	5
Tennelec LB5100	Gas proportional	Tennelec	N/A	Beta smear measurement	N/A	10
44-10	NaI	Ludlum	2221	Gamma scan	10 pCi/g	N/A

cm^2 – square centimeter, dpm – disintegrations per minute, g – gram., pCi – picocuries.

Effects of surface conditions on measurements are integrated into the overall instrument response through use of a "source efficiency" factor, in accordance with the guidance in ISO-7503-1, *Evaluation of Surface Contamination – Part 1: Beta Emitters and Alpha Emitters* (First Edition) (ISO, 1988), and NUREG/CR-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Fields Conditions* (NRC, 1997b). Default source efficiency factors, of 0.5 for beta-emitters > 0.4 MeV E_{max} and 0.25 for beta-emitters between 0.150 MeV and 0.400 MeV E_{max} (per ISO-7503-1) are generally applicable to anticipated FNR contaminants and surface conditions. However, if contaminants or conditions are not consistent with use of these default values, specific source efficiency factors will be determined and documented in the final status survey design.

Detection sensitivities are estimated, using the guidance in NUREG-1575 (NRC, 2000b) and NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Fields Conditions* (NRC, 1997b). Instrumentation and survey techniques are chosen with the objective of achieving detection sensitivities of ≤ 25 percent of the criteria for structure surfaces, for both scanning and direct measurement. This assures identification of areas of elevated activity, having a size and activity level that could adversely impact the average for the survey units.

Calibration procedures shall identify or reference required accuracy. Methods and checking accuracy shall be defined in procedures and shall follow American National Standard ANSI N323-1978 (HPS, 1978). The calibration method and interval of calibration for instruments shall be defined, based on the type of equipment, stability characteristics, required accuracy, intended use, manufacturer's recommendations and other conditions affecting capability, and shall follow ANSI N323-1978 (HPS, 1978). Out of calibration and defective instruments will be removed from service and not used until they have been repaired and recalibrated. Instruments consistently found to be out of calibration shall be repaired or replaced.

Measuring instruments shall be calibrated at prescribed time periods or usage and whenever the accuracy of the equipment is suspect. Calibration shall be performed using standards traceable to NIST or an equivalent standard organization.

Instruments found to be out of calibration, shall require a documented evaluation, commensurate with the significance of the condition, of the validity of data obtained with that instrument since its previous acceptable performance.

Instruments shall be properly handled and stored to maintain accuracy and shall follow ANSI N323-1978 (ANSI, 1978).

Instruments shall be suitably marked or otherwise identified to indicate calibration status.

Operational and background checks will be performed at the beginning of each day of final status survey activity and whenever there is reason to question instrument performance. These checks should follow ANSI N323-1978 (ANSI, 1978).

4.2.7 Audits and Corrective Actions

Project audits will be planned and conducted using criteria that describe acceptable work practices, including performance. Audits will verify compliance with applicable requirements of the FSS QA program and will determine its effectiveness. The scheduling of audits and allocation of resources will be based on the work status, risk, and complexity of the item or process being assessed. Audits will be performed as described in Section 2.4. Audit results will be reported to and reviewed by management as discussed in Section 2.4.

Conditions adverse to quality shall be identified to the Reactor Manager promptly and corrected as soon as practicable. Significant condition adverse to quality shall be identified to the review committee as soon as practicable along with the cause of the condition, when known, and corrective actions taken to prevent recurrence.

4.3 Isolation following Remediation

4.3.1 Isolation Criteria

The following criteria will be satisfied prior to acceptance of a survey unit for final status survey. The physical aspects of these criteria are verified during the walk-down.

- Planned dismantlement activities within the post remediation survey unit are completed.
- Planned dismantlement activities affecting or adjacent to the post remediation survey unit are completed, or are evaluated and determined to not have a reasonable potential to introduce radioactive material into the post remediation survey unit.

- An operational radiation protection survey of the post remediation survey unit is completed and all outstanding items are addressed.
- Planned physical work in, on, or around a post remediation survey unit, other than routine surveillance or maintenance, is complete.
- Tools, non-permanent equipment, and material not needed for survey data collection are removed.
- Housekeeping, clean up, and remediation of the survey unit are completed.
- Scaffolding, temporary electrical and ventilation equipment and components, and other material or equipment needed for survey data collection is radiologically clean and left in place.
- Transit paths to or through the post remediation survey unit are eliminated or re-routed.
- Appropriate measures are instituted to prevent the re-introduction of radioactive material into the isolated area from ventilation systems, drain lines, system vents, and other potential airborne and liquid contamination pathways.
- Measures are instituted to control access and egress and otherwise restrict radioactive material from entering the survey unit.

4.3.2 Transfer of Control

Once a walk-down has been performed and the isolation criteria are met, control of activities within the post remediation survey unit is transferred solely to the Reactor Manager and the RSO. The need for localized remediation within the isolated area may be identified after transfer of control. Localized remediation may be performed under the control of the Reactor Manager or the RSO. However, if large areas require remediation, the isolated area may be returned for further decontamination.

4.3.3 Isolation and Control Measures

Prior to performing the final status survey, the post remediation survey unit is isolated and controlled. Routine access, equipment removal, material storage, and worker and material transit through the area without proper controls are no longer allowed. One or more of the following administrative and physical controls will be established to minimize the possibility of introducing radioactive material from ongoing decommissioning activities in adjacent or nearby areas:

- Personnel training
- Installation of barriers to control access to the area(s)
- Installation of postings with access and egress requirements
- Locking or otherwise securing entrances to the area

4.4 Data Quality Objectives

The objective of the final status survey is to demonstrate that the radiological conditions of the facility satisfy the decommissioning criteria (see Section 2.1.5). The DQOs permit demonstration at the 95 percent confidence level that these criteria are met. Decision errors are 5 percent for both Type I and Type II errors. Such a Type I (alpha) decision error provides a confidence level of 95 percent that the statistical tests do not incorrectly determine that a

surveyed area satisfies criteria when, in fact, it does not. The Type II (beta) decision error provides a confidence level of 95 percent that the statistical tests do not incorrectly determine that a surveyed area does not satisfy criteria when, in fact, it does. Measurement sensitivities ≤ 25 percent of DCGLs enable quantification of contaminants at or below the guideline values at the 95 percent confidence level.

Data quality indicators for precision, accuracy, representativeness, completeness, and comparability, are as follows:

- Precision is determined by comparison of replicate values from field measurements and sample analyses; the objective is a relative percent difference of 20 percent or less at 50 percent of the guideline value.
- Accuracy is the degree of agreement with the true or known value; the objective for this parameter is +/- 20 percent at 50 percent of the guideline value.
- Representativeness and comparability do not have numeric values. Performance is assured through selection and proper implementation of sampling and measurement techniques.
- Completeness refers to the portion of the data that meets acceptance criteria and is thus acceptable for statistical testing; the objective for this survey is 90 percent.

4.5 Classifications of Areas by Contamination Potential

For the purposes of guiding the degree and nature of final status survey coverage, MARSSIM (NRC, 2000b) first classifies areas as *impacted*, i.e., areas that may have residual radioactivity from licensed activities, or *non-impacted*, i.e., areas that are considered unlikely to have residual radioactivity from licensed activities. Non-impacted areas do not require further evaluation. For impacted areas MARSSIM (NRC, 2000b) identifies three classifications of areas, according to contamination potential.

- Class 1 Areas: Impacted areas that, prior to remediation, are expected to have concentrations of residual radioactivity that exceed the guideline value.
- Class 2 Areas: Impacted areas that, prior to remediation, are not expected to have concentrations of residual radioactivity that exceed the guideline value.
- Class 3 Areas: Impacted areas that have a low probability of containing residual activity. Typically levels will not exceed 25-35 percent of the guideline value.

