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March 31, 2009

Document Control Desk

U.S. Nuclear Regulatory Commission

11555 Rockville Pike

Rockville, MD 20852

Attn. Mr. Ted Carter

Dear Ted,

Please accept the enclosed Decommissioning Plan for the Leslie C. Wilbur Nuclear Reactor Facility at Worcester Polytechnic Institute. Should you have any questions regarding this submission, please feel free to contact me in my office at (508) 831-6919 , or e-mail mjcurley@wpi.edu.

Thank you.

Sincerely,

Michael J. Curley

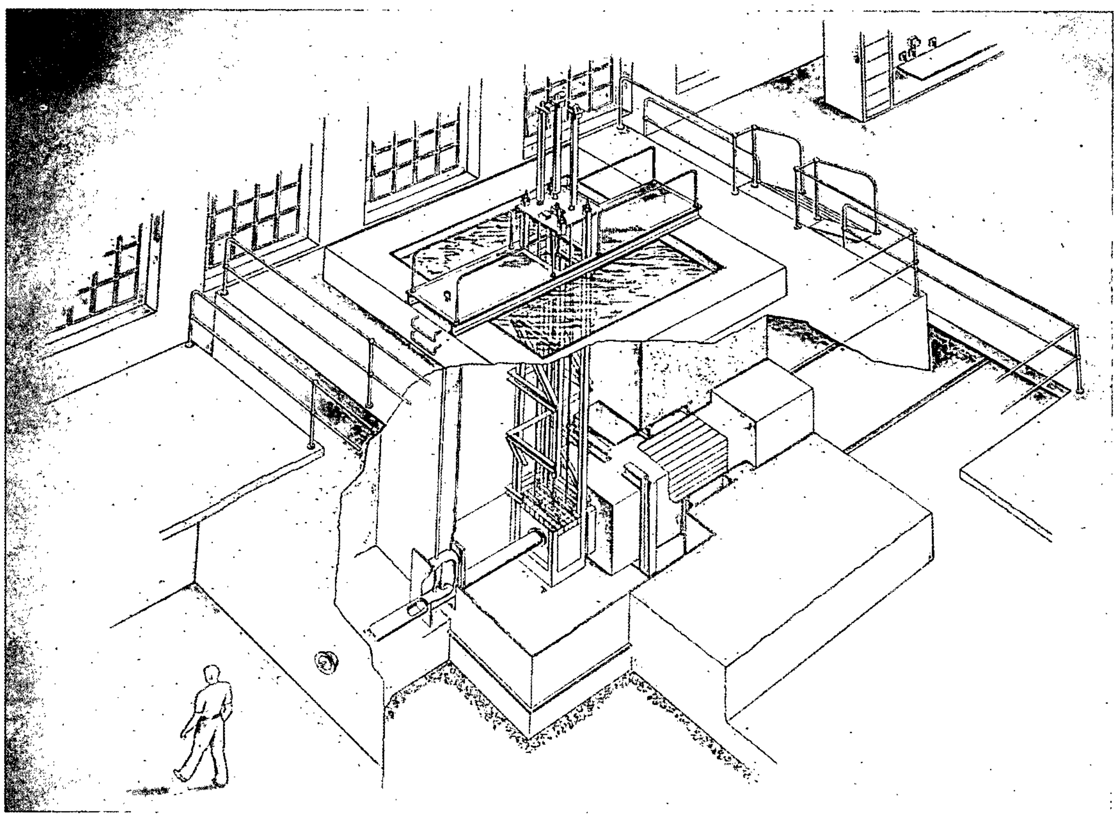
Director, Nuclear Reactor Facility

NM5501
A020
FSME

Document W19-1579-003, Rev. 0

DECOMMISSIONING PLAN
for the
LESLIE C. WILBUR NUCLEAR REACTOR FACILITY
at the
WORCESTER POLYTECHNIC INSTITUTE

Operating License No. R-61
Docket No. 50-134



Prepared for:

WORCESTER POLYTECHNIC INSTITUTE

Prepared by:

TLG Services, Inc.
Bridgewater, Connecticut

March 2009

APPROVALS

Radiation Safety Officer	David S. Adams (signature) David Adams	3-30-09 Date
Reactor Facility Director	(signature) Michael Curley	3/30/09 Date
Radiation, Health & Safety Committee	David Messier (signature) David Messier	3/30/09 Date

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REVISION LOG

Rev. No.	CRA No.	Date	Item Revised	Reason for Revision
0	N/A	03/27/2009	N/A	Original Issue

ACRONYMS and ABBREVIATIONS

ALARA	As low As Is Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
cm	centimeter
COMPASS	COMPASS Computer Code Version 1.0.0 (development sponsored by the NRC)
cpm	counts per minute
D&D	Decontamination and Decommissioning
DAC	Derived Air Concentration (see 10 CFR 20)
DCGL^{w}	Derived Concentration Guideline Levels
DCGL^{emc}	Elevated Measurement Comparison DCGL
DECON	Decontamination Decommissioning Option
Director	WPI Reactor Director
dpm	disintegrations per minute
DQO	Data Quality Objective
EDE	Eye Dose Equivalent (see 10 CFR 20)
ENTOMB	Entombment Decommissioning Option
FSS	Final Status Survey
g	gram, a unit of mass
GM	Geiger-Mueller
GST	General Site Training
HEPA	High Efficiency Particulate Air (Filter)
HP	Health Physics
LCWNRF	Leslie C. Wilbur Nuclear Reactor Facility
MARSSIM	*Multi-Agency Radiation Survey and Site Investigation Manual, NUREG-1575*
MA	Massachusetts
mR	milli-Roentgen, 10^{-3} Roentgen
mrem	millirem, 10^{-3} rem
MSHA	U.S. Mine Safety and Health Administration
mSv	milli-Sievert (unit of dose equivalence, see 10 CFR 20), 10^{-3} Sievert
MW	Megawatt
NIOSH	National Institute for Occupational Safety and Health
NQA	Nuclear Quality Assurance
NRC	U.S. Nuclear Regulatory Commission
OSHA	Federal Occupational Safety and Health Acts
pCi	pico-curie, a unit of radioactivity (2.22 disintegrations per min.) 10^{-12} curie
POL	Possession-Only License
ppm	parts per million
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control

ACRONYMS and ABBREVIATIONS
(Continued)

rem	Roentgen equivalent man (rem) is a unit of absorbed radiation dose: 100 rem = 1 Sv
RHSC	Radiation Health and Safety Committee
RSO	Radiation Safety Officer
RWP	Radiation Work Permit
SDE	Shallow Dose Equivalent (see 10 CFR 20)
Sv	Sievert (unit of dose equivalence, see 10 CFR 20)
TEDE	Total Effective Dose Equivalent (see 10 CFR 20)
WPI	Worcester Polytechnic Institute
WPI-RDP	WPI LCWNRF Reactor Decommissioning Project

1.0 SUMMARY OF PLAN

1.1 INTRODUCTION

1.1.1 Overview

This Decommissioning Plan (DP) is for the Leslie C. Wilbur Nuclear Reactor Facility (herein referred to as "LCWNRF", or "Reactor Facility"). The LCWNRF is situated on the Worcester Polytechnic Institute (WPI) campus in Worcester, Massachusetts, approximately 40 miles west of Boston. The LCWNRF is housed in a portion of the Washburn Shops and Stoddard Laboratories Building, located between West and Boynton Streets. The geographical location of WPI campus is shown on Figure 1.1. Figure 1.2 provides an overall map of the WPI Campus. An exterior photograph of the Washburn Shops-Stoddard Laboratories Building, which houses the reactor facility, is presented in Figure 1.3. Figures 1.4 and 1.5 present photographic views from the two levels of the LCWNRF. Figure 1.6 presents a rendering of the reactor pool, biological shield and experimental facilities.

The LCWNRF provided graduate and undergraduate students with reactor operating experience and experimental practice in the fields of nuclear engineering, metallurgy, chemistry and physics, as well as irradiation services for other teaching, medical and industrial institutions. WPI discontinued routine operation of the reactor on June 30, 2007 (Ref. 1) and submitted an application to the Nuclear Regulatory Commission (NRC) for a possession-only license (Refs. 2 and 3). On August 26, 2008 NRC subsequently granted Amendment No. 11 to Operating License No. R-61, placing the LCWNRF reactor in a Possession-Only License status (Ref. 4). WPI desires to proceed with decommissioning the LCWNRF and the termination of the associated reactor license. Accordingly, WPI has filed the appropriate decommissioning amendment requests with NRC along with this DP.

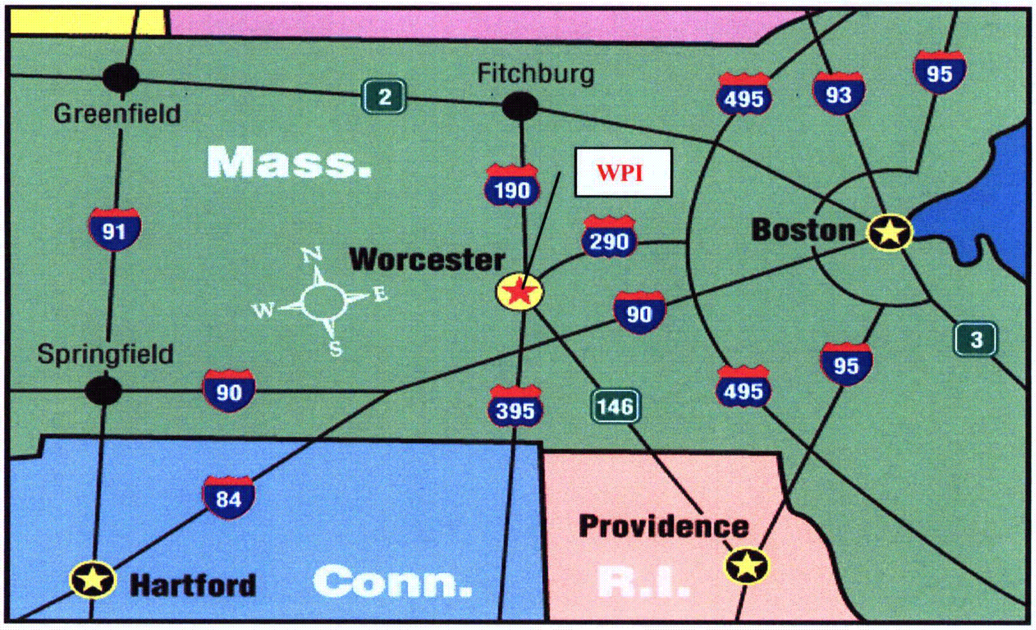
The LCWNRF is contaminated with limited amounts of radioactive material, principally contained in activated metallic core components. Scoping surveys indicate that practices employed by WPI to minimize the spread of contamination were effective and contamination was not found outside of the reactor pool.

The goal of the proposed D&D activities is termination of WPI's NRC License R-61, Document No. 50-134 and release of the LCWNRF for "unrestricted use." The term "unrestricted use" means that there will be no future restrictions on the use of the Washburn Shops and Stoddard Laboratories building other than those imposed by the City of Worcester zoning ordinances. WPI holds a separate Massachusetts Agreement State byproduct material license for possession and use of radioactive materials at other campus locations: that license will not be affected by this decommissioning.

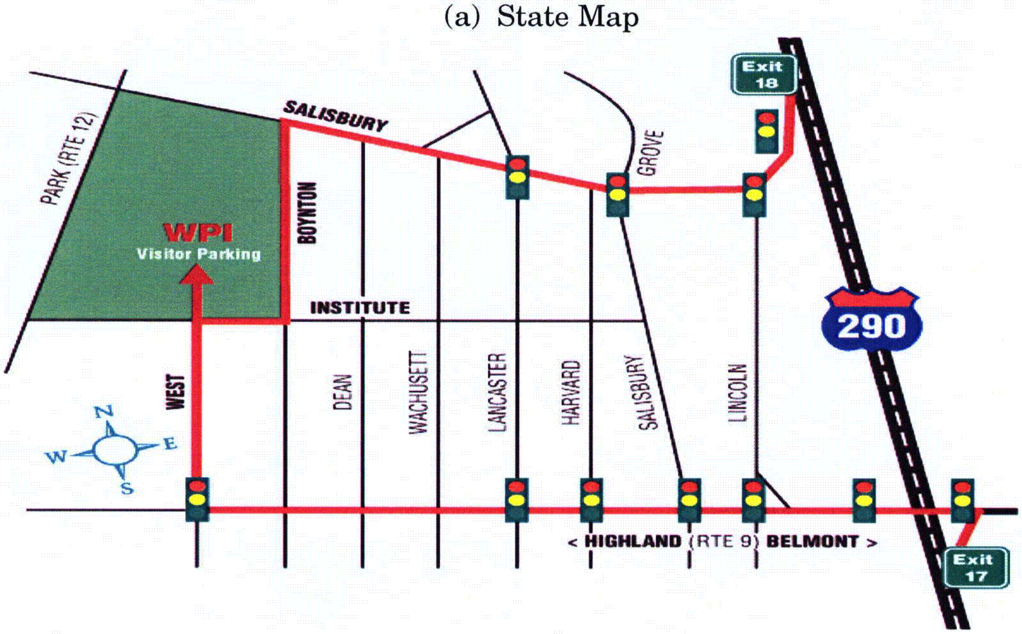
This DP has been prepared using the guidance and format recommended in NUREG-1537, Rev. 0, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors* (Ref. 5) and incorporates additional guidance from NUREG-1757, *NMSS Decommissioning Standard Review Plan* (Ref. 6). A summary profile for the LCWNRF reactor is provided in Tables 1.1 and 1.2.

FIGURE 1.1

STATE AND LOCAL MAPS OF THE AREA SURROUNDING WPI



(a) State Map



(b) Local Map

FIGURE 1.2
WPI CAMPUS MAP

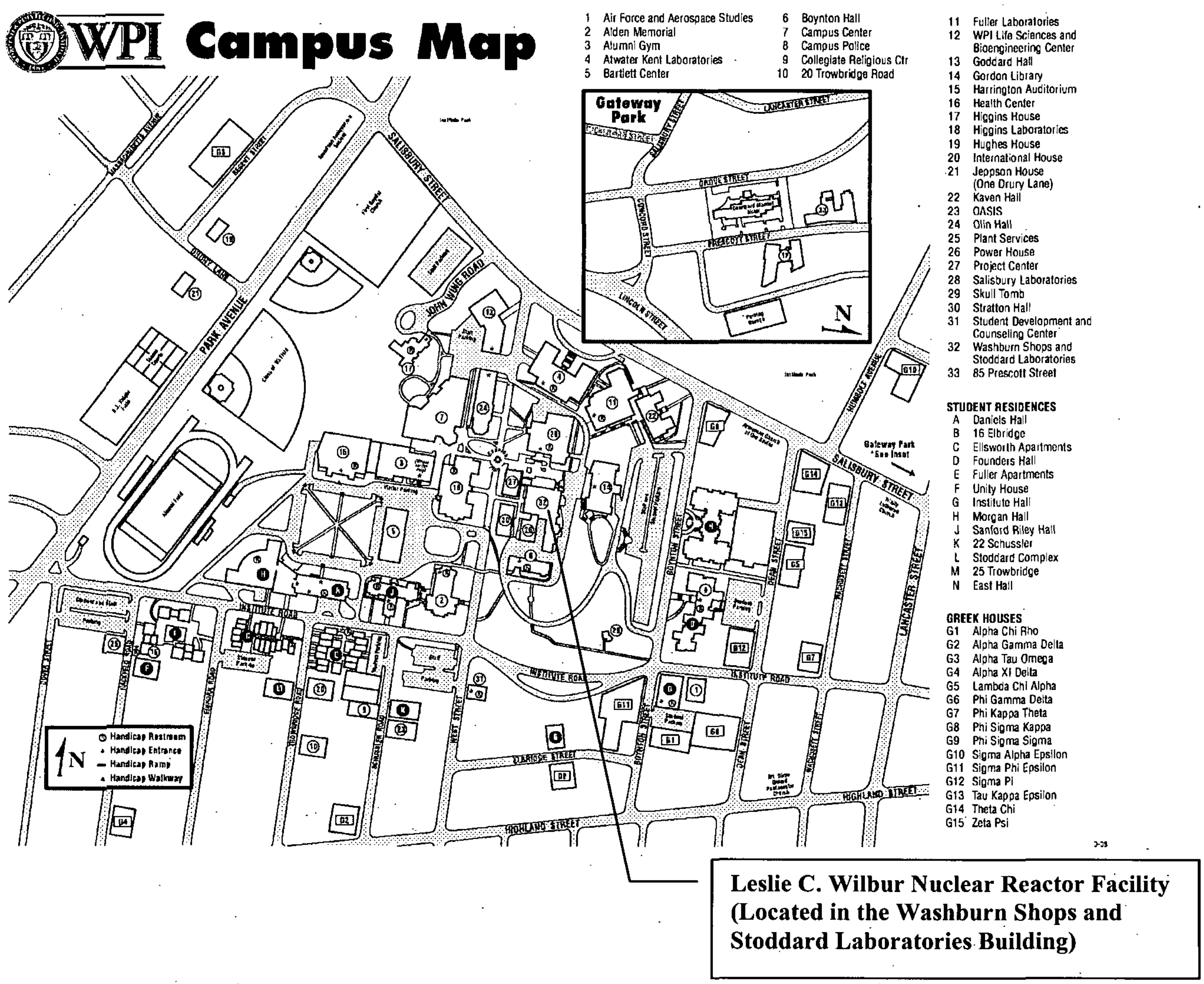


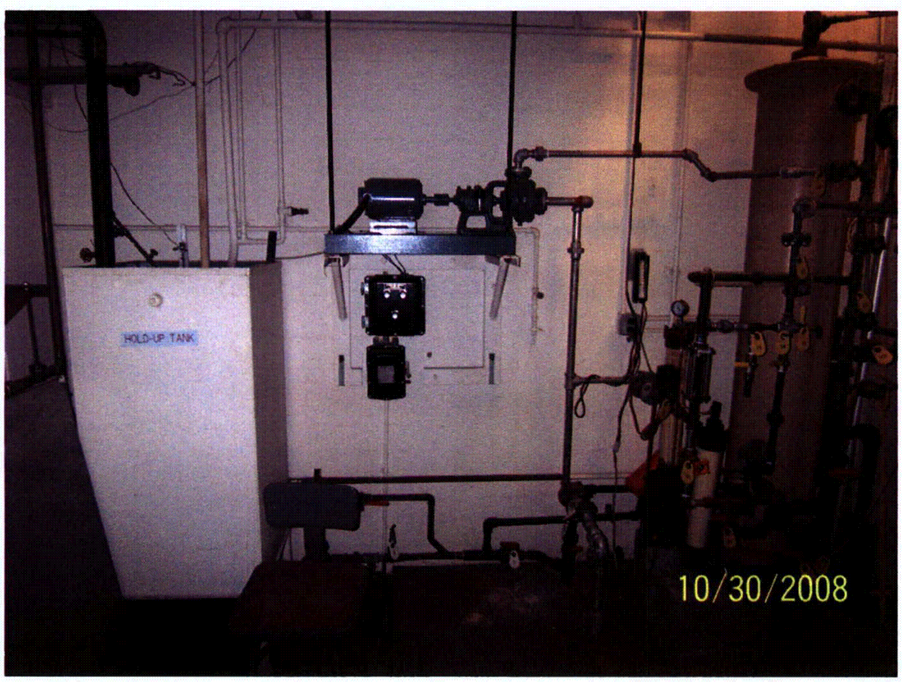
FIGURE 1.3

PHOTOGRAPHIC VIEW OF THE REACTOR FACILITY BUILDING EXTERIOR

FIGURE 1.4

PHOTOGRAPHIC VIEWS OF THE REACTOR FACILITY LOWER LEVEL

(a) Thermal Column Shield Door



(b) Pool Water Treatment System & Effluent Hold-up Tank

FIGURE 1.4 (continued)

PHOTOGRAPHIC VIEWS OF THE REACTOR FACILITY LOWER LEVEL

(c) Beam Port Access Plug

FIGURE 1.5

PHOTOGRAPHIC VIEWS OF THE REACTOR FACILITY OPERATING LEVEL

(a) Operating Floor – East View

(b) Operating Floor – West View

FIGURE 1.6

RENDERING OF THE REACTOR POOL, BIOLOGICAL SHIELD AND EXPERIMENTAL FACILITIES

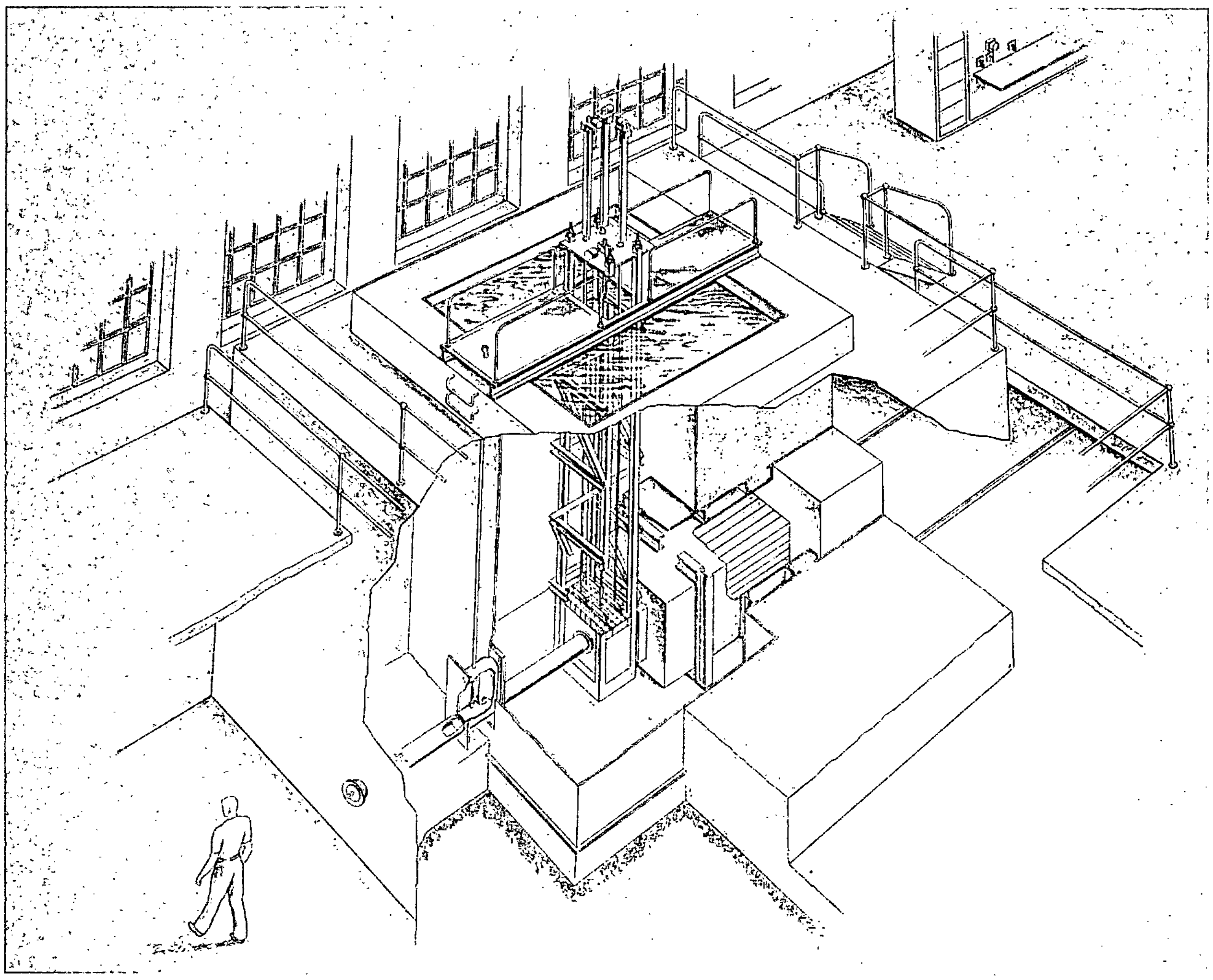


TABLE 1.1

PROFILE OF THE WPI LCWNRF REACTOR

REACTOR MATERIALS	
Fuel	Uranium aluminum alloy, 19.75% enriched
Moderator	High purity light water
Reflector	High purity light water and graphite
Coolant	High purity light water
Control	Boral and stainless steel
Structural material	Aluminum
Shield	Water filled aluminum lined concrete pool
STRUCTURAL DIMENSIONS	
Pool	8 feet by 8 feet by 15 feet deep
Core (active portion)	15 inches by 15 by 24 inches high
Grid box	9x6 array of 3-inch modules
Beam port	one, 6 inches diameter
Thermal column	One, 40 inches by 40 inches in cross-section
STRATEGIC MATERIALS	
Burn-up	Approximately 1% U-235
THERMAL CHARACTERISTICS (Calculated)	
Operating power	1 kW (maximum 1959-1967) and 10 kW (maximum 1968-2007)
Temperature, water	130°F (maximum)
Maximum power peaking factor	2.17
Maximum hot channel factor	1.51
Maximum heat flux	400 Btu/hr-ft^2
Specific power (clean, cold)	2.5 Watt/g U-235
Maximum gamma heat in core	11 Watt/liter
NUCLEAR CHARACTERISTICS (Calculated)	
Average thermal flux	8.3x10^{10} n/cm^2 sec
Average fast flux	11x10^{10} n/cm^2 sec
Maximum operating excess reactivity	0.5% delta k/k$_{eff}$
Critical mass	4.0 kg U-235
Temperature coefficient	-1.63x10^{-4} delta k/k$_{eff}$ per degree C
Void coefficient	-2.4x10^{-3} delta k/k$_{eff}$ per 1% void
Prompt neutron lifetime	61.2 microseconds
Delayed neutron fraction	0.0077
CONTROL SAFETY ELEMENTS	
Number	3 vertical blades
Dimensions	10.5 inches wide by 40.5 inches long by 0.375 inches thick
Material	Boral
Reactivity Control	3.5% delta k/k$_{eff}$ each, minimum
Total worth, 3 blades	12% delta k/k$_{eff}$
Maximum withdrawal rate	7.5 inches/minute