Facility history (including the Historical Site Assessment [CH2M HILL, 2003]) and radiological monitoring conducted during characterization and remedial activities are the bases for classification.

Once approval for the Final Status Survey is obtained through a subsequent license amendment request to the NRC, the UM may make changes to the classification of an area as long as the classification is changed to one of higher contamination potential. A license amendment pursuant to 10 CFR 50.90 shall be obtained if the change would decrease an area classification (i.e., impacted to non-impacted, Class 1 to Class 2, Class 2 to Class 3, or Class 1 to Class 3), see Section 9.0.

4.6 Identification of Survey Units

Impacted areas are divided into survey units for implementing the final status survey. A survey unit is a portion of a facility with common contaminants and contamination potential and contiguous surfaces or areas. **Error! Reference source not found.** Table 4-2 lists the survey unit areas suggested by MARSSIM (NRC, 2000b) for application at the FNR facility. The area of individual survey unit will follow these suggested maximum sizes. Impacted structure surfaces of ≤ 10 m^2 and impacted land surfaces of ≤ 100 m^2 will not be designated as survey units. Instead, a minimum of 4 measurements (or samples) will be obtained from such areas, based on judgment, and compared individually with the DCGL.

TABLE 4-2, MARSSIM – RECOMMENDED SURVEY UNIT AREAS

Class	**Recommended Survey Unit Area**	
	Structures	Land
1	up to 100 m^2	up to 2000 m^2
2	100 to 1000 m^2	2000 to 10,000 m^2
3	no limit	no limit

m^2 – square meter

Survey units will be identified following remediation, at the time of final survey design. Based on a historical assessment, preliminary survey data obtained in November 2002, and the characterization survey in April 2003 (see Appendix A) a listing of facility areas that are currently expected to be included in the final status survey, the estimated areas, anticipated contamination potential classifications, and the projected number of survey units within each area is provided in Table 4-3. Actual survey unit boundaries and classifications will be determined at the time of final status survey design, and survey unit classifications and surface areas may change as characterization and remedial activities proceed. If classifications and boundaries change, surveys will be redesigned and the survey and data evaluation will be repeated, as necessary. Classifications and survey unit boundaries may change, based on results as the final status survey progresses. If classifications or boundaries change, the survey of the survey unit will be redesigned and the survey and data evaluation repeated.

TABLE 4-3, MARSSIM – RECOMMENDED FNR SURVEY AREAS AND INITIAL FINAL STATUS SURVEY CLASSIFICATIONS

Room or Area	**Surface**	**Class**	**Approx. Surface Area (m^2)**	**No. of Survey Units**	**Remarks**
Basement	Floor and walls	1	290	3	
Basement	Ceiling	1	190	2	
Basement	Pits and sumps	1	20	1	Smaller pits and sumps not surveyed as Survey Units, due to small areas
1st Floor (includes JC 1103)	Floor and lower walls	1	650	7	
1st Floor	Upper walls and ceiling	2	650	1	
1st Floor	Pool wall (remaining)	1	100	1	May be removed
1st Floor	Source storage ports	1		N/A	Not surveyed as Survey Units, due to small area

TABLE 4-3, MARSSIM – RECOMMENDED FNR SURVEY AREAS AND INITIAL FINAL STATUS SURVEY CLASSIFICATIONS

Room or Area	Surface	Class	Approx. Surface Area (m^2)	No. of Survey Units	Remarks
2nd Floor Rm 2111	All	2	250	1	
2nd Floor Rm 2109	All	2	70	1	
2nd Floor Rms 2106/2107/2108	All	2	260	1	
2nd Floor Rms 2105/2102	All	2	140	1	
2nd Floor Rms 2103/2104	All	2	55	1	
2nd Floor Corridor 2101	All	2	175	1	
3rd Floor Rm 3102	All	2	145	1	
3rd Floor Rm 3103	Floor and lower walls	1	60	1	
3rd Floor Rm 3103	Upper walls and ceiling	2	40	1	
3rd Floor Rm 3104	Floor and lower walls	1	75	1	
3rd Floor Rm 3104	Upper walls and ceiling	2	50	1	
3rd Floor Rm 3106J	All	1	25	1	
3rd Floor Corridor 3101	All	2	160	1	
3rd Floor Rms 3108/3109	All	2	170	1	
3rd Floor Rm 3110	Floor	1	90	1	
3rd Floor Rm 3110	South wall (lower)	1	40	1	
3rd Floor Rm 3110	West wall (lower)	1	30	1	
3rd Floor Rm 3110	All other wall sections and ceiling	2	210	1	
3rd Floor Rm 3110	Pool wall (remaining)	1	150	2	
4th floor cooling tower	All	3	350	1	
Stair No. 2	All	3	180	1	
Stair No. 1	All	3	230	1	
Reactor stack Plenum	All	1	100	1	
Remaining ventilation systems	All surfaces	1	TBD	TBD	
Inside drains and piping	Interior surfaces	1	TBD	TBD	
Outside drains and piping	Interior surfaces	2	TBD	TBD	
Building exterior	Walls and Roof	3	TBD	TBD	Doors, vents, stacks
Soil beneath reactor basement	N/A	1	100	1	About 120 m^3 soil volume
Outside areas	Soil and concrete	1	150	2	Storage pad areas

4.7 Demonstrating Compliance with Guidelines

MARSSIM (NRC,2000b) recommends the use of non-parametric statistical tests for demonstrating that radiological conditions satisfy the established project guideline levels. One of the recommended tests is the Wilcoxon Rank Sum (WRS) test. This WRS test may be used when a specific radionuclide of concern is present in background at a concentration greater than 10 percent of the guideline level and when the measurement is not radionuclide specific, e.g., for direct measurements of total surface activity. The other recommended test is the Sign test, which is used when the radionuclide of concern is not present in background at a significant fraction (i.e., <10 percent) of the guideline level. The Sign test is also used when evaluating data based on the Unity Rule and may be used for surface activity data representing multiple surface media. Both of these tests are applicable to FNR facility final status survey. The selection of a specific test method will be designated at the time of specific final status survey design. MARSSIM (NRC, 2000c) Section 8 and NUREG-1505, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys* (NRC, 1997a), contain details on data assessment, interpretation and application of these statistical tests. Also refer to Section 4.14.4 of this plan.

The Null Hypothesis (H_o) for each survey unit is that residual activity exceeds the guideline levels. Rejection of the Null Hypothesis by the statistical test therefore concludes that the residual activity does not exceed guidelines and the survey unit satisfies requirements for unrestricted release.

4.8 Background Reference Areas and Materials

In addition to the instrumentation background response, many construction materials and environmental media (e.g., soil, sediment) contain naturally occurring levels of radioactive materials, which contribute to a survey measurement. Background contributions must therefore be determined, if 1) the residual contamination includes a radionuclide that occurs in background, or 2) measurements are not radionuclide-specific. Multiple reference areas and materials are anticipated to be required for the final status survey. For applications involving the WRS test, reference areas must be of the same material as the survey unit being evaluated, but without a history of potential contamination by licensed operations; the number of reference data points must be the same (+/- 20 percent) as the number of data points required from the survey unit. A set of reference measurements must be obtained for each instrument being used for survey unit evaluation. For applications involving the Sign test, sufficient background determinations should be made for each media or surface material and with each instrument to provide an average background level that is accurate to within +/- 20 percent; this usually requires 8 to 10 measurements, which are then evaluated using the procedure described in draft NUREG/CR-5849, *Manual for Conducting Radiological Surveys in Support of License Termination* (NRC, 1992a) and additional data points obtained, as necessary. Reference area and background requirements will be identified at the time of individual survey unit final status survey design.

4.9 Survey Reference Systems

A grid system will be established on surfaces to provide a means for referencing measurement and sampling locations. On Class 1 and 2 structure surfaces, a 1-m interval grid will be established; a 5-m interval grid will be established on Class 3 structure surfaces; and a 10-m

interval grid will be established for land area surfaces. Grid systems typically originate at the southwest corner of the survey unit, but specific survey unit characteristics may necessitate alternate grid origins. Grids are assigned alphanumeric indicators to enable survey location identification. Structure grids are referenced to building features; open land grids are referenced to the state or federal planar grid system. Maps and plot plans of survey areas will include the grid system identifications. Systems and surfaces of less than 20 m^2 will not be gridded, but survey locations will be referenced to prominent facility features.

4.10 Determining Data Requirements

Data needs for statistical tests will be determined as follows:

1. Calculate the relative shift (Δ/σ)

   Δ/σ = DCGL - LBGR

   The DCGL is the gross or nuclide specific guideline
   The LBGR (Lower Bound of the Gray Region) is initially selected as half of the DCGL as recommended by MARSSIM (NRC, 2000b).
   σ should be determined empirically from actual survey data; however, for planning purposes, a value of 25 percent of the DCGL will be used.
   The resulting relative shift is 2, which is within the range of 1 to 3, recommended by MARSSIM (NRC, 2000b).

2. Determine decision errors

   The DQOs for this project establish decision errors of 0.05 for both Type I and Type II errors.

3. Determine the number of data points required

   The number of data points required for statistical testing is obtained from MARSSIM (NRC, 2000b) Tables 5.3 (WRS test) and 5.5 (Sign test). For a relative shift of 2 and decision errors of 0.05, the number of data points for the WRS test is 13 and the number for the Sign test is 15. These numbers of data points include an additional 20 percent to allow for potential sample loss and quality control.

The number of data points will be determined in this manner for each survey unit undergoing final status survey and documented in the final status survey design, applicable to that survey unit.

4.11 Determining Data Point Locations

MARSSIM (NRC, 2000b) recommends a triangular measurement or sampling pattern to increase the probability of identifying small areas of residual activity. This type of triangular pattern will be used for this final status survey, except where dimensions and/or other factors related to a specific survey unit require use of an alternate pattern. The spacing (L) between data points on a triangular pattern is determined by:

$$L = [(\text{Survey Unit Area})/(0.866 \times \text{number of data points})]^{1/2}$$

To simplify the designation of data points while assuring a sufficient number of data points are obtained for statistical purposes, the value of L is rounded to the nearest whole meter. If the

systematic pattern does not provide sufficient data points to satisfy the number determined in Section 4.9, additional data points will be identified, using a random-number technique.

4.12 Integrated Survey Strategy

Data collected for final status survey of structure surfaces will consist of scans to identify locations of residual contamination, direct measurements of beta surface activity, and measurements of removable beta surface activity. Final status survey of open land (soil) areas will consist of scans to identify locations of residual contamination and samples of soil, analyzed for potential contaminants. Additional measurements and samples will be obtained, as necessary, to supplement the information from these typical survey activities. Survey techniques are described in more detail in this section.

4.12.1 Beta Surface Scans

Beta scanning of structure surfaces will be performed to identify locations of residual surface activity. Gas-flow proportional detectors will be used for beta scans. Floor monitors with 580 cm^2 detectors will be used for floor and other larger accessible horizontal surfaces; hand-held 125 cm^2 detectors will be used for surfaces not assessable by the floor monitor. Scanning will be performed with the detector within 0.5 cm of the surface (if surface conditions prevent this distance, the detection sensitivity for an alternate distance will be determined and the scanning technique adjusted accordingly). Scanning speed will be no greater than 1 detector width per second. Audible signals will be monitored and locations of elevated direct levels identified for further investigation.

Minimum scan coverage will be 100 percent for Class 1 surfaces, 25 percent for Class 2 surfaces, and 10 percent for Class 3 surfaces. Coverage for Class 2 and Class 3 surfaces will be biased towards areas considered by professional judgment to have highest potential for contamination.

4.12.2 Gamma Surface Scans

Gamma scanning surfaces will be performed on structure and land surfaces to identify locations of residual surface activity. NaI gamma scintillation detectors (2 inch x 2 inch) will be used for these scans. Scanning will be performed by moving the detector in a serpentine pattern, while advancing at a rate of approximately 0.5 m per second. The distance between the detector and the surface will be maintained within 5 cm of the surface. Audible signals will be monitored and locations of elevated direct levels identified for further investigation.

Minimum scan coverage will be 100 percent for Class 1 surfaces, 25 percent for Class 2 surfaces, and 10 percent for Class 3 surfaces. Coverage for Class 2 and Class 3 surfaces will be biased towards areas considered by professional judgment to have highest potential for contamination.

4.12.3 Surface Activity Measurements

Direct measurement of beta surface activity will be performed at designated locations using a 125-cm^2-gas flow detector. Measurements will be conducted by integrating the count over a 1-minute period. Where adverse surface conditions may result in underestimating activity by direct measurements, surface samples will be obtained for laboratory analyses. Need for such sampling will be identified in final status survey design for specific survey units.

4.12.4 Removable Activity Measurements

A smear for removable activity will be performed at each direct surface activity measurement location. A 100 cm^2 surface area will be wiped with a 2 inch diameter cloth or paper filter, using moderate pressure. Dampened smears will be used for removable tritium activity.

4.12.5 Soil Sampling

Samples of surface (upper 15 cm) soil will be obtained from selected locations using a hand trowel or bucket auger. Approximately 500 to 1000 g of soil will be collected at each sampling location.

4.13 Ground Water Survey Strategy

Data collected from ground water will consist of measurements of gross beta activity, gamma activity and tritium activity taken from the monitoring well immediately south of the Phoenix Memorial Laboratory already taken and a final sample taken after the pool is drained. The results of these ground water samples will be provided in the Final Status Survey Report, see Section 4.15.

4.14 Data Evaluation and Interpretation

4.14.1 Sample Analysis

Smears for removable activity will be analyzed by the onsite laboratory for gross alpha and gross beta activity. Analyses of samples of soil and other volumetric media may include gamma spectrometry and/or wet chemistry analyses, depending on radionuclides anticipated. Individual final status survey designs will describe analyses to be performed.

4.14.2 Data Conversion

Measurement data will be converted to units of dpm/100 cm^2 or pCi/g for comparison with guidelines and/or for statistical testing. Where appropriate for Sign tests, data will be adjusted for material and instrument background contributions; data for WRS tests will not be corrected for background, but, instead, will be compared with the data from a reference area.

4.14.3 Data Assessment

Data will be reviewed to assure that the type, quantity, and quality are consistent with the survey plan and design assumptions. Data standard deviations will be compared with the assumptions made in establishing the number of data points. Individual and average data values will be compared with guideline values and proper survey area classifications will be confirmed. Individual measurement data in excess of the guideline level for Class 2 areas and in excess of 25 percent of the guideline for Class 3 areas will prompt investigation. Patterns, anomalies, and deviations from design assumption and plan requirements will be identified. Need for investigation, reclassification, remediation, and/or resurvey will be determined; a resolution will be initiated and the data conversion and assessment process repeated for new data sets.

4.14.4 Determining Compliance with Guidelines

4.14.4.1 WRS Test

For a structure surface survey unit to be evaluated using the WRS test, individual survey unit net total activity measurements and the average of the total net activity measurements will be calculated using the average reference area level; also, the difference between the highest survey unit and lowest reference area measurements will be calculated.