(Note: Table 1.1 is continued on next page)

TABLE 1.1 (continued)
PROFILE OF THE WPI LCWNRF REACTOR

CONTROL REGULATING ELEMENTS	
Number	1 vertical blade
Dimensions	10.65 inches wide by 40.5 inches long by 0.125 inches thick
Material	Stainless steel
Reactivity Control	0.7% delta k/k_{eff}
Maximum withdrawal rate	3.8 inches/minute
STANDARD FUEL	
Number of elements	21 + 1/3 for minimum critical loading
Fuel alloy	Uranium-Aluminum
Clad material	Aluminum
Fuel enrichment	U-235 19.75% enriched
Number of plates per element	18
Plate thickness	0.060 inches (1.52mm)
COOLING SYSTEM	
Coolant	Pool water
Type cooling	Natural convection
Temperature	130°F (maximum)
Purity required	1 ppm – $0.5x10^6$ ohm-cm

1.1.2 Decommissioning Plan Provisions

This DP provides the following information associated with the WPI LCWNRF Reactor Decommissioning Project (WPI-RDP):

- A description of the present radiological condition of the LCWNRF and site environs
- A description of the planned approach to be employed
- Descriptions of the methods that will be utilized to ensure protection of the health and safety of the workers and to protect the environment and the public from radiological hazards
- A description of the physical security and material accountability controls that will be in place during the various project phases
- A description of radioactive waste management and disposal
- A cost estimate for decommissioning the LCWNRF and a discussion of the source of funding for these activities
- A decommissioning project schedule
- A description of the applicable quality assurance program

- A description of the training program to be established for personnel performing work in support of the project
- An Environmental Report concerning the expected impact of performing the decommissioning activities

1.2 BACKGROUND

1.2.1 Site and Facility History

The LCWNRF was made possible by a grant of $150,000 from the U.S. Atomic Energy Commission in June 1958. The LCWNRF reactor was constructed by the General Electric Company as a standard 1-kW (thermal) open-pool training reactor, and first achieved criticality on December 18, 1959. The reactor license was upgraded to 10-kW (thermal) in 1967, and the LCWNRF reactor first achieved 10-kW operation on January 31, 1968.

The LCWNRF reactor license was renewed in 1982, which extended the operating license for 20 years. In 1989, the reactor was converted from HEU fuel (93% U-235) to LEU fuel (19.75% U-235). No modifications were made to the reactor other than the fuel itself. In February 1992, a new solid-state-based control panel replaced the original vacuum-tube-based control panel. The control rod drive motors and position sensors were also upgraded at that time.

The primary use of the LCWNRF reactor was in support of WPI Nuclear Engineering-based laboratory activities and student projects. The reactor provided undergraduate and graduate students, under close supervision of qualified personnel, with reactor operating experience and experimental practice in the fields of nuclear engineering, metallurgy, chemistry, and physics.

The LCWNRF is located in the Washburn Shops and Stoddard Laboratories Building, an academic facility on the WPI campus about one mile from the center of Worcester, Massachusetts. That building is located at the top of a small hill, surrounded by other academic buildings. The nearest dormitories are located more than 500 feet away in various compass directions.

The reactor is a light-water cooled, and moderated, heterogeneous reactor, fueled with approximately 4 kg of 19.75% enriched U-235. The core is located in the center of a pool of de-mineralized water eight feet square by 15 feet deep. There is a minimum of ten feet of water above the top of the core.

A typical core configuration is based on 21 full elements, plus one-third of an element, in a rectangular array. The one-third element consists of a removable-plate element with six plates loaded (of the normal 18 plates). A one-curie Pu-Be start-up source occupies one module adjacent to the active core. Three Boral safety blades and a single, manually actuated stainless-steel regulating blade control the reactivity. The blades move vertically in two shrouds, extending the length of the core. Figure 2.1 depicts the configuration of the reactor core components.

The core is moderated, reflected, and cooled, by light water that is circulated by natural convection. The thermal column side of the core is also reflected by graphite. Core elements are contained in a grid box enclosed on four sides to confine the flow of cooling water to the channels between and surrounding the elements. The grid box and contents, as well as the blade drive mechanisms, are supported by a suspension frame from a reactor bridge. The cold, clean, core with control blades has less than 0.5% excess reactivity. The safety blades, because of their location and large surface area, have a total shutdown reactivity worth of approximately 12% delta k/k_{eff}.

There is an in-core sample irradiation device, which consists of a sample holder on a track arranged so that the contents of the sample holder may be remotely placed adjacent to the core, from the bridge above. An additional area radiation detector is located on the reactor bridge approximately one meter above the pool surface.

There are two experimental facilities, external to the reactor core: the beam port, and thermal column, depicted in Figure 2.2. The beam port is provided as a means of access to fast neutrons. The beam port is an air-filled, six-inch diameter aluminum tube, which extends from the reactor core face and terminating in a flange at the biological shield face. In its now permanent inactive state, it is shielded by a shutter inside the pool and a shield plug in the tube at the biological shield. An area radiation detector is located near the exterior of the beam port. Provisions are made for venting gasses generated in the port to the building exhaust system, and for collecting and draining any seepage that may accumulate between the port tube and surrounding biological shield concrete. Such collecting and draining occurs via tubes embedded in the biological shield.

The thermal column is a graphite-filled, horizontal penetration through the biological shield that provided a means of access to thermal neutrons. The column consists of an aluminum case, 40 inches square in cross section, and about six feet long. Access to the interior of the thermal column is by a dense, 5.5-foot thick, 2.5-foot square concrete shield door at the biological shield face. The shield door is mounted on wheels that run on rails set in the concrete floor, and can be moved perpendicular to the shield face. Provisions are made for venting gasses generated in the thermal column to the building exhaust system, via tubes embedded in the biological shield. An area radiation detector is located on the external pool wall adjacent to the thermal column door.

1.2.2 Radiological Status

An historical site assessment and radiological surveys were conducted at LCWNRF during the summer of 2008. Results of those efforts indicated that radioactivity is essentially limited to the interior of the reactor pool and the associated experimental facilities. The maximum exposure rate encountered was at the geometric center of the reactor core, predominantly from the activated stainless-steel regulating blade, with an exposure rate of 125 milliRem per hour at 13 cm through water (with a calculated in-air exposure rate of about 17 milliRem per hour at one meter).

Aside from the regulating blade, the radiation exposure rates encountered were low, and originated from other activated core components, a portion of the beam port liner near the reactor core and small metallic hardware items within the thermal column, limited to no more than a few milliRem per hour. Actual exposure rates from these other items may be lower, as radiation from the regulating blade and the spent fuel still stored in the pool may have interfered with the radiation measurements.

Based upon the direct radiation exposure rate measurements, a radiological inventory was calculated for the activated materials. The majority of the residual radioactivity at the LCWNRF is contained in the activated regulating blade, with an estimated 56 mCi of radioactivity; a mixture of Mn-54 (<1%), Fe-55 (68%), Ni-59 (<1%), Co-60 (27%) and Ni-63 (4.2%). Based on direct radiation measurements, the radioactivity in the remaining structures, systems and components is not expected to exceed a small fraction of that total.

Further radiological characterization of these radioactive items cannot be completed until the start of decommissioning, when destructive sampling techniques can be employed.

Radioactive surface contamination was not found on building or structural surfaces exterior to the reactor pool, the reactor pool's gutters and drains, floor drains, nor was it detected on a limited number of accessible interior surfaces of the ventilation and reactor water cleanup systems. Additionally, no elevated radioactivity levels were observed from NaI detector gamma scans made on these items or areas.

1.2.3 Reactor Facility Status

The Reactor is currently permanently shut down, with the fuel removed from the core and stored in racks in the reactor pool. By letter dated June 27, 2007 (Ref. 1), WPI certified to NRC that it was progressing toward decommissioning and that it would defuel and not operate the reactor after June 30, 2007. By letter dated August 13, 2007 (Ref. 2), and subsequent correspondence (Ref. 3), WPI submitted a request to NRC for an amendment to the Facility Operating License No. R-61 (including Appendix A, thereto, *Technical Specifications for the Worcester Polytechnic Institute Reactor*), to place the LCWNRF reactor in Possession-Only-License status. On August 26, 2008 (Ref. 4), NRC granted WPI's requested License Amendment No. 11, for Possession-Only License status. WPI has notified the

Department of Energy (DOE) of the Institute's desire to return the used reactor fuel to DOE (Ref. 7).

1.2.4 Reactor Decommissioning Overview

Prior to implementing the decommissioning actions described herein, WPI will have cleared LCWNRF of extraneous furnishings and materials not directly associated with the reactor. In addition, the used nuclear fuel will have been returned to the DOE. The majority of the remediation will focus on components within the reactor pool, the biological shield surrounding the reactor, and connected support systems (e.g., exhaust ventilation system, pool water treatment system and floor drains adjacent to the reactor pool). No structural contamination in other areas of the facility is known or suspected, such that only minor remediation efforts, if any, are anticipated for the building structures. Additionally, no contamination of soil or ground water is known or suspected; therefore, decommissioning of the LCWNRF can be accomplished without dismantlement or breaching of the building. After termination of the NRC license (thus allowing unrestricted use), the affected portions of the building will be restored for reuse as an educational facility. The general activities planned to complete the DP objectives are:

- Remove the activated regulating blade from the reactor core
- Drain the reactor pool
- Remove the remaining reactor core components from within the reactor pool
- Remove the beam port extension, liner and shutter
- Remove the graphite blocks and activated portions of the aluminum liner from the thermal column
- Remove the activated / contaminated portion of the aluminum pool liner, and survey and remove any affected underlying biological shield concrete
- Remove the reactor pool water treatment system
- Remove the exhaust ventilation ducts associated with the beam port and thermal column
- Survey and remediate the floor drains, if required
- Perform additional decontamination and dismantlement of the structure and equipment in accordance with this DP
- Prepare the decommissioning-generated material for free-release or disposal, as appropriate (i.e., either decontaminate and release the material as non-radioactive waste, or package the material for transport as radioactive waste)
- Ship all radioactive waste off-site to a licensed waste processor or disposal facility
- Perform and document a Final Status Survey and submit a request to NRC for termination of the reactor license

1.2.5 Estimated Cost

This estimated to cost for the radiological decommissioning of the WPI LCWNRF, including a 25% contingency, is $2,594,000. A cost breakdown is given in Table 1.2.

TABLE 1.2

RADIOLOGICAL DECOMMISSIONING COST SUMMARY
(2009 dollars*)

OPERATION	PERSON-HOURS	LABOR PLUS TRAVEL & PER DIEM	EQUIPMENT, CONTRACTS & SUPPLIES	RADWASTE SHIPPING & DISPOSAL**	TOTAL COST
Mobilization	400	$64,000	***	***	$64,000
Decontamination and Dismantling	9,060	$1,317,000	$316,000	$131,000	$1,764,000
Demobilization	***	***	***	***	***
License Termination Activities	1,440	$247,000	***	***	$247,000
Totals	10,900	$1,628,000	$316,000	$131,000	$2,075,000
				25% Contingency	**$519,000**
				GRAND TOTAL	**$2,594,000**

*** Costs have been rounded to the nearest thousand; person-hours have been rounded to the nearest ten.***
**** The estimate for low-level radioactive waste disposal is based upon the assumption that the radioactive waste will be buried at the EnergySolutions, LLC disposal site (formerly the Envirocare of Utah site).***
****** Included in Decontamination and Dismantling operations.***

1.2.6 Availability of Funds

WPI is committed to providing the funding for decommissioning of the WPI LCWNRF as evidenced by the letter of financial commitment shown in Appendix A.

1.2.7 Program Quality Assurance

A quality assurance program will be implemented throughout the decommissioning effort to assure that work is performed as intended in a controlled manner and does not endanger public safety, and to assure the safety of the decommissioning staff.

A Quality Assurance Project Plan (QAPP) will be developed to incorporate the applicable portions of Code of Federal regulations (CFR) 10 CFR 50, Appendix B (Ref. 8) and 10 CFR 71, Subpart H (Ref. 9). In addition, the QAPP will utilize a graded approach that bases the level of controls on the intended use of the results and the degree of confidence needed in their quality. ANSI/ASQC E4-1994 (ASQC 1995, Ref. 10) and NUREG-1575, Appendix K (MARSSIM, Ref. 11) will be used to provide guidance in quality-related activities, including the development of a QAPP and the collection and evaluation of environmental data.

Quality Assurance (QA) efforts during the decommissioning period will include the following:

- Performing QA functions for procurement
- Qualifying suppliers
- Auditing project activities
- Monitoring worker performance for compliance with work procedures
- Verifying compliance of radioactive shipments with appropriate procedures and regulations
- Performing dimensional, visual, nondestructive examinations or other required inspection services to assure compliance with work plans
- Maintaining auditable files

The QAPP will be prepared and implemented by a Decommissioning Consultant and/or contractors, and will be approved and overseen by WPI's management. The program will be implemented by written procedures throughout the decommissioning project. The management staff of those organizations participating in the QAPP shall regularly review the status and adequacy of that part of the QAPP that they are implementing. All changes to the QAPP shall be governed by measures commensurate with those applied to the original issue. The QAPP will incorporate the items discussed in the following subsections.

1.2.7.1 Quality Assurance Responsibilities

WPI, its Decommissioning Consultant and contractors shall have the responsibility, authority and organizational freedom to:

- Identify quality problems
- Take action to stop unsatisfactory or unsafe work and control further processing, delivery, installation or use of nonconforming items
- Initiate, recommend or provide solutions to correct quality issues, unsafe practices or non-conformances
- Verify implementation of solutions

1.2.7.2 Quality Requirements for Instrumentation

Calibration

Field instruments and associated detectors shall be calibrated on a semi-annual basis using National Institute of Standards and Technology (NIST) traceable sources. Calibration equipment and laboratory instruments shall be calibrated on an annual basis. Equipment calibrations will be performed by a qualified vendor. Calibration labels showing instrument identification number, calibration date and calibration due date shall be attached to all field and laboratory instrumentation.

Response Testing

All instrumentation will be inspected and source-checked daily prior to use to verify calibration status and proper operation. Control checks and/or source-check criteria will be established prior to the initial use of the instruments.

Maintenance

Limited maintenance, such as changing Mylar windows, high-voltage cables, etc., may be performed on-site by qualified personnel. Following any limited maintenance, calibration checks may be performed on site by qualified personnel. Major repairs and ensuing re-calibrations will be performed by a qualified vendor.

Record Keeping

Calibration and maintenance records, or copies of these records, shall be maintained on site where they will be available for review. The results of the daily instrument functional checks will be recorded on separate log sheets for each instrument and maintained on site.

1.2.7.3 Sampling and Analysis Quality Control

Sample Collection

Direct surface beta measurements, removable contamination measurements, gamma exposure rates, soil sampling and any specialized measurements will be performed to provide data required to meet the regulatory guidance provided in:

- Title 10, Code of Federal Regulations, Part 20, Section 1402 (10 CFR 20.1402), *Radiological Criteria for Unrestricted Use* (Ref. 12)
- NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (Ref. 11)
- NUREG-1727, *NMSS Decommissioning Standard Review Plan* (Ref. 6)

Sample QC

Quality Control (CQ) samples will be obtained for minimum of 10% of all samples collected for radionuclide-specific analysis. The QC samples will be a combination of split, duplicate, blank, and/or spiked samples. Independent replicate QC measurements for direct measurements will be provided for a minimum of 5% of the measurements associated with the FSS.

Sample Identification

Direct surface beta measurements, removable contamination samples, exposure rates, and any specialized measurements will be identified as to location, type of measurement, specific instrument and probe used, sample time and date (as appropriate) and name of the person collecting the data.

Volumetric samples (e.g., concrete, soil, etc.) will be identified with a unique sample number, sample location, depth of sample, sample time and date as appropriate, and the name of the person collecting the sample.

Sample Chain-of-Custody

Sample chain-of-custody shall be initiated for those samples being sent off site for analysis or transferred to another organization for analysis. A sample Chain-of-Custody Record will be generated which will document the sample identification and sample transfer and will accompany the sample during shipping to the new custodian of the sample.

Sample Analysis

Vendor laboratories shall be on a QA Approved Suppliers List for WPI or its contractors for the type of analytical services being provided. WPI has the ultimate responsibility for ensuring that decommissioning sample analysis specifications and laboratory capabilities meet data quality requirements.

Sample Documentation

Sample identification information, sample Chain-of-Custody Records, sample analysis results, vendor laboratory qualification records, or copies of these records, shall be maintained on site where they will be available for review.

1.2.7.4 Record Keeping

Measures shall be established to control the issuance of documents and changes to documents that prescribe activities affecting quality, such as procedures, drawings and specifications. These measures shall ensure that documents and changes to documents are reviewed for adequacy, approved for release by authorized personnel and distributed to and implemented at the location where the prescribed activity is performed.

Procedure Control

Procedures shall be controlled to ensure that current copies are provided to personnel performing the prescribed activities. Procedures shall be independently reviewed by a qualified person and shall be approved by a management member of the organization responsible for the prescribed activity. Significant changes to procedures shall be reviewed and approved in the same manner as the original.

Radioactive Shipment Package Documents

All documents related to a specific shipping package for radioactive material shall be controlled by appropriate procedures. All significant changes to such documents shall be similarly controlled.

Final Survey Documents

All documents related to the FSS shall be controlled by appropriate procedures. All significant changes to such documents shall be similarly controlled. This documentation would normally include a Survey Plan, Survey Packages, Survey Results and Survey Report.

1.2.7.5 Handling, Storage and Shipping

Approved procedures shall be utilized to control the handling, storage and shipping of radioactive materials.

Radioactive Material Storage

Areas shall be provided in the Reactor facility for storage of radioactive material to ensure physical protection and control of the stored material. The handling, storage and shipment of radioactive material shall be controlled through the following requirements:

- Procedures shall be provided for handling, storage and shipping operations.
- Established safety requirements concerning the handling, storage and shipping of packages for radioactive material shall be followed.
- Shipments shall not be made unless all tests, certifications, acceptances and final inspections have been completed.

Shipping and Packaging

Shipping and packaging documents for radioactive material shall be consistent with pertinent regulatory requirements.

1.2.7.6 Quality Assurance Records

Sufficient records shall be maintained to furnish evidence of activities important to safe decommissioning as required by code, standard, and specification or project procedures. Records shall be identifiable, available and retrievable. The records shall be reviewed to ensure their completeness and ability to serve their intended function. Requirements shall be established concerning record collection, safekeeping, retention, maintenance, updating, location, storage, preservation, administration and assigned responsibility. Requirements shall be consistent with applicable regulations and the potential for impact on quality and radiation exposure to the workers and the public. Typical records would include:

- Decommissioning Plan
- Procedures
- Reports

- Personnel qualification records
- Radiological and environmental site characterization records, material free-release records and FSS records
- Dismantlement records
- Inspection, surveillance, audit and assessment records

Health and Safety Related Activities

Records that have a potential for impact on quality and radiation exposure to the workers and the public include the following:

- Work Permits
- Work Procedures
- Contamination Survey Reports
- Radiation Survey Reports
- Airborne Survey Reports
- Counting data or air samples and gamma spectrum analysis
- Instrument calibrations
- Source inventory and storage
- Radioactive material inventory and storage
- Shipment records
- Incidents and accidents
- Confined space entry permits
- Monitoring records for oxygen deficient and explosive atmospheres

Personal Records

Typical records containing personal information that may impact quality and radiation exposure to the workers and the public are as follows:

- Dosimetry Records
- Bioassay analysis
- Respiratory protection qualifications (medical/clearance and fit tests, if required)
- Training records
- Visitor logs and exposure information

1.2.7.7 Audits

Audits shall be implemented to verify compliance with appropriate requirements of the Quality Assurance Project Plan and to determine the effectiveness of the plan. The audits shall be performed in accordance with written procedures or checklists by trained and qualified personnel not having direct responsibility in the areas being audited.

Audit Reports

Reports of the results of each audit shall be prepared. These reports shall include a description of the area audited, identification of the individual responsible for implementation of the audited provisions and for performance of the audit, and identification of discrepant areas. The audit report shall be distributed to the appropriate level of management and to those individuals responsible for implementation of audited provisions.

Audit Corrective Action

Measures shall be established to ensure that discrepancies identified by audits are resolved. These measures shall include notification of the manager responsible for the discrepancy and verification of satisfactory resolution. Discrepancies shall be resolved by the manager responsible for the discrepancy. Higher levels of management shall resolve disputed discrepancies. Follow-up action, including re-audit of deficient areas, shall be taken as indicated.

2.0 DECOMMISSIONING ACTIVITIES

The decommissioning alternative selected for the WPI Reactor is the removal of the facility from service and reduction of the residual radioactivity to a level that will permit unrestricted termination of the reactor license and beneficial reuse of the affected rooms within the building. The facility will be surveyed and left in place.

2.1 DECOMMISSIONING ALTERNATIVE

The objective of the decommissioning is the regulatory release of the Reactor facility located within the Washburn Shops and Stoddard Laboratories building, allowing for their unrestricted use. On this basis the safe storage (SAFSTOR) and entombment (ENTOMB) decommissioning options were considered inappropriate to WPI's future plans, and DECON is the decommissioning alternative selected by WPI.

SAFSTOR poses essentially the same potential risks and environmental impacts as the proposed project, but for a much greater period of time. This alternative would necessitate continued surveillance and maintenance of the Reactor facility over a substantial time period during which the radiological risks would continue to exist.

ENTOMB would necessitate continued surveillance and maintenance of the Reactor facility over a substantial time period. During this period, the radiological risks would continue to exist.

DECON is the decommissioning option chosen by WPI. To the extent possible, decontamination of facility equipment and structural components will be conducted so as to minimize radioactive waste. Structural portions of the building found to be radiologically contaminated and/or activated shall be dismantled and/or decontaminated, as necessary. This would be followed by an extensive and comprehensive final radiation and contamination survey demonstrating that the LCWNRF meets NRC criteria for release to unrestricted use. The results of this final survey will be documented in a report to be submitted to NRC in support of a request that the site be released for unrestricted use and the reactor license terminated.

2.2 FACILITY RADIOLOGICAL STATUS

2.2.1 Facility Operating History

- Initial reactor criticality: December 12, 1959
- Reactor operation (at 1KW): 1959 to 1967
- Conversion from 1 KW to 10KW: 1968
- Reactor operation (at 10 KW): 1968 to 2007

- Typical reactor operation: ~ 100 hours per year
- Conversion from HEU to LEU fuel: 1989
- Certification of permanent reactor shut-down: 2007
- Possession-only License Amendment: August 26, 2008
- Request return of reactor fuel to DOE: September 2008
- Return of reactor fuel to DOE: To Be Determined

The LCWNRF was constructed at WPI to provide graduate and undergraduate students with reactor operating experience and experimental practice in the fields of nuclear engineering, metallurgy, chemistry and physics, as well as irradiation services for other teaching, medical and industrial institutions. The integrated power generated during operation of the WPI Reactor is estimated at approximately 41 MW-hours.

2.2.2 Current Radiological Status of the WPI Facility

2.2.2.1 General

Routine radiological surveys show that the radiation levels and contamination levels measured at the LCWNRF have consistently been low and limited in extent. Radiological measurements performed in the summer of 2008, confirmed that only minor quantities of residual radioactivity or radioactive contamination are present. The information indicates that the radioactive portions of the facility are primarily confined to the reactor internals and reactor pool. The majority of the residual radioactivity at the facility contained in structures, systems and components, is found in a single item, an activated stainless-steel regulating blade (as discussed in section 1.2.2. Other than the exposure rate emanating from the regulating blade (125 mR/hr at 13 cm, through water) radiation exposure rates at the reactor facility were found to be very low and limited to portions of the reactor core box and grid, the reactor core end of the beam port and the reactor side of the graphite thermal column interior, that were slightly neutron activated. Maximum radiation levels on these non-regulating blade components were found to be in the range of one to three mrem per hour on contact. Radiation and radioactive contamination was not detected outside of the reactor pool or experimental facilities in the biological shield.

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 the energies of the incident neutrons. However, in the case of the WPI Reactor, with its low and intermittent usage, direct measurements are generally more reliable and will be used to guide the actual removal and/or dismantlement of components. Characterization of radionuclide mixtures and concentrations will be performed at the time of actual removal and dismantlement, as this requires destructive sampling of structures, systems and components that are currently needed to support safe storage of the used fuel, and would not be permitted by the possession only license. This information will be obtained during the process of decommissioning and will further define the basis for specifying the necessary safety measures and procedures for the various dismantling, removal, decontamination, and waste packaging and storage operations so that exposure to personnel is maintained ALARA.

2.2.2.2 <u>Principal Radioactive Components</u>

This listing is based upon process knowledge and direct radiation measurements. The most radioactive components to be handled and processed during the WPI LCWNRF Reactor Decommissioning Project (WPI-RDP) are those that have become neutron activated.

The reactor core structure was principally constructed of Aluminum, with a small amount of stainless-steel (i.e., the single regulating blade and small fasteners, such as bolts and pins). Based upon the elemental constituents of these materials, only the stainless steel items should show a significant amount of neutron activation. In the case of the biological shield and experimental facilities, there are elemental constituents within the concrete, steel rebar and graphite that can become activated; however due to their distance from the reactor core, and the low power level and low usage of the reactor, there should not be a significant amount of activation. This pattern of neutron activation was confirmed by the gamma exposure rates measured in and around the reactor core and pool structures.

A summary of the most significant radiological characteristics found at the LCWNRF are as follows (see Figures 2.1 and 2.2 for location and configuration of these items):

- The gamma exposure rate at the geometric center of the reactor core, between the three Boral Control Blades and one stainless-steel Regulating Blade was measured at 125 mR/hr (at 13 cm), through water). The Regulating Blade is believed to be the principal the source, and based on its composition and geometry, an in air exposure rate of 17 mR/hr at 1 meter is estimated. That 6.9 Kg (884 cubic centimeter) blade is estimated to contain 56 mCi of radioactivity; Mn-54 (<1%), Fe-55 (68%), Ni-59 (<1%), Co-60 (27%) and Ni-63 (4.2%).
- Ion Chambers and lower Suspension Posts, located above the top corners of the active core region were found to have contact gamma exposure rates of about 2 mR/hr (under water).
- Gamma exposure rates from the Reactor Core Box, at mid-core plane elevation, on the exposed West, South and North sides, were found to be 4, 7 and 15 mR/hr, respectively, measured through 30 cm of water. Based on the core configuration and the radiation level pattern these radiation levels are likely influenced by the activated stainless steel regulating control blade and the spent fuel that is currently stored in racks on the North and South pool walls, and not due to significant activation of the Aluminum Core Box.

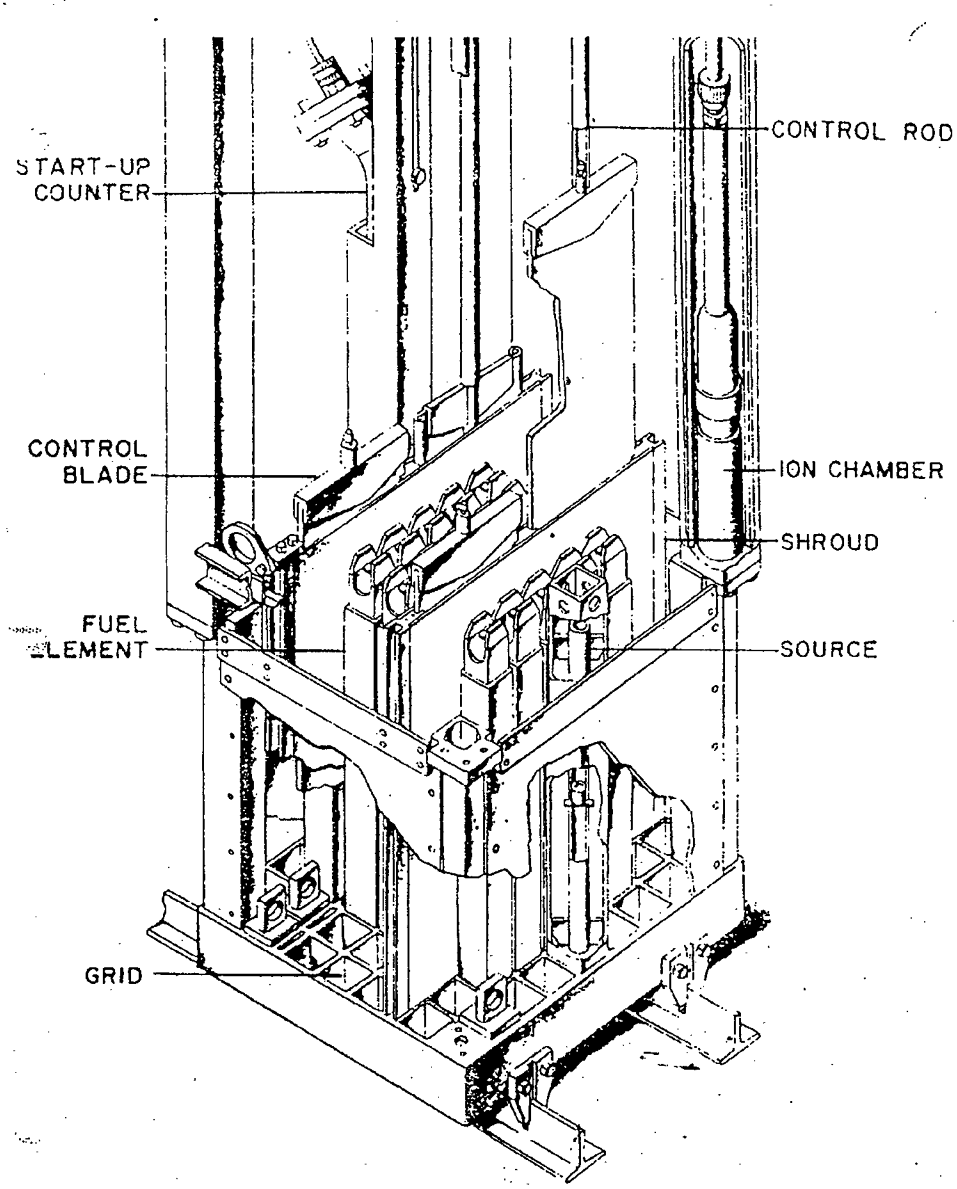
- Gamma radiation levels were measured, in air, inside the six-inch diameter Beam Tube Extension, with 0.4mR/hr where the tube penetrates the pool wall and 4 mR/hr where the extension tube terminates at the reactor core face.
- Activated stainless steel Heli-Coil hardware embedded in thermal column graphite blocks were found with gamma exposure rates of 200 microR/hr on contact.
- Gamma exposure rate measured on contact with select Thermal Column graphite blocks, ranged from background-to-10 microR/hr above background. As expected, the graphite blocks closest to the reactor core were those found with the detectable radiation levels.

2.2.2.3 Radionuclides

The radionuclides expected to be present, or that may possibly be present, in the WPI Reactor at the time of decommissioning, are listed in Table 2.1. This list of radionuclides is principally based on the assumption that reactor operation resulted in neutron activation of reactor core components and other integral hardware or structural members situated adjacent to, or in close proximity to, the reactor core. Specific materials that are considered to have been exposed to neutron activation include principally aluminum, and small amounts of stainless steel, graphite and concrete. Co-60 and Fe-55 are the principal radionuclides expected to be present in detectable quantities at the WPI LCWNRF. Table 2.1 also contains other radionuclides that generically can be present at a research reactor, but are not expected in detectable quantities at the LCWNRF, such as fission products from the fuel and Europium radioisotopes from activated graphite and concrete. The determination of actual radionuclides and concentrations present as residual activity in structures surrounding the reactor, and there content in waste materials will be based upon direct measurements and sampling performed during the decommissioning process.

FIGURE 2.1

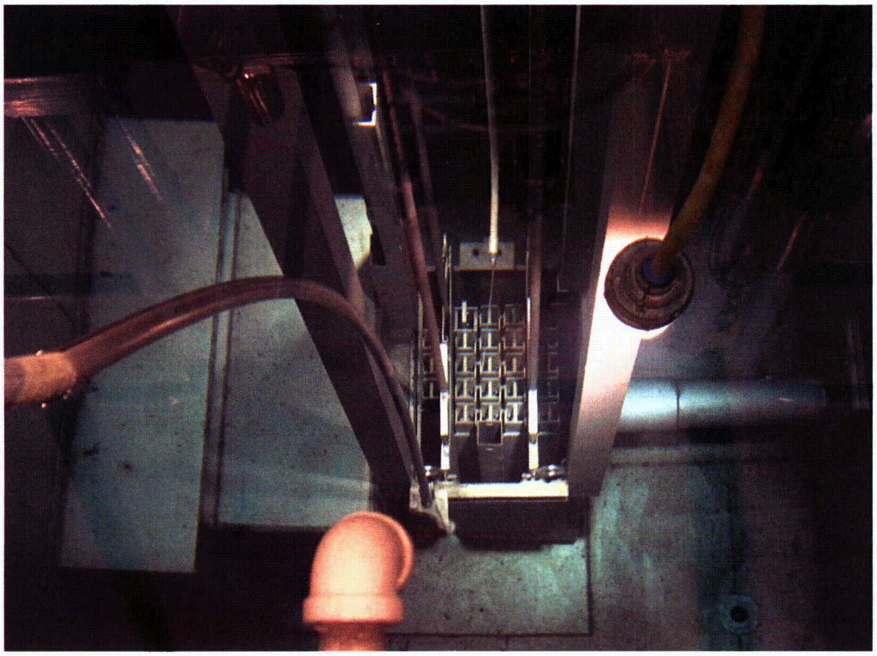
REACTOR CORE DRAWING AND PHOTOGRAPHIC VIEW



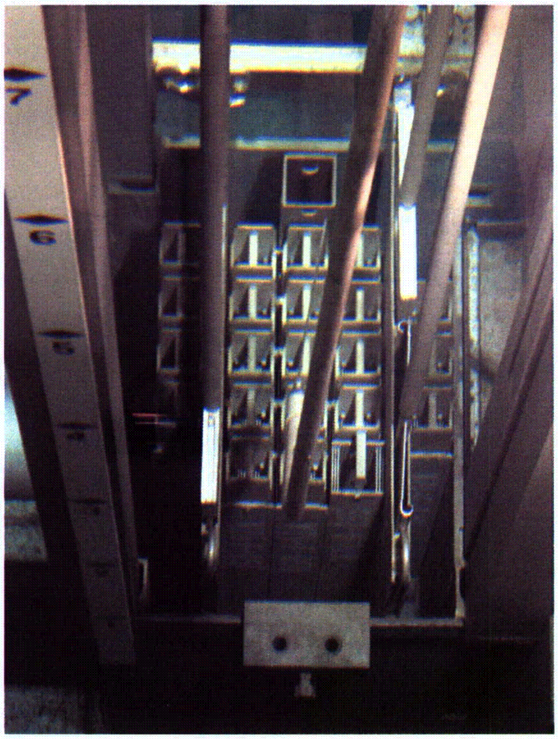
(a) Reactor Core - Artist Rendering

FIGURE 2.1 (continued)

REACTOR CORE DRAWING AND PHOTOGRAPHIC VIEW



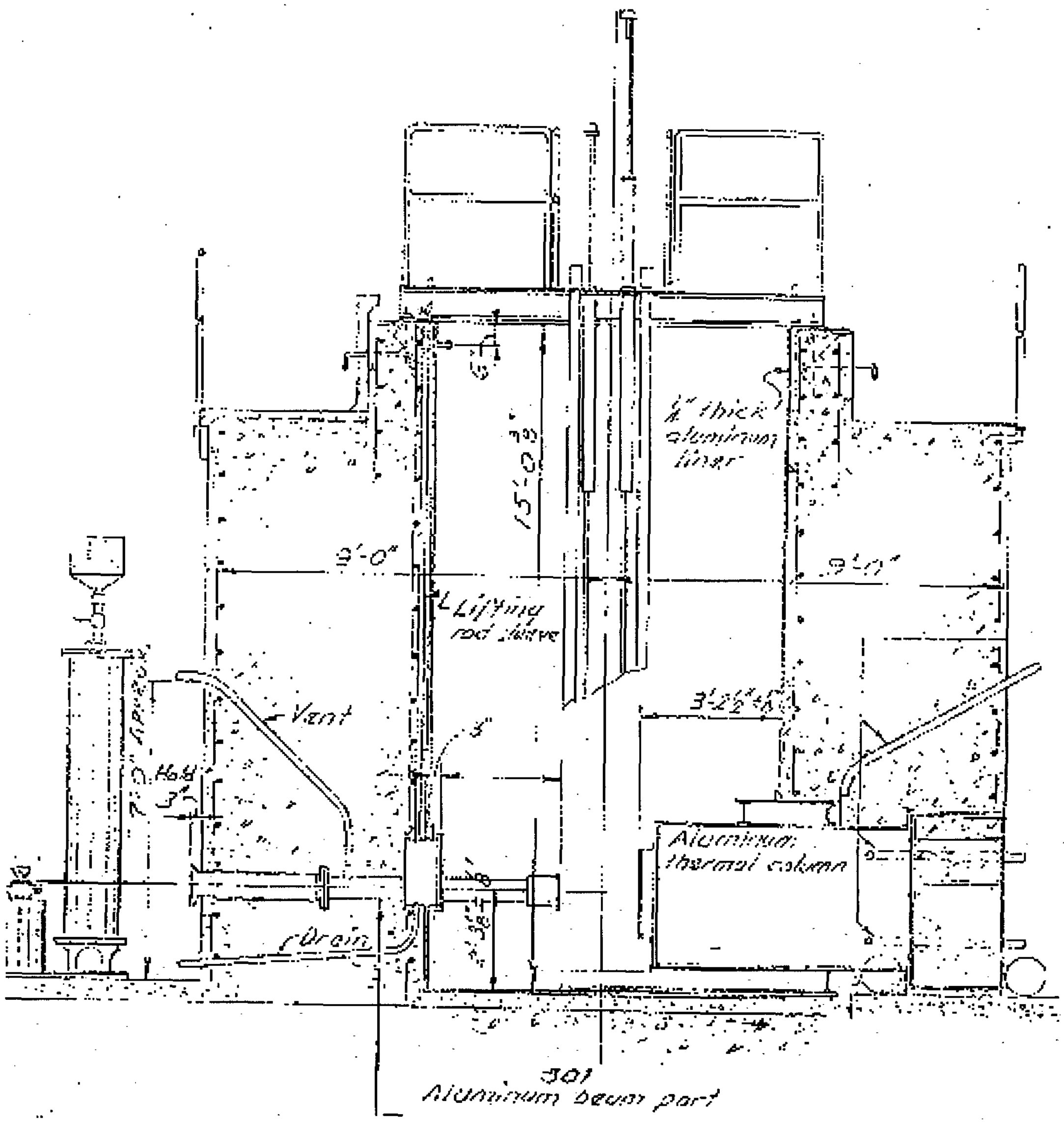
(a) Reactor Core Photo – Overall View



(b) Reactor Core Photo – Close-up View

FIGURE 2.2

REACTOR POOL AND BIOLOGICAL SHIELD CROSS-SECTION DRAWING



2.3 DECOMMISSIONING TASKS

2.3.1 Activities and Tasks

WPI Reactor facility decommissioning activities will be conducted in four phases, as follows:

- Phase One: Pre-Decommissioning Activities
- Phase Two: Decommissioning Mobilization Activities
- Phase Three: Decontamination and Dismantling Activities
- Phase Four: Site Restoration Activities

TABLE 2.1

LIST OF EXPECTED AND POTENTIAL RADIONUCLIDES

NUCLIDE	HALF-LIFE* (years)	MEANS OF PRODUCTION
^{3}H	12.28	Activation
^{14}C	5,730	Activation
^{54}Mn	0.856	Activation
^{55}Fe	2.73	Activation
^{60}Co	5.27	Activation
^{59}Ni	76,000	Activation
^{63}Ni	100	Activation
^{63}Zn	0.67	Activation
^{90}Sr	29.1	Fission
^{94}Nb	20,000	Activation
^{99}Tc	213,000	Fission
^{125}Sb	2.76	Fission
^{129}I	15,700,000	Fission
^{134}Cs	2.07	Activation
^{137}Cs	30.17	Fission
^{152}Eu	13.48	Activation
^{154}Eu	8.8	Activation
^{155}Eu	4.96	Activation

** Radionuclide Half-Life values are from "The Health Physics and Radiological Health Handbook" (Ref. 13).*

Phase One and Phase Four activities are considered outside of the scope of this DP: however, they are described briefly below to provide an overall context to this DP.

Phase One

Phase One activities are those tasks performed to prepare the facility for future decommissioning activities. This work will be performed under the authority of the existing possession-only license and may be performed before NRC approval to decommission is granted. This work will include general cleanup of the facility, such as removal of non-contaminated equipment and materials (non-reactor related) situated throughout the WPI LCWNRF: such materials will be collected, surveyed, and appropriately dispositioned in accordance with established procedures. This would generally include removal of no-longer needed furniture, laboratory and office supplies. Non-radiologically contaminated ceiling and floor tiles and insulation, which may contain asbestos, may be removed during this phase to facilitate future decommissioning activities. At no time during Phase One will any of the reactor's structures, systems or components be dismantled or decontaminated. It is also expected that the used nuclear fuel would be returned to DOE during this phase.

Phase Four

Phase Four activities are those tasks performed after the facility's NRC license has been terminated, allowing for unrestricted use. During this phase, the facility will be restored and remodeled for educational use, such as classroom or office space. While detailed plans have not yet been devised, it is expected that the remaining portions of the biological shield, the decontaminated reactor bridge and control console will be demolished, and disturbed structural surfaces repaired.

Phases Two and Three activities encompass the radiological decommissioning tasks that will be performed in accordance with the provisions of this plan, as described herein. WPI anticipates that this work will be performed by experienced and qualified contractors, managed by a Decommissioning Consultant, and overseen by the RHSC (see Sections 2.4.3 Decommissioning Consultant and 2.4.4 Contractor Assistance).

2.3.1.1 Phase Two: Decommissioning Mobilization Activities

Phase Two will commence after a Decommissioning Order has been granted by NRC and the used nuclear fuel has been removed from the reactor pool. The tasks in this phase will include preparatory work needed to fulfill the administrative and physical requirements of this DP, prior to the conduct of Phase Three decontamination and decommissioning (D&D) activities. WPI anticipates that Phase Two will be implemented in two sub-phases; off-site and on-site mobilization.

Off-site mobilization activities will include detailed project planning and engineering, preparation of detailed operating procedures, and procurement of specialty equipment, supplies and subcontractors.

On-site mobilization will include training and qualification of workers and physical preparation of the facility for phase three decommissioning activities. Training and qualification of workers will be performed as described in Section 2.5 Training Program.

Systems and components to be dismantled will be de-energized and isolated at this time in order to assure the safety of the decommissioning workers. Temporary electric power, lighting and other utilities will be installed as needed. Material-handling equipment, such as ramps, gantry cranes, etc., will be installed as required to support the decommissioning work.

General radiological controls will be installed to support the decommissioning work. Contamination control boundaries and protective barriers, personnel contamination frisking stations and step-off pads will be setup to isolate the active work areas from the un-involved portions of the facility. Facilities for workers to change into protective clothing will be set up. Portable air samplers, continuous air monitors, area radiation monitors, containment tents and High-Efficiency Particulate Air (HEPA) filtered ventilation units will be setup as required to support the decommissioning work. A Health Physics (HP) counting room will be set up for sample analysis to support radiation protection, waste management and final site survey (FSS) work. A secure radioactive material storage area will be setup, where waste material can be accumulated, packaged and prepared for shipment to the disposal facility. Additionally, a low-background area will be set up for conducting free-release surveys and secure storage of materials prior to release.

During this phase, characterization of structures, systems and components will be performed on those areas that could not be performed prior to receipt of the decommissioning order (e.g., destructive sampling of activated reactor core and biological shield materials).