If the difference between the highest survey unit and lowest reference area measurements is less than the guideline level, the survey unit satisfies the criterion and no further evaluation will be necessary.

If the average net surface activity value is greater than the guideline, the survey unit does not satisfy the criterion, and further investigation, remediation, and/or resurvey is required.

If the average net surface activity value is less than the guideline value, but the difference between any survey unit and reference area activity measurement is greater than the guideline, data evaluation by the WRS test proceeds, as follows:

- List each of the survey unit measurements and reference area measurements; do not correct these data for background.
- Add the guideline value to each reference area measurement (for surface activity add the calculated instrument response equivalent of the guideline to the reference area measurements); these are known as adjusted reference area measurements.
- Rank all (survey unit and reference area) measurements in order of increasing size from 1 to N, where N is the total number of pooled measurements.
- If several measurements have the same value, assign them the average ranking of the group of tied measurements.
- If there are "less-than" values, they are all assigned the average of the ranks from 1 to t, where t is the number of "less-than" values.
- Sum the ranks of the adjusted reference area measurements; this value is the test statistic, WR.
- Compare the value of WR to the critical value in MARSSIM (NRC, 2000b) Table I.4 for the appropriate sample size and decision level.

If WR is greater than the critical value, the null hypothesis is rejected, and the survey unit meets the established criteria. If WR is smaller than the critical value, the null hypothesis is accepted, and the survey unit does not meet the established criteria; investigation, remediation, reclassification, and/or resurvey should be performed as appropriate.

4.14.4.2 Sign Test

For an open land or structure surface survey unit to be evaluated using the Sign test, individual activity values and the average activity value will be calculated.

If all values for a survey unit are less than the guideline level, that survey unit satisfies the criterion and no further evaluation is necessary.

If the average activity value is greater than the guideline, the survey unit does not satisfy the criterion, and further investigation, remediation, and/or resurvey is required.

If the average activity value is less than the guideline level, but some individual values are greater than less than the guideline, data evaluation by the Sign test proceeds, as follows:

- List each of the survey unit measurements.
- Subtract each measurement from the guideline level.
- Discard all differences which are “0”; determine a revised sample size.
- Count the number of positive differences; this value is the test statistic, S+.
- Compare the value of S+ to the critical value in MARSSIM (NRC, 2000b) Table I.3 for the appropriate sample size and decision level.

If S+ is greater than the critical value, the null hypothesis is rejected, and the survey unit meets the established criteria. If S+ is smaller than the critical value, the null hypothesis is accepted, and the survey unit does not meet the established criteria; investigation, remediation, reclassification, and/or resurvey should be performed, as appropriate.

4.14.4.3 Unity Rule Sign Test

For an open land or structure surface survey unit to be evaluated using the Unity Rule Sign test, individual activity values and the ratios of the activity values to their respective guideline values will be calculated. For each data location add the ratios together to determine the Sum of Ratios.

If all Sum of Ratios values for the survey unit are less than 1, that survey unit satisfies the criterion and no further evaluation is necessary.

If the average Sum of Ratios value is greater than 1, the survey unit does not satisfy the criterion, and further investigation, remediation, and/or resurvey is required.

If the average Sum of Ratios value is less than 1, but some individual values are greater than 1, data evaluation by the Sign test proceeds, as follows:

- List each of the survey unit Sum of Ratios value.
- Subtract each value from 1.
- Discard all differences which are “0”; determine a revised sample size.
- Count the number of positive differences; this value is the test statistic, S+.
- Compare the value of S+ to the critical value in MARSSIM (NRC, 2000b) Table I.3 for the appropriate sample size and decision level.

If S+ is greater than the critical value, the null hypothesis is rejected, and the survey unit meets the established criteria. If S+ is smaller than the critical value, the null hypothesis is accepted, and the survey unit does not meet the established criteria; investigation, remediation, reclassification, and/or resurvey should be performed, as appropriate.

4.15 Final Status Survey Report

A report describing the survey procedures and findings will be prepared for submission to the NRC in support of license termination. The survey report will provide a complete record of the facility's radiological status and a comparison to the site release criteria. The survey report will provide a summary of any ALARA analysis, survey data results, and overall conclusions, which demonstrate that the FNR Facility meet the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical test results will be included in the report. The survey report will contain additional detail to enable an independent or third party re-creation and evaluation of the survey results and a determination as to whether the site release criteria have been met.

The following outline illustrates a general format that may be used for the final status survey report and may be adjusted to provide a clearer presentation of the information. The level of detail will be sufficient to clearly describe the final status survey program and certify the results.

Information to be submitted (NRC, 2003, Vol1, Appendix D, XIVe):

- A summary of the results of the final status survey.
- A discussion of any changes that were made in the final status survey from what was proposed in the LTP or other prior submittals.
- A description of the method by which the number of samples were determined for each survey unit (NRC, 2000b, Section 5.5.2).
- A summary of the values used to determine the numbers of samples and a justification for these values (NRC, 2000b, Section 5.5.2).
- The results for each survey unit including:
    1. Number of samples taken for the survey unit.
    2. A map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units, and random locations shown for Class 3 survey units and reference areas.
    3. Measured sample concentrations.
    4. Statistical evaluation of the measured concentrations (NRC, 2000b, Section 8.3, 8.4 and 8.5).
    5. Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation.
    6. Discussion of anomalous data including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the DCGLw.
    7. A statement that a given survey unit satisfied the DCGLw and the elevated measurement comparison if any sample points exceeded the DCGLw.
- A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity.

- A description of the investigation conducted when a survey unit fails to ascertain the reason for the failure and a discussion of the impact that the failure has on the conclusion that the facility was ready for final radiological surveys.
- A description of the impact of a survey unit failure has on other survey unit information and the reason for the failure.

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5.0 License and Technical Specifications

5.1 License

This decommissioning plan is intended to become part of the Safety Analysis Report for the facility in accordance with the regulations. A license condition is requested consistent with the methodology used by the NRC for the termination of the license for the UVa license (NRC, 2002):

> The license is amended to approve the decommissioning plan described in the licensee's application date <insert date>, as supplemented on <insert date>, and authorizes inclusion of the decommissioning plan as a supplement to the Safety Analysis Report pursuant to 10 CFR 50.82(b)(5).

The outline for the final status survey plan provided in Section 4.1 is intended to provide information to the NRC in determining the adequately of the licensee's understanding of the final status plan as it pertains to the adequacy of the goal of remediation in a manner which will meet the radiological criteria for license termination. Additional details on the radiological status of the facility, only available following activities conducted under an approved decommissioning plan, are necessary to fully describe the final status survey plan. The completed final status survey will be formally submitted to the NRC for approval at a later date and will be accompanied by reports of any additional characterization surveys performed. This is consistent with the methodology used by the NRC for the termination of the license for the UVa license (NRC, 2002).

The following license condition is requested to license R-28 for the Ford Nuclear Rector:

> The licensee shall submit reports of any characterization surveys performed that are not part of a license amendment application and shall submit the completed final status survey plan for review prior to performing the final status survey.

Other license conditions are presented in Section 9.0.

5.2 Technical Specifications

As outlined in Section 2.4, the licensee's organization for decommissioning is changing substantially. To support these changes the following revisions to the Technical Specifications are requested with the approval of this decommissioning plan.

NOTE: The paragraph numbers and figure number listed below are taken from the Technical Specification for the Ford Nuclear Reactor and are not intended to follow the number system utilized in the Decommissioning Plan.