2.3.1.2 <u>Phase Three: Decontamination and Dismantling Activities</u>

Phase Three activities will be comprised of ten discrete tasks involving the physical dismantling of contaminated and activated equipment and removal or decontamination of structures, plus three additional tasks associated with:

- The preparation, packaging and transportation of radioactive waste for disposal (see Section 3.2 Waste Management)
- Demobilization and preparation of the building for final surveys
- Performance of a FSS to verify compliance with facility release criteria (see Sections 2.7 Release Criteria and 4.0 Proposed Final Radiation Survey Plan)

The general philosophy concerning the dismantling and removal of contaminated or activated equipment will be to use non-destructive disassembly techniques, such as un-bolting, wherever possible. Such techniques eliminate creation of contaminated particles and debris that can create airborne and contamination migrations risks. If un-bolting can not be performed (e.g., if frozen bolts are encountered) only mechanical segmentation techniques will be employed in order to minimize the creation of airborne dusts, mists or fumes: examples of such mechanical segmentation techniques include sawing and shearing.

In addition, whenever possible, equipment will be removed as whole items to minimize exposure times and creation of loose contamination. The thirteen main D&D tasks are discussed in the following pages.

Task 1: Remove Reactor Core Structure

The objective of this first task will be the removal of the reactor core support structure from the pool. The major items include the following:

- Reactor Core Box and Grid
- Control Blade Drive Shafts and Drive Mechanisms
- Regulating Blade
- Safety Blades
- Ion Chambers and associated instrumentation hardware
- Start-up Counter Assembly and Guide Tube Assembly
- Sample Irradiation
- Reactor Suspension Posts

The reactor core structure is suspended by a four-post aluminum suspension frame, attached to the bridge spanning the top of the pool. Activation levels and the resulting radiation exposure rates emanating from the core structure are low – in the range of a few mrem/hr. Because of the low radiation levels, remote underwater disassembly of the structure will not be required. Additionally, because the core structure is small, it will not need to be segmented to fit into waste containers.

The first items to be removed from the core will be the regulating blade, which contain the bulk of the reactor's radioactivity and are the largest contributor to radiation exposure rates. The four safety / control blade drive mechanisms and drive shafts will be decoupled at the reactor bridge. The control drive shafts and the attached blades will be lifted out of the core box by workers latching onto the blades with long-handled tools. The blades and shafts will be raised to the surface of the pool and decoupled. The control blades, one of which is the regulating blade which has the highest estimated radiation levels of 17 mR/hr at one meter, will be manually placed into steel waste container boxes. The control blade drive shafts will be manually cut by mechanical means (saws or shears), and sized to fit into the waste boxes.

Exposure rates from the remaining pool items will be sufficiently low as to safely allow draining of the pool prior to further disassembly of the reactor core. The pool will be drained in a controlled manner using the existing liquid effluent discharge protocols. Pool water will be drained to temporary batch tanks, sampled, analyzed and treated (if necessary) and then released to the sanitary sewer system. Radiation levels above the pool water will be monitored as the water level is lowered to assure that no un-expectedly higher sources of radiation are exposed. Loose surface contamination levels on the exposed surfaces will be checked as the pool water is drained, to assure that no sources of potential airborne contamination are exposed. If significant loose contamination is indicated, the exposed surfaces will be washed down with water sprays and the loose contamination will be collected along with the drained pool water.

After the pool has been drained, the reactor core box will be removed. The core box consists of the core box sides, grid plate, control blade shrouds, and other small, miscellaneous hardware items. The entire reactor core box structure is small in size (about 36 inches high by 21 inches wide by 28 inches long) and will be removed whole, without the need for segmentation.

After removal of the core box structure, radiation levels should be low enough to allow manned entry into the pool. The reactor core structure will be unbolted from the suspension posts and floor locating lugs using hand tools, and slid laterally to clear the suspension frame assembly. A gantry crane will be set up to span the top of the reactor pool and will be used to hoist the core box out of the pool. Once hoisted out of the pool, the core box will be placed directly into a waste container staged nearby.

After removal of the core box, workers will then remove the suspension frame, ion chambers and start-up counter / guide assembly. These items form one integral structure, with the ion chambers being located inside of three of the four hollow corner posts that comprise the lower portion of the suspension frame assembly. While the suspension assembly is hanging from the reactor bridge above the pool, workers will manually segment the assembly into manageable-sized pieces using saws or shears. Segments will be hoisted out of the pool using the previously installed gantry crane, and placed into waste containers staged nearby.

<u>Task 2: Remove Contents of Thermal Column</u>

This task removes the contents of the thermal column. The thermal column, a horizontal, aluminum-lined penetration through the biological shield, is filled with graphite blocks. The individual graphite blocks have an approximately four-inch square cross-section, and vary in length up to about three feet long. The graphite blocks are stacked in alternating perpendicular rows; some of the blocks contain embedded stainless Helicoils to facilitate their removal using a threaded T-bar.

During scoping surveys, radiation exposure rates emanating from the graphite and Helicoils were found to be very low – 0.01 and 0.2 mR/hr on contact, respectively. Smear samples taken on the graphite did not reveal loose graphite dust or radioactive surface contamination. The graphite blocks will be manually un-stacked and placed into waste containers staged nearby.

Task 3: Remove Reactor Pool Equipment

The objective of this task will be to remove the remaining activated equipment from the reactor pool. The equipment is expected to exhibit very low radiation levels, generally less than one mR/hr. Much of pool equipment is either attached to the aluminum pool liner or embedded into the underlying concrete structure. Reactor pool equipment includes the following items:

- Beam port extension
- Beam port shutter and shutter housing
- Core box locating rails
- Spent fuel racks

The beam port extension is a six-inch diameter aluminum tube that extends between the interior face of the biological shield and the reactor core box. It will be unbolted from the beam port shutter assembly for removal from the pool. The beam port shutter is a movable shield inside the beam port shutter housing: the beam port shutter and its attached lifting rod will be removed from the housing. The beam port extension and the beam port shutter (and lifting rod) will then be hoisted out of the pool using a gantry crane and placed into a waste container staged nearby.

The remaining items will require destructive removal techniques that have a potential for creating loose debris. Accordingly, engineering controls will be used as required to contain and collect any potentially contaminated debris. Such controls may include enclosing the top of the pool with tarpaulins and using HEPA-filtered ventilation to maintain inward air flow and the use of HEPA vacuums for collection of debris. The sequence of equipment removal will be determined by an ALARA evaluation, with the items with the highest exposure rates being removed first.

The beam port shutter housing is an aluminum box welded to the aluminum pool liner and partially embedded in the biological shield concrete. The aluminum liner surrounding the housing will be cut and removed by sawing, drilling or chisel cutting. Jack hammering and concrete drilling will be used to break away surrounding concrete to free the housing. The housing will then be saw-cut from the remaining beam port tube still embedded in the biological shield.

The core box locating rails are two aluminum T-channels attached to the pool liner floor that provided support for the reactor core box. The rails will be detached from the pool liner by sawing, drilling or chisel cutting. The two spent fuel racks are also attached to the liner on the pool floor, and will also be detached from the pool liner by sawing, drilling or chisel cutting.

Task 4: Remove Reactor Pool Water Treatment System

The objective of this task will be the disassembly and removal of piping and equipment comprising the pool water treatment system. This system is external to the reactor pool, and is comprised of the following items:

- Ion exchange column
- Filter housing
- Pump
- Hold-up tank
- Piping (metal and plastic, all less than two-inch diameter)

Radiological surveys conducted in 2008 did not reveal any radiation exposure rates above background from this equipment, and internal contamination levels are expected to be low, if at all detectable. These are small items, the largest of which is the ion-exchange column, approximately 7.5 feet long by 1.5 feet in diameter.

The equipment will be drained (and will have been previously de-energized during initial decommissioning set-up activities). Generally, the equipment will not need to be segmented to fit into waste containers. Connecting piping and anchor bolts will be saw-cut, sheared or un-threaded. Catch basins, drop cloths and HEPA vacuums will be used to control and capture any potentially contaminated debris.

Task 5: Remove Liners and Embedments from Biological Shield

The objective of this task will be the removal of contaminated or activated portions of the liner from within the pool, thermal column, beam port and gutter drains; and removal of contaminated piping embedded within the biological shield. The exact amount and location of liner that will need to be removed cannot be determined at this time. Characterization will take place after the pool has been drained and activated equipment removed, at which time direct radiation and contamination measurements can be made to direct the work activities.

At the time of construction, the liners were pre-fabricated from ¼-inch thick sheets of aluminum, and used as the interior form for casting-in-place the concrete biological shield. Removal will entail pre-cutting the liner in place and prying the pieces free from the underlying concrete. The aluminum liner will be mechanically cut (e.g., sawing, drilling or chisel cutting). The manageable-sized pieces will be mechanically pried free using pry bars, jack hammers or hydraulic spreaders.

The beam port liner embedded in the biological shield (and potentially activated concrete surrounding it) will be removed by over coring around the beam port liner, at a diameter of approximately 16-to-18 inches. A contractor specializing in concrete cutting and coring will be utilized for this operation. Once over cored, the core containing the liner will be withdrawn from the biological shield using chain falls and a portable gantry crane. The resulting core, approximately 5.5 feet in length, may be cross-cut into smaller, more manageable segments. The segments will then be placed into waste containers.

Concrete coring and cross cutting will necessitate the use of water for cooling and lubrication, and to prevent the generation of any potential airborne dust. The resulting slurry will be controlled, collected and captured as a waste material. Berms, catch basins and plastic drapes, as well as wet / dry HEPA vacuums will be used to capture and collect the slurry. The resulting slurry will be allowed to congeal and further dried by mixing with cement, if required, and placed into waste containers.

Potentially contaminated embedded pipes within the biological shield include the thermal column and beam port vents and drains, the inlet and outlet pipes for the pool water treatment system, gutter drain lines and the beam port shutter lifting rod sleeve. There may be other, non-system-related embedments present (such as electrical conduits) that will be evaluated for removal on a case-by-case basis. Such pipes (or conduits) may or may not be contaminated. The interior of the embedded pipes will be checked for contamination at accessible openings, using swabs and small-diameter detectors. If contamination is detected, or they can not be adequately accessed or contamination levels quantified, the embedded pipes will be assumed to be contaminated above release criteria. If these items require removal, they will either be over cored or the surrounding concrete jack hammered to expose the items for removal.

<u>Task 6: Decontaminate Concrete Biological Shield</u>

The objective of this task will be to determine if activated or contaminated biological shield concrete is present, and if found removed to meet release criteria. By the start of Task 6, all contaminated and activated items will have been removed from the biological shield. The remaining concrete will be surveyed and sampled for activation and surface contamination. Added attention for characterizing activation will be given near the beam port, thermal column and floor beneath the core box. Due to the reactor's low power level and limited use, extensive and deep neutron activation is unlikely, if at all present. Added attention for characterizing surface contamination will be given if signs of water leakage are observed, and near locations having higher probabilities of leakage, such as liner seams and penetrations.

While activated or contaminated concrete is not expected, if it is found above release criteria, it will be removed by mechanical means, such as surface chipping, jack hammering or hydraulic splitting. This process would be iterative, where material would be removed, resulting surfaces characterized, and the procedure repeated until achievement of release criteria.

These removal techniques have a potential for creating loose or airborne dust. Accordingly, engineering controls will be used as required to control, contain and collect any potentially contaminated dust and debris. Such controls may include enclosing the top of the pool and other openings, use of HEPA-filtered ventilation for inward air flow, HEPA vacuum collection of debris, and water misting.

Task 7: Remove Liquid Effluent Discharge Equipment

The objective of this task will be to gain access to the interior of the liquid effluent discharge equipment for the purpose of radiological characterization, to determine if remediation is required, and if necessary removing them as radioactive waste.

The liquid effluent discharge equipment consists of a few feet of small-bore piping, which connects the hold-up tank (removed in Task 4) and reactor pool water discharge pipe to a locked valve, which in-turn connects to a pipe that discharges to the sanitary sewer system. The piping is partially embedded in the concrete floor of the building's lower level, with the valve located in a floor pit.

The radiological condition of the interior of this piping and valve is not known at this time. Based on historical site assessment information and scoping survey results, these items may not be radiologically contaminated.

Pipes and the valve will be mechanically segmented, at non-embedded locations, to provide access to interior pipe surfaces. The interior of the pipes will be checked for contamination at accessible openings, using swabs and small-diameter detectors. If contamination is detected, or they can not be adequately accessed or contamination levels quantified, it will be assumed that the piping is contaminated above release criteria. Otherwise, they will be abandoned in-place. If these items require removal, they will either over cored or the surrounding concrete jack hammered to expose the items for removal. Additionally, if radioactive contamination is found, the first sewer manhole receiving discharges from this piping will also be checked for radioactive contamination and decontaminated or removed as required.

Task 8: Remove Exhaust Ventilation Equipment

The objective of this task is the removal of ventilation ducts that could possibly containing internal contamination, and the evaluation of down-stream building ventilation equipment to determine if additional remediation is warranted.

The exhaust ventilation equipment consists of about 50 linear feet of small-diameter (< 4-inch), light-gauge metal duct work, with two inline electric booster fans, that are connected to the beam port and thermal column vent pipes embedded in the biological shield. During reactor operation the ducts drew a vacuum on the beam port and thermal column such that any gasses generated (e.g., Ar-41, half-life 1.8 hours) would be routed to the building's main exhaust ventilation system.

The ventilation duct will be segmented and removed using mechanical methods, such as saws or shears, and placed into waste containers. The duct will be removed back to the point where it enters the building's main exhaust ventilation system.

The interior of the building's main ventilation system will be checked for contamination at accessible openings, using swabs and small-diameter detectors. While not expected, if contamination is detected, or the system's interior can not be adequately accessed or contamination levels properly quantified, the ventilation system will be assumed to be contaminated above release criteria and removed. Otherwise, it will be left in-place. If radioactive contamination is found in the ventilation ducts, the discharge stack on the roof (and surrounding roofs) surfaces will also be checked for radioactive contamination and decontaminated or removed as required.

Task 9: Remove Remaining Equipment

The objective of this task is to survey, evaluate and/or remove any fixtures or equipment remaining within the WPI LCWNRF, where surface contamination could have been missed or hidden in, under or behind equipment or fixtures. Such remaining items may include sinks, old tools, work benches, etc.

Task 10: Decontaminate Building Interior Surfaces

The objective of this task is to provide added assurance that the facility is ready for conducting the FSS. This task will principally involve taking radiological measurements to identify any contamination within the facility that may have been missed during prior tasks. If any remaining surface contamination is identified it will be remediated at this time. Based upon the results of the historical site assessment and scoping surveys, any additional structural decontamination is expected to be minimal.

Task 11: Prepare Waste Packages for Transportation and Disposal

The objective of this task will be the final disposition of the radioactive wastes generated during the decommissioning. Due to the relatively small amount of waste expected to be generated (less than 600 cubic feet), only one truck shipment should be required. As such, containers of waste materials will be staged and prepared for shipment at the conclusion of the D&D activities. Section 3.2 describes the waste management activities to be performed during this task.

Task 12: Demobilize and Prepare Reactor Rooms for Final Status Survey

During this task, the WPI facility will be prepared for the FSS. Contractors' tools and equipment used during the decommissioning will be surveyed, decontaminated (if necessary), and removed from the facility. Final housekeeping will be performed to remove debris, dust or protruding objects from surfaces, which would otherwise interfere with the final survey. Ladders, scaffolding and equipment to be used for the final survey will be staged.

Task 13: Perform License Termination Activities

During this task, the FSS Plan will be prepared, implemented and the final report prepared and submitted to NRC with a request for termination of the R-61 license. Section 4.0, Proposed Final Radiation Survey Plan, describes these activities.

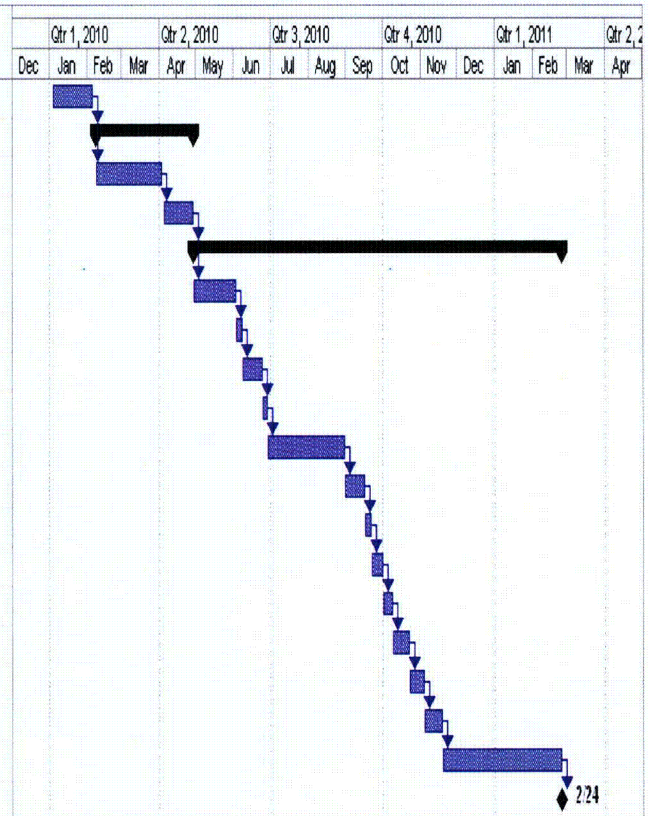
2.3.2 Schedule

The decommissioning project schedule is presented as Figure 2.3. The January 2010 start date shown on the schedule was chosen for illustration purposes only, with the actual start date based on the timing of NRC approval of this DP and the DOE removal of the fuel. Due to the facility's small size, the tasks listed above will essentially be performed in series, by a single work crew. Some task work may be performed in parallel to level work loads, where radiation and industrial safety interferences do not exist. The duration of actual D&D work is expected to be less than 12 months. However, due to WPI being an active educational institution, some gaps in the work may need to be scheduled so as to avoid disruption of WPI's normal activities. Additionally, changes to the schedule may be made at WPI's discretion as a result of resource allocation, availability of a radioactive waste burial site, ALARA optimization and force majeure considerations. Notwithstanding the foregoing, WPI does intend to complete the decommissioning project as soon as possible. Based on the information in this Decommissioning Plan (DP), WPI anticipates that a formal request for termination of Facility License No. R-61 will be submitted to NRC within 24 months after the approval of the DP is received from NRC.

FIGURE 2.3

LCWNRF DECOMMISSIONING SCHEDULE

ID	WBS	Task Name
1	1	Phase 1: Pre-Decommissioning Activities (Performed prior to NRC DP Approval)
2	2	Phase 2: Decommissioning Mobilization Activities
3	2.1	Off-site Mobilization
4	2.2	On-site Mobilization
5	3	Phase 3: Decontamination and Dismantling Activities
6	3.1	Task 1: Remove Reactor Core Structure
7	3.2	Task 2: Remove Contents of Thermal Column
8	3.3	Task 3: Remove Reactor Pool Equipment
9	3.4	Task 4: Remove Pool Water Treatment System
10	3.5	Task 5: Remove Reactor Pool Liners and Embedded Drains and Pipe
11	3.6	Task 6: Decontaminate Concrete Biological Shield
12	3.7	Task 7: Remove Liquid Effluent Discharge Equipment
13	3.8	Task 8: Remove Exhaust Ventilation System
14	3.9	Task 9: Remove Remaining Equipment
15	3.10	Task 10: Decontaminate Building Interior Surfaces
16	3.11	Task 11: Waste Management
17	3.12	Task 12: Demobilize and Prepare Building for FSS
18	3.13	Task 13: Perform License Termination Activities
19	4	Phase 4: Site Restoration (Post Lic. Term.)



2.4 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

WPI is committed to, and retains ultimate responsibility for, full compliance with the Institute's existing NRC reactor licenses and applicable regulatory requirements during decommissioning. WPI's policies and goals will be followed to ensure high standards of performance in accomplishing the decommissioning tasks.

WPI intends to utilize experienced and qualified consultants and contractors to perform the physical decommissioning work. WPI will select the contractors after NRC's approval of the DP. The Reactor Director (from the Office of the Vice President for Finance and Operations) and the WPI Radiation Safety Officer (RSO), with assistance of the Radiation Health and Safety Committee (RHSC) will monitor performance of the consultants and contractors to ensure that the decommissioning work is being performed safely and according to federal, state and local regulatory requirements (NRC, EPA, DOT, etc.).

Consistent with WPI policy, the RHSC has certain responsibilities to review and approve policies, procedures, programs and facilities pursuant to the safe use of radiological materials and radiation-producing equipment. The RHSC's jurisdiction will extend to all decommissioning activities dealing with radioactive material and radiological controls. The RSO and the RHSC will approve consultant and contractor plans, policies and procedures used during the decommissioning as described in this DP.

The planned organization for the WPI LCWNRF Reactor Decommissioning Project (WPI-RDP) is shown in Figure 2.4. Individuals performing the functions may vary over the project duration; however, the specified functions will be maintained. Consultants, contractors and subcontractors performing work under this DP will be required to comply with applicable WPI policies, procedures and regulatory commitments.

2.4.1 WPI Reactor Director

The WPI Reactor Director functions include:

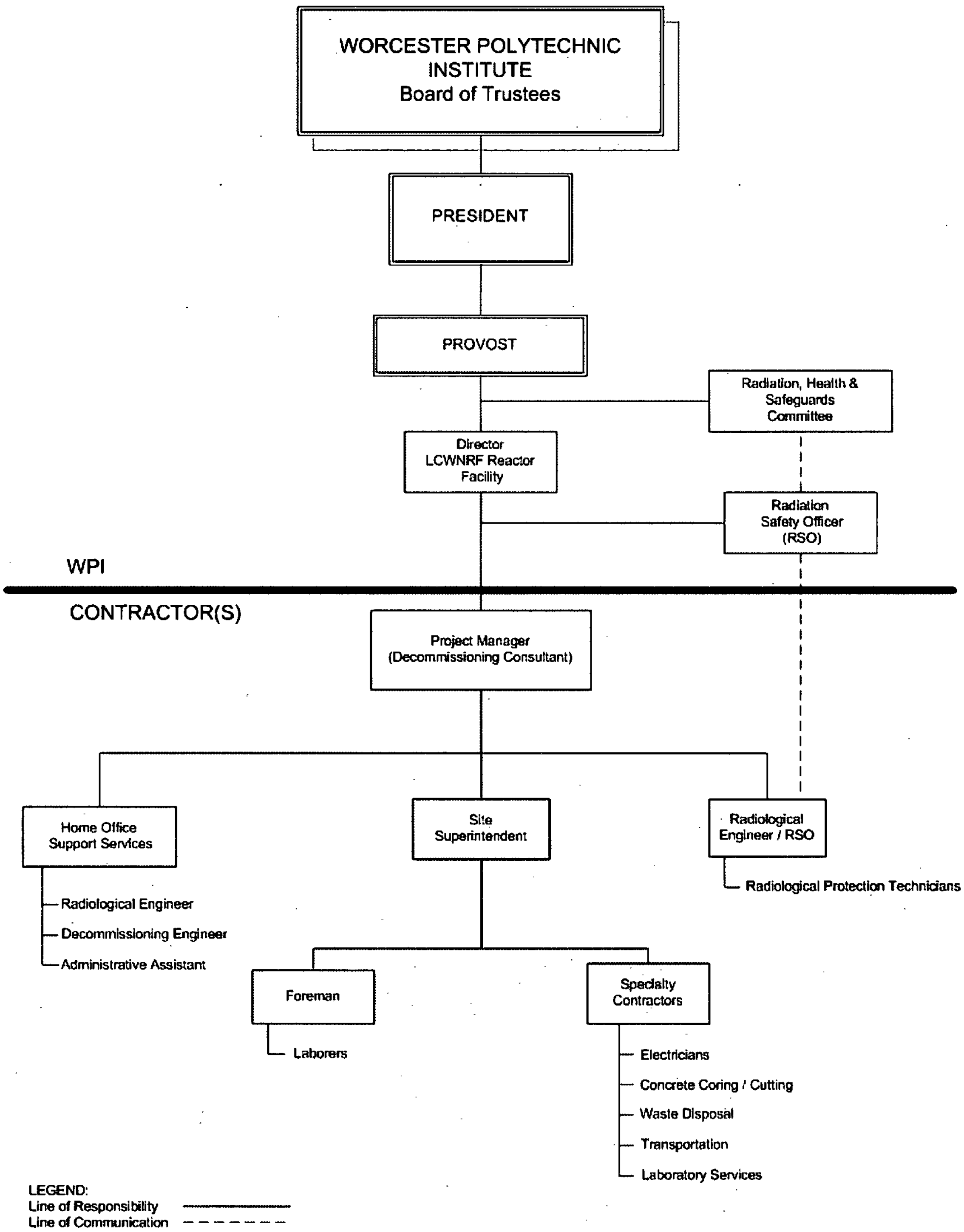
- Controlling and maintaining safety during decommissioning activities and protecting of the environment
- Reporting of performance
- Approving minor changes to the DP and procedures (which do not change the original intent and do not involve an un-reviewed safety question)
- Oversight and coordination of WPI functional groups, and contractors and consultants
- Ensuring that the conduct of decommissioning activities complies with applicable regulations and is in accordance with WPI licenses

The minimum qualifications for the WPI Reactor Director are:

- Currently holds (or previously held) position of Reactor Director
- Ten years of management experience in the field of regulatory compliance
- Familiarity with the WPI LCWNRF
- Appropriate training in radiation protection, nuclear safety, hazard communication, environmental protection and industrial safety

FIGURE 2.4

WPI-RDP DECOMMISSIONING ORGANIZATION



2.4.2 WPI Radiation Safety Officer

The WPI Radiation Safety Officer (RSO) shall be responsible for providing radiological support in the decommissioning of the WPI LCWNRF. This function ensures that the activities involving potential radiological exposure are conducted in compliance with the applicable licenses, federal and state regulations, and WPI standard operating procedures. The position includes responsibility for maintaining the surveillance and monitoring program and for HP radiological protection procedures.