6.0 **ADMINISTRATIVE CONTROLS**

6.1 **Organization**

1. The organizational structure of the University of Michigan relating to the Ford Nuclear Reactor (FNR) shall be as shown in Figure 6-1

2. The Reactor Manager shall be responsible for the safe decommissioning of the Ford Nuclear Reactor. He shall be responsible for assuring that all activities are conducted in a safe manner within the limits prescribed by the facility license, including the technical specifications and facility procedures. During periods of his absence, the responsibilities of the Reactor Manger may be delegated to an individual who satisfies the qualification requirements for the Reactor Manager.
3. In all matters pertaining to the decommissioning of the Ford Nuclear Reactor and these technical specifications, the Reactor Manager shall report to and be directly responsible to the Director of Occupational Safety and Environmental Health.
4. A Radiation Safety Officer or health physicist, who is organizationally independent of the Ford Nuclear Reactor staff and contractors performing decommissioning activities, shall be responsible for radiation and industrial safety at the facility. During periods of his absence, the responsibilities of the Radiation Safety Officer may be delegated to an individual who satisfies the qualification requirements for the Radiation Safety Officer.
5. Qualifications:
    1. At the time of appointment to the position, the Reactor Manager shall have a minimum of six years of nuclear experience. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job related may be substituted for a degree on a case-by-case basis. The degree may fulfill four years of the six years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate facility specific training based upon a comparison of the individual's background and abilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required.

    2. At the time of appointment, the Radiation Safety Officer shall have a minimum of six years of radiation safety experience. The individual shall have a recognized baccalaureate or higher degree in health physics, nuclear engineering, or scientific field. Education or experience that is job related may be substituted for a degree on a case-by-case basis. The degree may fulfill four years of the six years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate facility-specific training based upon a comparison of the individual's background and abilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required.

6.2 Review

1 A Decommissioning Review Committee (DRC) shall review decommissioning activities and advise the Director of Occupational Safety and Environmental Health in matters relating to the health and safety of the UM community, the public, and the safety of decommissioning activities.

2 The Decommissioning Review Committee shall be composed of a minimum of three members and an unspecified number of alternates of which only a minority shall be from the line organization shown in Figure 6-1. The members and alternates shall be appointed by the Vice President for Research. The review committee chair shall be appointed from the UM tenured faculty with a degree in engineering or a scientific field. The review committee chair shall receive, at the time of appointment, briefings sufficient to provide an understanding of the decommissioning project. The

remaining members of the review committee and alternates shall collectively represent a broad spectrum of expertise appropriate for the decommissioning of FNR and may be either from within or outside the UM. Alternates may attend and vote on matters, regardless of the absence of regular members.

3 The review committee shall meet at least semiannually through the completion of the final status survey. After the completion of the final status survey the review committee shall meet as necessary to review or approve such matters as desired by the committee chair, the Director, Reactor Manager or the Radiation Safety Officer.

4 A quorum shall consist of not less than one-half the regular review committee membership, not including alternates, where the FNR Decommissioning Project staff does not constitute a majority and a representative of UM management at the Associate or Assistant Vice President level or higher.

5 Approval of items by the review committee must be by a majority of the full review committee membership. Approval of items by the review committee may be cast at meetings or via individual polling of the regular review committee members.

6 The review committee chair may appoint subcommittees to facilitate targeted reviews or audits. The subcommittee chair shall be a regular committee member or alternate and shall not be a member of the FNR Project Staff. The subcommittee shall forward items to the review committee chairman with recommendations. The full review committee shall approve all products of the subcommittee.

7 The minutes of the review committee shall be distributed to the Director, Reactor Manager, Radiation Safety Officer, Project Manager, Health Physics Supervisor, the regular members of the review committee, and such others as the chairman may designate.

8 The Decommissioning Review Committee shall approve:
   1. Proposed changes in the license or technical specifications.
   2. Proposed changes to the facility that can be implemented without the prior approval of the NRC as authorized by the license conditions implementing 10 CFR 50.59.
   3. Proposed changes in the Decommissioning Plan that can be implemented without the prior approval of the NRC as described in the *Decommissioning Plan,* Section 9.0, *Changes to the Decommissioning Plan* and authorized by license condition.
   4. New procedures and changes to the procedures involving licensed activities and required by Section 6.5 of these specifications.

9 The Decommissioning Review Committee, as a review function, shall review:
   1. Violations of technical specifications and reportable occurrences made pursuant to the requirements of the technical specifications.
   2. Audit reports issued by a member or subcommittee as required by Section 6.3 of these specifications.
   3. Plans for the following decommissioning activities prior to their implementation:
      1 Any activity which could compromise the structure and integrity of the reactor pool or the primary coolant system while pool water is relied upon for shielding of irradiated reactor components;
      2 The dismantlement of the irradiated reactor components in preparation for disposal;
      3 The movement of any heavy objects, greater than 5 tons in weight;
      4 Any activity which could compromise the structural integrity of the post and beam structure which supports the reactor building;

5 Any activity that will result in the direct release of radioactivity from the facility to the sanitary sewer or a navigable waterway;
6 The draining of the reactor pool;
7 The decontamination or dismantlement of the reactor pool structure;
8 Any activity for which it is estimated that the cumulative radiation exposure for the activity will exceed 1 person-rem, or an individual radiation exposure to either an occupationally exposed person or a member of the public will exceed 20% of any applicable exposure limits of 10 CFR 20; and
9 Any activity, known or anticipated by the review committee, which the review committee requests to review, subject to the approval of the Director.

6.3 Audit

1 The Decommissioning Review Committee as an audit function, shall ensure that the following are independently monitored or audited:
    1 Decommissioning operations to ensure they are being performed safely and in accordance with all applicable licenses and registrations held by the University and in compliance with applicable federal and state regulatory requirements (Radiological Protection Plan, Environmental Safety and Health Plan, etc.).
    2 The quality assurance to verify that performance criteria are met as well as to determine the effectiveness of the program in satisfying the quality assurance requirements.
2 Each monitoring or audit report shall describe each reported adverse finding and shall be distributed to the Director, Reactor Manager, all review committee members, and others at the direction of the Director.
3 Monitoring or audits shall be performed annually, at a minimum, and should be scheduled by the Chair of the Decommissioning Review Committee, in a manner to provide coverage and coordination with ongoing activities, based on the status and importance of activities.
4 The lead auditor and the audit team (if utilized) shall be selected by the Chair of the Decommissioning Review Committee, shall not be directly associated with decommissioning activities, shall not be a member of the FNR Decommissioning Project Team, and shall be familiar with quality assurance requirements applicable to the decommissioning of nuclear facilities.

6.4 Action to Be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence, as defined in these technical specifications, the following action shall be taken:

1 The Reactor Manager shall be notified of the occurrence. Corrective action shall be taken to correct the abnormal conditions and to prevent its recurrence. All other ongoing licensed activities shall be ceased until the occurrence has been resolved.
2 A report of such occurrence shall be made to the Decommissioning Review Committee the Director and the Nuclear Regulatory Commission in accordance with Section 6.7. The report shall include an analysis of the causes of the occurrence, the effectiveness of corrective actions taken, and recommended measures to prevent or reduce the probability or consequences of recurrence.

6.5 Procedures

1 Written procedures, including applicable check lists, reviewed and approved by the Decommissioning Review Committee shall be in effect and followed for the following licensed activities:
    1. Normal operation of all systems structures or components described in these technical specifications or which are important to safety.

2. Actions for responding to emergency conditions involving the potential or actual release of radioactivity, including provisions for evacuation, reentry, recovery, and medical support.
3. Actions to be taken to correct specific and foreseen malfunctions of systems, structures or components described in these technical specifications or which are important to safety.
4. Activities performed to satisfy a surveillance requirement contained in these technical specifications.
5. Radiation and radioactive contamination control.
6. Physical security of the facility.
7. Implementation of the quality assurance program for the calibration and response testing of radiation instrumentation utilized for direct measurement in support of characterization, release, final status survey, or other quality assurance activities.

2 Substantive changes to these procedures shall be made only with the approval of the Decommissioning Review Committee. Non-substantive changes to these procedures may be made with the approval of the Reactor Manager. All non-substantive changes made to procedures shall be documented and subsequently reviewed by the Decommissioning Review Committee.

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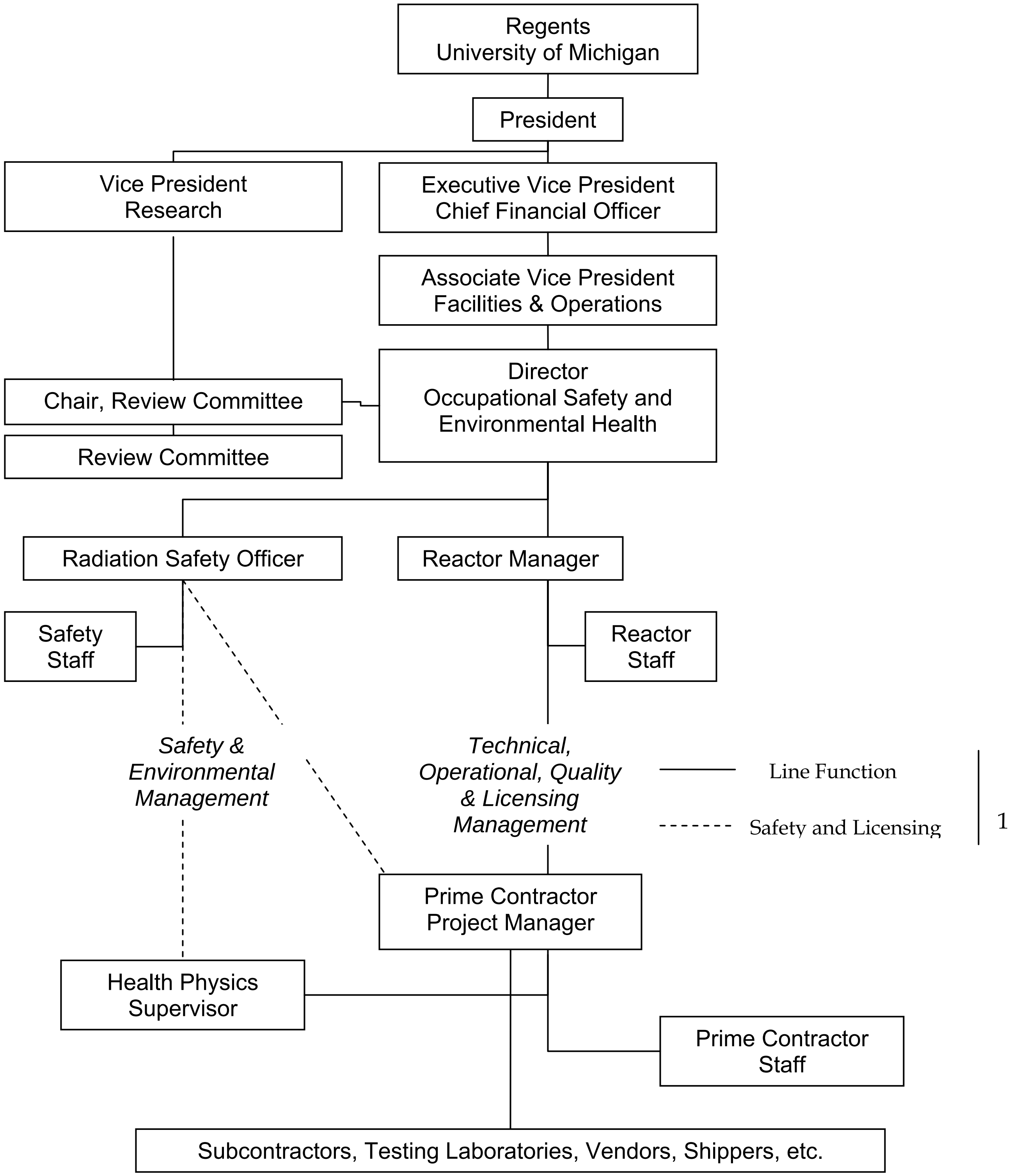
FIGURE 6-1
Organization Chart for the FNR Decommissioning Project



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. As all special nuclear material in the form of reactor fuel covered by the license for the Ford Nuclear Reactor has been removed and the license has been amended for no possession of reactor fuel (Amendment No. 47) a physical security plan is not required.

It is recognized that the regulations in Sub Part I, Storage and Control of Licensed Material" of Part 20 are applicable to the remaining byproduct and special nuclear materials possessed by the FNR. All FNR licensed materials that are in storage will be secured from unauthorized access or removal; and licensed materials that are not in storage will be under the control and constant surveillance of authorized FNR personnel as required by 10 CFR 20.

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7.0 Emergency Plan

FNR has an NRC approved emergency plan that satisfies the requirements of 10 CFR 50 Appendix E, *Emergency Planning and Preparedness for Production and Utilization Facilities*. FNR may make changes without NRC approval only if these changes do not decrease the effectiveness of the plan and continues to meet the requirements of 10 CFR 50 Appendix E [10 CFR 50.54 (q)]. FNR will maintain a record of each change to the emergency plan for a period of three years from the date of the change.

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8.0 Environmental Report

The Environmental Report is provided in Appendix B, in accordance with 10 CFR 51.45.

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9.0 Changes to the Decommissioning Plan

The existing requirements governing the authority of production and utilization facility licensees to make changes to their facilities and procedures, or to conduct tests or experiments, without prior NRC approval are contained in 10 CFR 50.59. Comparable provisions exist in Sec. 72.48 for licensees of facilities for the independent storage of spent nuclear fuel and high-level radioactive waste. Section 50.59(b) of Part 50 states that Section 50.59 "applies to each holder of a license authorizing operation of a production or utilization facility, including the holder of a license authorizing operation of a nuclear power reactor that has submitted the certification of permanent cessation of operations required under Section 50.82(a)(1) or a reactor licensee whose license has been amended to allow possession of nuclear fuel but not operation of the facility." As the applicability of Section 50.59 of Part 50 appears not to apply, the UM requested and received in the possession only license amendment issued by the NRC (NRC, 2004), a new license condition which continues authority to make changes to the facility and procedures, or to conduct tests or experiments without prior NRC approval as contained in 10 CFR 50.59.

By license condition, this Decommissioning Plan will become a supplement to the Safety Analysis Report in accordance with the regulations. As a result, through this license condition, 10 CFR 50.59 with additional conditions provides the necessary authority for the UM to make changes to the Decommissioning Plan as provided by the license conditions described below:

> A license amendment pursuant to 10 CFR 50.90 shall be obtained for changes to this Decommissioning Plan if the change would:
>
> - Require a license amendment pursuant to 10 CFR 50.59,
> - Increase the radioactivity level, relative to that applicable derived concentration guideline level, at which an investigation occurs,
> - Allow the use of a statistical test other than the Sign test or Wilcoxon Rank Sum test for evaluation of the final status survey,
> - Reduce the coverage requirements for scan measurements,
> - Decrease an area classification (i.e., impacted to non-impacted, Class 1 to Class 2, Class 2 to Class 3, or Class 1 to Class 3),
> - Increase the Type I decision error,
> - Result in more than a minimal increase in the environmental consequences not previously evaluated in the final safety analysis report (as updated) or
> - Foreclose the release of the site for possible unrestricted use.
>
> The provisions above do not apply to changes to the Decommissioning Plan when applicable regulations, such as 10 CFR 50.54, 10 CFR 20, etc., establish more specific criteria for accomplishing such changes.