The RSO will have oversight of all D&D operations. The scope of his oversight will include all D&D operations that involve work with systems or materials that have a radiological component. The minimum qualifications for this position are:

- An advanced degree in health physics or a related field
- Ten years operational experience related to radiation safety
- Currently holds (or previously held) position of RSO

The RSO is responsible for ensuring that:

- Radiological controls are in place prior to and during any work involving radiation
- Applicable license conditions are satisfied
- Applicable state and federal regulations are met

The RSO has the authority to:

- Implement any actions necessary to ensure that radiological controls are implemented and followed
- Immediately stop or modify radiological work determined to be unsafe

2.4.3 Decommissioning Consultant

WPI may elect to use a decommissioning consultant to assist WPI with the management of the decommissioning.

The consultant would provide project management and day-to-day oversight of the contractors, as well as providing WPI with technical expertise needed for the project. While more than one individual may provide assistance to WPI, the minimum qualifications for the lead consultant are:

- An advanced degree in health physics, radiological engineering or a related field
- 10 years supervisory experience related to radiation safety

- 10 years operational experience related to radiological decommissioning, including completion of a minimum of one NRC-licensed research reactor decommissioning project

2.4.4 Contractor Assistance

WPI management, with assistance from its Decommissioning Consultant, will select qualified contractors to perform the physical decommissioning of the LCWNRF. In addition to planning and implementing the physical decommissioning activities, the contractors (which may include the Decommissioning Consultant's organization) will provide required HP support, radiation surveys, and waste packaging, processing, and shipping.

Contractors will be selected after WPI's receipt from NRC of the license amendment authorizing decommissioning. The selection process will be based on a defined scope of work and selection criteria for contractor qualifications, experience, and reputation. The required contractor qualifications and experience shall include the following:

a. Demonstration of experience in the performance of the following tasks, as applicable to their roles and responsibilities:

   - Integration of decommissioning, dismantlement and demolition plans
   - Waste management and other methods used to minimize final waste disposal costs
   - Radiological decontamination and remediation of facilities and equipment
   - Use of survey equipment and protocols suitable for compliance with current NRC or MARSSIM survey criteria
   - Development and execution of radiation protection, environmental protection and industrial safety programs that will be used during the D&D
   - Selection, design and/or procurement of appropriate containers and packaging for radioactive and hazardous waste, and transportation to approved treatment and disposal facilities
   - Performing license termination surveys on a project of similar size and scope
   - Waste management (packaging, manifesting and transportation) of radioactive waste

b. The contractor selected for waste management must be able to provide a QA program that meets the requirements under Code of federal Regulations (CFR) 10 CFR 71, *Packaging and Transportation of Radioactive Material*, Subpart H, *Quality Assurance*.

c. The contractors will be required to provide qualified personnel and programs, including but not limited to the following areas of expertise, as applicable to their roles and responsibilities:

- Project management
- Health Physics
- Radiological engineering with MARSSIM survey experience
- Radioactive waste management
- Industrial and occupational safety
- Civil and mechanical engineering
- Quality assurance
- Construction supervision
- Project controls
- Decontamination and waste handling
- Radiological safety engineering
- Worker training and qualification

2.5 TRAINING PROGRAM

Individuals (employees, contractors and visitors) who require access to the work areas or a radiologically restricted area will receive training commensurate with the potential hazards to which they may be exposed.

Radiation protection training will be provided to personnel who will be performing remediation work in radiological areas or handling radioactive materials. The training will ensure that decommissioning project personnel have sufficient knowledge to perform work activities in accordance with the requirements of the radiation protection program and accomplish ALARA goals and objectives. The principle objective of the training program is to ensure personnel understand the responsibilities and the required techniques for safe handling of radioactive materials and for minimizing exposure to radiation.

Training records will be maintained which will include trainee's names, dates of training, type of training, test results, authorization for protective equipment use, and instructor's name. Radiation protection training provides the necessary information for workers to implement sound radiation protection practices. The following are examples of the training programs applicable to remediation activities.

2.5.1 General Site Training

WPI will be implement a general training program designed to provide orientation to project personnel and meet the requirements of 10 CFR Part 19, *Notices, Instructions and Reports to Workers: Inspection and Investigations* (Ref. 14). General Site Training (GST) will be required for all personnel assigned on a regular basis to the decommissioning project. GST will include:

- Project orientation/access control
- Introduction to radiation protection
- Quality assurance
- Industrial safety
- Emergency procedures

2.5.2 Radiation Worker Training

All individuals directly associated with the WPI-RDP will be required to undergo Radiation Worker Training (RWT), which will include the following topics:

- Fundamentals of radiation
- Biological effects of radiation
- External radiation exposure limits and controls
- Internal radiation limits and controls
- ALARA Program (program, objectives, investigational limits, keeping doses ALARA)
- Contamination limits and controls
- Management and control of radioactive waste
- Use of personal protective equipment

Personnel who have documented equivalent RWT from another site may be waived from taking training except for training on site specific requirements such as, WPI's administrative limits and emergency response actions. Personnel will be required to pass the written examination and practical exercises.

2.5.3 Respiratory Protection Training

Generation of airborne radioactivity and extensive loose surface contamination during the decommissioning activities is not expected, and use of respiratory protective devices is unlikely to be needed. However, if personnel whose work assignments do require the use of respiratory protection devices, they will receive respiratory protection training in the devices and techniques that they will be required to use. The training program will follow the requirements of:

- 10 CFR 20 Subpart H, *Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas* (Ref. 15)
- Regulatory Guide 8.15, *Acceptable Programs for Respiratory Protection* (Ref. 16)
- NUREG 0041, *Manual of Respiratory Protection Against Airborne Radioactive Materials* (Ref. 17)
- 29 CFR 1910.134, *Respiratory Protection* (Ref. 18)

Training would be provided by either the Decommissioning Consultant or a specialty contractor and would consist of a lecture session and a simulated work session. Personnel who have documented equivalent respiratory protection training may be waived from this training. Use of respiratory protective devices will require specific medical evaluation and respiratory device fit testing.

2.6 DECONTAMINATION AND DECOMMISSIONING DOCUMENTS AND GUIDES

Health physics, industrial safety criteria and other standards that guide the activities described in this DP are discussed in the following sections of this DP:

- Section 3.1 Radiation Protection
- Section 3.2 Radioactive Waste Management
- Section 3.3 General Industrial Safety Program
- Section 3.4 Radiological Accident Analyses

Relevant documents and guides used are noted therein.

2.7 FACILITY RELEASE CRITERIA

The proposed decommissioning alternative that has been presented in this DP does not necessitate the dismantlement of the building that houses the WPI reactor. The results of the historical site assessment and radiological surveys indicate that the building structures (exclusive of the reactor pool / biological shield structure) may be directly releasable, without the need for decontamination.

This section provides the specific criteria for unrestricted release of the WPI LCWNRF. The final release survey will use the Derived Concentration Guideline Levels (DCGL's) developed from the current NRC guidance for license termination in 10 CFR Part 20, Subpart E, *Radiological Criteria for License Termination, Standards of Protection Against Radiation* (Ref. 19).

10 CFR 20.1402, Subpart E, *Radiological Criteria for Unrestricted Use* (Ref. 12), allows termination of a license and release of a site for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a

total effective dose equivalent to an average member of a critical group that does not exceed 25 millirem (0.25 mSv) per year and the residual radioactivity has been reduced to levels that are as low as is reasonably achievable (ALARA).

The current NRC guidance for acceptable license termination screening values (meeting the 10 CFR 20.1402 criteria) for common radionuclides for building surface contamination and surface soil contamination are presented in NUREG-1757, Volume 1, Appendix B, *Consolidated NMSS Decommissioning Guidance, Decommissioning Process for Materials Licensees,* (Ref. 20). As such, an ALARA analysis is not needed. As stated in NUREG-1727, Appendix D, *NMSS Decommissioning Standard Review Plan* (Ref. 6), "in light of the conservatism in the building surface and surface soil generic screening levels developed by the NRC staff, the staff presumes, absent information to the contrary, that licensees or responsible parties that remediate building surfaces or soil to the generic screening levels do not need to demonstrate that these levels are ALARA."

Upon completion of the decontamination and remediation activities (e.g., see Section 2.3 Decommissioning Tasks), WPI will complete a FSS of the WPI reactor facility using the method described in NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (Ref. 11). To support the license termination request, the results of the survey(s) will be summarized in a report which will be submitted to NRC, as required by NUREG 1537 (Ref. 5).

If it is impractical or not possible to satisfy release criteria (or conclusively demonstrate that they have been met), the location/item will be treated as radioactively contaminated and dispositioned as low-level radioactive waste.

Based upon radiological surveys and historical site assessment, there is no known or suspected potential for surface soil contamination and release criteria for soil will likely not be needed. However, if small amounts of concrete with volumetric residual radioactivity need to be released, the volumetric soil release criteria may be used for that purpose.

The release criteria will be based upon use of the sum-of-fractions unity rule, based on relative concentrations of radionuclides on or in the material and their respective release criteria if more than one radionuclide is present. If additional screening values are required for nuclides not included in Table 2.2 or Table 2.3, they will be calculated using the NRC's COMPASS computer code (Ref. 21) for planning and assessing MARSSIM site surveys, with default values.

TABLE 2.2

LICENSE TERMINATION SCREENING VALUES FOR BUILDING SURFACE CONTAMINATION

RADIONUCLIDE	SYMBOL	ACCEPTABLE SCREENING LEVELS* FOR UNRESTRICTED RELEASE (dpm/100 cm²)**
Hydrogen-3 (Tritium)	^{3}H	1.2e+08
Carbon-14	^{14}C	3.7E+6
Sodium-22	^{22}Na	9.5E+03
Sulfur-35	^{35}S	1.3E+07
Chlorine-36	^{36}Cl	5.03+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

** 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 surface contamination is removable, and therefore the screening levels should be decreased by a factor of 10. Alternatively, users having site-specific data on the fraction of removable contamination, based on site-specific resuspension factors, (e.g., within 10 percent to 100 percent range) may calculate site-specific screening levels using COMPASS Version 2 (Ref. 21).*

*** Units are disintegrations per minute (dpm) per 100 square centimeters (dpm/100 cm²). One dpm is equivalent to 0.0167 Becquerel (Bq). Therefore, to convert to units of Bq/m² 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 / year) 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).*

TABLE 2.3

LICENSE TERMINATION SCREENING VALUES FOR SURFACE SOIL

RADIONUCLIDE*	SYMBOL	SURFACE SOIL SCREENING VALUES** FOR UNRESTRICTED RELEASE (pCi/g)***
Hydrogen-3 (Tritium)	^{3}H	1.1E+02
Carbon-14	^{14}C	1.2E+1
Sodium-22	^{22}Na	4.3E+00
Sulfur-35	^{35}S	2.7E+02
Chlorine-36	^{36}Cl	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+4
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	5.8E00
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	2.8E+4
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	^{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

(Note: Table 2.3 is continued on next page)

TABLE 2.3 (continued)

LICENSE TERMINATION SCREENING VALUES FOR SURFACE SOIL

RADIONUCLIDE*	SYMBOL	SURFACE SOIL SCREENING VALUES** FOR UNRESTRICTED RELEASE (pCi/g)***
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}Am	2.1E+00
Curium-242	^{241}Cm	1.6E+02
Curium-243	^{243}Cm	3.2E+00

** Plus Chain (+C) indicates a value for a radionuclide with its decay progeny present in equilibrium. The values care 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).*

*** These values represent surface soil concentrations of individual radionuclides that would be deemed in compliance with the 25 mrem/year (0.25 mSv/year) 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).*

**** Screening values are in units of (pCi/g) equivalent to 25 mrem/year (0.25 mSv/year). To convert from pCi/g to units of Becquerel per kilogram (Bq/kg) divide each value by 0.027. These values 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 the mean of the distribution of the assumed critical group. The metabolic parameters were set at "Standard Man" or at the mean of the distribution for an average man.*

3.0 OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 RADIATION PROTECTION

3.1.1 Ensuring As Low As Reasonably Achievable (ALARA) Radiation Exposures

Decommissioning activities at the WPI LCWNRF involving the use and handling of radioactive materials will be conducted in a manner such that radiation exposure will be maintained As Low As Reasonably Achievable (ALARA), taking into account the current state of technology and economics of improvements in relation to the benefits.

ALARA Program

A documented ALARA evaluation will be required for specific work evolutions if it is likely that 5% of the applicable dose limits (collective dose) for any of the following may be exceeded:

- Total Effective Dose Equivalent (TEDE) (5 rem)
- 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 (50 rem)
- Eye Dose Equivalent (EDE) (15 rem)
- Shallow Dose Equivalent (SDE) to the skin or any extremity (50 rem)

The ultimate responsibility for assuring radiation protection and maintaining exposures ALARA during decommissioning remain with the WPI Reactor Director and Radiation Safety Officer (RSO).

Methods for Occupational Exposure Reduction

Various methods will be utilized during the Decommissioning Project to ensure that occupational exposure to radioactive materials is kept ALARA. The methods include the use of Radiological Work Permits (RWPs), use of protective equipment, radioactive material handling techniques, and work practices as described in the following subsections. Work will be performed in accordance with NRC regulations and this Decommissioning Plan (DP), and implemented in accordance with approved written procedures.

Radiological Work Permits (RWPs)

A Radiation Work Permit (RWP) will be used for the administrative control of personnel entering or working in areas that have radiological hazards. Work techniques will be specified in such a manner that the exposure for all personnel, individually and collectively, are maintained ALARA. RWPs will not replace work procedures, but will supplement procedures. Radiation work practices will be considered when procedures are developed for work that will take place in a radiologically controlled area.

Project RWPs will describe the job to be performed, define protective clothing and equipment to be used, and personnel monitoring requirements. RWPs will also specify any special instructions or precautions pertinent to radiation hazards in the area. Such information will include listing the radiological hazards present, area dose rates and the presence and intensity of hot spots, loose and fixed surface radioactivity, and other hazards as appropriate. The Health Physics (HP) organization will ensure that radiation, surface radioactivity and airborne surveys are performed as required to define and document the radiological conditions for each work activity.

RWPs for work activities with low dose commitments may be approved by the HP technician or HP supervisory personnel. RWPs for work activities with potentially high dose commitment or significant radiological hazards will be approved by the RSO or Decommissioning Consultant. Use of RWPs will be governed by written and approved procedure. Examples of topics covered by RWP implementing procedures include:

- Requirements, classifications and scope for RWPs
- Initiating, preparing and using RWPs
- Extending expiration dates of an RWP
- Terminating RWPs

Respiratory Protection and TEDE ALARA Evaluations

WPI anticipates that dismantling activities for the more radioactive components can be performed by unbolting or unlatching, without the need for segmentation. However, if it is required, segmentation activities will be minimized and limited to mechanical cutting (e.g., sawing or shearing) which does not generate airborne dust, fumes or smoke. If there ever is a potential for generating airborne radioactivity, the use of engineering controls will be the first choice with respect to controlling the hazard. Engineering controls will be used to mitigate the airborne radiological hazard at the source. Such controls may include contaminant capture with HEPA-filtered vacuums.

While extremely unlikely, if circumstances were to exist where engineering controls are not practical or may not be sufficient to prevent airborne concentrations that constitute an airborne radioactivity area, and worker access is required to such areas, respiratory protective equipment will be utilized to limit internal exposures. Any situation wherein workers are allowed access to an airborne radioactivity area, or allowed to perform work that has a high degree of likelihood to generate airborne radioactivity exceeding a value of 0.1 Derived Air Concentration (DAC), the decision to allow access will be accompanied by the performance of representative measurements of airborne radioactivity to assess potential worker intake. The results of DAC-hour tracking and air sample results for intake will be documented in accordance with appropriate regulations and implementing procedures.

Control and Storage of Radioactive Materials

The WPI-RDP's HP Program will establish radioactive material controls that ensure:

- Deterrence of inadvertent release of licensed radioactive materials to unrestricted areas
- Confidence that personnel are not inadvertently exposed to licensed radioactive materials
- Minimization of the volume of radioactive wastes generated during the decommissioning

All material leaving the Restricted Area will be surveyed to ensure that radioactive material is not inadvertently released from the WPI LCWNRF. Refer to Section 3.1.3 Radioactive Material Controls, for a description of the specific survey methods that will be used.

3.1.2 Health Physics Program

General

WPI will use written, approved procedures to control the implementation of work during the WPI-RDP. All procedures will be developed and approved in accordance with written policy and procedure.

The WPI RSO, with the assistance of a Decommissioning Consultant, will have full authority to act in all aspects to ensure protection of workers and the public from the effects of radiation. Procedures for conduct of the decommissioning HP program will be will reviewed and approved according to WPI policy.

Audits, Inspections, and Management Review

Physical decommissioning work will be subject to continuous oversight by WPI's RSO or Decommissioning Consultant. Additionally, aspects of the project may be evaluated and reported by an applicable contractor's Quality Assurance program through audits, assessments and inspections of various aspects of decommissioning performance, including HP, as described in Section 1.2.5 Program Quality Assurance.

Audits of the decommissioning HP program will be conducted in accordance with the requirements of Code of Federal Regulations (CFR) 10 CFR 20 (Ref. 15).

Additional independent assessments or management reviews may be performed when deemed appropriate by the WPI Reactor Director.

Health Physics Equipment and Instrumentation

Project HP equipment and instrumentation will be suitable to permit ready detection and quantification of radiological hazards to workers and the public. They will be chosen so as to ensure the validity of measurements taken during remediation and final release surveys. The selection of equipment and instrumentation to be utilized will be based upon detailed knowledge of the radiological contaminants, and concentrations that are expected to exist based on scoping survey measurements or as known from process knowledge of the working history of the WPI LCWNRF. Equipment and instrumentation selection also takes into account the working conditions, contamination levels and source terms that are reasonably expected to be encountered during the performance of decommissioning work, as presented in this DP.

The following sections present examples of the typical equipment and instrumentation that may be used during the decommissioning. It is anticipated that the Decommissioning Consultant and contractors will be supplying equipment for their own use, and while capabilities will remain equivalent, specific manufacturers and model designations may vary from those examples listed herein.

Criteria for Selecting Survey and Monitoring Equipment

A sufficient inventory and variety of instrumentation will be maintained on site to facilitate effective measurement of radiological conditions, control worker exposure consistent with ALARA, and to evaluate the suitability of materials for release for unrestricted use. Instrumentation and equipment will be capable of measuring the range of dose rates and radioactivity concentrations expected to be encountered during the LCWNRF decontamination and decommissioning activities, including implementation of a FSS.

The project HP staff will select instrumentation that is sensitive to the minimum detection limits for the particular task being performed, but also with sufficient range to ensure that the full spectrum of anticipated conditions for a task or survey can be met by the instrumentation in use. Consumable sampling supplies (e.g., smears, air filters, etc.) will conform to manufacturer and/or regulatory recommendations to ensure that measurements meet desired sensitivity and are valid for the intended purpose.

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

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

Specific Health Physics Equipment and Instrumentation Use and Capabilities

Table 3.1 provides details of typical HP equipment and instrumentation that is planned for use in the WPI-RDP. This list is not intended to be all-inclusive or exclusive, and equivalent makes and models may be substituted based on availability.

Policy, Method, Frequency and Procedures

The existing framework of WPI's current HP Program will be used as the basis for constructing the WPI-RDP's HP Program. WPI's existing program will be augmented as necessary using plans and procedures provided by the Decommissioning Consultant and contractors.

Airborne Effluent Monitoring – During decommissioning activities where a temporary enclosure is in use with a HEPA exhaust system, the ventilation system exhaust points from the temporary barrier will be sampled continuously downstream of the HEPA filtration system. Airborne concentrations within the uncontrolled areas of the building will be maintained at less than environmental airborne concentration effluent limits.

Radiation Surveys – Radiation, airborne radioactivity and contamination surveys during decommissioning will be conducted in accordance with approved HP procedures. The purposes of these surveys will be to (1) protect the health and safety of workers, (2) protect the health and safety of the general public, and (3) demonstrate compliance with applicable license, federal and state requirements, as well as DP commitments. HP personnel will verify the validity of posted radiological warning signs during the conduct of these surveys. Surveys will be conducted in accordance with procedures utilizing survey instrumentation and equipment suitable for the nature and range of hazards anticipated. Equipment and instrumentation will be properly calibrated and, where applicable, operationally tested prior to use in accordance with procedural requirements. Routine surveys will be conducted at a specified frequency to ensure that contamination and radiation levels in unrestricted areas do not exceed license, federal, state or site limits. The HP staff will also perform surveys during decommissioning whenever work activities create a potential to impact radiological conditions.

TABLE 3.1

TYPICAL HEALTH PHYSICS EQUIPMENT AND INSTRUMENTATION

INSTRUMENT MODEL	DETECTOR TYPE	INSTRUMENT RANGE	APPLICATION
Bicron Micro-Rem meter	1"x1" Tissue equivalent scintillation	0-5000 Micro rem/hr	Low gamma exposure rates
Ludlum Model 12 count rate meter, with Ludlum 44-10	2"x2" NaI scintillation	0- 500K cpm	Low-level gamma scanning
Ludlum Model 12 count rate meter, with Ludlum 44-2	1"x1" NaI scintillation	0-500K cpm	Low-level gamma scanning interior of equipment and pipes
Ludlum Model 17	Ionization chamber	0-50 R/hr	Gamma and Beta exposure rates
Bicron labtech dual channel scaler / ratemeter	Alpha and beta scintillation, 100cm^2	0-1,000K cpm	Measurement of surface contamination
Ludlum Model 300 Portable Area Radiation Alarm	GM tube	0.1-1000 mr/hr	Portable area radiation alarm
Ludlum Model 177 Alarming Ratemeter	GM tube, pancake, 15 cm^2	0- 500K cpm	Personnel contamination frisking
Ludlum 2200 ratemeter-scaler, with Ludlum 44-1	Beta scintillation, ~12 cm^2	0-500K cpm, 6 sec. to 990 min.	Beta-counting smear and air samples
Ludlum 2200 ratemeter-scaler, with Ludlum 44-2	Alpha scintillation, ~12 cm^2	0-500K cpm, 6 sec. to 990 min.	Alpha-counting smear and air samples
Ludlum Model 12 count rate meter, with Ludlum 44-7	Thin end window GM tube	0-500K cpm	Beta-gamma contamination checks, interior equipment and pipes
Hi-Q Environmental Products TFIA	N/A	0-70 CFM	High-volume grab air samples
Apex Pro 200	N/A	0.8-4 LPM	Personnel breathing zone air sampler
F&J Model LV-14M Gooseneck "Lo-Vol"	N/A	0.35-3.5 cfm	Low volume air sampling for long-term continuous air sampling
Ludlum Model 333-2 air monitor	GM	10-10^3 cpm	Local continuous particulate airborne monitor with alarm capability

Personnel Monitoring – Internal and External – External monitoring will be conducted in accordance with approved procedures. Prospective external exposure evaluations will be performed prior to initiating decommissioning activities and whenever changes in conditions warrant. Visitors to the WPI LCWNRF will be monitored in accordance with requirements specified in WPI's HP procedures and according to the radiological hazards of areas to be entered.

Internal monitoring will be conducted in accordance with approved procedures. This prospective internal exposure evaluation will be performed on an annual basis, at a minimum, or whenever significant changes in planned work evolutions warrant additional evaluation. A comprehensive air-sampling program will be conducted during the WPI-RDP to evaluate worker exposures regardless of whether or not internal monitoring is specified. The results of this air-sampling program will be utilized to ensure validity of specified internal monitoring requirements for decommissioning personnel. If hazards are encountered, at any time during the decommissioning, that may not be readily detected by the preceding measures, appropriate special measures or bioassay will be instituted to ensure the adequate surveillance of worker internal exposure.

Monitoring will be required if the prospective dose evaluation shows that an individual's dose is likely to exceed 10% of the applicable limits, and for individuals entering a high or very high radiation area.

Respiratory Protection – If required, the WPI-RDP respiratory protection program will include direction for use of National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) certified equipment. This program will be reviewed and approved by the HP staff to ensure adherence to the requirements of 10 CFR 20 (Ref. 15).

NIOSH/MSHA-approved air purifying respirators include full-face-piece assemblies with air purifying elements to provide respiratory protection against hazardous vapors, gases, and/or particulate matter to individuals in airborne radioactive materials areas. The RSO or Decommissioning Consultant will also ensure that the respiratory protection program meets the requirements of 10 CFR Part 20, subpart H (Ref. 15).

> *Maintenance* – When respiratory protection equipment requires cleaning, the filter cartridges will be removed. The respirator will be cleaned and sanitized after use with a cleaner/sanitizer and then rinsed thoroughly in plain warm water in accordance with HP procedures.
>
> *Storage* – Respiratory protective equipment will be kept in proper working order. When any respirator shows evidence of excessive wear, distortion or has otherwise failed inspection, it will be repaired or replaced. Respiratory protective equipment that is not in use will be stored in a clean dry location.

Contamination Control – Contamination control measures that will be employed include, as appropriate, the following:

- Worker training will incorporate methods and techniques for the control of radioactive materials, and proper use and donning/doffing of protective clothing
- D&D procedures will incorporate control techniques to minimize spread of contamination during work

- Radiological surveillance surveys conducted by HP
- Containment devices (e.g., designed barriers, containers and plastic bags, etc.) will be used to prevent the spread of radioactive material
- Physical decontamination of areas or items to mitigate potential contamination migration
- Physical barriers such as Herculite sheeting, strippable paint, and tacky mat step-off pads to limit contamination spread
- Posting, physical area boundaries and barricades
- Step-off pads at the exit point from contaminated areas

Personnel entries into radiological contaminated areas will require the use of protective clothing. Depending upon the conditions outlined in the RWP, such clothing may consist of suitable combinations of the following items:

- Cloth or Tyvek lab coat
- Cloth or Tyvek coveralls
- Cloth or Tyvek hoods or skull caps
- Plastic, cloth or Tyvek calf-high booties
- Rubber, plastic or cloth shoe covers
- Plastic or rubber gloves, which may require cloth liners
- Additional layers of disposable Tyvek coveralls or plastic rain suits
- Face shields, safety glasses, hard hats, safety shoes, ear muffs or other industrial protective devices

Access Control – A Restricted Area will be established and properly posted and monitored to prevent unauthorized access.

Engineered Controls – Personnel exposure to airborne radioactive materials will be minimized by utilizing engineering controls such as the following:

- Ventilation devices – in-place or portable HEPA-filtered ventilation units or vacuum cleaners
- Containment devices – designed containment barriers, containers, plastic bags, tents, and glove-bags
- Airborne source term reduction – application of fixatives prior to handling, misting of surfaces to minimize dust and resuspension

Airborne Radioactivity Monitoring – Air sampling will be performed in areas where airborne radioactivity is present or likely. 10 CFR 20.1502(b) requires monitoring for the intake of radioactive material if the intake during the year is likely to exceed:

- Ten percent of the annual limit on intake (ALI) for an adult worker, or
- The committed effective dose equivalent 0.1 rem (1.0 mSv) for an occupationally exposed minor or declared pregnant woman

WPI will establish monitoring requirements for the project based on prospective estimates of worker intakes and air concentrations. Such estimates will consider the following:

- The quantity of material(s) based upon its physical form and use
- The annual limit on intake for the nuclides of interest
- The release fraction for the radioactive material(s) being handled
- Other factors that may be applicable

HP personnel will use technical judgment in determining the situations that necessitate air sampling regardless of results of generalized, prospective evaluations performed for the WPI-RDP.

The purpose of the radiological air sample will be identified prior to specifying a location for an air sampler. The following are a few examples of the variety of reasons that exist for collecting air samples:

- Estimation of worker intakes
- Verification of confinement of radioactive materials
- Early warning of abnormal airborne concentrations of radioactive materials
- Determining the existence of criteria for posting an Airborne Radioactivity Area in accordance with 10 CFR 20

Smoke tubes and buoyant markers (ribbons) may be used to determine airflow patterns in the area.. Airflow patterns may be reevaluated if there are changes at the WPI LCWNRF that may impact the validity of the sampling locations. Such factors might include the following:

- Changes in the work process
- Changes in the ventilation system
- Use of portable ventilation that might alter earlier assessments

After identifying the purpose for the air sample and establishing flow patterns, air sample locations will be chosen as follows:

- For verification of confinement of radioactive materials:

  - Locate samplers in the airflow near the actual or potential release point
  - More than one sampling point may be appropriate when there are more than one potential or actual release points

- For estimation of a worker's intake:
  - Sampler intakes will be located as close as practical to the worker's breathing zone, without interfering with the work or worker

General workplace air sampler intakes will not be placed in or near ventilation exhaust ducts unless their purpose is to detect system leakage during normal operation, and if quantitative measurements of workplace concentrations are not required. Locations or number of air samplers will be changed when dictated by modifications to facility structure, changes in work processes, or elimination of potential sources.

WPI will maintain a sufficient inventory and variety of operable and calibrated portable and semi-portable air sampling equipment to allow for effective collection, evaluation, and control of airborne radioactive material and to provide backup capability for inoperable equipment. Air sampling equipment will be calibrated at prescribed intervals or prior to use: calibration will be against certified equipment having known valid relationships to nationally recognized standards. Table 3.1 provides a listing of anticipated air-sampling equipment.

When the work being performed is a continuous process, a continuous sample with up to a weekly exchange frequency is appropriate. Longer sample exchange frequencies may be approved by HP management for situations where airborne radioactive material and nuisance dust are expected to be relatively low. Grab sampling for continuous processes may also be approved by HP management based upon consideration of variability of the expected source term for the facility and process. Grab sampling is the appropriate means of airborne sampling for processes conducted intermittently, and for short duration radiological work that involves a potential for airborne release.

<u>Potential Sources of Radiation or Contamination Exposure</u>

Potential sources of radiation or contamination exposure to workers and public, as a result of decommissioning activities, may be assessed by several methods, including:

- Process and historical knowledge
- Surveys performed during characterization
- Pre-work RWP surveys
- Previous and current job coverage surveys

- Daily, weekly and monthly routine surveillance surveys

Classification of potential sources may also be identified by radionuclide, physical properties, volatility and radioactivity.

Worker exposure to significant external deep-dose radiation fields is considered unlikely during this project, due to the nature and extent of the contaminants present and/or the work precautions and techniques employed. Worker exposure to airborne radioactivity is also unlikely to occur during most dismantling operations/work evolutions unless they involve use of abrasive or thermal cutting methods on activated materials: such cutting methods can volatilize or create loose dispersible contamination. WPI does not plan to use those types of cutting methods on activated materials.

Exposure of the public to external or internal radiation from the WPI-RDP is not considered credible because of the limited levels and extent of radioactivity present and the confinement provided by the facility.

Control of potential sources of radiation exposure to workers and public, as a result of decommissioning activities, will be achieved through, but not limited to, the use of administrative, engineering and physical controls. Administrative controls consist of, but are not limited to:

- Administrative dose limits that are lower than regulatory limits
- Training
- Radiological surveys

In general, physical barriers (such as lockable doors/gates, and, or other applicable barriers) may be used in combination with radiological postings to preclude inadvertent exposure of the general public to radiation or radioactive materials.

<u>Environmental Monitoring</u>

During the conduct of decommissioning activities, measurements will be made for radiation and radioactive contamination at locations where direct radiation exposure or contaminant release pathways could exist that could credibly impact the environment or the general public. While releases of radioactive material or increased radiation levels in uncontrolled areas are not expected during decommissioning activities, monitoring will be performed to verify that expectation.

Generally, environmental monitoring will entail placement of dosimeters at selected locations to monitor for direct radiation exposure. Such locations will include nearby classrooms, offices or dormitories adjacent to the LCWNRF frequently occupied by WPI students, staff or visitors. Periodic measurements will be made for radioactive surface contamination at locations where radioactive contamination could migrate out of the facility (e.g., at the LCWNRF's doorway exits and outward air flow paths). Continuous particulate air sampling will be made at selected locations representative of nearby adjacent uncontrolled areas.

Health Physics Policies for Non-WPI Personnel

WPI intends to use non-WPI personnel (contractors) to assist in the decommissioning. Contractors who will work with licensed radioactive materials will be required to:

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

3.1.3 Radioactive Materials Controls

The WPI-RDP radiation protection program will establish radioactive material controls that ensure the following:

- Prevention of inadvertent decommissioning radioactive waste (licensed) material release to uncontrolled areas
- Assurance that personnel are not inadvertently exposed to radiation from licensed radioactive decommissioning waste materials
- Minimization of the amount of radioactive waste material generated during decommissioning.

Pool water releases will be processed using existing pool water treatment equipment and analysis protocols to ensure that discharges to sanitary sewerage will meet the requirements of 10 CFR 20.2003 (Ref. 22).

The potentially contaminated equipment, materials, instrumentation, and tools that are used or encountered during the decommissioning will be handled as described below:

- Items may be surveyed and released on site as clean if the residual radioactivity is less than the values specified in Table 1 of NRC Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*, June 1974 (Ref. 23)
- Items may be shipped directly for disposal as radioactive waste
- Items may be shipped to a licensed radioactive material processing facility for survey and release, decontamination followed by survey and release, or shipment for disposal as radioactive waste

- Items may be shipped to a licensed facility for holding until they are utilized on another project involving radioactive materials
- Items may be transferred to WPI's Agreement-State radioactive materials license

3.1.4 Dose Estimates

The total projected occupational exposure to complete the decommissioning of the Reactor facility is conservatively estimated to be less than 0.5 person-rem.

This estimate is provided for planning purposes only. Detailed exposure estimates and exposure controls shall be developed during detailed planning of the decommissioning activities. Area dose rates used for this estimate are based on 2008 scoping survey data, process knowledge and current survey maps (where available).

The dose estimate to members of the public as a result of decommissioning activities is estimated to be negligible: site perimeter controls will restrict members of the public from entering the area where decommissioning activities are taking place. This is consistent with the estimate given for the "reference research reactor" in NUREG-0586, *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities* (Ref. 24). In NUREG-0586, the dose to the public during decommissioning (DECON) and truck transport transportation of radioactive waste from the reference research reactor is estimated to be "negligible (i.e., less than 0.1 person-rem)."

3.2 RADIOACTIVE WASTE MANAGEMENT

3.2.1 Radioactive Waste Processing

The processes of decontamination and dismantlement of the WPI LCWNRF will result in solid low-level radioactive waste, consisting principally of activated metals, and a small amount of contaminated metals and concrete debris. No mixed or hazardous waste is expected to be generated. Lead paint coatings and asbestos-containing materials are likely to be present in the facility, but are not anticipated to be significantly present on materials likely to become radioactive waste. No soil remediation is anticipated which would result in solid radioactive waste. Solid, low-level radioactive waste will be handled (processed and packaged), stored and disposed of in accordance with applicable sections of the Code of Federal Regulations (CFR), disposal site Waste Acceptance Criteria, Massachusetts Department of Environmental Quality requirements, WPI Licenses and Permits, and the applicable implementing plans and procedures. Radioactive waste processing includes waste minimization or volume reduction, radioactive and hazardous waste segregation, waste characterization, neutralization, stabilization, solidification and packaging.

3.2.2 Radioactive Waste Disposal

Low-level radioactive waste will be processed and packaged for disposal at a licensed low-level waste site such as the Energy*Solutions*, LLC (formerly Envirocare of Utah) site. The volume of low-level radioactive waste is estimated at 600 cubic feet (packaged volume). All wastes resulting from and dismantling and decontamination are estimated to have concentrations of radioactive materials well below 10 CFR 61 Class A waste limits. The most highly radioactive item of waste arising from the decontamination and dismantling of structures, systems and components will be the activated regulating control blade. The blade is estimated to contain a total of 56 mCi, representing the majority of the waste radioactivity to be generated during the decommissioning program. Table 3.2 provides the estimated concentrations of radioactivity in the regulating control blade.

TABLE 3.2

ESTIMATED CONCENTRATIONS OF RADIOACTIVITY IN THE REGULATING CONTROL BLADE (as of December 2008)

RADIONUCLIDE	CONCENTRATION, Ci/Cubic Meter
Mn-54	0.61
Fe-55	43.22
Ni-59	0.05
Co-60	16.91
Ni-63	2.69

All other wastes materials are expected to have radioactive concentrations a small fraction of those listed for the regulating blade. The one Curie Pu-Be source, which would exceed 10 CFR 61 Class A limits, will either be returned to DOE or a source manufacturer, or transferred to WPI's agreement state license until a suitable disposal option becomes available.

10 CFR 61, *Licensing Requirements for Land Disposal of Radioactive Waste, Subpart D – Technical Requirements for Land Disposal Facilitie*s (Ref. 25), establishes minimum radioactive waste classification, characterization and labeling requirements. WPI's adherence to those requirements will be ensured through the implementation of project packaging and characterization procedures, Disposal Site Waste Acceptance Criteria for the selected disposal site(s) and the Project-Specific Quality Assurance Plan. Training/Qualifications will be provided for project waste management personnel to assure conformance to applicable 10 CFR 61 requirements (Ref. 25) as stated in the specific implementing procedures and plans. Audits and surveillances will be conducted per the Project-Specific Quality Assurance Plan based on American Society of Mechanical Engineers (ASME) NQA-1 (Ref. 26) and the requirements of 10 CFR 71 (Ref. 9).

10 CFR 71, *Packaging and Transportation of Radioactive Material*, establishes requirements for packaging, shipment preparation and transportation of licensed material. WPI is licensed by the NRC/ State of Massachusetts to receive, possess, use and transfer licensed byproduct and source materials. 10 CFR 71 requirements will be met through the implementation of the WPI-RDP approved packaging and shipping procedures. Training will be provided to assure conformance to applicable 10 CFR 71 requirements. Quality Assurance for waste packaging and transportation will conform to 10 CFR 71 Subpart H (Quality Assurance) requirements through the implementation of a WPI-RDP project-specific quality assurance plan.

10 CFR 20.2006, *Transfer for Disposal and Manifests*, establishes requirements for controlling transfers of low-level radioactive waste intended for disposal at a land disposal facility; establishes a manifest tracking system; supplements requirements concerning transfers and record keeping; and requires generator certification that transported materials are properly classified, described, packaged, marked and labeled, and are in proper condition for transport. These requirements will be met through the implementation of WPI-RDP packaging and shipping procedures with the oversight of WPI's RHSC and Decommissioning Consultant.

Radiological wastes will be disposed of at disposal sites per the applicable disposal site's Waste Acceptance Criteria. Associated implementing plans and procedures will reflect the characterization, processing, removal of prohibited items, packaging and transportation requirements. Appropriate documentation will be submitted to designated disposal sites including, as required, certification plans, qualification statements, assessments, waste stream analysis, evaluations and profiles, transportation plans, and waste stream volume forecasts. Waste characterization, waste designation, waste traceability, waste segregation, waste packaging, waste minimization, and quality assurance and training requirements of the designated disposal sites will be incorporated in implementing procedures to assure conformance to disposal site requirements.

Generator State (Massachusetts) and Treatment/Storage/Disposal Facility State (Utah) requirements for radioactive waste management will be incorporated into plans and procedures to assure conformance with applicable state regulations, licenses and permits. Applicable state regulations include Massachusetts Department of Environmental Quality requirements and Utah Department of Environmental Quality rules (R313) for the control of ionizing radiation reflected in Envirocare's Utah Radioactive Material License, UT 2300249.

Radioactive waste will be staged in designated controlled areas at the LCWNRF in accordance with the requirements of 10 CFR Part 19 (Ref. 14) and 10 CFR Part 20 (Ref. 19). Measures will be implemented through plans and procedures to prevent the spread of contamination, limit radiation levels, prevent unauthorized access, prevent unauthorized material removal, prevent tampering, and prevent weather damage. The designated areas will be controlled by Radiological Work Permits (RWP).

Radioactive waste material will be packaged for shipment per 10 CFR 61 (Ref. 25) and 49 CFR (Ref. 27), and the designated disposal site's Waste Acceptance Criteria and held in interim storage (staged) until shipped. Due to the relatively small amount of waste to be generated during the WPI-RDP, it is expected that only one truck shipment will be needed. As such, the waste will be staged on-site until all the waste products have been generated. Radioactive material storage areas will be contained inside posted restricted areas according to WPI-RDP procedures and consistent with 10 CFR 20.

3.3 GENERAL INDUSTRIAL SAFETY PROGRAM

WPI, along with its contractors, shall be responsible for ensuring that the WPI-RDP complies with all applicable occupational health and safety requirements. The primary functional responsibility is to ensure compliance with the Federal Occupational Safety and Health Acts (OSHA) of 1973 (Ref. 28). Specific responsibilities include:

- Conducting an industrial training program to instruct employees in general safe work practices
- Reviewing decommissioning project procedures to verify adequate coverage of industrial safety and industrial hygiene concerns and requirements
- Performing periodic inspections of work areas and activities to identify and correct any unsafe conditions and work practices
- Providing industrial hygiene services as required
- Advising Project Management personnel on industrial safety matters and on the results of periodic safety inspections

All personnel working on the WPI-RDP will receive Health and Safety training in order to recognize and understand the potential risks involving personnel health and safety associated with the work at the WPI LCWNRF. The Health and Safety training implemented on the WPI-RDP is to ensure compliance with the requirements of NRC (10 CFR), the EPA (40 CFR), and OSHA (29 CFR). Workers and regular visitors will be familiarized with plans, procedures and operation of equipment to conduct themselves safely. In addition, each worker must be familiar with procedures that provide for good quality control. Section 2.5 Training Program, provides additional information.

3.4 RADIOLOGICAL ACCIDENT ANALYSES

The inventory of radioactive material expected to be present at the start of decommissioning is significantly smaller than that postulated for the accident scenarios applicable for when the reactor was operational. The reactor fuel will have been removed prior to the start of decommissioning, and the majority of the residual radioactivity (less than 60 mCi) is in the form of activated metals which are not readily dispersible. As such, the existing operational accident analysis continues to bound and by far exceeds the consequences of any accident that could credibly occur during the decommissioning period. As such, a new accident analysis is not required.

4.0 PROPOSED FINAL RADIATION SURVEY PLAN

Based upon the results of an historical site assessment and scoping surveys of the Leslie C. Wilbur Nuclear Reactor Facility (LCWNRF), Worcester Polytechnic Institute (WPI) intends, to the extent practicable, to completely remove equipment and items that had become neutron activated or that contain radioactive contamination. Such removed materials will be disposed as radioactive waste. WPI anticipates that most – if not all – of the building structures will be free of residual radioactivity, and will not require decontamination.

The Final Status Survey (FSS) Plan (and subsequent FSS Report) discussed in this section deals with release of the WPI reactor building structure and surrounding environs for unrestricted use.

4.1 DESCRIPTION OF FINAL STATUS SURVEY PLAN

The purpose of the FSS is to demonstrate that the radiological condition of the LCWNRF structures is at or below established release criteria (see Section 2.7). It is anticipated that NRC will then terminate the R-61 reactor license, thereby allowing unrestricted use of all areas of the former LCWNRF.

WPI will develop its FSS Plan using the guidance provided in NUREG-1757, Volume 2, *Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria,* (Ref. 29) and NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM,* Ref. 11).

A FSS Plan will be developed after decommissioning contracts are awarded. The basis for the design of the FSS is MARSSIM. The overall goal of the FSS design is to ensure surveys are planned and conducted in such a manner that ensures the proper decision is made as to whether or not to accept or reject the Null Hypothesis (see further discussion on Null Hypothesis later in this section). The COMPASS computer code (Ref. 21) will also be used as part of the survey planning and survey assessment process.

The major inputs into the FSS planning process are:

- The development of Data Quality Objectives (DQOs) (see Section 4.1)
- The designation of survey units within each area classification (see Section 4.4.2)
- The review of contaminants and establishment of Derived Concentration Guideline Levels (DCGL) (see Sections 2.7 and 4.3)
- The selection of appropriate survey instrumentation (see Section 3.1)

The survey plan will serve as the guidance document for development of the survey package instructions used during implementation of the FSS. The facility will be sectioned into survey units according to guidance provided in MARSSIM. Plots, diagrams, and facility layout drawings will be developed to illustrate the classification of the survey units. In addition to a FSS Plan, survey work packages will be developed for each survey unit. Each survey package will include survey-unit-specific instructions, describe the survey unit size, grid spacing, and scan area prescribed, and specify a prescribed number of static measurements (including the location and spacing).

As the survey progresses, reevaluation of the survey plan may be necessary based on newly acquired survey data. If a condition is discovered that is not encompassed by the survey plan, the survey plan may undergo revision to address the condition. The condition with the revised survey plan will be fully disclosed and provided to the Decommissioning Consultant and RSO for review and concurrence prior to further performance of the FSS as it applies to the revised information.