All definitions contained within 10 CFR 50.59 apply, as does the non-power reactor applicability of Regulatory Guide 1.187, *Guidance for Implementation of 10 CFR 50.59 Changes, Tests, and Experiments* (NRC 2000c).

10 CFR 50.59 and the additional conditions above establish the conditions under which changes to this decommissioning plan can be made without prior NRC approval. Thus 10 CFR 50.59 and the additional conditions above provide a threshold for regulatory review – not the final determination of safety.

10 CFR 50.59(a)(4) defines the Final Safety Analysis Report [FSAR] (as updated) as the FSAR submitted in accordance with 10 CFR 50.34, as amended and supplemented and as updated per the requirements of 10 CFR 50.71(e) or 50.71(f), as applicable. The FNR license was reissued in July 1985 based on the Safety Analysis dated November 30, 1984 and subsequent responses to NRC requests for additional information. As a non-power reactor, FNR was not required to maintain its Safety Analysis Report per the requirements of 10 CFR 50.71(e) or 50.71(f) . Thus the FSAR (as updated), or better the Current Licensing Basis, for FNR consists of the Safety Analysis as revised and the safety analyses supplied by the UM to support each license amendment after and including license amendment 34 dated August 4, 1989. The license amendment requesting approval of this Decommissioning Plan will become part of the FSAR (as updated) upon approval by the NRC.

10 CFR 50.59(c)(2) provides eight criterion against which changes to the Decommissioning Plan would be evaluated. The criterion: *Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered* focuses on the fission product barriers and on the critical design information that supports their continued integrity. License amendment 47 (NRC, 2004) removed the possession of reactor fuel. Changes to the decommissioning plan would not need to be evaluated against this criterion.

The proposed license established criterion: *Results in more than a minimal increase in the environmental consequences not previously evaluated in the final safety analysis report (as updated)* is based upon license conditions from previous decommissioning amendments issued by the NRC (NRC, 2002) which required approval of any change which could “Result in significant environmental impacts not previously reviewed.” The proposed license established criterion utilizes the guidance provided by NEI 96-07 (NEI, 2000) for “more than minimal increase in consequences” in the evaluation of changes with environmental consequences (dose).

The screening process described in NEI 96-07 (NEI, 2000) may be initially used to determine if a change should be evaluated against the criteria of 10 CFR 50.59(c)(2). Changes to the decommissioning plan which do not require evaluation against the criteria of 10 CFR 50.59.(c)(2) can be made with the approval of the Reactor Manager and subsequently reviewed by the review committee. Changes to the decommissioning plan which fail the screening process and are evaluated against the full criteria of 10 CFR 50.59 may be made with the approval of the review committee.

A brief description of any changes to the decommissioning plan will be submitted as required by Technical Specifications. The records of changes to the decommissioning plan will be maintained until the termination of the license.

10.0 References

10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," *Code of Federal Regulations*, as amended.

10 CFR Part 20, "Standards for Protection Against Radiation," *Code of Federal Regulations*, as amended.

10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," *Code of Federal Regulations*, as amended.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," *Code of Federal Regulations*, as amended.

10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," *Code of Federal Regulations*, as amended.

10 CFR Part 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulation,* as amended.

10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," *Code of Federal Regulations*, as amended.

29 CFR Part 1910, "Occupational Safety and Health Standards," *Code of Federal Regulations*, as amended.

29 CFR Part 1926, "Safety and Health Regulations for Construction," *Code of Federal Regulations*, as amended.

49 CFR Parts 100-177, "Research and Special Programs Administration, Department of Transportation," *Code of Federal Regulations*, as amended.

American Nuclear Society (ANS), 1976. *American National Standard for Quality Assurance Program Requirements for Research Reactors,* ANS-15.8. American Nuclear Society, LaGrange, IL.

American Nuclear Society (ANS), 1993. *American National Standard for Radiation Protection at Research Reactor Facilities,* American Nuclear Society, LaGrange Park, Illinois.

American Society of Mechanical Engineers (ASME). 2001. *Quality Assurance Requirements for Nuclear Facility Applications,* American Society for Mechanical Engineers, New York, NY.

American Society for Testing and Materials (ASTM). 1989. *Standard Test Methods for Particulate and Dissolved Mater, Solids, or Residue in Water*, D1888-78, American Society for Testing and Materials, West Conshohocken, PA.

American Society of Mechanical Engineering (ASME). 2001. *Quality Assurance Requirements for Nuclear Facility Applications,* NQA-1-2000. The American Society of Mechanical Engineers, New York, NY.

Atomic Energy Commission, 1974. *Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors,* Directorate of Regulatory Standards, U.S. Atomic Energy Commission, Washington, D.C.

Daniel Bullen. 1994. *Pathway Analysis of the Tritium Contamination Plume from the Phoenix Memorial Laboratory*, Consultant Report, Daniel B. Bullen, Associate Professor, Iowa State University, Ames, Iowa.

CH2M HILL. 2003. Historical Site Assessment, Ford Nuclear Reactor, North Campus, University of Michigan, Final Report, CH2M HILL, Inc., Richland, Washington.

Environmental Protection Agency (EPA). 2002. Memorandum from M. Cook to Addressees, Subject: *Distribution of Memorandum of Understanding between EPA and the Nuclear Regulatory Commission* dated 9 Oct 2002. Office of Solid Waste and Emergency Response, U.S. Environmental Protection Agency, Washington, D.C.

General Atomis. 1999. *General Atomics TRIGA® Reactor Facility Decommissioning Plan*, General Atomic, San Diego, California.

Gerhard Erdman. 1976. *Kernchemie in Einzeldarstellugen Volume 6: Neutron Activation Tables*, Weinheim, New York: Verlag Chemie.

Grove Engineering. 1996. *MicroShield User's Manual* (Version 5), Rockville, MD 20852.

GTS Duratek. 2000. *University of Virginia Reactor Decommissioning Plan* (prepared for University of Virginia), Radiological Engineering and Field Services, GTS Duratek, Kingston, TN.

Health Physics Society (HPS). 1978. *American National Standard Radiation Protection Instrumentation Test and Calibration,* ANSI N323-1978. The Institute of Electrical and Electronics Engineers, Inc., New York, NY.

International Organization for Standardization (ISO). 1988. *Evaluation of Surface Contamination – Part 1: Beta Emitters and Alpha Emitters* (First Edition), ISO-7503-1.

W. K. Luckow & R.B. Mesler. 1957. Erection *of a High Density Concrete Nuclear Reactor Shield,* Phoenix Memorial Laboratory, University of Michigan, Ann Arbor, MI.

Michigan Occupational Safety and Health Act, Act 154 of 1974 as amended.

National Aeronautics and Space Administration (NASA). 2001. *Decommissioning Plan for the Plum Brook Reactor Facility* (Rev 2) National Aeronautics and Space Administration, Glenn Research Center, Sandusky, OH.

Nuclear Energy Institute (NEI). 2000. *Guidelines for 10 CFR 50.59 Implementation*¸ Nuclear Energy Institute, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1973a. *Guide for Administrative Practices in Radiation Monitoring,* Regulatory Guide 8.2, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1973b. *Direct-Reading and Indirect-Reading Pocket Dosimeters,* Regulatory Guide 8.4, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1974. *Termination of Operating Licenses for Nuclear Reactors,* Regulatory Guide 1.86, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1976. *Acceptable Programs for Respiratory Protection*, Regulatory Guide 8.15, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1977. *Quality Assurance Program Requirements for Research Reactors,* Regulatory Guide 2.5. Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1981. *IE Circular No, 81-07: Control of Radioactively Contaminated Material*, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1982. *Standard Radiological Effluent Technical Specifications for Boiling Water Reactors,* NUREG-0473, Rev 3, U.S. Nuclear Regulatory 1
Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1983. *Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1985. *IE Information Notice No. 85-92: Surveys of Wastes Before Disposal from Nuclear Reactor Facilities*, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1988. *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,* NUREG-0586. Office Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1991a. *Technical Position on Waste Form, Revision 1,* Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1991b. Letter to R.F. Fleming, Director Michigan Memorial Phoenix Project from L.R. Greger, Chief Reactor Programs Branch dated July 22, 1991.

Nuclear Regulatory Commission (NRC). 1992a. *Manual for Conducting Radiological Surveys in Support of License Termination*, NUREG/CR-5849, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1992b. *Residual Radioactive Contamination for Decommissioning*, NUREG/CR-5512, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1994. *NRC Information Notice 94-07: Solubility Criteria for Liquid Effluent Releases to Sanitary Sewerage Under Revised 10 CFR Part 20.* U.S. Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1995. *NRC Branch Technical Position on Concentration Averaging and Encapsulation*, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1996. *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,* NUREG-1537-PT-1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1997a. *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*, NUREG-1505, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1997b. *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Fields Conditions*, NUREG/CR-1507, Final, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1988. *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,* NUREG-0586, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1998. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination*, Federal Register: November 18, 1998 (Volume 63, Number 222; pages 64132-64134).

Nuclear Regulatory Commission (NRC). 1999. *Acceptable Programs for Respiratory Protection,* Regulatory Guide 8.15, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 2000a. *NMSS Decommissioning Standard Review Plan*, NUREG 1727, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 2000b. *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM), NUREG-1575, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 2000c. *Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,* Regulatory Guide 1.187, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 2000d. *Quality Assurance Program Approval for Radioactive Materials Packages*, Approval No. 0348, Rev 7, Transportation and Storage Safety and Inspection Section, U.S. Nuclear Regulatory Commission, Washington, DC.

Nuclear Regulatory Commission (NRC). 2000e. Ltr to Dr. Mulder, University of Virginia, Nuclear Reactor Facility, Subject: *University of Virginia – Request for Additional Information, RE: Decommissioning Amendment Request (TAC No. MA8186)* dated November 2, 2000.

Nuclear Regulatory Commission (NRC). 2002. *Safety Evaluation Report Related to the Decommissioning of Virginia Research Reactor, University of Virginia*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 2003. *Consolidated NMSS Decommissioning Guidance,* NUREG 1757, U.S. Nuclear Regulatory Commission, Washington, D.C.

Nuclear Regulatory Commission (NRC). 2004. letter to A.H. Francis, Subject: *Issuance of Possession Only License Amendment No. 47 for University of Michigan Ford Nuclear Reactor (TAC No. MC 1369)*

Smith Hinchman & Grylls (SH&G). 1955. *Second & Third Floor Framing Plans, Sheet No. S-2,* Smith Hinchman & Grylls, Inc., Detroit, MI.

University of Michigan (UM). 1960. *Agenda for the Reactor Advisory Committee Meeting May 5, 1960.*

University of Michigan (UM). 1961. Memorandum from H.W. Henry to C.W. Ricker, Subject: *Silver 100m Contamination Incident in PML on April 21, 1967,* dated May 2, 1961.

University of Michigan(UM). 1963. Memorandum from H.W. Henry to W. Kerr, Subject: *Radiation Safety Activities in Phoenix Memorial Laboratory and Phoenix Radiation Facility During June, 1963,* dated July 19, 1963.

University of Michigan (UM). 1966. Memorandum from J.D. Jones to R.D. Martin, Subject: *Health Physics Report Regarding Transfer Operations of D_2O from D_2O Tank to Storage Drum and Inventory of $H^3{}_2O$ in D_2O Tank and F.N.R. Pool,* dated March 17, 1966

University of Michigan (UM). 1967a. *Agenda for the Reactor Advisory Committee Meeting, September 14, 1967.*

University of Michigan (UM). 1967b. Memorandum from H.L. Millis to Airborne Activity File, Subject: *Airborne Activity @ Startup-for Cycle No. 22,* dated September 14, 1967.

University of Michigan (UM). 1967c. *Minutes of the Thirty-Third Meeting of the Reactor Advisory Committee Meeting September 14, 1967.*

University of Michigan (UM). 1967d. Memorandum from N.L. Millis to Airborne Activity File, Subject: *Analysis of Dust from Absolute Filter Removed from the BHF (Beam Hole Floor) on 9-15-1967,* dated October 19, 1967.

University of Michigan (UM). 1969. Memorandum from H.L. Millis to Airborne Activity Incidents File, Subject: *Fission Product Release During Core Unloading May 12, 1969,* dated May 13, 1969.

University of Michigan (UM). 1971. Memorandum from R.L. Beethe to Airborne Activity File, Subject: *Flooding of J-port 13:30, 10/15/71,* undated.

University of Michigan (UM). 1977a. Ltr to the U.S. Nuclear Regulatory Commission from W. Kerr, *Reportable Occurrence No. 1,* dated January 21, 1977.

University of Michigan (UM). 1977b. Ltr to the U.S. Nuclear Regulatory Commission from W. Kerr, *Reportable Occurrence No. 5,* dated December 5, 1977.

University of Michigan (UM). 1985. *Demography, Topology, Geology, and Meteorology.* Ford Nuclear Reactor, Michigan Memorial – Phoenix Project.

University of Michigan (UM). 1987. Ltr to U.S. Nuclear Regulatory Commission from W. Kerr dated April 16, 1987.

University of Michigan (UM). 1989. Ltr to the U.S. Nuclear Regulatory Commission from W. Kerr, *Reportable Occurrence No. 12: Release of Fission Products from the Ford Nuclear Reactor,* dated January 16, 1989.

University of Michigan (UM). 1991. Ltr to the U.S. Nuclear Regulatory Commission from R. Fleming, *Reportable Occurrence No. 14: Fission Product Leaks from Babcock and Wilcox Fuel Elements,* dated November 7, 1991.

University of Michigan (UM). 1993. *Safety Analysis*. Ford Nuclear Reactor, Michigan Memorial–Phoenix Project.

University of Michigan (UM). 1998. Ltr to U.S. Nuclear Regulatory Commission from R. F. Fleming dated June 3, 1998, Subject: *Report of Non-routine Occurrence – Hot Demineralizer Leak Causes 75 gallon Loss of Pool Water to Radioactive Retention Tank System.*

University of Michigan (UM). 1999. *Report of Reactor Operations, January 1, 1998 – December 31, 1998.* Ford Nuclear Reactor, Michigan Memorial–Phoenix Project.

University of Michigan (UM). 2001. Agenda for the Radiation Policy Committee Meeting March 20, 2001

University of Michigan (UM). 2004. *Minutes for the February Meeting, 2004 of the Board of Regents for the University of Michigan*. University of Michigan, Ann Arbor, MI http://www.umich.edu/~regents/meetings/minutes/pdfs/2004/Feb04.pdf

University of Virginia (UVa). 2000. *University of Virginia Reactor Decommissioning Plan*. University of Virginia, Charlottesville, VA, prepared under contract by GTS Durateck, Radiological Engineering and Field Services Division, Kingston, TN. 1

C. Yu, D.J. LePoire, J.-J. Cheng, et. al. 2003. *User's Manual for RESRAD-BUILD Version 3,* Environmental Assessment Division, Argonne National Laboratory, Argonne, IL.

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Appendix A
Characterization of the Ford Nuclear Reactor Facility

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Appendix B
Environmental Report for the Ford Nuclear Reactor Decommissioning

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