To prevent recontamination of the clean areas, WPI will institute administrative and physical access controls of surveyed areas deemed radiologically clean. Administrative control of surveyed areas will be accomplished by written instruction contained in the final survey plan and by training of project personnel. Physical control of surveyed areas may be accomplished by placing rope barriers, locking doors where possible, etc. and placing signs to notify personnel regarding the condition of an area.

The FSS Plan will contain the criteria used to assess all final survey data including the statistical tests performed. The Plan will also state conclusions to be drawn based upon statistical test results.

The FSS Plan will be developed according to the guidance provided in MARSSIM and based upon the following assumptions:

- The interior rooms of the LCWNRF are impacted (determined from the results of operational history, scoping survey data and professional judgment).
- The screening activity levels for building surfaces are based on the assumption that the fraction of removable surface contamination is equal to 0.1. The fraction of activity that is removable will be verified. If the removable fraction exceeds 0.1 then more decontamination will be performed or site-specific DCGL values will need to be applied.
- Decision Errors – there are two types of decision errors applied to analytical results: Type I (α) and Type II (β) errors. A Type I error, or false positive, is the probability that Null Hypothesis is rejected when it should be accepted. A Type II error, or false negative, is the probability of determining that Null

Hypothesis is accepted when it should be rejected. The probability of making decision errors can be controlled by adopting an approach called hypothesis testing. The Null Hypothesis is treated like a baseline condition and is defined by MARSSIM as:

Null Hypothesis (H_o) = residual radioactivity in the survey unit exceeds the release criterion.

This means that the site or survey area is assumed contaminated until proven otherwise. For the purpose of this final survey, Type I or α error will be set at 0.05 (or 5 percent) and Type II or β error will be set at 0.05 (or 5 percent).

- For the LCWNRF it is likely that WPI will not need to evaluate survey data relative to a reference background area. However, if there were a need to evaluate survey data relative to a reference background area, WPI would utilize either the Sign test or the Wilcoxon Rank Sum (WRS) test to determine if any residual contamination in a survey unit exceeds the DCGL. The selection process for determining which test method to use will be documented.
- Any background count rate information required for input into the survey planning process will be obtained prior to performing the FSS. The standard deviation in the background count rate will be calculated based on the data obtained. With the COMPASS code used to automatically calculates the standard deviation. The background information is stored as part of the general site information and not on a survey unit specific basic.
- Once the final survey has been performed, survey data will be converted to DCGL units and compared to the DCGLs. Individual measurements and sample concentrations will be compared to DCGL levels for evidence of small areas of elevated activity. Data will then be evaluated using the appropriate statistical test method (i.e., Sign test or WRS test) to determine if they exceed the release criterion.
- If the release criterion has been exceeded (i.e., Null Hypothesis is proven true) WPI management, with assistance from the Decommissioning Consultant, will determine appropriate further actions. If all data points are less than the DCGL levels, no statistical test need be performed.

4.1.1 Review and Approval of Final Status Survey Plan

WPI is utilizing Method 1 from NUREG-1757, Volume 2, *Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria* (Ref. 29), to submit information to NRC on facility radiation surveys. WPI has submitted information to NRC on release criteria, characterization surveys, and operational surveys as part of this DP. In addition, WPI has committed to using the MARSSIM approach in developing the final radiological survey. Once the design of the final radiological survey for the site has been completed, WPI will submit information to NRC on FSS Design.

The FSS Plan will include sufficient information to allow NRC to determine that the FSS Design is adequate to demonstrate compliance with the radiological criteria for license termination. The information will include:

- A brief overview describing the FSS Design.
- A description and map or drawing of impacted areas of the site, area, or building classified by residual radioactivity levels (Class 1, Class 2, or Class 3) and divided into survey units, with an explanation of the basis for division into survey units. Maps will indicate compass orientation.
- A description of the background reference areas and materials, if they will be used, and a justification for their selection.
- A summary of the statistical tests that will be used to evaluate the survey results, including the elevated measurement comparison, if Class 1 survey units are present, a justification for any test methods not included in MARSSIM, and the values for the decision errors (and) with a justification for values greater than 0.05.
- A description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide.
- For in-situ sample measurements made by field instruments, a description of the instruments, calibration, operational checks, sensitivity, and sampling methods, with a demonstration that the instruments, and methods, have adequate sensitivity.
- A description of the analytical instruments for measuring samples in the laboratory, including the calibration, sensitivity, and methodology for evaluation, with a demonstration that the instruments and methods have adequate sensitivity.
- A description of how the samples to be analyzed in the laboratory will be collected, controlled, and handled.
- A description of the FSS investigation levels and how they were determined.
- A summary of any significant additional residual radioactivity that was not accounted for during site characterization.
- A summary of direct measurement results and/or concentration levels in units that are comparable to the DCGL and, if data is used to estimate or update the survey unit.
- A summary of the direct measurements or sample data used to both evaluate the success of remediation and to estimate the survey unit variance.

4.1.2 Means for Ensuring that all Equipment, Systems, Structures and Site are Included in the Survey Plan

Any items left in the LCWNRF after remediation will be radiologically surveyed prior to removal to ensure that radioactive (i.e., licensed) materials are not inadvertently removed from the facility (see Section 3.1.3). When it is impractical or not possible to decontaminate an item such that it exhibits no discernable facility-related activity when surveyed following methods presented in Section 3.1.3, the item will be treated as radioactive waste. Items that exhibit no discernible facility-related activity will be disposed of as uncontaminated waste. The systematic approach to the WPI LCWNRF Reactor Decommissioning Project (WPI-RDP) will ensure that every item or structural component in the LCWNRF is specifically evaluated for release before beginning the FSS. The FSS will group the LCWNRF into three classes (as suggested in MARSSIM) to ensure adequate survey coverage in support of a license termination request and subsequent release of the property for unrestricted use.

4.1.3 Means for Ensuring that Sufficient Data is Included to Achieve Statistical Goals

WPI will develop the WPI-RDP FSS Plan using the guidance presented in NUREG-1757, Volume 2, Rev. 1, *Consolidated Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria* (Ref. 29) and NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM,* Ref. 11). Using such regulatory guidance will ensure that NRC-recommended statistical goals are satisfied.

4.2 BACKGROUND SURVEY RESULTS

The FSS Guideline values for residual activity are taken to be levels above the naturally occurring background radiation. However, if the FSS results are significantly below the release guideline levels, the regulations allow the licensee to opt not to use background subtraction. WPI has opted to not subtract background for direct surface contamination measurements due to naturally occurring radioactive materials that may be present in construction materials. However, WPI will account for background when gamma exposure rate measurements are made.

4.3 FINAL RELEASE CRITERIA – RESIDUAL RADIATION AND CONTAMINATION LEVELS

The criteria for release of the LCWNRF for unrestricted use, after completion of the decommissioning activities described in this DP, are presented in Section 2.7. In summary, they are:

a. The generic screening values provided in Tables H.1 and H.2 of Volume 2 in NUREG 1757 (Ref. 29), and

b. A limit specified in 10 CFR 20.1402, *Radiological Criteria for Unrestricted Use* (Ref. 12)

4.4 MEASUREMENTS FOR DEMONSTRATING COMPLIANCE WITH RELEASE CRITERIA

4.4.1 Instrumentation – Type, Specifications and Operating Conditions

Instrumentation utilized during the Final Release Survey (and equipment and materials survey) will be selected based upon the need to ensure that site residual radiation will not exceed the release criteria. In order to achieve this goal, WPI will select instrumentation that is sensitive to the isotopes of concern and that is capable of measuring levels below the guideline values for those isotopes. Instrumentation available for the FSS and its respective detection range capability is presented in Table 3.1 of this DP. Instrumentation sensitivities will be determined following the guidance of NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions* (Ref. 30) using nominal literature values for background, response and site conditions. Refinements to these detection sensitivity estimates will be made, as necessary, on the basis of actual instrument response and background data gathered during site survey activities. Instrumentation used in the surveys will be calibrated against sources and standards that are traceable to National Institute of Standards and Technology (NIST), and representative of the isotopes encountered at the LCWNRF. When used, instruments will be operationally tested daily, or prior to each use, whichever is less frequent. Instruments will not be used in conditions that are not in conformance with manufacturer recommendations.

4.4.2 Measurement Methodology for Conduct of Surveys

This DP has been developed under the presumption that the LCWNRF will have been decontaminated to the extent practicable prior to the FSS. The WPI-RDP FSS Plan will include several steps to calculate the number of measurements and samples required, according to MARSSIM guidance, to release the site without restrictions. These steps include:

- Classify survey units
- Specify the decision error
- Determine the DCGLs
- Calculate the relative shift (COMPASS, Ref. 21, may be utilized)
- Obtain the number of samples per survey unit (COMPASS, Ref. 21, may be utilized)
- Estimate the sample grid spacing
- Perform evaluation for small areas with elevated radioactivity

- Determine if the number of samples is reasonable

Classify Survey Units

The impacted areas of the LCWNRF will be divided into three classes of survey units. Class 1 is an area with the highest potential for contamination. Class 2 is an area that has a low potential for delivering a dose above the release criteria and has little or no potential for containing small areas of elevated activity. Class 3 is an area with the lowest potential for contamination.

The FSS Plan and survey work packages for each survey unit will include a discussion regarding relevant historical, characterization and in-process survey information and evaluations, to support each survey units' classification.

Specify the Decision Error

There are two types of decision error (applied here to analytical results): Type I (alpha) and Type II (beta). A Type I error is described as the probability of determining that a result is above a criterion when it actually is not (false positive). A Type II error is described as the probability of determining that a result is below a criterion when it actually is above it (false negative). Both types of error will be set at 0.05 (5%).

Determine the DCGL

MARSSIM defines the derived concentration guideline level (DCGL) as the radionuclide-specific concentration within a survey unit corresponding to the release criterion. The radionuclides known or potentially existing at the LCWNRF include ^{60}Co and ^{55}Fe, the presumed predominant radionuclides, and possibly traces of ^{3}H, ^{14}C, ^{54}Mn, ^{63}Ni, ^{152}Eu, ^{154}Eu, and ^{155}Eu. The DCGL values are discussed in Section 2.7.

As stated in NUREG-1757 (Ref. 29), "in light of the conservatism in the building surface and surface soil generic screening levels developed by the NRC staff, the staff presumes, absent information to the contrary, that licensees or responsible parties that remediate building surfaces or soil to the generic screening levels do not need to demonstrate that these levels are ALARA."

Gross activity DCGLs and $DCGL_{EMC}$ (elevated measurement comparison DCGLs) will be developed during the planning stage for the FSS. Such planning will include consideration of "hard-to-detect" radionuclides for surfaces. The site-specific DCGLs will be calculated based on the relative fraction of each radionuclide in the expected radionuclide mix. Once developed, the values will be included in the FSS plan.

Calculate the Relative Shift

The relative shift is defined as Δ/σ where Δ is the DCGL – LBGR (Lower Bound of the Gray Region) and σ is the standard deviation of the contaminant distribution. In order to calculate the relative shift, the DCGL must be determined and two assumptions must be made to estimate the LBGR and the standard deviation of the measurement distribution. MARSSIM suggests that the LBGR be set at 50% of the DCGL but can be adjusted later to provide a value for the relative shift between the range of 1 to 3. The standard deviation may be calculated from preliminary survey data, prior surveys of similar areas and materials or the standard deviation of a reference background area.

It should be noted that σ represents the standard deviation prior to release after all area decontamination is thought to be complete. If no reference data is available to make a reasonable estimate of the background standard deviation, MARSSIM suggests using 30% of the mean survey unit background. For the LCWNRF, data from the facility characterization or from post-remediation surveys will be used to calculate the standard deviation value for each survey unit. The value for LBGR is input into COMPASS, and the relative shift is automatically calculated and reported.

Obtain the Number of Samples per Survey Unit

Once the relative shift is determined the calculated value, Δ/σ, will be used to obtain the minimum number of measurements or samples necessary to reject the Null Hypothesis based upon the initial assumptions and justify that the survey unit meets the requirements for release for unrestricted use. MARSSIM Table 5-3 contains the number of samples or measurements necessary for the given decision errors (α and β), and the calculated relative shift (Δ/σ), when dealing with non-radionuclide-specific measurements or when the radionuclide is present in the background. The value N/2 from MARSSIM Table 5-3 represents the number of samples or measurements to be collected in each survey unit and the reference background area. MARSSIM Table 5-5 provides the number of measurements of samples for the case in which the radionuclide is not in the background.

Estimate the Sample Grid Spacing

The grid spacing for the measurement and samples is estimated in two ways, depending upon the shape of the grid (i.e., either triangular or rectangular grid). If a triangular grid is used, the grid spacing is estimated as follows:

$$L = (A/0.866N)^{1/2}$$

Where:

L = Distance between measurement locations
A = Survey unit Area
N = Number of measurements

If a square grid is used, the spacing is estimated as follows:

$$L = (A/N)^{1/2}$$

Perform Evaluation for Small Areas with Elevated Radioactivity

After the grid spacing has been calculated, the area between samples can also be calculated. For example, if for a square grid the grid spacing is 10 m, then there could be an undetected pocket (area) of elevated radionuclide concentrations 100 m^2. Adjustments to the grid spacing (i.e., additional sampling) may be necessary depending on the following three factors:

- The class of the survey unit
- The ability to scan for the radionuclide
- The minimum potential size of the elevated activity that could produce an exposure above the dose or risk criterion

Determine if the Number of Samples is Reasonable

After the number of samples per unit has been calculated, it will be evaluated to determine if the number is reasonable. It is possible, even if MARSSIM guidance is strictly followed, that there are too few samples to produce the desired level of comfort. The site managers and health physicists will be responsible for evaluating whether the number of samples is reasonable. If it is determined that the number of samples is inadequate or excessive, the data quality objectives will be reevaluated.

4.4.3 Scan Surveys

Following remediation and prior to conducting sampling, beta scans for surfaces and structures and gamma scans for environs will be performed over 100% of the Class 1 surfaces, 50% of the Class 2 surfaces and 25% of the Class 3 surfaces. A scanning response exceeding an action level (based on Section 6.8.2 of NUREG-1507) will be investigated/sampled/re-surveyed and, if necessary, remediated. If remediation is performed, scanning shall be repeated to demonstrate effectiveness of the remediation.

4.4.4 Soil Sampling

While soil sampling is not anticipated, if soil samples are later determined to be necessary, they will be obtained to a depth of 15 cm; samples will be packaged and uniquely identified in accordance with chain-of-custody and site-specific procedures.

4.4.5 Sample Analysis

Samples will be transferred to a radio-analytical laboratory for analyses in accordance with documented laboratory-specific standard methods. In accordance with MARSSIM, analytical techniques will provide a minimum detection level of 50% of the individual radionuclide $DCGL_W$ (or $DCGL_{EMC}$) values for all primary contaminants. If analyses indicate residual activity exceeding guideline levels, further remediation will be performed, as required, and scans and sampling of the remediated area will be repeated.

4.4.6 Investigation Levels

Radiation levels identified by scans that indicate potential residual radioactive contamination above background will be investigated to identify the source, level and extent of such residual activity. Areas that contain residual radioactivity concentrations of individual radionuclides, or sum-of-ratio concentrations above respective guideline values, will be remediated, reclassified (as necessary) and re-surveyed.

4.5 METHODS TO BE EMPLOYED FOR REVIEWING, ANALYZING, AND AUDITING DATA

4.5.1 Laboratory/Radiological Measurements Quality Assurance

During decommissioning survey activities, many direct and indirect measurements and sample media samples will be collected, measured and analyzed for radiological contaminants. The results of these surveys will be utilized to evaluate the suitability of the material or item for release to unrestricted use, or whether decontamination of structures, components, and the surrounding site have achieved the desired result. Sample collection, analysis, and the associated documentation will adhere to written procedures and meet NRC guidance, as well as comply with recognized industry recommendations and good practices. Laboratories selected to analyze decommissioning samples will be approved by WPI and listed on the contractor's QA Approved Suppliers List.

The selected contractor will need to implement a QA program that meets the requirements under 10 CFR 71, *Packaging and Transportation of Radioactive Material,* Subpart H, *Quality Assurance* (Ref. 9). In addition, the contractor's QA program must meet the applicable criteria from 10 CFR 50, Appendix B (Ref. 8) and ASME NQA-1 (Ref. 26). One of the applicable criteria that must be included is a QA Approved Suppliers List; the contractor will be required to maintain the QA Approved Suppliers List. WPI will assess the effectiveness of the QA program either through direct audits performed by WPI's Reactor Director, RSO, or Decommissioning Consultant, or by the acceptance of audits performed by other organizations.

Organizations that perform radiological monitoring measurements recognize the need to establish quality assurance programs to assure that radiological monitoring measurements are valid. Such programs are established to:

(1) Readily identify deficiencies in the sampling and measurement processes to those individuals responsible for these activities so that prompt corrective action can be taken, and

(2) Routinely monitor the survey and laboratory measurement results in order to assure that results and conclusions are valid

4.5.2 Supervisory and Management Review of Results

Health Physics technicians who are trained and qualified will conduct radiological surveys. In addition, senior-level Health Physicists other than the individual that performed the survey will review radiological surveys and sample results. FSS data will also be independently reviewed by the Decommissioning Consultant.

5.0 TECHNICAL SPECIFICATIONS

In August 2008, NRC granted WPI a license amendment to remove its authorization to operate the reactor (Ref. 4). That amendment authorizes WPI's possession only of the reactor and used fuel, and amended the Technical Specifications to remove operational requirements not needed for a Possession-Only License (POL).

After the nuclear fuel is removed from the reactor and shipped off-site and NRC issues the Decommissioning Order, WPI can commence decommissioning operations. At that time, many of the technical specifications for the POL will no longer apply. Proposed revisions to the Technical Specifications for the decommissioning period are given in Appendix B.

6.0 PHYSICAL SECURITY PLAN

The existing physical security plan that is approved by NRC will continue to be implemented during the decommissioning period. That existing plan meets the requirements in NUREG-1537, Chapter 17 (Ref. 5). All radiologically controlled areas are secured from unauthorized entry. The LCWNRF is normally locked during non-working hours, if unattended. WPI maintains routine, periodic police surveillance of the reactor site.

7.0 EMERGENCY PLAN

As required by NRC, WPI has a Reactor Facility Emergency Plan for responding to emergencies at the reactor facility. The purpose of the Emergency Plan is to minimize the effect of any emergency on the public, personnel, reactor facility and the environment surrounding the facility. Removal of spent fuel from the site will significantly reduce the potential for significant release of radioactive material off site. Any airborne or liquid releases due to decommissioning activities would have negligible impact off-site. The most likely accident scenario is a contaminated and/or injured individual. This scenario is adequately addressed by the existing emergency plan. Training will be provided to key personnel to ensure their familiarity with the Emergency Plan and their expected responses.

8.0 ENVIRONMENTAL REPORT

This section presents a summary of information relevant to the Environmental Review process, as described in NUREG-1748 (Ref. 32). Based upon the work scope and approach described in this decommissioning plan, the potential for observing negative impacts to the environment during the decommissioning of the WPI research reactor are small or not applicable. As such, the information presented herein has been intentionally limited, in scope, to that necessary for demonstrating the lack of significant environmental impact. In that context, impacts are defined as being effects upon the following environmental resources: land use, aquatic ecology, terrestrial ecology, water use and quality, air quality, waste, human health, socioeconomic, transportation, aesthetics, historical and archeological resources, and environmental justice.

Due to several factors, many of the potential environmental impacts that would normally be associated with a decommissioning project are not be applicable to the WPI decommissioning program. Such factors include: the small size of the facility, the limited scope of the planned D&D work, the short duration of the proposed work, and the small radiological inventory within this facility.

Decommissioning work at the WPI reactor facility will be limited to the reactor's structures, systems and components associated with the reactor's biological shield / pool, that are housed within a classroom area with a 1,200 ft^2 footprint. No changes will be made to the building's structure and no soil or ground water contamination is present. Decommissioning will principally entail disassembly of small metallic components, with a very limited amount of destructive removal work. As such, there will be no disruption or contact with the outside environment. This decommissioning work is not expected to cause any perceivable negative environmental impacts. The rational for this conclusion is based on the fact that the decommissioning work scope is small and of limited extent, has very low inherent radiological hazards due to a small inventory of radioactive materials, and will produce a small amount of radioactive waste materials. These statements are supported by the following determinations obtained during preparation of the Decommissioning Plan:

<u>Nature and Extent of the Decommissioning Work Scope</u>

- The scope of the decontamination and dismantling work is limited to an 8-foot by 8-foot by 15-foot deep reactor pool, within a five-foot thick concrete biological shield.
- Radioactive contamination within the facility is low and limited to the interior of the reactor pool and a small demineralizer system.
- No decontamination is expected to be required for the building structures.

- There will be no radiological work external to the room which houses the reactor and there will be no outdoor radiological work.
- Soil and ground water contamination is not present at the facility.
- The majority of the D&D work will entail disassembly of small, slightly activated or contaminated components by non-destructive method (e.g., unbolting or unthreading) and simple saw cutting of slightly contaminated metals. No thermal cutting or otherwise airborne-producing cutting techniques will be used or permitted.
- No structural changes will be made to the building that houses the reactor room.
- During the course of the D&D work, no perceivable impacts (e.g., noise or staging of materials and equipment) external to the building are expected.
- Radiological D&D work is estimated to require less than 160 work-days of effort.
- Decommissioning work can be accomplished by a very small, hands-on work crew, generally 3 or 4 individuals, with an estimated 6,800 person-hours of work; plus 4,100 person-hours of associated technical and management support.

Facility Radiological Conditions

- The reactor was of very low power, with limited use (approximately 40 MW-hours over 40 years) which resulted in the production of very little residual radioactivity by neutron activation.
- The extent of radioactivity within the facility is limited, with the majority of the radioactivity contained within a single, activated stainless steel regulating control blade having a mass of about 7,000 grams, with an estimated 56 mCi of radioactivity (15 mCi of Co-60). Radiation levels from the blade are estimated to be 17 mrem/hour at one meter (air), based on a measured 125 mrem/hr at 13 cm (through water).
- All other radioactive reactor core components and experimental facilities are estimated to contain significantly lower levels of radioactivity.
- No significant activation of the concrete biological shield is expected.
- Worker radiation exposure is estimated to total less than 0.5 person-rem.
- Generation of airborne radioactivity is not expected nor will it be permitted, and no air effluents containing radioactivity are anticipated.

Radioactive Waste Generation and Disposal

- Less than 600 cubic feet of solid radioactive waste are estimated to be produced, containing less than 60 mCi of radioactivity. The majority of this radioactivity (56 mCi) is contained in a single 7 Kg stainless steel control blade, comprised of activated metals (Mn-54, <1%; Co-60, 27%; Fe-55, 68%, Ni-59, <1%; and Ni- 63, 4.2%).
- All radioactive wastes are expected to be 10 CFR 61 Class A wastes.

Based upon the preceding information, WPI has assessed the potential for environmental impacts on environmental resources. The results of that assessment are presented in the Table 8-1. The results of this assessment use following definitions as a measure of environmental impact:

NONE- Not applicable, environmental effects on the resource are not possible or credible given the site specific conditions.

SMALL- Environmental effects are not detectable or are so minor that they will neither destabilize important attributes of the resource.

MODERATE- Environmental effects are sufficient to alter noticeably, but not destabilize important attributes of the resource.

LARGE- Environmental effects are clearly noticeable and are sufficient to destabilize important attributes of the resource.

TABLE 8.1

SUMMARY OF ENVIRONMENTAL IMPACT ASSESSMENT

ENVIRONMENTAL RESOURCE	IMPACT	JUSTIFICATION
Land Use	None	No outdoor work will be performed. Current use of the property – a university campus – will not change as a result of the decommissioning.
Ecology: Aquatic	None	No outdoor work will be performed. No contact will be made with the aquatic environment.
Ecology: Terrestrial	None	No outdoor work will be performed. No contact will be made with the terrestrial environment.

(Note: Table 8.1 is continued on next page)

TABLE 8.1 (continued)
SUMMARY OF ENVIRONMENTAL IMPACT ASSESSMENT

ENVIRONMENTAL RESOURCE	IMPACT	JUSTIFICATION
Water Use and Quality	Small	The decommissioning will require disposal of approximately 7,000 gallons of pool water via the public sewer system. Historically the pool water has not contained any detectable radioactivity. As a precaution, pool water will be filtered, sampled and analyzed for radioactive content, and batch released to assure that 10 CFR 20 effluent limits are not exceeded.
Air Quality	None	Radioactive contamination levels are very low within the facility, with the majority contained in solid, activated metallic components that can be dismantled without the need for segmentation. Dismantling techniques that could produce dust, mists or fumes will be prohibited. Air quality within the reactor classroom will be monitored and sampled for airborne radioactivity to assure and verify that no emissions of radioactivity to the environment are possible. Use of internal combustion powered equipment, such as compressors or generators, that would generate emissions are not planned.
Waste	Small	Less than 600 cubic feet of radioactive waste is estimated to be generated. No hazardous or mixed wastes are expected to be generated. All waste materials are estimated to be a small fraction of 10 CFR 60 Class A upper limits. Wastes are planned to be disposed of at the Energy*Solutions*, LLC facility in Clive, Utah.
Human Health	Small	Occupational radiation exposure to five decommissioning workers is estimated at less than 0.5 person-rem: to be incurred over 6,800 person-hours. No radiation exposure to the general public is expected via direct radiation or air and liquid effluents. There is a small risk of worker injury during performance of the decommissioning work, similar to that of construction work. The probability of an injury leading to a fatality is very small. Based upon the frequency of fatalities for construction workers, 10.3 per 100,000 workers per year, as reported by the Bureau of Labor Statistics for 2007 (Ref. 33), the probability for a fatality for this decommissioning project, with 6,800 person-hours of construction type work, is estimated to be 0.00035.

(Note: Table 8.1 is continued on next page)

TABLE 8.1 (continued)

SUMMARY OF ENVIRONMENTAL IMPACT ASSESSMENT

ENVIRONMENTAL RESOURCE	IMPACT	JUSTIFICATION
Socioeconomics	Small	No workers will be displaced by shutdown and decommissioning of the reactor. The reactor facility has a minimal part time staff (an RSO and an SRO), who have other duties at WPI. Decommissioning of the reactor will likely have a positive socioeconomic impact, in that these staff will be available for increased educational activities. Additionally, a small and temporary positive economic impact may be provided to the local area through employment of decommissioning workers, purchase of materials, supplies and equipment, and spending on living expenses at hotels and restaurants.
Transportation	Small	Only one truck shipment is anticipated: to transport the radioactive waste to the disposal site.
Aesthetics	None	All decommissioning work will be limited to the interior of the reactor classroom. There will be no outdoor work performed. No outdoor storage of material or equipment is anticipated. No excessively noise producing dismantling techniques is anticipated.
Historical and Archeological	None	No structural or appearance changes will be made to the building. Decommissioning work is limited to one classroom within the building.
Environmental Justice	None	The decommissioning work will not involve contact with, or disruption of, any surrounding neighborhoods.

9.0 CHANGES TO THE DECOMMISSIONING PLAN

WPI requests that changes to the Decommissioning Plan (DP) be allowed with local approval by the Reactor Director and RHSC, and without prior NRC approval, unless an un-reviewed safety question is involved. An un-reviewed safety question involves:

1. The increase in probability of occurrence or the increase of consequences of an accident or malfunction of equipment important to safety compared to that situation previously evaluated in the original Safety Analysis Report (SAR) for the Leslie C. Wilbur Nuclear Reactor Facility, or

2. The possibility for a type of accident or malfunction differing from those previously analyzed in the SAR, or

3. The reduction in margin of safety as defined in the SAR.

Reports and records of changes to the DP, and retention of documents, will be in accordance with the applicable portions of 10 CFR 50.59 (Ref. 31).

10.0 REFERENCES

1. WPI discontinued routine operation of the reactor on June 30, 2007. Certification letter sent to the NRC on 6-27-07 (copied to Alexander Adams) signed by David Adams, Mike Curley, and Jeffrey Solomon.

2. WPI submitted an application to NRC for a possession-only license. Certification letter sent to the NRC on 8-13-07 (copied to Alexander Adams) signed by David Adams, Mike Curley, and Jeffrey Solomon.

3. WPI submitted additional correspondence to NRC in support of an application for a possession-only license. Certification letters sent to the NRC on 11-19-07, 4-10-08, and 5-5-08 (copied to Alexander Adams) signed by David Adams, Mike Curley, and Jeffrey Solomon.

4. NRC Granted WPI a Possession-Only License Amendment No. 11 to R-61. Received at WPI 9-2-08.

5. Nuclear Regulatory Commission (U.S.), "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, NUREG-1537 Rev. 0, February 1996.

6. Nuclear Regulatory Commission (U.S.), "NMSS Decommissioning Standard Review Plan," NUREG-1727, September 2000.

7. WPI requests DOE take possession of the fuel Email from David Adams to Douglas K. Morrell on 6-25-07; verification email from Douglas K. Morrell to David Adams 6-25-07.

8. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Code of Federal Regulations.

9. 10 CFR 71, Subpart H, "Packaging and Transportation of Radioactive Material – Quality Assurance," Code of Federal Regulations.

10. ANSI/ASQC E4-1994, "Specifications and Guidelines for Quality Systems for Environmental Data Collection and Environmental Technology Programs," American Society for Quality Control, 1995.

11. Nuclear Regulatory Commission (U.S.), "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," NUREG-1575, Rev. 1, August 2000.

12. 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use," Code of Federal Regulations.

13. B. Shleien (editor), "The Health Physics and Radiological Health Handbook, Revised Edition, Scinta, Inc, Silver Spring, MD, 1992.

14. 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," Code of Federal Regulations.

15. 10 CFR 20 Subpart H, "Standards for Protection Against Radiation - Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas," Code of Federal Regulations.

16. Nuclear Regulatory Commission (U.S.), "Acceptable Programs for Respiratory Protection," Regulatory Guide 8.15, Rev. 1, October 1999.

17. Nuclear Regulatory Commission (U.S.), "Manual of Respiratory Protection Against Airborne Radioactive Materials," NUREG-0041, Rev. 1, January 2001.

18. 29 CFR 1910.134, Subpart I, "Occupational Safety and Health Standards - Personal Protection Equipment - Respiratory Protection," Code of Federal Regulations.

19. 10 CFR 20 Subpart E, "Standards for Protection Against Radiation - Radiological Criteria for License Termination," Code of Federal Regulations.

20. Nuclear Regulatory Commission (U.S.), "Consolidated NMSS Decommissioning Guidance, Decommissioning Process for Materials Licensees," NUREG-1757, Volume 1, Rev. 2, September 2002.

21. COMPASS Code Version 1.0.0, Developed by Pacific Northwest National Laboratory, Richland, Washington, under the sponsorship of the U.S. Nuclear Regulatory Commission for implementation of MARSSIM in support of the decommissioning license termination rule (10 CFR Part 20, Subpart E). BNWL-1264, March 1970.

22. 10 CFR 20.2003, Subpart K, "Standards for Protection Against Radiation - Disposal by Release into Sanitary Sewerage," Code of Federal Regulations.

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

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

25. 10 CFR Part 61, Subpart D, "Licensing Requirements for Land Disposal of Radioactive Waste - Technical Requirements for Land Disposal Facilities," Code of Federal Regulations.

26. ASME NQA-1-2000, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers, New York, NY, 2001.

27. 49 CFR Parts 100-177, "Research and Special Programs Administration, Department of Transportation," Code of Federal Regulations.

28. Federal Occupational Safety and Health Acts (OSHA) of 1973, Title 29 United States Code, Chapter 15, "Occupational Safety and Health."

29. Nuclear Regulatory Commission (U.S.), "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria," NUREG-1757, Volume 2, Rev. 1, 2003.

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

31. Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Domestic Licensing of Production and Utilization Facilities - Changes, Tests and Experiments," U.S. Nuclear Regulatory Commission, Washington, D.C., November 2000.

32. NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs," U.S. Nuclear Regulatory Commission, August 2003.

33. "National Census of Fatal Occupational Injuries in 2007," Bureau of Labor Statistics, United States Department of Labor.

APPENDIX A

LETTER OF FINANCIAL COMMITMENT

Office of the Executive
Vice President

100 Institute Road
Worcester, MA 01609-2280, USA
508-831-5288, Fax 508-831-5064
www.wpi.edu

March 25, 2009

U. S. Nuclear Regulatory Commission,

I, Jeffrey S. Solomon, Executive Vice President and Treasurer of Worcester Polytechnic Institute (WPI), which is located at 100 Institute Road, Worcester, MA 01609-2280. This letter is being provided to assure WPI's financial commitment to fund the radiological decommissioning of the Leslie C. Wilbur Nuclear Reactor Facility that is described in the Decommissioning Plan to which this letter is attached.

As documented in WPI's Decommissioning Plan, the radiological decommissioning cost, including a 25% contingency, is estimated to be $2,600,000. These funds are available and will be obtained from WPI's unrestricted capital project funds. As Executive Vice President and Treasurer of WPI, I commit to spending these funds to accomplish the decommissioning.

As Executive Vice President and Treasurer of WPI, I have the authority to commit and spend these funds for the decommissioning of the Leslie C. Wilbur Nuclear Reactor Facility. I certify the statements made in this letter are true and made freely under penalty of perjury.

Sincerely,

Jeffrey S. Solomon

Executive Vice President and

Treasurer

Worcester Polytechnic Institute

APPENDIX B

REVISED TECHNICAL SPECIFICATIONS FOR DECOMMISSIONING

(Edited from the 9-2-08 Possession Only License Amendment)

APPENDIX A

TO LICENSE NO. R-61

TECHNICAL SPECIFICATIONS FOR THE

WORCESTER POLYTECHNIC INSTITUTE REACTOR

DOCKET NO. 50-134

TABLE OF CONTENTS

The actual values of dimensions, measurements, and other numerical values may differ from values given in these specifications to the extent of normal construction and manufacturing tolerances, or normal accuracy of instrumentation.

1.0 DEFINITIONS

~~Cold, Clean, Critical Condition: Since xenon and samarium effects are negligible for this reactor in its normal operations, the term cold, clean, critical shall refer to the condition of the reactor core when it is at the normal ambient water temperature of 70 to 75°F and free of any experiments that could affect reactivity.~~

~~Critical Reactor Operation: Critical reactor operation shall refer to any situation when more than 12 fuel elements are loaded in the core and any control blade is withdrawn more than 6 in.~~

~~Experiment: An experiment shall mean any apparatus, device, or material installed in the core or in external experimental facilities that is not a normal part of those facilities.~~

~~Movable Experiment: A movable experiment is one that may be inserted, removed, or manipulated while the reactor is critical.~~

Operable: An instrument or channel is operable when the instrument or channel will be operational once it is energized.

Operational: An instrument or channel is operational when that instrument or channel is installed, energized, and in all other respects performing the monitoring and safety functions for which it was intended.

~~Reactor Operation: Reactor operation shall be any condition wherein either the reactor key is inserted into the console lock or the reactor is not in a shutdown condition.~~

~~Reactor Safety System: The reactor safety system is that combination of control channels and associated circuitry that forms the automatic protective system for the reactor or provides information that requires manual protective action to be initiated.~~

~~Reactor Scram: Reactor scram shall be the rapid insertion of the three control blades into the core by either of the following methods:~~

~~(1) Relay (slow) scram: Reactor relay scram (slow scram) shall be instigated by the relay scram circuits which control current inputs for the trip amplifier. Interruption of this current shall de-energize the scram magnets.~~

~~(2) Electronic (fast) scram: Reactor electronic scram (fast scram) shall be caused by the application of sufficient negative bias in the trip amplifier to terminate current to the scram magnets.~~

Readily Available on Call: Readily available on call shall mean an individual on duty within a reasonable driving time (1/2 hr) from the reactor building, that can be contacted to fulfill duties as specified in the implementing plans and procedures.

Reportable Occurrence: A reportable occurrence is any of the following conditions:

(1) a safety system setting less conservative than the limiting setting established in the Technical Specifications;

(2) operation in violation of a limiting condition for operation established in the Technical Specifications;

(3) a safety system component malfunction or other component or system malfunction that during operation could, or threatens to, make the safety system incapable of performing its intended safety functions;

~~(4) release of fission products from a failed fuel element;~~

(5) an uncontrolled or unplanned release of radioactive material from the restricted area of the facility;

(6) an uncontrolled or unplanned release of radioactive material that results in concentrations of radioactive materials within the restricted area in excess of the limits specified in Appendix B, Table 1 of 10 CFR Part 20;

~~(7) an uncontrolled or unanticipated change in reactivity in excess of 0.5% $\Delta k/k$;~~

(8) conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the facility; and

(9) an observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

~~Shutdown Condition:~~
~~The reactor shall be deemed to be in the shutdown condition if no control or regulating blade is withdrawn from its fully inserted position or if there are less than 12 fuel elements loaded on the grid plate.~~

2.0 SAFETY LIMITS AND OPERATING RESTRICTIONS

2.1 Safety Limits

Radiation Alarms: Upon indication of radiation levels in excess of 50 mrems/hr ~~(20 mrems/hr for fuel storage)~~ area monitors shall actuate audible evacuation alarms in the reactor room and in the second and third floor areas above the reactor pool.

Radiation Levels: The maximum radiation levels 1 m above the pool surface and at the surface of the concrete shield, when the beam port and thermal column are closed, shall be less than 50 mrems/hr.

Water Level: ~~The minimum depth of water above the top of the end box of the fuel elements in the reactor pool shall be 10 ft. When fuel is not in the pool, the water level shall be maintained to keep radiation levels as low as reasonably achievable (ALARA). The pool water level detector and alarm will be operational, with the alarm set-point being a water drop of less than or equal to 1 foot. The pool water level detector and alarm may be by-passed provided a licensed senior reactor operator is present in the facility.~~

~~Water Purity: Corrective action shall be taken promptly if the following limits for the pool water are not met.~~

~~(1) pH less than 8.0 and greater than 6.0~~
~~(2) resistivity greater than 5 x 10^{-5} ohm-cm~~
~~(3) pool water activity less than 10^{-5} uCi/ml~~

~~Water Temperature: The maximum bulk water temperature of the reactor pool shall be 110°F and the minimum shall be 40°F, when fuel is in the pool.~~

3.0 SURVEILLANCE REQUIREMENTS

3.1 Frequency of Surveillance

Quarterly: The area radiation monitoring systems ~~and the pool water level switch~~ shall be checked and ensured to be operational quarterly.

~~Semiannually: At least semiannually, during possession of the facility, a reactor inspection shall be performed consisting of~~

~~(1) Pool water pH shall be measured and conductivity and pH devices shall be calibrated.~~

3.2 Action to be Taken

If maintenance or recalibration is required for any of the items, it shall be performed and the instrument shall be rechecked before being placed back in service.

3.3 Radiation Detection

Area Monitors: Area radiation sensors capable of detecting gamma radiation in the range of 0.1 to 100 mrems/hr shall be installed near the beam port, demineralizer, thermal column door, ~~fuel storage area~~, and less than 1 m above the core pool surface, until these items have been removed or decontaminated. Upon indication of radiation levels in excess of 50 mrems/hr ~~(20 mrems/hr for fuel storage)~~ these monitors shall actuate audible alarms in the reactor room and in the second and third floor areas above the reactor pool. Area monitors shall be calibrated on a semi-annual basis.

Portable area monitors capable of detecting gamma radiation in the range of 0.10 to 50 mrems/hr may temporarily replace fixed area monitors described above provided that the required alarms are operational.

Portable Monitors: During fuel handling or other operations involving or potentially involving sources of radiation, operable portable survey instruments shall be readily available to the reactor operator for measuring beta-gamma exposure rates in the range 1.0 mrems/hr to 50 rems/hr, ~~and fast plus thermal neutron dose rates from 0.04 to 1,000 mrems/hr~~. One or more portable survey instruments for measuring beta-gamma exposure rates in the range 10 mrems/hr to 50 rems/hr will be kept available to the reactor staff in an external location (normally the security office) to facilitate obtaining radiation readings if a reactor radiation alarm should be activated.

4.0 SITE AND DESIGN FEATURES

4.1 Site

The reactor and associated equipment is housed in the Washburn Laboratories located between West Street and Boynton Street on the campus of Worcester Polytechnic Institute in Worcester, Massachusetts.

4.2 Restricted Area and Exclusion Area

The reactor room shall constitute a restricted area as defined in 10 CFR Part 20 and shall be controlled by partitions and normally locked doors. In addition, two small areas, one each on the second and third floors of Washburn Laboratories, directly above the reactor control drives, shall become restricted areas whenever the radiation levels in any of the rooms exceed those specified in 10 CFR 20.1301. The exclusion areas, as defined in 10 CFR Part 100, shall consist of the reactor room and the areas above the reactor.

4.3 Reactor Building and Ventilation System

The reactor shall be housed in a closed room that is designed to restrict leakage. The ventilation system shall provide at least two changes of air per hour in the reactor room whenever the reactor is operating.

4.4 Reactor Core

~~Fuel Elements: Standard fuel elements shall be flat plate type consisting of uranium-aluminum alloy clad with aluminum. The width and depth of each fuel element shall be 3 in. x 3 in. Each element shall have an active length of 24 in. There shall be a maximum of 10 g of U-235 in each fuel plate and not more than 170 g of U-235 in any fuel element. The fuel shall be enriched to less than 20% U-235. Standard fuel elements have 18 fuel plates, each plate 1.52mm thick with a clad thickness of 0.38 1 mm on each side. No fuel elements may be installed in the core.~~

5.0 ADMINISTRATIVE AND PROCEDURAL REQUIREMENTS

5.1 Facility Administrator

The Director of the Nuclear Reactor Facilities shall have full responsibility for maintaining the facility in a safe configuration. The Director shall report to the Provost and shall be responsible to the Radiation, Health, and Safeguards Committee for conformance to the facility license provisions and all local and NRC safety regulations. The Director also shall be responsible for proper maintenance of such records and operating practices as the Committee may deem necessary for the safe storage of the facility.

5.2 Radiation, Health, and Safeguards Committee

A Radiation, Health, and Safeguards Committee (RHSC) shall review, approve and document all proposed modifications affecting reactor safety and procedures, pursuant to 10 CFR 5O.59. This committee also shall conduct, at least quarterly, reviews of operations, equipment performance, records, and procedures. The Committee shall establish written procedures regarding review methods, quorums, and subcommittees, and it shall maintain written records of its activities. The members of the Committee shall be appointed by the Provost of Worcester Polytechnic Institute (WPI) and a majority shall be WPI faculty members.

5.3 Radiological Safety Officer

A Radiological Safety Officer shall be appointed to serve on the Radiation, Health, and Safeguards Committee and to review and approve all proposed procedures ~~and experiments~~ concerning radiological safety. The Radiological Safety Officer shall advise the Director of the Nuclear Reactor Facilities of rules, regulations, and procedures relating to radiological safety and shall routinely conduct radiation surveys.

5.4 Fire Protection

The licensee shall provide heat or ionization-type smoke detectors, which will alarm when there is a fire in the reactor room. At least two such detectors shall be operable at all times.

5.5 Procedures

Detailed written procedures shall be provided for all normal decommissioning operations ~~of the reactor~~, operation of supporting facilities, maintenance operations, radiation protection, ~~experiments,~~ and emergency plans and operations. These procedures shall be approved by the Radiation, Health, and Safeguards Committee before they are implemented.

Temporary procedures that do not change the intent of the initial approval procedures may be authorized by two members of the facility staff at least one of whom shall be a licensed senior operator. Such procedures shall be subsequently reviewed by the Radiation, Health, and Safeguards Committee,

5.6 Operating Records

In addition to records required elsewhere in the license application, the following records shall be generated and kept of:

(1) maximum radioactivity released or discharged into the air or water beyond the effective control of the licensee as measured at or before the point of such release or discharge;

~~(2) maintenance operations involving substitution or replacement of reactor equipment or components; and~~

(3) tests and measurements performed pursuant to the Technical Specifications.

Old records pertaining to operation of the reactor, including power levels, emergency shutdowns, inadvertent scrams, experiments, and in core irradiations, shall be kept for purposes of decommissioning.

5.7 Reports

In addition to reports otherwise required under this license and applicable regulations—

(1) The licensee shall inform the Commission of any incident or condition relating to the safe storage of the facility that prevented or could have prevented a safety system from performing its safety function as described in the Technical Specifications. For each such occurrence, WPI shall promptly notify, by telephone or telegraph, the Administrator of the appropriate NRC Regional Office Listed in Appendix U of 10 CFR Part 20 and shall submit within 10 days a report in writing

to the Director, Division of Waste Management and Environmental Protection (DWME.P), with a copy to the Regional Office.

(2) The licensee shall report to the Director, DWMEP, in writing within 30 days, any observed occurrence of substantial variance of conditions from performance specifications contained in the Safety Analysis Report or the Technical Specifications.

(3) The licensee shall report to the Director, DWMEP, in writing within 30 days, any occurrence of significant changes in transient or accident analysis as described in the
SAR.

5.8 Annual Operating Reports

A report covering the previous year shall be submitted to the Administrator of the appropriate
Regional Office not later than March31 of each year. It shall include:

(1) Operations Summary: a summary of issues having safety significance occurring during the reporting period, including:

(a) changes in facility design
(b) performance characteristics (e.g., equipment and fuel performance)
(c) changes in operating procedures that relate to the safety of facility operations
(d) any abnormal results of surveillance tests and inspections required by these Technical Specifications
(e) a brief summary of those changes, tests, and experiments that did not require authorization from the Commission pursuant to 10 CFR 50.59 (d)(2)
(f) changes in the plant staff serving in the positions of Reactor Facility Director, SRO, RSO, ARSO, or Radiation, Health, and Safety Committee members;

(2) Maintenance: a discussion of corrective maintenance (excluding preventative maintenance) performed during the reporting period on safety related systems and components;

(3) Changes, Tests, and Experiments: a brief description and a summary of the safety evaluation for those changes, tests, and experiments that were carried out without prior Commission approval, pursuant to the requirements of 10 CFR 50.59 (d)(2); and

(4) Radioactive Effluents Releases: a statement of the quantities of radioactive effluents released from the plant.

~~5.9 Fuel Storage~~

~~Two fuel storage racks are located on opposite sides of the reactor pool. Each rack shall be designed to contain not more than 18 fuel elements. When the reactor contains a critical mass, all additional fuel elements not in the core shall be locked in place except as authorized by the licensed senior operator in charge.~~

~~A fuel element shall not be stored outside of the reactor pool unless it produces radiation dose levels of less than 100 mrems/hr at the storage container surface. Storage containers of fuel elements shall be locked closed when unattended.~~

~~All fuel element transfers shall be conducted by a staff of not less than three persons, which shall include a licensed senior operator in charge and a RSO. Staff members will continuously monitor the operations using appropriate radiation monitoring and core nuclear instrumentation.~~